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Limited Scope Sustainability Assessment of Planned Nuclear Energy Systems Based on BN-1200 Fast Reactors

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LIMITED SCOPE SUSTAINABILITY ASSESSMENT OF PLANNED NUCLEAR ENERGY SYSTEMS BASED ON BN-1200 FAST REACTORS

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

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INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was launched in 2000 in response to resolution GC(44)/RES/21 of the IAEA General Conference. One of the INPRO objectives is to help to ensure that nuclear energy is available in the 21st century in a sustainable manner.

To meet this objective, INPRO developed a methodology for assessing the sustainability of a national or international nuclear energy system. This entailed establishing a set of INPRO basic principles pertaining to system sustainability, a set of INPRO user requirements in support of each INPRO basic principle and a set of criteria for meeting each INPRO user requirement. The basic principles, user requirements and associated criteria contained in the INPRO methodology are for use for INPRO assessments of system sustainability only. The INPRO methodology does not establish any safety requirements or criteria, or provide guidance on safety; IAEA safety requirements and guidance are issued only in the IAEA Safety Standards Series.

The INPRO methodology was published in 2008 as nine manuals providing an overview and covering the areas of economics, infrastructure, waste management, proliferation resistance, physical protection, environment, safety of nuclear reactors and safety of nuclear fuel cycle facilities. Since its initial publication, the INPRO methodology has been updated to take into account the experience of Member States that have performed national and international nuclear energy system sustainability assessments using the INPRO methodology as well as recommendations developed by the INPRO steering committee and IAEA experts. All the proposals were evaluated by internal and external experts at several IAEA meetings, including five technical meetings in 2013, 2014, 2016 (2) and 2017.

At the 20th INPRO Steering Committee meeting, in 2013, several Member States expressed interest in performing limited scope INPRO assessments of liquid metal cooled fast reactor designs. At the 21st INPRO Steering Committee meeting, in 2014, a new activity was endorsed and included in the INPRO action plan. Three consultancy meetings were organized in 2015 and 2017 to perform the assessment and review the results. The final draft report on the assessment of BN-1200 reactors, compiled by experts from the Russian Federation and submitted to the IAEA in 2019, served as the basis of the case study presented in the present publication. The case study applied the INPRO methodology areas of economics and reactor safety to an innovative fast reactor design at the full criteria level. The information in this publication is not to be used for formal or authoritative safety assessment or safety analysis to address compliance with the IAEA safety standards or for any national regulatory purpose associated with the licensing or certifications of nuclear power plants.

The IAEA officer responsible for this publication was A. Korinny of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Strategic planning for energy supply system evolution including nuclear energy therein requires a sound understanding of all issues in the areas of energy demand evaluation, general energy supply[1], and nuclear energy[2]. The deployment of a nuclear power programme has intergenerational implications and obligations extending well beyond 100 years. The IAEA offers tools and support for long-term energy system planning and nuclear energy system assessment. INPRO methodology is a tool for the assessment of sustainability of nuclear energy systems that was originally created between 2001 and 2003 under the aegis of the IAEA using broad philosophical outlines of the concept of sustainable development. INPRO basic principles, user requirements and criteria have been defined for assessing nuclear energy system sustainability in different areas, i.e. economics, infrastructure (legal and institutional measures), waste management, proliferation resistance, environmental impact of stressors, environmental impact from depletion of resources, safety of nuclear reactors and safety of nuclear fuel cycle facilities. The INPRO basic principles establish goals that should be met in order to achieve long term sustainability of a nuclear energy system. An INPRO user requirement of sustainability defines what different stakeholders (users) in a nuclear energy system should do to meet the goal defined in the basic principle. A criterion enables the assessor to check whether a user requirement has been met. Full scope application of the INPRO methodology is a holistic metric of sustainability of the nuclear energy system, defining a list of gaps to be closed to achieve sustainability.

In the last two decades the INPRO methodology has been revised several times and the latest full edition of the INPRO manuals was published in 2008 [1]. The latest revision and update of the methodology manuals has been commenced in 2012 and by 2020 the IAEA has published seven new INPRO methodology manuals for sustainability assessment in the areas of economics [2], infrastructure [3], waste management [4], environmental impact of stressors [5], environmental impact from depletion of resources [6], safety of nuclear reactors [7] and safety of nuclear fuel cycle facilities [8].

One possible output from the INPRO sustainability assessment of a nuclear energy system is the identification of areas where a given system needs to be improved. Given the comprehensive nature of an assessment using the INPRO methodology, such an assessment would be expected to indicate clearly the specific attributes of a nuclear energy system that need to be improved.

Unlike the full scope INPRO sustainability assessment the limited scope study can be more focused on the selected INPRO methodology areas and / or selected nuclear energy system installations under development checking whether this installation can meet the sustainability requirements in a given assessment area. Limited scope INPRO assessments are expected to be performed at relatively early stages of a nuclear energy system development and deployment when they can help to identify new complementary R&D studies and technical or institutional improvements which can be implemented with no excessive costs.

INPRO methodology has been applied several times to the nuclear energy systems based on evolutionary water cooled reactors [9]. A few early attempts to perform INPRO assessment of

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[1] For example, resources, climate change effects and national development.

[2] For example, nuclear safety, infrastructure, economics, waste management.

innovative reactors including sodium cooled fast reactor did not involve a full depth consideration of the design related criteria mainly because of lack of input data [10].

1.2. OBJECTIVE

This publication provides an example of the limited scope INPRO sustainability assessment of an innovative nuclear energy system using the fast reactor BN-1200 as a case study. The INPRO assessment performed at the full depth criteria level helped to identify actions, including potential research development and demonstration, for sustainable long term deployment of sodium cooled fast reactors.

This publication discusses the application of the INPRO sustainability assessment method to the innovative nuclear energy system based on fast reactor BN-1200 in the areas of economics and safety of nuclear reactors. The case study is intended to verify readiness of the updated INPRO Methodology for assessment of the sodium cooled fast reactors and to develop recommendations for further improvements and updates of the INPRO assessment method.

This publication is intended for use by organizations involved in the development and deployment of the nuclear energy systems including planning, design, modification, technical support and operation for nuclear power plants. Data provided in this publication can be used in further detailed INPRO sustainability assessments of the nuclear energy systems based on BN-1200 reactors, sustainability assessments of other fast reactors and in scenario modelling studies involving fast reactors which can be carried out by the technology holders and technology users.

1.3. SCOPE

Limited scope INPRO sustainability assessment of sodium cooled fast reactors was performed in 2015–2019 in parallel as a series of bilateral studies between the developers of fast reactors and the IAEA in a few countries developing such reactors. Every study was conducted as a self-assessment exercise performed by the national designer experts focused on their own design and supported by the IAEA staff.

This publication presents the results of the case study of the INPRO assessment of BN-1200 reactor in the INPRO areas of economics and reactor safety. The BN-1200 assessment has been performed by the Russian Federation experts from the Institute of Physics and Power Engineering with the support provided by JSC Afrikantov OKB Mechanical Engineering. It is based on assessors' experience and publicly available data, taking into account proprietary information concerns.

This INPRO methodology sustainability assessment study is focused on the nuclear power plants that produce primarily electricity, heat or combinations of the two[3]. This publication does not explicitly consider economics and safety issues related to other non-electric applications (hydrogen production, desalination, etc.) or to cogeneration involving such energy products. It is expected that as more detailed information is acquired on the safety of interactions between a reactor and industrial facilities located on the same site, the INPRO criteria and the assessment studies may be modified accordingly.

[3] For example, all NPPs of Russian design provide small amounts of energy in the form of heat used for district heating, greenhouse heating, etc., by local communities.

The INPRO methodology user requirements and criteria for sustainability assessment are formulated in a generic manner to make them applicable to both evolutionary and innovative reactors based on different technologies. However, innovative reactors with a lower level of design maturity may require modifications or clarifications of selected criteria. Such potential changes will be considered in future revisions of the INPRO methodology after sufficient experience has been accrued from INPRO assessments of such reactors.

The INPRO methodology manual for sustainability assessment does not establish any specific safety requirements, recommendations or guidance. IAEA safety requirements and guidance are only issued in the IAEA Safety Standards Series. Therefore, the basic principles, user requirements and associated criteria contained in the INPRO methodology have been used for sustainability assessments only. This INPRO assessment study should not be used for formal or authoritative safety assessments or safety analyses to address compliance with the IAEA Safety Standards or for any national regulatory purpose associated with the licensing or certification of nuclear facilities, technologies or activities. Data presented in this report do not represent the national practices required in the regulations and should be considered as a proposal by the experts for possible approaches to assess sustainable development of nuclear energy systems.

The sustainability issues relevant to safety of the facilities of closed nuclear fuel cycle and waste management are outside of the scope of this publication.

1.4. STRUCTURE

This publication follows the relationship between the concept of sustainable development and INPRO methodology areas of economics and reactor safety. Section 2 provides general information on the fast reactor development programme in the Russian Federation, to set the context. Section 3 presents the INPRO sustainability assessment of BN-1200 in the area of economics. This includes an overview of the application of the INPRO methodology area of economics to the fast reactors under development, information on the improvement of economic characteristics of sodium cooled fast reactors in the Russian Federation, basic results of analysis and sustainability assessment of BN-1200 in the area of economics. Section 4 presents the INPRO sustainability assessment on the criterion level in the area of reactor safety including assessment of the design robustness, detection and interception of anticipated operational occurrences, design basis accidents, severe plant conditions, independence of levels of defence in depth, inherent safety characteristics, passive safety systems, human factors related to safety and necessary research, development and demonstration. Section 5 summarises the discussion and suggests conclusions on the performed study.

2. GENERAL INFORMATION ON THE FAST REACTORS CONSIDERED IN THE ASSESSMENT STUDY

Liquid metal cooled fast reactors development programme in the Russian Federation involves the following commercial sodium cooled fast reactors of BN lineage[4]:

- BN-350 prototype reactor constructed and operated (1973–1999) in Kazakhstan;
- BN-600 reactor operated since 1980 as unit 3 of the Beloyarsk NPP, Russian Federation;
- BN-800 reactor operated since 2016 as unit 4 of the Beloyarsk NPP, Russian Federation;
- BN-1200 reactor design; pilot unit is planned to be constructed at the Beloyarsk NPP site as unit 5.

Besides that, this programme includes two more types of the liquid metal cooled fast reactors:

- Lead-cooled fast reactor design concept BREST- OD-300 with on-site manufacturing of nuclear fuel and reprocessing of spent nuclear fuel;
- Lead-bismuth cooled reactor SVBR-100 (installed electrical power rating – 100 MW(e)) which is being developed under the public-private partnership framework.

The associated closed nuclear fuel cycle facilities / technologies have been developed in the Russian Federation:

- MOX fuel experimental fuel assemblies tested in BN-350 and BN-600 reactors (approx. 50 bundles with pellet fuel and approx. 30 bundles with vibro-packed fuel);
- Mining and Chemical Combine (GHK) commercial MOX fuel fabrication facility for BN-800;
- RT-1 spent fuel reprocessing facility (PUREX technology) for uranium oxide fuel from VVER-440 (water cooled reactors) and BN-600;
- Experimental technology for the nitride fuel fabrication;
- Laboratory level technology for the pyrochemical reprocessing of irradiated nuclear fuel.

This INPRO sustainability assessment focuses on BN-1200 reactor as an example case study. In the area of reactor safety, the BN-800 reactor was selected as a reference design for BN-1200 and in a few cases (e.g. references to the accrued operational experience) the BN-600 reactor data have been used. In the INPRO area of economics the BN-800 and BN-600[5] reactor data have been used for estimation of trends in economic characteristics of BN reactors.

General information on BN-600, -800, -1200 reactors is presented in this section for broader context of the study.

2.1. BN-600 REACTOR

The original plan for the development of BN-600 reactor was based on the following basic assumptions:

- BN-600 would have higher steam temperature and pressure (540ºC, 140 MPa) and increased electrical power rating compared to BN-350;

[4] BN in Russian stands for 'fast sodium'. BN lineage is a series of commercial sodium cooled fast reactors of Russian design.

[5] BN-600 data have been used mostly for evaluation of specific mass of metal used and specific capital costs at different stages of the BN technology development programme.

– The original basic concept and layout of BN-600 reactor were originally expected to be similar to BN-350 (loop type reactor);
– The highest electrical power rating of the standard turbo-generators designed for these parameters of steam and available at that time was 200 MW(e). The electrical power rating of BN-600 was defined as 3×200 MW(e). The reactor concept used three completely independent heat transport loops in the secondary and tertiary circuits.

The BN-600 design process modified the original conceptual ideas. Hence, the BN-600 reactor constructed at Beloyarsk NPP site contains significant improvements [11] in comparison to the original requirements.

Loop type reactor designs have been broadly used in many countries at the early stages of development of the fast reactors.[6] However, as sodium cooled fast reactor designs grew in installed power rating a loop type layout was revealed to have a number of complicated engineering problems[7].

Loop type fast reactor designs used relatively long pipelines of large cross section connecting the reactor vessel and intermediate heat exchangers. High temperature and radioactive primary sodium circulating in these pipelines change temperature at changes of reactor power level and creates essential tension/stress loads on the pipes and welds. Compensation of the thermal expansions by bending of the pipelines increases the primary circuit piping length and impedes the primary coolant natural circulation which is important for reactor cooling under the postulated station blackout conditions. Besides that, this does not provide an efficient solution since the mechanical stress in bends may become very close to the yield stress.

Moreover, the main pipeline tension/stress loads compensation forces propagate to the relatively thin walled casings of the pipes, vessels and nozzles occasionally deteriorating their stress-strain characteristics. The areas of the reactor nozzles seemed to be the most vulnerable parts and belonged to the group of most challenging components to fabricate since the stresses caused by the reactor coolant parameter cycles in these locations were the most frequent and the ranges of changing values were broad. Eliminating the nozzles provides an inherent safety feature associated with this hazard and seemed to be the most effective way to increase the robustness of the reactor design. Hence, a high power rating pool type reactor design seemed to have more reliability than a loop type designs.

The electric heating system associated with the primary coolant pipelines and the systems connected to the primary circuit of a loop type reactor had to involve sophisticated and expensive components for large isolating valves and casings of the pipes that cannot be isolated in the case of a leak. This equipment aimed to rule out any dangerous decrease in the reactor sodium level in case of the coolant leakage through the failed pipeline.

More challenges occur implementing fire and radiation protection from the sodium leaks, in particular those caused by the guillotine break of the primary pipeline – an accident scenario postulated to be accounted for by the national regulatory requirements.

Pool type layout has significant advantages compared against loop type sodium cooled fast reactors. The main reactor vessel design can avoid pipeline connections under the normal operation sodium level and accommodation of the primary circuit systems structures and

[6] France, Russian Federation, India and China currently develop the pool type designs of the fast reactors. Japan studies loop type layout (Monju and Joyo).

[7] Many of these problems, such as leaks of primary sodium, exist in small reactors either, however their significance/danger steeply grows with the increase of reactor power rating.

components within the reactor vessel essentially reduces the probability of radioactive sodium leaks to the confinement/ containment premises and makes the solution of leak-tightness problems much easier. The reactor vessel walls, bottom and support structures are designed to withstand mechanical forces caused by the weight of reactor vessel, sodium weight and the weight of reactor internals and fuel.[8] In the pool type design, forces coming from the thermal expansion of reactor pipelines and stressing the reactor vessel nozzles do not exist.

Placing the primary circuit systems, structures and components inside of the reactor vessel reduces the surface-to-volume ratio of the radioactive sodium and the length of welds where in the case of failure a leakage may occur. In addition to the apparently positive safety effects the cost of materials and the reactor fabrication efforts reduce significantly.

In the pool type fast reactor, accommodation of the primary circuit equipment within the large volume of sodium increases the system inertia and makes the system parameters more stable. All internals are immersed in sodium and small sodium leaks through the detachable joints between the reactor internals caused by the pressure difference on different sides can be accepted. Some of the walls of systems, structures and components placed inside the reactor vessel are not required to withstand significant strain, unlike those in the loop type reactors, and can be thinner and/or easier to fabricate. Thinner walls of reactor internals further reduce thermal stresses during the reactor transients.

Elimination of primary circuit pipelines or at least effective minimization of their length essentially reduces the cost of materials used, cost of equipment manufacturing and cost of NPP construction. Sealed compartments of the loop type design primary circuit can be eliminated, reducing the associated costs of ventilation and fire-protection systems, costs of electric heating system, costs of guard casings, thermal insulation and drains. The size and costs of several other systems can be essentially reduced, e.g. the biological shielding is required only for the reactor vessel and a few remaining pipelines and systems containing radioactive sodium. The in-vessel neutron shielding installed in the pool type reactors reduces the radiation dose to the reactor vessel and internals. It also allows for in-vessel spent fuel storage.

Another important feature of the pool type reactor is the possibility of passive removal of the residual heat in emergency situations [11]. Passive heat removal depends on natural circulation of the primary and secondary sodium achieved through complex thermal-hydraulic design considerations in the reactor, heat exchangers and steam generators. At the time of BN-600 design development the passive removal of residual heat through the main circuits (three channels) was considered to be reliable[9].

The deficiencies of pool type designs are mostly related to the size of reactor vessel and to the mass of in-vessel systems, structures and components. Pool type reactor vessels are normally too large to be manufactured at the fabrication facility and transported in one piece to the NPP site. In this situation the reactor vessels need to be manufactured on the site which makes this process more expensive and challenging. The size and mass of reactor can make it vulnerable to the seismic loads and may require special arrangements different from other types of reactors. Other challenges are associated, for example with the introduction of compact and reliable in-vessel neutron shielding having sufficiently long lifetime, with the reliable reactor vessel support structure and with the neutron flux monitoring necessary for the NPP power control [11].

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[8] Impacts from earthquakes and other dynamic loads are accounted for.

[9] BN-800 and BN-1200 use different residual heat removal schemes in emergency situations.

Several major concepts have been incorporated in the final design of BN-600 as follows:

- Pool type layout of the primary circuit. This configuration helps to simplify the design, resolve several engineering problems associated with large fast reactors, and provides the designer with conditions and tools which allow both safety improvements and cost reductions;
- New neutron absorbers. New shim rods with higher absorbing efficiency have been installed to compensate fuel burnup effects. It allows an extension of the time between refuelling and improvement of economic characteristics for the plant;
- New design of primary coolant pumps. The primary coolant pumps use a new design with a bottom hydrostatic bearing working under sodium. This design allows control of the pump speed, reduction of the pressure in the reactor vessel gas plenum and further simplification of the layout of the primary and secondary circuits;
- New design of steam generators. Once-through sectional-modular steam generators and sodium-steam reheating scheme provide robustness for the higher temperature and pressures of steam generated in BN-600. In the case of malfunctions of the heat exchanging components the sectional design of steam generators allows isolation of a given section and uninterruptable operation of others;
- Independent heat removal loops in secondary and tertiary circuits. This layout allows for higher flexibility of the operating regimes, e.g. several reactor start-up procedures were performed consecutively and separately in every loop. The reactor can operate at power levels less than 67% of full power using only two pairs of secondary and tertiary loops[10]. Operation of reactor at any power levels with only one pair of secondary and tertiary loops is not allowed.

The BN-600 reactor's first criticality was achieved in 1980. The power plant unit construction used general civil industrial type building construction requirements. BN-600 has three steam generators PGN-200M, three turbines of the K-200-12.8-3 type and three electric generators of the TGV-200M type. The BN-600 has a thermal power rating of 1470 MW(th) and the electrical power rating amounted 600 MW(e).

The reactor core fuel, blankets, neutron reflectors, the control and protection system including actuators, three primary coolant pumps, most of the primary coolant pipelines, six intermediate heat exchangers and associated structures and components are placed in the main reactor vessel filled with liquid sodium (primary coolant). The volume of primary sodium exceeds 800 m^3. The BN-600 uses enriched uranium oxide fuel, however it was designed to generate the 'secondary' nuclear fuel material (plutonium isotopes) in the reactor core and blankets.

The main reactor vessel is enclosed inside the guard vessel with the gap between these two vessels chosen to keep the sodium level in the main vessel from dropping too low in the case of main reactor vessel leak[11]. The guard vessel sits within a concrete chamber lined with a 10 mm thick steel. The top side of this chamber has a cover of an upper biological shielding.

Each secondary loop includes two intermediate heat exchangers located in the reactor vessel, a buffer tank compensating for sodium volume changes, a secondary coolant pump, pipelines and a sectional-modular steam generator. The volume of secondary sodium in every loop equals 280–300 m^3.

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[10] BN-600 has three loops in the secondary circuit and three corresponding loops in the tertiary circuit.

[11] Too low dropping of the sodium level in the main reactor vessel may cause losing of the coolant natural circulation through the reactor core.

The pressure of gas in the reactor vessel gas plenum is normally maintained at values lower than 0.2 MPa. The characteristics of the secondary circuit including the geometry of the loops have been selected in such a way that the static pressure (including gravity pressure set by the sodium level) on the secondary side of the intermediate heat exchangers exceeds the pressure on the primary side. These arrangements help to prevent potential accidental leaks of the primary sodium into the secondary circuit through the intermediate heat exchangers.

Every once-through type steam generator has an eight-section arrangement. Every section consists of the evaporation module, steam superheater module and reheater module. Each section can be disconnected from the secondary (sodium) and tertiary (steam/ water) loops when necessary.

The basic design of the BN-600 tertiary circuit is similar to fossil power plants or secondary circuits of pressurized water cooled reactors. Every loop of the tertiary circuit includes a steam-water part of the steam generator, a turbine with its auxiliary equipment, a condenser, a deaerator, three feedwater pumps with electric drives and an emergency feedwater pump.

Three independent pairs of secondary and tertiary loops in the BN-600 reactor provide for reactor cooldown during normal operation and in the case of emergency for a safety function of residual heat removal. This three-train design may create delays in the reactor maintenance processes. Hence, a special complementary cooldown system was considered for BN-600. When introduced, this system may be connected to the secondary loops through the sodium-air heat exchangers.

BN-600 operates in a base-load mode. The average load factor estimated in the reactor design documentation amounts 76% and this value corresponds to the actual performance achieved during the reactor operation once the necessary level of technology maturity had been achieved.[12]

2.2. BN-800 REACTOR

The long lasting and successful operation of the BN-600 reactor preceded the design of the BN-800, which uses most of the technologies developed and mastered at the design, commissioning and operation of the BN-600. Unlike BN-600, the BN-800 reactor design objectives involved the demonstrations of BN technology competitiveness against other energy supply options and the feasibility of industrial scale implementation of the closed fuel cycle technology.

Apart from the generation of electricity and district heating the BN-800 reactor has these design objectives:

- Operation of the reactor using mixed uranium-plutonium oxide (MOX) fuel which is important part of the closed fuel cycle technology deployment;
- Preservation and continuity of knowledge, practical skills and technologies in design, construction and operation of sodium cooled fast reactors;
- Support the research development and demonstration programmes of the sodium cooled fast reactor technologies, and the development of new fuels and of the reactor core structural materials in particular;
- Testing and validation of new systems, structures and components and new computer codes.

[12] Load factor achieved at the BN-600 operation exceeds 76% if the energy used for district heating is accounted for

BN-800 is an evolutionary upgrade of BN-600. The BN-800 based its normal operation systems including the sodium circuits design and sodium systems, safety systems (except new safety systems and numerous improvements), instrumentation and control systems including reactor monitoring systems on similar systems used in the BN-600. Operating and maintenance conditions, and procedures are similar to those in BN-600, and incorporate operational approaches and experience accrued over the 60 years of the national fast reactor programme.

Commissioned in 2016 (see Figure 1), the BN-800 reactor is equipped with three steam generators N-272, a turbine of the K-800-130/3000 type and an electric generator of the TZV-800-2 type. The BN-800 has a thermal power rating of 2100 MW(th) and an electrical power rating of 880 MW(e).

FIG. 1. The BN-800 power unit at the site of Beloyarsk NPP (courtesy of Rosatom).

BN-800 is a pool type reactor using a similar layout as the BN-600 reactor described above. The volume of primary sodium in BN-800 equals 1100 m^3. BN-800 can use enriched uranium oxide fuel and MOX fuel.

Like the BN-600, BN-800 uses the three circuit configuration for energy transfer and conversion. Sodium is the coolant in the primary and secondary circuits and water is the coolant in the tertiary circuit. Each circuit consists of three identical parallel loops.

Each secondary loop includes two intermediate heat exchangers located in the reactor vessel, a buffer tank compensating for sodium volume changes, a secondary coolant pump, an emergency heat removal system heat exchanger, pipelines and a sectional-modular steam generator. The volume of secondary sodium in every loop amounts approx. 350 m^3. The emergency heat removal system connects to every secondary loop through the sodium-sodium heat exchanger which is connected in parallel to the steam generator.

Every steam generator consists of ten sections. Every section consists of the evaporation module and the steam superheater module. BN-800 uses steam-steam reheating scheme. Each section can be disconnected from the secondary (sodium) and tertiary (steam/water) loops when necessary.

The layout of tertiary circuit and its systems structures and components have similarity to superheated steam turbines[13]. The tertiary circuit through the primary and secondary circuits and steam generators provides the removal of heat from the reactor at normal operation conditions, deviations from normal operation and in most accident scenarios. However, this heat removal scheme may fail in the cases of loss of feedwater, station blackout scenarios, earthquakes, etc. Unlike BN-600, the BN-800 uses a single feedwater supply system for all three steam generators which stipulates an emergency heat removal system connection in parallel to the steam generators[14].

The most extensive and sophisticated improvements of the BN-800 design were introduced in the systems, structures and components important to safety, and mostly incorporating new regulatory requirements. The design of BN-800 has been developed to be in compliance with the new or updated national regulatory documents on nuclear safety such as OPB-88/97, PBYa RU AS-07, SP-AS-03, etc. Classification, layout, design and construction of the NPP buildings and premises in accordance with the fire and explosion protection requirements complied with Russian fire safety standards norms documented in NPB 105-95. Nevertheless, all systems modifications and functions included the operational experience of previous installations [11].

BN-800 protective safety systems include:

- Emergency heat removal system;
- Reactor protection system;
- Reactor loss-of-coolant protection system;
- Reactor overpressure protection system;
- Secondary circuit overpressure protection system;
- Heat removal system for fuel assemblies for use during their reloading from the reactor to the spent fuel ‘drum’ storage;
- Spent fuel ‘drum’ storage heat removal system;
- Spent fuel ‘drum’ storage casing overpressure protection system;
- Guard casings of primary pressure gas pipelines and guard shell of the pressure chamber.

BN-800 confinement safety systems include:

- Reactor guard vessel;
- Reactor core catcher;
- Reactor confinement compartments and leak-tight enclosure;
- Guard casings of primary auxiliary systems pipelines;
- Primary sodium systems, structures and components compartments ventilation system and spent fuel ‘drum’ storage compartment ventilation system;
- Sodium systems, structures and components compartments fire protection system;
- Spent fuel ‘drum’ storage casing;

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[13] The BN-800 full power normal operation steam characteristics at the turbine entrance are 13 MPa and 495°C.

[14] To avoid loss of heat sink caused by the loss of feedwater

– Guard casings at the pipeline sections from the spent fuel ‘drum’ storage to the overflow vessel;
– Exterior lining of the spent fuel cooling pond.

In comparison with the BN-600, the modifications of the BN-800 reactor design include the following features[15] increasing the reactor systems reliability and improving safety:

– Sealed cover was introduced around the reactor pressure chamber;
– A reactor vessel bottom part and the reactor support structure[16] have been redesigned to improve seismic characteristics;
– The thickness walls of the reactor main vessel and guard vessel was increased from 20 to 30 mm;
– Reactor core catcher has been introduced to protect the reactor vessel from the molten fuel effects in the case of severe accident;
– Control rods with the passive actuation principles have been added to the reactor protection system;
– The primary coolant purification system has a stationary arrangement installed for separation and removal of caesium from sodium;
– System protecting the reactor from overpressure or accidental depressurization has been improved;
– New ionization chambers have been introduced to monitor the reactor core in the subcritical state;
– New reactor core design was proposed to minimize the value of sodium void reactivity effect;
– One rotating plug was added to the fuel reloading system[17]; however, the reloading system and reloading procedures have been simplified and one in-vessel elevating machine was eliminated;
– In-vessel fuel cladding leaks detection system has been installed and interlocked with the reloading system;
– Entrainment of primary sodium from the reactor vessel to the systems and pipelines located outside of reactor vessel in the case of depressurization was ruled out;
– Fire and explosion protection of the steam generators and protection of other systems, structures and components located in the steam generator compartments have been improved;
– New emergency heat removal system has been installed and connected to the secondary circuit loops in parallel to the steam generators. The residual heat removal is provided through the independent sodium-air heat exchangers;
– Unlike the BN-600 electric generators TGV-200M that uses hydrogen for cooling, the cooling system of the BN-800 electric generator TZV-800-2 uses water as coolant;
– Filters removing radioactive aerosols from the combustion products have been added to the fire protection and ventilation systems.

[15] This is not the full list of modifications increasing the reactor safety.

[16] BN-600 and BN-800 reactor vessels are supported at the lower cylindrical part (from the bottom side).

[17] BN-600 reloading system has two rotating plugs, BN-800 has three.

2.3. BN-1200 REACTOR

The Russian Federation nuclear power strategy in the first half of 21 century determines tasks for the new generation nuclear power plant development and deployment [12]:

- Eliminate accidents that require public evacuation or relocation;
- Use effectively fissile and fertile materials provided by natural uranium;
- Multi-recycling of nuclear material minimizing amount of high level radioactive waste;
- Strengthening of non-proliferation characteristics of materials and technologies;
- Maintain competitiveness of nuclear power.

Minimum excess reactivity in the reactor core can be achieved through the appropriate breeding characteristics. Effective use of natural uranium resources can be achieved using a closed fuel cycle and renouncing of direct disposal of the spent fuel. Multiple recycling of nuclear materials in closed fuel cycle can help to minimize the minor actinide content in the high level waste.

The development of large sodium cooled fast reactors started in the Russian Federation a few decades ago. Design work on the BN-1200 reactor started in 2007 as part of the JSC Concern Rosenergoatom programme. Approval of the terms of reference for the development of BN-1200 reactor installation occurred in 2012. Later the development of BN-1200 moved ahead as part of the national 'Breakthrough' programme.

By the time of this INPRO assessment study, the basic design of the reactor installation and systems, structures and components had been developed including the design of steam generators and the core and fuel design of both MOX fuel and high density (nitride) fuel. The development of BN-1200 detailed design has been a continuous project.

The 'Breakthrough' Project Office and the national nuclear utility JSC Concern Rosenergoatom coordinate the BN-1200 project. The 'Breakthrough' Technical Committee reviews major modifications, new features and other technical issues related to the reactor development. JSC OKBM is the BN-1200 design organization responsible for the implementation of RD&D programme and design developments required. Organizations participating in the development of systems and technologies for BN-1200 have many years of experience in this area and established cooperation mechanisms, which are necessary for the assurance of design quality and safety [13–16].

The 'new generation' reactor designs which are being developed in the Russian Federation, including BN-1200, aim to resolve the following tasks:

- Competitiveness with other advanced nuclear power plants and with power plants using fossil fuel;
- Enhanced safety level corresponding to the requirements formulated for the Gen IV reactors, and elimination of the necessity of public evacuation/relocation in the case of potential accidents;
- Multi-recycling of plutonium isotopes in fast reactors and nuclear fissile material breeding for long term fuel supply to other types of reactors (water cooled reactors);
- Duration of the construction period up to 48 months for 'N-th-of-a-kind' commercial power plants;
- A feasibility of commissioning of a series of NOAK commercial power plants in 2–3 years after the FOAK power plant commissioning.

The following basic conceptual provisions have been defined as a basis for the BN-1200 design development project [17]:

- Proven technologies and experience acquired in the design, commissioning and operation of BN-600 and BN-800 reactors should have broad use in the BN-1200 design to the greatest extent possible [18, 19];
- Testing and validation of improvements in the reactor safety, economic competitiveness and effectiveness of fuel management incorporating innovative technologies through the appropriate RD&D activities using existing and newly developed research facilities;
- Optimization of the infrastructure requirements can be achieved through the selection of the appropriate value of the BN-1200 electrical power rating [18] and may involve unification of requirements for the NPP siting and unification of the electric generators and electric components used for the plant connection to the grid;
- Transportation of the NPP components to the construction site by railroad.

BN-1200 is a pool type reactor and it has an integral layout of the primary circuit like BN-800 and BN-600 where the primary circuit and radioactive primary coolant are in the main reactor vessel enclosed in the guard vessel as described above.

Like the BN-600 and BN-800, the BN-1200 reactor vessel is supported at the lower cylindrical part from the bottom side. Cooling of the reactor vessel has been worked out in a way that shielding structures provide the possibility of compact layout of the in-vessel systems, structures and components, and allows to get relatively small vessel diameter.

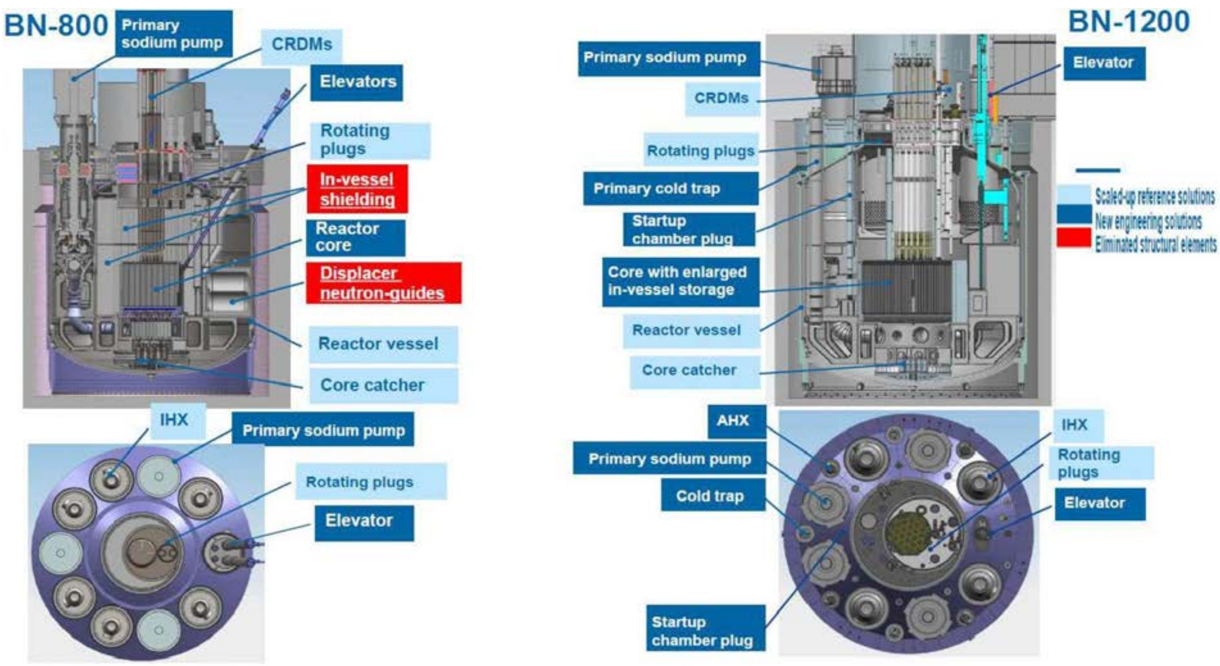

FIG. 2. BN-800 and BN-1200 reactor vessels [20].

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[18] Installed electrical power rating of BN-1200 was defined as equal to the electrical power rating of the advanced Russian water cooled reactors (AES-2006 or VVER-TOI). Based on the optimization studies performed in the Russian Federation for both sodium cooled fast reactors and water cooled reactors, the concept of 1200 MW(e) monobloc was adopted for the BN-1200 power plant.

Unlike the BN-600 and BN-800, the BN-1200 reactor maintains the level of primary sodium below the tapered upper part of the main vessel. This modification eliminates the need of guard vessel and bellows in the tapered part of the reactor vessel. Use of the main reactor vessel upper tapered part for the support of reactor equipment allows accommodating the four primary coolant pumps, four intermediate heat exchangers, the emergency heat removal system heat exchangers and a cold filter trap of the primary circuit within the BN-1200 reactor vessel without increasing the vessel diameter.[19] The latter modification eliminates components containing primary sodium outside of the reactor vessel and eliminates potential leaks of primary sodium from the reactor vessel external system. Figure 2 summarises basic similarities and distinctions between BN-1200 and BN-800 reactor vessel designs.

The four in-vessel ionization chambers improve the BN-1200 neutron flux monitoring and eliminates the need of neutron guides used in BN-600 and BN-800. A rotating roof planned for installation above the BN-1200 reactor vessel protects the reactor systems, structures and components from the potential falling of heavy objects.

Like BN-600 and BN-800, the BN-1200 uses the three-circuit configuration for the energy transfer and conversion. Sodium is used as a coolant in primary and secondary circuits, and water in tertiary circuit. However, in the BN-1200 every circuit contains four loops.

Each secondary loop is physically separated from three other secondary loops and located in a separate compartment. The loop includes a single intermediate heat exchanger located in the reactor vessel, a secondary coolant pump, pipelines and a sectional-modular steam generator. In BN-1200 the guard casings is introduced in the most of secondary pipelines. The buffer tanks compensating for the secondary sodium volume changes is combined with the tanks of secondary coolant pumps.

In order to reduce the length of pipelines in the secondary circuit and to minimize the number of installed isolating valves the bellows, compensators have been introduced in the BN-1200 reactor. Further reducing the volume of building and materials used the shell-and-tube type steam generators have been introduced in BN-1200 instead of sectional-modular type steam generators in BN-600 and BN-800 (see Table 1 and Figure 3).

TABLE 1. VOLUME OF THE FAST REACTOR BUILDINGS

Specific volume	BN-800	BN-1200
Main building, m^3/MW(e)	750	525
Steam generators boxes, m^3/MW(e)	32	16

FIG. 3. BN-800 and BN-1200 reactor building schemes [20].

[19] List of other systems, structures and components placed in the BN-1200 reactor vessel is similar to the BN-800 and BN-600 systems described previously.

The tertiary circuit layout of BN-1200 includes the K-1200-16.0/50 turbine and steam-steam reheating scheme. To increase the plant thermal efficiency to 43.6% (gross), the temperature and pressure of live steam, and feed water have been increased compared to the BN-800 parameters. BN-1200 uses electric generator of the type which is used in other advanced water cooled NPPs of that power rating designed in the Russian Federation (AES-2006).

In comparison with BN-600 and BN-800, other than the discussed above modifications of BN-1200 reactor design for increasing the reactor systems reliability and improving safety include the following [21, 22]:

- Use of uranium-plutonium nitride fuel helps to reduce the excess reactivity at full power conditions mitigating the potential consequences of RIA type of accidents. The BN-1200 excess reactivity at full power conditions will not exceed 0.5% $\Delta k/k$;
- Simultaneous withdrawal of more than one control rods from the reactor is prevented by multiple independent measures;
- Passive high temperature actuated control rods system is introduced in BN-1200 in addition to the passive hydraulically suspended control rods system[20] and active shut down system;
- Maximum power density in the reactor core is reduced to 380 MW/m^3;
- Duration of spent fuel storage within the reactor vessel is extended to two years reducing the spent fuel assembly power density to 2 W/cm^3, simplifying the fuel reloading process and improving safety of spent fuel management;
- Emergency heat removal system is based on the passive[21] removal of heat from the reactor to the environment through the sodium-sodium and sodium-air heat exchangers, and designed in such a way that in the case of emergency cooling the primary sodium circulates through the reactor core fuel assemblies driven by natural circulation flow;
- The above the reactor premise is used for confinement of radioactive gases and aerosols prior to subsequent filtration in the special ventilation system reducing the radioactivity of released gases.

Calculated core damage frequency of the BN-1200 reactor (approx. 5×10^{-7} 1/a) is essentially reduced compared with those of BN-800 (approx. 2×10^{-6} 1/a) and BN-600 (approx. 10^{-5} 1/a).

[20] Similar hydraulically suspended control rods system was installed in BN-800.

[21] Forced circulation of sodium in the emergency heat removal system is possible.

3. INPRO SUSTAINABILITY ASSESSMENT OF BN-1200 IN THE AREA OF ECONOMICS

This section presents an overview of the assessment method used in the INPRO area of economics and modifications to the method that have been introduced in the study. It provides input parameters used in the calculations and basic assumptions made by the national experts performing such calculations. Finally, the results of calculations presented in this section may be considered as an input for further research activities focused on optimization of the fast reactors designs and fuel cycles, however they should not be considered in relation to any commercial activities.

3.1. OVERVIEW OF THE APPLICATION OF THE INPRO METHODOLOGY AREA OF ECONOMICS TO THE FAST REACTORS UNDER DEVELOPMENT

The INPRO methodology has been developed for the sustainability assessment of the nuclear energy systems comprising different types of reactors and fuel cycle facilities. Sustainability assessment using INPRO methodology covers several different areas including economics, reactor safety, safety of fuel cycle facilities, waste management, etc. However, in the area of economics the INPRO methodology focuses on economic characteristics of energy products, i.e. electricity, heat, etc, generated by an NPP, rather than on economic parameters of fuel cycle facilities which are used as an input in a form of ‘cost of a given service’. For example, unless such calculations are the only way of obtaining costs of services the assessor is not required to calculate the figures of merit of the enrichment or reprocessing facility for the purpose of INPRO sustainability assessment. Instead, the costs of enrichment or reprocessing services are aggregated along with other costs in the figures of merit of the NPP producing final energy product. These figures of merit are used for the assessment against INPRO criteria in the area of economics.

INPRO methodology recommendations have the same hierarchy in all areas of assessment. On top of this structure a basic principle of sustainability is defined for every area determining a goal to be achieved to make the nuclear energy system sustainable [2]. The necessary actions to be taken to achieve the goal defined in the basic principle are introduced on the second level and called ‘user requirements’. At the next and third level, every user requirement splits into a few criteria. Criteria are the tools for assessment. Every criterion consists of an indicator (e.g. parameter) and an acceptance limit defining the range of values satisfying this criterion.

The INPRO assessment is normally done on the criteria level. When all criteria comprising a given user requirement are met, it means that necessary action occurred in a correct manner to meet the user requirement. When all given user requirements in a given area are met, the goal defined in the corresponding INPRO basic principle is also met [2]. However, the INPRO methodology assessment is not only used for the confirmation of nuclear energy system sustainability.[22] The INPRO assessment criteria which are not met provide valuable information on the gaps in the nuclear power programme and help to define follow-up actions (e.g. R&D) necessary to close these gaps. Comparison of the different assessment results may help to estimate the potential advantages of the assessed nuclear energy systems.

[22] At the moment of report preparation (2019) there were no known results of INPRO assessments confirming that a nuclear energy system meets all INPRO methodology criteria.

Judgement on the system sustainability can be derived from the results of full scope INPRO assessment in all INPRO methodology areas. However, preliminary results of the assessment in selected areas can be used for the optimization of nuclear energy system or modification of the selected system components. Although such limited scope assessments can provide valuable information on the system sustainability, the judgement on sustainability can be concluded only in the full scope INPRO sustainability assessment study in all areas.

This limited scope assessment study omits INPRO areas involving the essential numbers of country specific criteria (waste management, infrastructure, proliferation resistance) and focuses on the two areas encompassing the most of design related information – economics and reactor safety.

The INPRO methodology in the area of economics sets up the goal of achieving ‘affordable and available’ produced energy and related products and services in a sustainable nuclear energy system [2]. The affordability of energy is understood as cost competitiveness with alternatives available in the country or region. The availability of nuclear energy is seen as the ability to finance the project at acceptably low investment risk.

Four user requirements have been introduced in the INPRO methodology area of economics to specify how the basic principle can be met [2]:

- The cost of energy supplied by nuclear energy systems, taking all relevant costs and credits into account, should be competitive with that of alternative energy sources, that are available for a given application in the same time frame and geographic region/jurisdiction;
- The total investment required to design, construct, and commission nuclear energy systems, including interest during construction, should be such that the necessary investment funds can be raised;
- The risk of investment in nuclear energy systems should be acceptable to investors;
- Innovative nuclear energy systems should be compatible with meeting the requirements of different markets.

The developer of sustainable energy technology is expected to meet these user requirements. The INPRO assessor has eight criteria which can be used for checking the status of nuclear energy system in relation to the INPRO user requirements:

- Cost competitiveness;
- Attractiveness of investment;
- Investment limit;
- Maturity of design;
- Construction schedule;
- Uncertainty of economic input parameters;
- Political environment;
- Flexibility.

Only five of these eight criteria have been assessed in this study as explained below.

The cost competitiveness criterion uses the cost of energy generated by the nuclear energy system as an indicator, i.e. characteristic to be assessed. The limit for acceptance of this characteristic was defined as:

$$C_N \leq k \times C_A \qquad (1)$$

where C_N is cost of nuclear energy, C_A is cost of energy from alternative source, and k is a factor based on strategic considerations.

The INPRO methodology uses the levelized unit energy cost (LUEC) concept for definition of the energy cost (both C_N and C_A). LUEC of an assessed NPP should be lower or at least comparable, within a factor of k, to the LUEC of a competing power plant type. LUEC of an NPP and of a given alternative power plant can be calculated using the NEST tool developed by the IAEA or other tools using similar approach for the calculation of economic functions of NPPs and alternatives. LUEC comprises three major constituents: capital costs, operation and maintenance costs and fuel costs. It was introduced as the equivalent of the electricity price at zero profit covering all the capital, operating and maintenance and fuel costs with regard for the discount rate.

LUEC calculations have to include contingency and all accountable external costs[23] at a discount rate associated with specific investment policy. Accountable external costs may include the costs of radioactive waste management, decommissioning or back-fitting costs for the NPP, greenhouse gas emission fees and charges for fossil or biomass fuelled power plants, cost of power banks or backup power plants for wind and photovoltaic power plants.

The assessment of NPP deployment differs from the assessment of NPP design under development. In the case of NPP deployment, the alternative power plant is normally selected among non-nuclear power plants available in the country and, if applicable, among NPPs operating there. In the case of development of a new NPP design the competing power plant could be a comparable NPP of different existing or developing design. Nevertheless, LUEC needs to be calculated for the 'N-th-of-a-kind' (NOAK) case (the 'first-of-a-kind', FOAK, case can be evaluated separately). In this study VVER-TOI (new water-cooled thermal reactor under development) and BN-800 (newest operating sodium cooled fast reactor) have been selected as alternative power plants.

Competitiveness does not mean that the assessed nuclear energy system has to produce the cheapest energy in the country or region. The strategic considerations factor k in Eq.(1) depends on particular circumstances in a given territory. It was introduced to account for factors which have not been evaluated in terms of costs and not included into the energy costs C_N and C_A, for example: energy supply security, energy costs stability, diversity of energy supply, unaccounted environmental impacts, utilization of domestic resources (mineral or labour) and industrial capacity, public and political support, etc. Such considerations may be used as justification for k factor values different from 1.

Since economics does not provide an aggregated universal function for the complete economic assessment of the energy system options and LUEC does not present a full economic picture of a given project the next INPRO criterion, attractiveness of investment requires to calculate and assess the economic (financial) figures of merit. There are three financial figures of merit recommended for the INPRO assessment:

- Internal rate of return (*IRR*);
- Return on investment (*ROI*);
- Net present value (*NPV*).

The limits for acceptance of these characteristics are defined as:

$$IRR_N \geq IRR_A \quad (2)$$

$$ROI_N \geq ROI_A \quad (3)$$

[23] Some of the external costs related to the environmental impact from water intake, rejected heat, noise, land committed (e.g. for photovoltaic or hydro power plants), chemical discharges, etc, may impose challenges at evaluation.

$$NPV_N \geq NPV_A \quad (4)$$

where IRR_N, ROI_N and NPV_N are internal rate of return, return on investment and net present value of the assessed NPP, and IRR_A, ROI_A and NPV_A are internal rate of return, return on investment and net present value of the alternative power plant (alternative NPP in the case of this study). Since net present value represents the total net value of the investment, discounted to time 0, its absolute value will depend on the size of the investment and it needs to be normalized to the initial (discounted) capital investment made up to the start of plant operation or to the power output of the plant.

When comparing the economic functions of assessed NPP with the alternative energy supply options using Eq.(1)–Eq.(4), if energy planning[24] has identified nuclear power as part of an optimized mix of generating options the comparison of C_N, IRR_N, ROI_N and NPV_N to C_A, IRR_A, ROI_A and NPV_A, is not, of itself a determining consideration. But if limits defined in Eq.(1)–Eq.(4) are not met, the assessor needs to set out the explanations of why the difference is acceptable in the circumstances.

The next INPRO criterion in the area of economics, investment limit, requires that the highest single plant total investment up to the commissioning of the reactor be compatible with the ability to raise capital in a given market climate. This criterion is relevant to the situations when the construction of NPP is financed by foreign or private investors. The assessment of this criterion involves calculation of two parameters:

- Total investment which consists of the overnight capital, the interest during construction, contingency allowances, owner's cost, back fitting and decommissioning costs;
- Investment limit – the maximum level of capital that could be raised by a potential investor in the market climate.

The calculation of the second parameter is based on a country specific set of input data. The INPRO assessment of BN-1200 is focused only on the domestic deployment of these reactors which assumes that necessary investments will not be provided by foreign or by private investors. Therefore, the assessment of criterion 'investment limit' was omitted in this study and necessary clarification was added.

To assess the risk of investment in the nuclear energy system the INPRO methodology provides a group of four generic criteria focused on different factors that can theoretically impinge the NPP project. The first criterion in this group, maturity of design, is focused on the evaluation of technical development status and licensing status of the design. At the time of development of this report (2019) the licensing process for BN-1200 has not been started yet and this criterion 'maturity of design' has not been assessed in the INPRO area of economics. It can be assessed at the later stages of the BN-1200 programme.

The next criterion, construction schedule, considers the background information used for the definition of times necessary for the reactor construction and commissioning. These times are among the most sensitive parameters for the energy cost calculations and their realistic definition is important for the investment risk minimization. Information on simplification of the BN-1200 design and improvement of the construction methods is discussed in this report section on the INPRO assessment of BN-1200 in the area of reactor safety. The construction period of the BN-1200 reactor is assumed to amount 6 years. However, due to the lack of practical experience on BN-1200 construction and commissioning, the criterion of

[24] An energy planning and modelling study is a recommended predecessor of the INPRO methodology sustainability assessment study.

‘construction schedule’ has not been assessed in the INPRO area of economics. It can be assessed at the later steps of BN-1200 programme.

The assessment of uncertainty of economic input parameters requires an analysis of the sensitivity of important input parameters. Sensitivity analysis can involve many different methods and approaches, e.g. calculation of robustness indexes, Monte-Carlo studies, cost-sensitivity diagrams, etc. In this study, the sensitivity analysis of BN-1200 economic parameters used cost sensitivity diagrams mainly for screening purposes. At the next stages of project development this analysis can be expanded to other methods.

The last criterion assessing the risk of investments, and political environment, requires checking the long term commitment to nuclear energy system development/deployment in the country.

The last criterion of the INPRO methodology area of economics, flexibility, evaluates the potential of the reactor to meet different market conditions including both electricity markets and fuel markets. Electricity market conditions may involve such characteristics as the grid size and requirements on participation in the grid regulation. Fuel market considerations include the possibility to use different types of fuel (e.g. UOX, MOX, nitride or metallic fuels) or fuels fabricated by different suppliers without major modification of the installation. To meet this criterion an NPP is expected to be sufficiently flexible to provide competitive energy for a wide range of markets.

3.2. IMPROVEMENT OF ECONOMIC CHARACTERISTICS OF SODIUM COOLED FAST REACTORS

Strategic planning of the nuclear energy system which is expected to contribute to the sustainable development in a global prospective or in a defined country/region involves several stages (e.g. system modelling, assessment, definition and implementation of the follow-up measures) and iterations at different levels of the nuclear energy system maturity. Information on the process of nuclear energy system development and optimization, on the requirements, boundary conditions, continuity of support and the trends related to improvement of the system economics can provide valuable background for the evaluation of potential investment risks, justification of basic assumptions used in cost calculations and input data reliability.

The fast reactor programme implemented in the Soviet Union and later in the Russian Federation has demonstrated scope and continuity (Figure 4). Passing from one stage of the programme to another, accumulating necessary experience and gradually improving the technology avoided overhasty decisions in and minimized potential risks from the introduction of innovative technology. R&D studies of the advanced systems and optimization of the reactor design are going on continuously. New reactors are characterized by improved operating parameters, higher fuel burnups, and improved safety. Prospective BN reactors with dense fuels allow an increase in the breeding of fissile material (total breading ratio and breading ratio in the core reaching the values of 1.45 and ~1 respectively). Different schemes of recycling minor actinides are under investigation with the objective to reduce the amount and radiotoxicity of HLW.

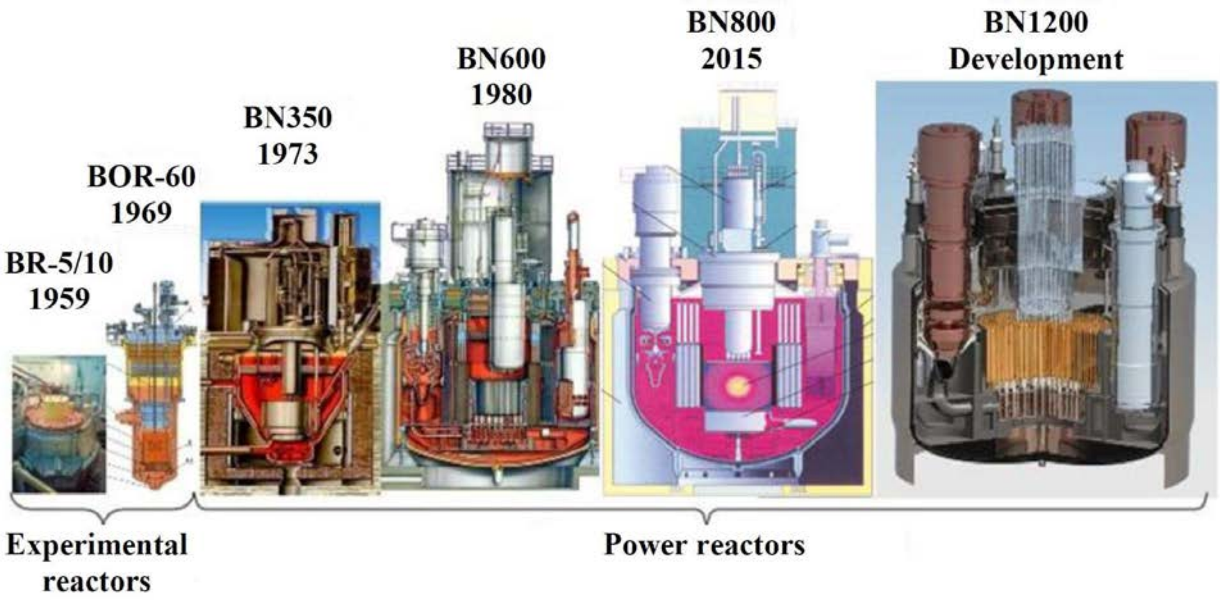

FIG. 4. Fast reactors developed in the Soviet Union and in the Russian Federation [20].

The fast reactor programme in the Soviet Union was launched in 1950s, when IPPE commenced the development of experimental fast reactors. In 1956–57 the design of sodium cooled fast reactor BR-5 was developed. This reactor commissioned in 1958–59 in IPPE originally had thermal power rating of 5 MW(th)[25]. Moving on into the 1960s, IPPE performed a comprehensive comparative analysis of different coolants and defined a preference for a fast reactor technology concept based on the sodium cooled fast reactor with steam-turbine cycle for the energy conversion.

The second Russian sodium-cooled fast reactor, BOR-60, was commissioned in 1969. For a long time, it was used as the main experimental facility to study sodium-cooled fast reactors. The convenience of BOR-60 design for conducting of a variety of research studies allowed to use this experimental facility extensively in the development and justification of the first Soviet commercial sodium-cooled power reactor BN-350.

The Soviet Union constructed the prototype commercial fast reactor BN-350 in early 1970s with its commissioning in 1973. The BN-350 was a loop-type reactor, having a designed lifetime of 20 years. It performed both electricity production and water desalination over its 25 year operating period (the BN-350 was sited in western Kazakhstan). From the beginning it provided valuable information on the real scale systems, structures and components behaviour which assisted the development of the BN-600 reactor constructed several years later.

The BN-600, commissioned in 1982, a commercial pool-type pilot reactor with an electrical power rating of 600 MW(e) demonstrated the results of significant design improvements and economic optimization. Like its predecessor BN-350, the BN-600 reactor uses enriched uranium oxide fuel. Over the next 30 years of operation its load factor achieved ~ 76% which is close to the load factors of traditional water cooled reactors. The average number of

[25] In 1973 the BR-5 reactor was upgraded increasing the thermal power rating. It was also renamed as BR-10.

unplanned total scrams per 7000 hours critical[26] during the period from 1990 to 2009 was 0.2 [23].

Further improvement of the technology and economic optimization was undertaken during the BN-600 operation. It involved the update or revision of the maintenance and replacement techniques for selected systems, structures and components, including the major power plant components such as pumps, heat exchangers and steam generators. Valuable experience was accrued on prevention and mitigation of potential sodium leaks[27], demonstrating effectiveness of the inspection methods, monitoring and confinement systems, and fire protection systems. More information on the BN-600 characteristics is provided in Section 2.

The newest operating reactor in the BN lineage, BN-800, was connected to the grid in 2016. The BN-800 design concept was developed in the period from 1983 to 1993 incorporating lessons learned from the successful operation of BN-600, lessons from the Chernobyl accident and the revised regulatory requirements introduced in the Russian Federation. Revised regulations required ensuring the safety of local population during the design basis accidents without such protective measures as evacuation or relocation. The BN-800 design was the first Russian reactor that obtained a construction license from the Federal Nuclear and Radiation Safety Authority after the Chernobyl accident.

Unlike its predecessors, the BN-800 reactor was designed to be operated with MOX fuel and in a few years after commissioning it has been using both UOX and MOX fuel assemblies in the core. This reactor is expected to play important role in the development and refinement of closed fuel cycle technologies in the Russian Federation. The objective is to obtain the MOX fuel burnup of 15% heavy atoms and higher, and to test fuel rods and fuel assemblies with the nitride fuel having a higher density than MOX. Improvements of the spent MOX-fuel reprocessing and the recycled fuel re-fabrication technologies are being carried out in parallel. The BN-800 will be used for the development of technology to burn minor actinides accumulated in the spent fuel from different types of reactors (fast and thermal). More information on the BN-800 characteristics is provided in Section 2.

The next step of development of the BN reactor family is the large commercial reactor BN-1200. The concept of BN-1200 envisages reactor core operation using different types of fuel and permits variation of fuel utilization parameters in accordance with system requirements providing significant flexibility and possibility to adapt the reactor to different market conditions.

The optimization of the BN-1200 reactor design [24] involved multiple modifications of the systems, structures and components layout, providing significant improvement of the reactor economics characteristics. Most of BN-1200 performance characteristics are similar to the characteristics of traditional large water cooled reactors which are currently under development in the Russian Federation. The installed power rating of BN-1200 reactor (1220 MW(e)) is similar to that of VVER-TOI (1255 MW(e)) and the design lifetime of both reactors is 60 years. Specific (per MW(e) installed) capital costs of the BN-1200 reactor are less than half of those for BN-350 and achieved the level of VVER-TOI. The reduction of mass of metal used in BN-1200 (per MW(e) installed) compared against BN-350 reactor is even lower. The evaluation of the improvement of the two latter characteristics is presented in Figure 5.

[26] Performance indicator used by the World Association of Nuclear Operators (sum of the number of unplanned automatic scrams and unplanned manual scrams for 7000 hours of operation).

[27] Prevention and effective mitigation of sodium leaks reduces the reactor idling periods and increases the load factor.

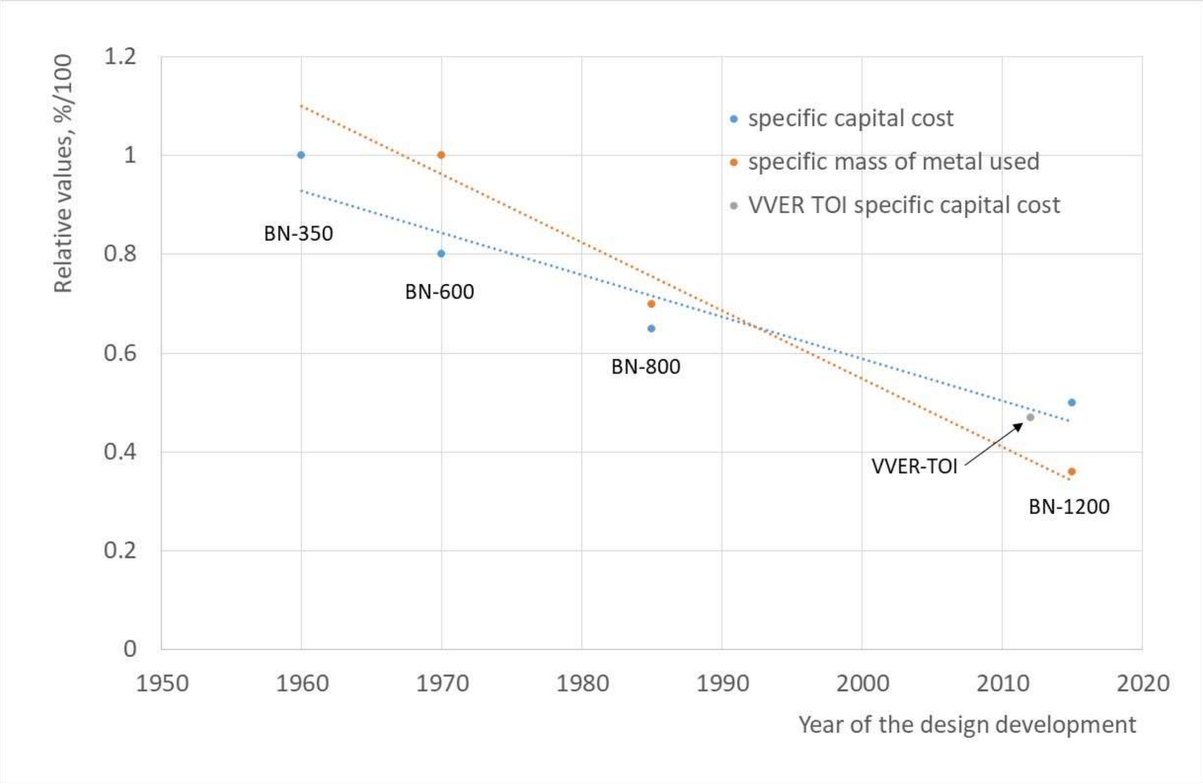

FIG. 5. Specific mass of metal used and specific capital costs for different BN reactors at different stages of the BN technology development programme.

The BN-1200 reactor core is designed for using either MOX fuel similar to BN-800 or a new type of fuel with a higher density and plutonium and uranium in nitride form. As in BN-800, the BN-1200 fuel assemblies are designed in such a way that sodium plenum is maintained in the upper part of the core. However, the increased fraction of fuel per unit of the core volume yields the breeding ratio of about ~1.2 and maximum fuel burnup of at least 15% heavy atoms.

At the time of this report preparation the optimization of BN-1200 design has not been fully completed and further improvement of the economics characteristics can be reasonably expected. However, the general trends used in this study and presented in this report remain valid and input information is sufficient for the limited scope INPRO assessment.

3.3. BASIC RESULTS OF ANALYSIS IN THE AREA OF ECONOMICS

This section presents the results of limited scope analysis of the planned system based on the sodium-cooled fast reactor BN-1200 in the Russian Federation.

Preliminary economic studies of an energy system development, deployment or modification involve several steps including system planning and modelling, cost study, profit characteristics or cashflow analysis, sensitivity study, etc, evaluating the system viability, competitiveness, and attractiveness which can be further studied in the comprehensive and somewhat cumbersome stages of financial analysis. Both the system planning and modelling study and the financial analysis are outside of the scope of the INPRO methodology sustainability assessment. The energy system planning and modelling and, more specifically, the nuclear energy system planning and modelling are the necessary prerequisites of the INPRO assessment study.

In this INPRO assessment of the BN-1200 reactor there is the assumption that the necessary scenario studies have been successfully performed, involving the fissile/fertile materials flow analysis, and the role of the nuclear energy system is understood. INPRO sustainability assessment in the area of economics consists of the consideration of the energy cost study results, profit characteristics and sensitivity study.

3.3.1. Cost of energy generation

3.3.1.1. Overnight capital cost

Projections of overnight capital costs associated with a given NPP design are sensitive to the assumption on the system characteristics at the different stages of technology maturity. Overnight costs of a FOAK reactors are different from NOAK reactors. The latter costs may depend on the assumed number of commercial reactors to be deployed in the home country and abroad. Prototype reactors normally have lower capacities than commercial designs. If the difference is too large the estimation of the effects of the economy of scale reducing the specific cost of energy, may introduce essential uncertainty. Smaller difference in the installed capacities allows better estimation of costs.

Another important assumption is related to the number of power units at a given site. Due to the common use of several systems, structures and components by all power units at the site, depending on the reactor design, the cost of a second power unit of the same design may be reduced by a factor of approx. 1.5 (estimated by the IPPE).[28] The cost reduction effects from construction of additional units are essentially smaller and can be estimated at less than 5%.[29]

Combinations of these assumptions may influence both the overnight costs of an NPP, and the cost of energy generated. For example, the overnight cost of a single 1 GW reactor can be lower than the capital costs of two 0.5 GW alternative units located at different sites, however, depending on the effects of economy of scale, it can be higher than the overnight costs of two 0.5 GW alternative units located at one site. The comparison of the two 1 GW reactors located at one site against alternative power plants of the same total capacity may yield different results.[30]

Cost calculations in this publication have been performed for the case of twin unit NPPs. However, these calculations do not account for the scenario of reactors deployment, i.e. the difference in dates of the twin reactors' commissioning and discounting of the costs between these dates has not been accounted for. Average energy costs have been calculated assuming that twin units had been constructed and commissioned simultaneously.

Overnight costs of the twin unit NPPs with BN-800 reactors, BN-1200 reactors and VVER-TOI reactors are presented in Table 2. A FOAK BN-1200 is planned to be constructed at the site of Beloyarsk NPP and the estimation of overnight cost was performed for that site. Note

[28] An NPP has a number of buildings (administrative buildings, laboratory buildings, personnel training centre, storage, emergency centres, waste management facilities, etc) and systems (some equipment cooling, some diesel generators, electric grid connection, etc) commonly used by all or at least by a few power units at the same site. These buildings and systems have to be in place at the commissioning of first power unit at the site. Following units (second, third etc) can use the same common buildings and systems. Some expenses still will be needed for connection and adjustments. IPPE estimates that the capital cost of unit 2 may be approx. 1.5 times lower than the cost of unit 1 of the same design at the same site.

[29] The cost of unit 3 is expected to be approx. 5% lower than that of unit 2 of the same design at the same site.

[30] It should be note that the effects of energy transmission costs and losses depend on the number of power plant sites and the sites location.

that there is no plan to deploy BN-800 reactors in the Russian Federation and the overnight cost of the twin BN-800 NPP was estimated numerically to make this comparison more convenient. Estimation of the overnight costs of VVER-TOI has not been made related to any specific site and potential site specific requirements.

TABLE 2. ESTIMATED OVERNIGHT CAPITAL COSTS AS OF 01.01.2013 [25]

Reactor type		Overnight cost per twin unit power plant, 10^9 USD	Specific overnight cost, 10^3 USD/kW(e)
BN-800 (2 x 880 MW(e))		6.77	3.8
BN-1200 (2 x 1220 MW(e))	FOAK	8.22	3.4
	NOAK	7.86	3.2
VVER-TOI (2 x 1255 MW(e))		7.72	3.1

Overnight capital costs of the FOAK BN-1200 are estimated at approx. 10% lower than the capital costs of BN-800 from the economy of scale and optimization of design surpassing the cost of BN-1200 safety improvements. Further optimization of BN-1200 design is expected to yield approx. 5% reduction of the NOAK BN-1200 overnight costs achieving the level of overnight costs of the new Russian water cooled reactors VVER-TOI.

3.3.1.2 Operation and maintenance cost

The operation and maintenance costs usually consist of the following expenses:

- NPP personnel salaries;
- Cost of services necessary for NPP operation provided by the external contractors including repair, maintenance, inspections, safety assessments, meteorological and environmental studies, etc;
- Cost of electricity, fossil fuels, chemical materials, etc, necessary for NPP operation/maintenance;
- Cost of equipment necessary for NPP operation/maintenance including cost of the NPP components which the NPP design calculates as necessary for occasional replacement;
- Cost of licensing related activities and services (peer reviews, assessments, inspections, knowledge management and personnel training, etc);
- Retrofit costs[31] during the NPP operation;
- Cost of management of radioactive waste other than spent fuel or waste from spent fuel reprocessing;
- Decommissioning and backfitting costs;[32]
- Insurance fees, cost of financial services, etc.

Operating and maintenance costs do not include the amortization costs, fresh fuel costs, cost of spent nuclear fuel management outside of NPP, cost of spent fuel reprocessing and management of radioactive waste arising from reprocessing. One part of operation and maintenance costs (e.g. NPP personnel salaries) does not depend on the amount of energy generated by NPP and

[31] Part of the retrofit costs can be considered in the backfitting costs.

[32] Decommissioning and potential backfitting funds are normally accrued in the form of regular contributions to the special funds. In the NPP economic analysis these costs can be accounted as the part of operation and maintenance cost or, alternatively, the decommissioning and backfitting costs can be considered as the part of total capital investment costs since they are evaluated as the portions of overnight costs. For example, decommissioning cost is often evaluated as 10% of overnight costs.

needs payment on a regular basis regardless of operational mode. Remaining operations and maintenance costs are variable and depend on the average amount of energy generated by an NPP. However, those dependencies can be different.[33]

At this stage of the BN-1200 development the detailed characteristics of the constituent pieces of operation and maintenance cost were not available to the assessor. For the purpose of INPRO sustainability assessment all operation and maintenance costs have been considered as fixed values independent from the NPP average load factors. The operation and maintenance costs of BN-800, BN-1200 and VVER-TOI are presented in Table 3.

TABLE 3. OPERATION AND MAINTENANCE COSTS

Reactor type	Annual specific operation and maintenance costs, USD/kW(e) per annum
BN-800 (2 x 880 MW(e))	134
BN-1200 (2 x 1220 MW(e))	122
VVER-TOI (2 x 1255 MW(e))	102

In this study the operation and maintenance costs of new fast reactors were estimated using the available data from water cooled reactors and the results of analysis of BN reactors design characteristics and operating/maintenance procedures. For example, BN-1200 personnel salaries are estimated as proportional to the number of employees per unit of installed power rating with a surcharge for MOX or nitride fuel management.

3.3.1.3 Fuel costs

Fuel cost relative contribution to the overall cost of energy generated in an NPP is relatively modest. It varies depending on the type of reactor and fuel cycle, national policies, company strategy, selected investor and vendors, however it normally remains within one fifth of the total energy cost. Theoretically, the type of fuel cycle can affect the sustainability areas other than economics, e.g. waste management or environment, and the fuel cycle consideration can involve issues other than costs. The closed fuel cycle fuel cost calculations can be quite sophisticated. However, in many cases their primary objective is rather to demonstrate that increased complexity of the process does not make the overall cost of fuel unacceptably high[34].

NPP fuel cost is the aggregated value of expenses born at different steps of the complete fuel cycle, both frontend and backend, including the final disposal of spent fuel in the case of once-through fuel cycle or the disposal of high level waste in the case of using reprocessing. The frontend cost of once-through fuel cycle (LEU) is combined from the cost of natural uranium and costs of fuel cycle services[35] necessary for obtaining the form of fuel which can be safely used in a reactor type. The backend involves costs of services necessary for obtaining the safe end state of spent fuel (deep geological disposal). The costs of fuel cycle services strongly depend on the scale of fuel cycle facility providing a given service (economy of scale) and the

[33] Cost of the NPP components that the NPP design determines needing occasional replacement can be proportional to the energy generated. Cost of maintenance services can be estimated as proportional to the maintenance period length when reactor is normally idle. The amount of radioactive waste generated during the maintenance period can be higher than at the full power operation.

[34] Cost associated with the increased complexity may be compensated (at least in part), e.g. by savings on the cost of natural uranium or, in case of transmutation of minor actinides, by savings on the cost of management of spent fuel from other reactors operated in once-through fuel cycle. Latter option is not considered in this report.

[35] Including the cost of materials, e.g. zirconium and stainless steel, transportation among facilities and storages.

scale of facility depend on the demand of this service, i.e. on the scale of nuclear power programme.

In the case of closed fuel cycle the backend services separate fissile and fertile materials from the waste, move waste to the end state and feed the frontend with fuel materials. Apparently, this feedback implies that the costs of backend services, e.g. cost of spent fuel reprocessing, and the reactor fissile material breeding characteristics affect the cost of 'fresh' fuel and can introduce additional uncertainty to the fuel cost calculation. A few other process step links need to be accounted for, e.g. better refining of fissile material may increase the cost of spent fuel reprocessing, however it can reduce the cost of other steps of the closed fuel cycle. The cost evaluations considering different types of the reactors and fuel cycles in a nuclear energy system may involve more links [26].

The evaluation of ranges of the costs of different fuel cycle services is provided in Ref. [25]. The ranges are quite broad, for example, the cost of MOX fuel fabrication varies from 1000 USD/kg to 6000 USD/kg and the cost of MOX fuel reprocessing – from 640 USD/kg to 2600 USD/kg. The discrepancies in cost estimations may be related to the lack of experience, different assumptions and criteria.

Data on the dependence of fuel cycle services costs from the production scale of a fuel cycle facility are relatively scarce and mostly relate to the once-through fuel cycle and uranium fuel production. Refs [27, 28] provide evaluations for the fuel fabrication facility producing MOX fuel for the light water reactors. Figure 6 presents cost vs production rate diagram for the light water reactor MOX fuel fabrication facility [29].

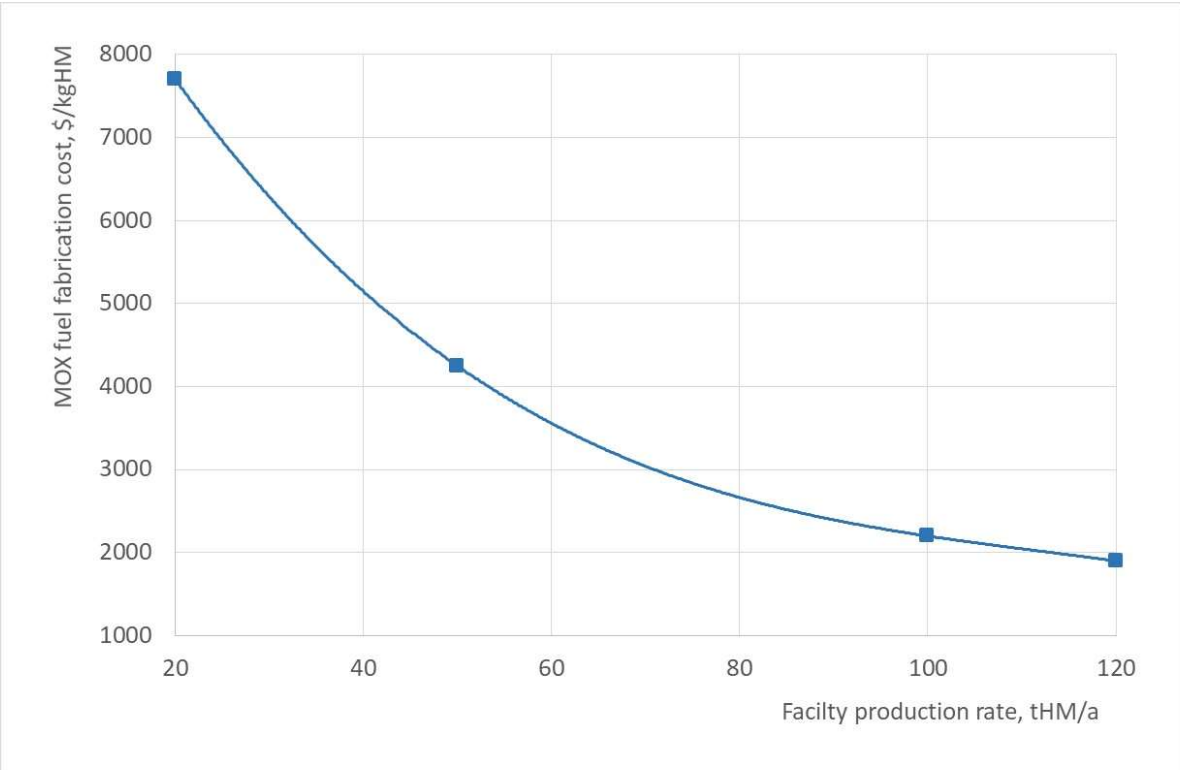

FIG. 6. MOX fuel fabrication cost as a function of facility production rate (modified from [29]).

Higher facility production rates allow fabricating MOX fuel at lower cost. Depending on the production rate the cost of MOX fuel fabrication can vary manifold. In the range between 40 and 120 tHM/a, raising the production rate with a factor of k reduces the cost of fuel fabrication at $\frac{1}{k} + \alpha$ rate. Here α is relatively small surcharge which can be estimated at about several percent. Every time further reduction of the fuel costs at the same absolute value requires a larger increase of the production rate which is limited by the fuel demand and the scale of nuclear power programme.

The same MOX pellet fabrication technology is normally used for thermal reactors and fast reactors.[36] Fast reactor fuel assembly materials and manufacturing technologies differ from thermal reactors; however, this difference is expected to be relatively small. For the purpose of this assessment study it was assumed that the cost data on MOX fuel fabrication for the light water reactors will remain valid for the core fuel of fast reactors using MOX.

TABLE 4. LEVELIZED COST OF FAST REACTOR CLOSED FUEL CYCLE SERVICES (MOX FUEL, 0% DISCOUNT RATE)

Fuel cycle service	Cost (mills/kWh)	
	BN-1200	BN-800
Core and axial blanket fuel fabrication	4.47	7.20
Radial blanket fuel fabrication	0.18	0.09
Transportation of fresh fuel	0.19	0.23
Transportation of spent fuel	0.19	0.23
Interim storage of spent fuel	0.02	0.02
Spent fuel reprocessing	1.42	1.83
Radioactive waste final disposal	1.58	2.04
Total fuel cycle cost	8.05	11.64

TABLE 5. LEVELIZED COST OF FAST REACTOR CLOSED FUEL CYCLE SERVICES (MOX FUEL, 5% DISCOUNT RATE)

Fuel cycle service	Cost (mills/kWh)	
	BN-1200	BN-800
Core and axial blanket fuel fabrication	5.22	7.33
Radial blanket fuel fabrication	0.23	0.10
Transportation of fresh fuel	0.23	0.25
Transportation of spent fuel	0.12	0.20
Interim storage of spent fuel	0.02	0.03
Spent fuel reprocessing	0.94	1.46
Radioactive waste final disposal	0.09[a]	0.13[a]
Total fuel cycle cost	6.85	9.50

[a] – 55 years' time lag was assumed for the disposal of radioactive waste from reprocessing.

Due to the different content of fissile material in the spent fuel of fast and thermal reactors the safety requirements at reprocessing can be different and the associated costs of reprocessing can be different either. However, at this early stage of fast reactor deployment, with their number remaining lower than existing pool of thermal reactors and the production rate of fast reactor fuel reprocessing being lower than necessary for competitive costs, one of the effective reprocessing scenarios uses a blending of spent fuel from fast and thermal reactors. In this scenario, the concentration of fissile material in the spent fuel mix can be maintained within safety limits of the reprocessing facility nominally seen in the spent fuel from thermal reactors.

[36] Similar 'cost vs production rate' studies performed in the Research Institute of Atomic Reactors (Russian Federation) have indicated that for fast reactors and closed fuel cycle the decline of cost in such a diagram may be shorter turning into a flat rate at relatively low production rates due to the higher concentration of fissile material and associated criticality safety requirements.

This scheme could keep the cost of fast reactor spent fuel reprocessing close to the cost of this service for thermal reactors. Previous discussion on the dependence of fuel cycle service cost on the facility production rate remains valid for the case of fuel reprocessing. More detailed consideration of the relation between the fuel cycle costs and production rates was provided in Ref. [26].

Evaluation of the levelized cost of fast reactor closed fuel cycle services was performed using FCCBNN [30] code developed in the IPPE. The results of costs calculation at 0% discount rate are seen in Table 4. Fuel fabrication, spent fuel reprocessing, and radioactive waste disposal are the principal contributors to the fuel cost. The results of calculation at 5% discount rate are provided in Table 5. All costs are discounted to the moment of fuel uploading to the reactor. In this case more than three quarter of the total fuel cost is contributed by the fuel fabrication.

The fuel cost structure of thermal reactors operated in once-through fuel cycle differs from the fast reactors operated in closed fuel cycle. The thermal reactor fuel cost normally combines costs of stages different from those in Tables 4 and 5, such as the cost of natural uranium, cost of uranium refining and conversion into hexafluoride form, cost of enrichment and cost of spent fuel direct disposal. There is normally no reprocessing in once-through fuel cycle. Principal contributors to the fuel cost in thermal reactors can be different from those in Tables 4 and 5.

Theoretically all costs of fuel cycle services and the cost of natural uranium can vary depending on the market prices. Quick and short term variations are normally not accounted for in the planning stages in economic analysis. Slow and long term variations are normally estimated through the annual escalation rates which can be added to the calculation of materials or services costs.

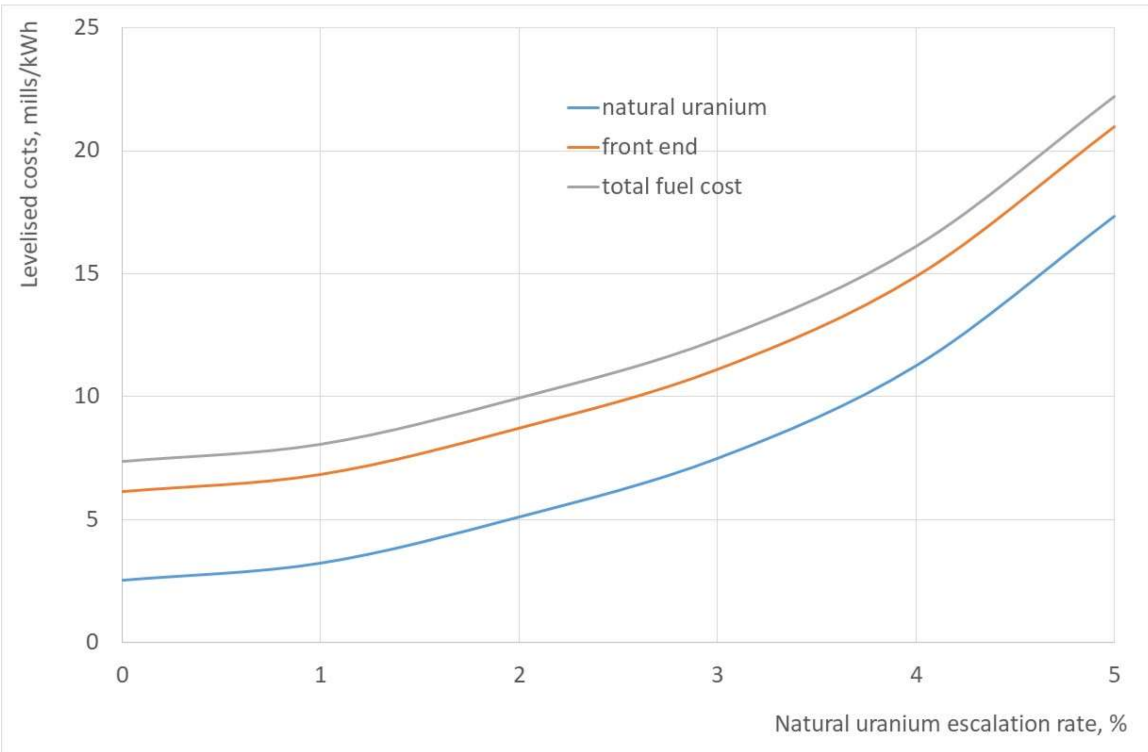

FIG. 7. Levelized fuel costs of thermal reactor at different natural uranium cost escalation rates.

Evaluation of the effects of cost escalation for different fuel cycle services and materials is a cumbersome task which is generally out of the scope of this study. Real costs of uranium refining, conversion, and enrichment, and fuel fabrication to vary and are assumed to be constant, e.g. the thermal reactor fuel fabrication cost, 350 USD/kgHM [25], is constant in these calculations. However, the primary objective of the closed fuel cycle is the reduction of natural uranium consumption per energy unit produced since it is considered as a limited resource. Long term strategic planning scenarios involve a broad range of projections and assumptions on the electricity demand, and on the share of nuclear power in the energy supply mix. The availability and cost of natural uranium in long term may both depend on the global nuclear energy system characteristics and affect that system.

The effects of potential escalation of natural uranium cost throughout the NPP lifetime onto the cost of electricity produced have been evaluated in this report. The results of VVER-TOI thermal reactor fuel cost calculation at 5% discount rate are presented in Figure 7. Calculations have been performed at the different natural uranium cost escalation rates using the FCCVVR tool [30] developed in IPPE.

Levelized costs of the fuel cycle services which do not depend on the cost of uranium and its escalation rate have been evaluated as follows:

- Front end cost:
  - Refining and conversion of uranium ~ 0.3 mills/kWh;
  - Enrichment of uranium ~ 2.4 mills/kWh;
  - Fuel manufacturing ~ 1.0 mills/kWh;
- Back end cost (1.2 mills/kWh):
  - Spent fuel reprocessing ~ 1.0 mills/kWh;
  - Radioactive waste management ~ 0.1 mills/kWh;
  - Transportation ~ 0.1 mills/kWh.

The effects of different escalation rates of the natural uranium cost on the levelized cost of natural uranium, levelized cost of front end and total levelized fuel cost include the following. The uranium cost annual escalation rate of 1% corresponds to the growth of cost from the accepted initial value[37] of 100 USD/kg (in 2015) to approx. 180 USD/kg during the NPP lifetime (60 years) and the average cost of approx. 140 USD/kg.[38] Levelized cost of natural uranium rises with a factor of 7 when escalation rate changes from 0 to 5%. At the same time the total fuel cost triples. At 0 escalation rate the cost of natural uranium contributes approx. 1/3 of the total fuel cost. At 5% escalation rate the share of natural uranium cost contribution exceeds 3/4.

3.3.2. Evaluation of INPRO sustainability assessment indicators in the area of economics

Calculation of basic economic functions presented in this section has been performed with the NEST tool [2].

[37] The uranium cost varies, and evaluation of the initial cost needs to be done carefully. High initial costs of uranium can make the reactors operated in closed fuel cycle more attractive. Low initial costs make them less attractive. The indicative value of 100 USD/kg was taken for the purpose of this study.

[38] 2% escalation rate corresponds to approx. 320 USD/kg after 60 years and approx. 190 USD/kg average value, 3% - approx. 570 USD/kg after 60 years and approx. 270 USD/kg average value, 4% - approx. 1000 USD/kg after 60 years and approx. 400 USD/kg average value, 5% - approx. 1800 USD/kg after 60 years and approx. 600 USD/kg average value.

3.3.2.1 *Input data for cost calculation*

Input data used in the INPRO sustainability assessment of BN reactors are summarized in Tables 6–8. Basic design data are presented in Table 6, investment characteristics including construction schedule – in Table 7, closed fuel cycle characteristics of BN reactors – in Table 8. Annual operation and maintenance costs of BN-1200 are equal to 122 USD/kW(e) and BN-800 – 134 USD/kW(e). Regular uniform contributions to the decommissioning fund are spread over the reactor lifetime and included in the operation and maintenance costs.

TABLE 6. BASIC DESIGN DATA OF BN REACTORS

No.	Characteristic	units	BN-1200	BN-800
1	Electric power rating	MW(e)	1220	880
2	Thermal efficiency	%/100	0.436	0.420
3	Load factor	%/100	0.90	0.85
4	Lifetime	a	60	60

TABLE 7. INVESTMENT CHARACTERISTICS

No.	Characteristic	units	BN-1200	BN-800
1	Discount rate	%/100	0.05	0.05
2	Project overnight investments	$/kW(e)	3200	3800
3	Construction time	years	6	6[a]
4	Investment schedule. Year of construction:			
	-1	%/100	0.03	0.03
	-2	%/100	0.10	0.10
	-3	%/100	0.28	0.28
	-4	%/100	0.38	0.38
	-5	%/100	0.18	0.18
	-6	%/100	0.03	0.03

[a] – Construction of BN-800 at the site of Beloyarsk NPP (Unit 4) took over 30 years (over 20 of that 30 years the project was on hold). In this report the construction schedule of BN reactors was estimated based on expert judgement and experience acquired at the construction of water cooled and fast reactors in the Russian Federation.

TABLE 8. FUEL CYCLE CHARACTERISTICS

No.	Characteristic	units	BN-1200	BN-800
1	Core fuel fabrication cost	$/kgHM	3500	3500
2	Blanket fuel fabrication cost	$/kgHM	300	300
3	MOX fuel transportation cost	$/kgHM	100	100
4	Spent fuel transportation cost	$/kgHM	100	100
5	Spent fuel storage cost	$/kgHM	14	14
6	Spent fuel reprocessing cost	$/kgHM	770	770
7	Cost of radioactive waste disposal	$/kgHM	860	860
8	Core fuel burnup	MWd/kgHM	152	104
9	Time from spent fuel discharge from reactor till reprocessing (including storage in spent fuel pool)	a	5	5
10	Time from spent fuel discharge from reactor till MOX fuel fabrication	a	6	6
11	Time from spent fuel discharge from reactor till HLW final disposal (including HLW storage)	a	55	55
12	Plutonium losses at reprocessing	%/100	0.001	0.001

Cost of depleted uranium was assumed equal to zero in the calculation.

3.3.2.2 Total cost of electricity production

The results of calculation of the Levelized Unit Energy Cost are demonstrated in Figure 8. All costs are calculated for a twin unit NPP of a given type. Three major contributors to the total electricity cost are presented in different colours. Capital investment costs[39] are blue, operation and maintenance costs are red, fuel costs are green. Costs are calculated for BN-800, BN-1200 and VVER-TOI reactors.

Costs of VVER-TOI reactor are calculated for six different assumed values of the natural uranium cost escalation rates. VVER-TOI (0) case corresponds to 0% escalation rate, VVER-TOI (1) – 1% escalation rate, VVER-TOI (2) – 2% escalation rate, VVER-TOI (3) – 3% escalation rate, VVER-TOI (4) – 4% escalation rate, VVER-TOI (5) – 5% escalation rate.

Cost of electricity from BN-800 amounts approx. 59 mills/kWh and from BN-1200 ~46 mills/kWh. Cost of electricity from the thermal reactor VVER-TOI at the constant cost of natural uranium 100 USD/kg is lower than from BN-1200. Cost of VVER-TOI energy grows depending on the escalation rate of natural uranium cost. At 2% escalation rate the cost of energy from thermal reactor may exceed the estimated cost of BN-1200.

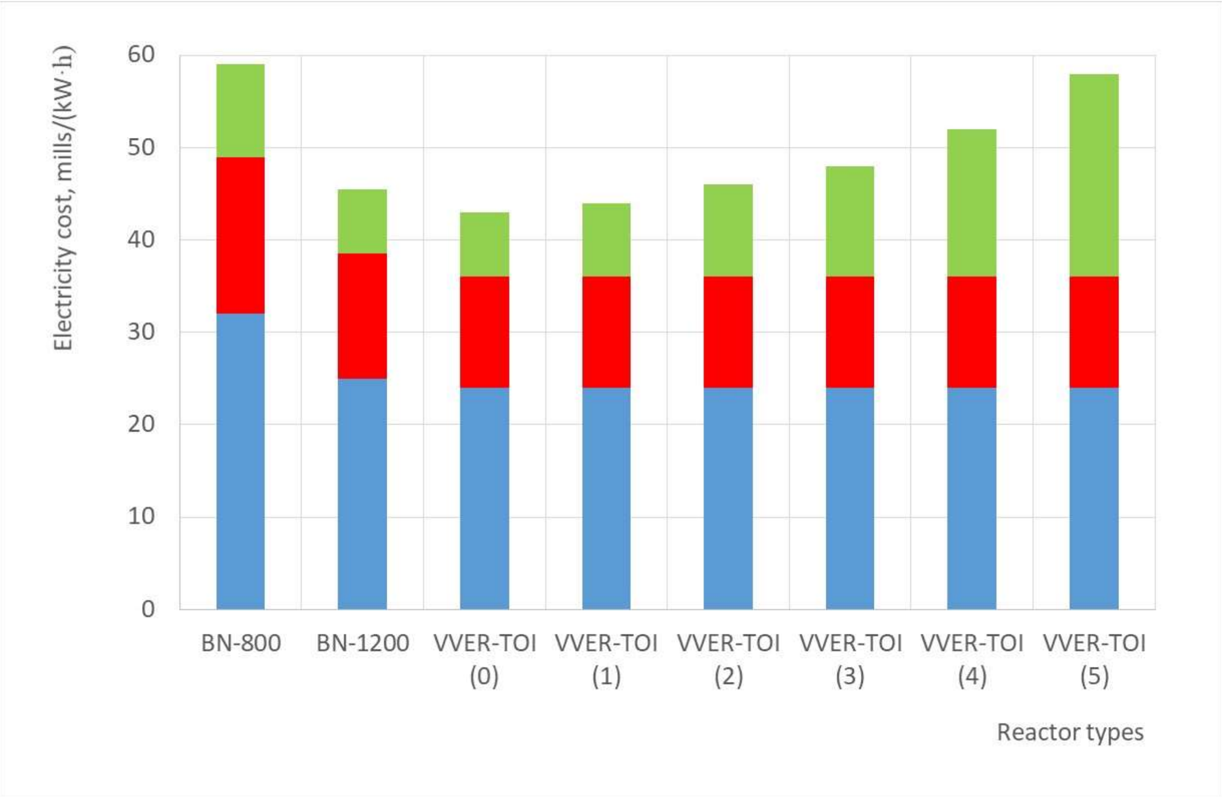

FIG. 8. Contributors to the total electricity costs of BN-800, BN-1200 and VVER-TOI at different uranium price escalation rates.

[39] Levelized unit lifecycle amortisation cost

3.3.2.3 Figures of merit

Figures of merit calculated in this sustainability assessment study include the internal rate of return, return on investment, net present value and total investment. Calculation of these parameters requires information on the price of electricity to be sold.

In the Russian Federation the price of electricity is normally determined through the bidding between electricity suppliers and customers in a given geographical and price zone. Prices include expenses necessary for electricity transfer and a surcharge for deployment of new power plants. In this study the price of baseload electricity sold by the NPP is evaluated at 52.8 mills/kWh value.

TABLE 9. FIGURES OF MERIT

Functions	BN-1200	VVER-TOI at different natural uranium price escalation rates					
		0	0.01	0.02	0.03	0.04	0.05
IRR, %/100	0.06	0.09	0.09	0.09	0.08	0.07	0.04
ROI, %/100	0.07	0.09	0.08	0.08	0.07	0.06	0.05
NPV, USD/kW(e)	1020	1530	1370	1060	750	120	-810
Total investment, 10^6 USD	4676	4641	4641	4641	4641	4641	4641

This price of electricity, 52.8 mills/kWh, is lower than the energy cost of BN-800, 59 mills/kWh. In this case the internal rate of return of BN-800 is lower than the discount rate, net present value is negative, and the quantitative results of calculation have limited value.

Internal rate of return, return on investment, net present value and total investment for BN-1200 and VVER-TOI have been calculated with NEST tool and results are presented in Table 9. Both IRR and ROI of the BN-1200 reactor are higher than the discount rate value used for cost calculations (0.05) which indicates that this reactor theoretically can be attractive for the investor. At the uranium cost escalation rates higher than approx. 2% the NPV of BN-1200 may exceed the NPV of VVER-TOI. The BN-1200 ROI may exceed the ROI of VVER-TOI at the uranium price escalation rates higher than 3%. At the uranium escalation rates higher than 4% the IRR of BN-1200 may exceed the IRR of VVER-TOI.

3.3.2.4 Sensitivity of electricity cost to the input data variation

Input data available at the early stages of development/ deployment of a nuclear energy system normally involve significant uncertainty. Some of the potential contributors to that uncertainty, e.g. effects of the economy of scale in different facilities, have been discussed above. Sensitivity of the levelized unit energy cost to the selected input parameters was studied using the cost sensitivity diagrams approach developed in INPRO.

Within this approach the sensitivity is defined as the levelized unit energy cost response to the variations of a selected parameter. Both the variations of selected parameter and the energy cost response are calculated in relative values which allows to compare effects and to plot several functions in one diagram. A given input parameter is varied while the rest of inputs are maintained constant. The cost sensitivity diagrams are calculated in four steps:

- Calculation of LUEC for a set of input data to be studied;
- Selection of input parameters to be varied (normally these parameters are selected among the most sensitive in different input categories);
- Calculation of LUECs for every variation of an input parameter;
- Conversion of all obtained LUECs into relative values by dividing by the value of LUEC calculated for original set of input data (see first bullet).

In this report the original value of LUEC (~46 mills/kWh) for the BN-1200 reactor has been calculated in section 3.3.2.2. The following parameters have been selected for the sensitivity study:

- Discount rate;
- Overnight cost;
- Operation and maintenance cost;
- Core fuel fabrication cost.

This set of parameters determines the BN-1200 electricity cost value calculated above.

Discount rate is normally the most sensitive parameter of the NPP electricity cost. It affects many cost constituents[40] including fuel costs and costs of individual fuel cycle services, however the most significant impact it provides is on the levelized unit lifecycle amortization cost which defines the interest to be paid on the investments. The value of discount rate depends on many conditions including the macroeconomic characteristics of a given country, national energy policy, the type of investor, energy market rules, risks associated with investment etc. Discount rates can be different for the same type of reactor planned to be constructed in different countries. They can be different for different types of power plants planned to be constructed in the same country. Some countries may provide a guidance on the evaluation of discount rates for different investment projects and in other countries the evaluation of discount rate can be based on academic studies.

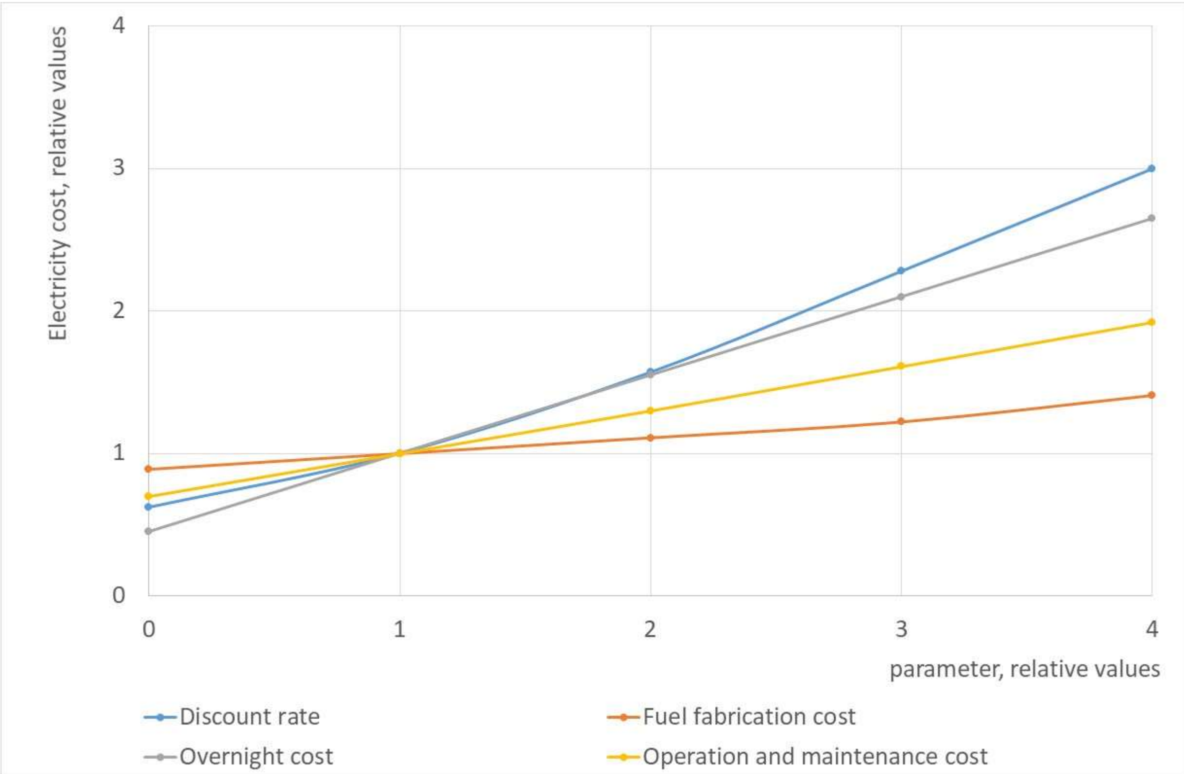

FIG. 9. Sensitivity of the electricity cost to the input parameters.

In this assessment study, the operation and maintenance costs were presented as a single parameter. Sensitivity of the energy cost was evaluated using this aggregated parameter.

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[40] Discount rate provides negligible effect on operation and maintenance cost.

In this assessment study, the MOX fuel manufacturing is the largest single service contributor to the fuel cost of BN-1200. Theoretically, changes in fuel cycle services cost data can recast the shares of their contribution to the fuel cost, however the shape of sensitivity curve is not expected to alter significantly.

The results of the sensitivity study are presented in Figure 9.

The growth of discount rate from 5% to 20% increases the levelized cost of electricity by a factor of three. Overnight cost is the second most sensitive parameter and its quadrupling raises the cost of energy by a factor of approx. 2.6. Operation and maintenance costs are less sensitive than discount rate and overnight cost. It should be noted that the MOX fuel fabrication cost is the least sensitive parameter in this group of four.

3.4. BASIC RESULTS OF BN-1200 SUSTAINABILITY ASSESSMENT IN THE AREA OF ECONOMICS

INPRO Economic Basic Principle: Energy and related products and services from nuclear energy systems shall be affordable and available.

3.4.1. User requirement UR1: Cost of energy.

The cost of energy supplied by nuclear energy systems, taking all relevant costs and credits into account, C_N, should be competitive with that of alternative energy sources, C_A, that are available for a given application in the same time frame and geographic region/jurisdiction

3.4.1.1. Criterion CR1.1: Cost competitiveness.

Indicator IN1.1: Cost of energy.

Acceptance limit AL1.1: $C_N \leq k \times C_A$ (C_N = cost of nuclear energy, and C_A = cost of energy from alternative source; factor k is usually ≥ 1 and is based on strategic considerations).

The cost of electricity from BN-800, BN-1200 and advanced water cooled reactor at different costs of natural uranium is discussed in section 3.3.2.2. BN-800 generates energy at the highest cost[41] among the considered reactors. The water cooled reactor has the lowest cost of electricity at the cost of natural uranium of 100 USD/kg. However, the difference between BN-1200 and water cooled reactor is very low (approx. 5% of the cost). At the uranium cost escalation rate higher than 2% the most affordable electricity may be produced by BN-1200.

3.4.2. User requirement UR2: Ability to finance.

The total investment required to design, construct, and commission nuclear energy systems, including interest during construction, should be such that the necessary investment funds can be raised.

3.4.2.1. Criterion CR2.1: Attractiveness of investment.

Indicator IN2.1: Financial figures of merit.

[41] BN-800 reactor is constructed as a part of BN development and deployment programme. It combines the generation of electricity with further improvements of SFR and closed fuel cycle technology.

Acceptance limit AL2.1: Figures of merit for investing in a nuclear energy system are comparable with or better than those for competing energy technologies.

The internal rate of return, return on investment and net present value of BN-1200 and advanced water cooled reactor at different costs of natural uranium are discussed in section 3.3.2.3. The water cooled reactor has the highest figures of merit at the cost of natural uranium of 100 USD/kg. However, the internal rate of return and return on investment of BN-1200 are higher than the discount rate which means that BN-1200 theoretically can generate reasonable profit. The BN-1200 return on investment may exceed water cooled reactor at the uranium cost escalation rates higher than 3%. The BN-1200 internal rate of return may exceed water cooled reactor at the uranium cost escalation rates higher than 4%. The BN-1200 net present value may exceed water cooled reactor at the uranium cost escalation rates higher than approx. 2%. Figures of merit for investing in BN-1200 and closed fuel cycle technology are comparable with those for water cooled reactors at the uranium cost escalation rates higher than 2%.

3.4.2.2. Criterion CR2.2: Investment limit.

Indicator IN2.2: Total investment.

Acceptance limit AL2.2: The total investment required should be compatible with the ability to raise capital in a given market climate.

The total investment of BN-1200 and advanced water cooled reactor are discussed in section 3.3.2.3. The difference between total investment in BN-1200 and water cooled rector is lower than 1%.

There are 36 commercial reactors operating in the Russian Federation. Several more commercial reactors are being constructed or planned. Rosatom plans to construct more than 30 commercial reactors abroad and in a few cases to invest its own funds into those international projects. The total investment required is compatible with the ability to raise capital in the Russian Federation market climate.

3.4.3. User requirement UR3: Investment risk.

The risk of investment in nuclear energy systems should be acceptable to investors.

3.4.3.1. Criterion CR3.1: Maturity of design.

Indicator IN3.1: Technical and regulatory status.

Acceptance limit AL3.1: Technical development and status of licensing of a design to be installed or developed are sufficiently mature.

At the moment of BN-1200 sustainability assessment the decision on reactor construction has not been made and the information on status of licensing was not available to the assessor. Assessment against this criterion is not complete.

3.4.3.2. Criterion CR3.2: Construction schedule.

Indicator IN3.2: Project construction and commissioning times used in economic evaluation.

Acceptance limit AL3.2: Times for construction and commissioning used in economic evaluation are sufficiently accurate, i.e. realistic and not optimistic.

In this assessment study the construction period of BN-1200 reactor was assumed to be 6 years. The prototype and demonstration reactors (BN-600 and BN-800) construction periods lasted longer than ten years in every case, however both reactors were used for the technology development including development of construction technologies. Improvements of the BN-1200 construction technologies are discussed in section 4.3.1.3 and more information can be found in the referenced materials. Ref. [17] states that BN-1200 reactor concept enables construction of 'N-th-of-a-kind' reactor in 48 months. In the case of BN-1200, shorter construction periods than for BN-600 and BN-800 can be reasonably expected, however the accuracy of construction time estimation cannot be assessed in this study.

3.4.3.3. Criterion CR3.3: Uncertainty of economic input parameters.

Indicator IN3.3: A sensitivity analysis of important input parameters for calculating costs and financial figures of merit has been performed.

Acceptance limit AL3.3: Sensitivity to changes in selected parameters is acceptable to investor.

The sensitivity analysis of important input parameters for calculating costs and financial figures of merit has been discussed in section 3.3.2.4. The most sensitive parameter is the discount rate which mostly depend on the investment conditions and interest rate associated with invested funds. The investment conditions in the State corporation Rosatom differ from the free market and private investors conditions. In the case of BN-1200 construction the discount rate changes can be expected to be moderate if any.

The basic cost structure of the BN-1200 energy is similar to the basic cost structure of water cooled reactor (Figure 8) and the sensitivity of the BN-1200 overnight capital cost and operation and maintenance cost is very close to those of water cooled reactor. Sensitivity of the fuel cycle services costs is relatively low and the effect of potential increase of these costs can be expected to be reasonably low.

Sensitivity to changes in selected parameters is deemed to be acceptable to investor.

3.4.3.4. Criterion CR3.4: Political environment.

Indicator IN3.4: Long term commitment to nuclear option.

Acceptance limit AL3.4: Commitment sufficient to enable a return on investment.

Long term commitment of the Russian Federation government to nuclear option and support of the sodium cooled fast reactor programme was discussed in section 3.2. Commitment is deemed sufficient to enable a return on investment.

3.4.4. User requirement UR4: Flexibility.

Innovative nuclear energy systems should be compatible with meeting the requirements of different markets.

3.4.4.1. Criterion CR4.1: Flexibility.

Indicator IN4.1: Are the innovative nuclear energy system components adaptable to different markets?

Acceptance limit AL4.1: Yes.

The BN-1200 is designed to use MOX or nitride fuels in a closed nuclear fuel cycle. In the case of domestic operation of BN-1200 or when nuclear fuel cycle facilities such as reprocessing and fabrication of nuclear fuel that use reprocessed fissile materials are located in the same country, the reactor can be fueled by different fuels providing additional benefits and flexibility to the operator. However, scenarios with import and export of closed fuel cycle services may require additional consideration of the potential proliferation resistance issues which are out of the scope of this assessment study.

The power rating of BN-1200 was selected to fit the Russian Federation electricity grid. Compatibility of the BN-1200 with small grids will need study on an ad hoc basis when necessary. Information on the BN-1200 design characteristics for reactor operation in non-baseload regime has not been available to the assessor. Assessment against this criterion is not complete.

4. INPRO SUSTAINABILITY ASSESSMENT IN THE AREA OF REACTOR SAFETY

4.1. INTRODUCTION

INPRO basic principle for sustainability assessment in the area of nuclear reactor safety [7]: The safety of the planned nuclear installation is superior to that of the reference nuclear installation[42] such that the frequencies and consequences of the accidents are greatly reduced. In the event of an accident, off-site releases of radionuclides[43] are prevented or mitigated so that there will be no need for public evacuation[44].

The INPRO methodology has developed seven INPRO user requirements for nuclear energy system sustainability assessment in the area of reactor safety to specify in more detail the main measures presented above. The role of the INPRO assessor is to check, based on evidence provided by the designer, whether the designer has implemented the necessary measures as required by the INPRO methodology. The assessor's product is therefore not an assessment of compliance with the IAEA Safety Standards but rather a sustainability assessment against the INPRO user requirements and criteria.

4.2. OVERVIEW OF THE APPLICATION OF THE INPRO METHODOLOGY AREA OF SAFETY TO THE FAST REACTORS UNDER DEVELOPMENT

INPRO methodology states that assessment can be carried out by a technology developer at any stage of the development of an advanced reactor design. Limited scope INPRO assessments can be focused on the specific areas and specific installations in a nuclear energy system. A developer can check whether its design under development meets the INPRO methodology sustainability criteria regarding nuclear safety. Limited scope studies may assess reactor designs under development and may help to highlight gaps to be closed by on-going R&D studies and to define the scope of data needed for making a future judgement on system sustainability. Design modifications can be initiated during early stages of development if necessary, to improve the safety level of its design. The extent and available level of detail of design and safety assessment information will increase as the design of an advanced reactor progresses to the detailed design. The amount of information available will be a significant factor in the uncertainty of the long term validity of conclusions drawn on whether an INPRO methodology criterion has been met by the advanced design.

INPRO methodology in the area of reactor safety sets up the goal to be achieved by the sustainable nuclear energy system as 'superiority of the reactor safety to the safety of reference

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[42] Within this publication, a reference reactor is a reactor of the latest design operating in 2013. It should preferably be designed by the same corporate designer as the reactor assessed and using the same technology. For innovative reactors that may have no operating prototypes in 2013, the latest design that has been licensed and safely operated, or at least licensed. can be used as the reference design.

[43] If significant amounts of toxic chemicals are used in the reactor design (e.g. as coolants or fuel forms in innovative reactors) or can be generated during the reactor operation or accidents the potential accidental releases of toxic chemicals has to be considered as part of the INPRO assessment. The INPRO criteria used for the assessment of potential releases of toxic chemicals should be similar to the INPRO criteria developed for the assessment of radioactive releases.

[44] Other protective measures may still be needed. Effective emergency planning, preparedness and response capabilities will remain a prudent requirement as discussed in the INPRO methodology manual for the area of infrastructure.

reactor and prevention / mitigation of off-site releases so that there will be no need for public evacuation'[7]. The user requirements specify how to achieve the goal:

- The nuclear reactor assessed is more robust than a reference design with regard to operation and systems, structures and components failures (robustness of design during normal operation);
- The nuclear reactor assessed has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions (detection and interception of AOOs);
- The frequency of occurrence of DBAs in the nuclear reactor assessed is reduced[45]. If an accident occurs, engineered safety features are able to restore the reactor to a controlled state, and subsequently to a safe shutdown state, and ensure the confinement of radioactive material. Reliance on human intervention is minimal, and only required after a sufficient grace period;
- The frequency of an accidental release of radioactivity into the containment / confinement is reduced. If such a release occurs, the consequences are mitigated, preventing or reducing the frequency of occurrence of accidental release into the environment. The source term of the accidental release into the environment remains well within the envelope of the reference reactor source term and is so low that calculated consequences would not require evacuation of the public;
- An assessment is performed to demonstrate that the defence in depth (DID) levels are more independent from each other than in the reference design. To excel in safety and reliability, the nuclear reactor assessed strives for better elimination or minimization of hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics and/or passive systems, when appropriate;
- Safe operation of the nuclear reactor assessed is supported by accounting for human factor requirements in the design and operation of the plant, and by establishing and maintaining a strong safety culture in all organizations involved;
- The development of innovative design features of the nuclear reactor assessed includes associated research, development and demonstration (RD&D) to bring the knowledge of plant characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for operating plants.

The developer of sustainable energy technology is expected to meet these user requirements. INPRO assessor employs 28 criteria which can be used for checking the status of nuclear energy system in relation to the INPRO user requirements. Four of those criteria are divided in 21 evaluation parameters for clarity. The assessor procures a total of 45 items for the INPRO assessment in the area of reactor safety:

- Design of normal operation systems:
  - Margins of design;
  - Design simplification;
  - Improved fabrication and construction;
  - Improvement of materials;
  - Redundancy of operational systems;
- Reactor performance:
  - Margins of operation;
  - Reliability of control systems;

[45] Here and further in this section when the improvement of BN-1200 characteristics is discussed the reactor is compared with the reference design.

  - Ageing management;
  - Impact from incorrect human intervention;
  - Sufficient technical documentation;
  - Appropriate training programmes;
  - Plant management organization;
  - Use of worldwide operating experience;
- Inspection, testing and maintenance;
- Failures and deviations from normal operation;
- Occupational dose;
- Instrumentation and control (I&C) system and inherent characteristics:
  - Continuous monitoring of plant health;
  - Capability of I&C system;
  - Compensation of deviations from normal operation;
- Grace periods after AOOs;
- Inertia;
- Frequency of DBAs;
- Grace period for DBAs;
- Engineered safety features;
- Barriers;
- Subcriticality margins;
- Frequency of release into containment / confinement;
- Robustness of containment / confinement design;
- Accident management;
- Frequency of accidental release into environment;
- Source term of accidental release into environment;
- Independence of DID levels;
- Minimization of hazards:
  - Stored energy;
  - Flammability;
  - Excess reactivity in the core;
  - Reactivity feedbacks;
  - Criticality outside the reactor core;
- Passive safety systems;
- Human factors;
- Attitude to safety;
- Safety basis and safety issues;
- RD&D;
- Computer codes;
- Novelty;
- Safety assessment.

Several INPRO criteria in this area require comparing the reactor under assessment to the reference design. In the INPRO methodology area of reactor safety, a reference reactor (or design) is a reactor of the latest design operating in 2013. It should preferably be designed by the same corporate designer as the reactor assessed and using the same technology. Based on previous experience with INPRO assessments, the definition of date for the selection of the reference design helps to avoid potential misinterpretations of terms. Note that 2013 was the date selected at the beginning of the latest methodology update. This date should be revised periodically along with the rest of the INPRO methodology.

The reference design selected for this assessment, the BN-800 reactor in Beloyarsk NPP, was commissioned in 2016. This may create a few challenges for the INPRO study of the BN-1200 reactor. For example, the accumulated operating experience of BN-800 may be insufficient for the assessment of reactor performance. In such cases the data from BN-600 can be used to support the reference reactor data.

At the time of this report preparation the BN-1200 design has not been fully optimised and finalised. Further improvement of the safety characteristics and more detailed data on the reactor characteristics could be reasonably expected. However, input data compiled in the framework of this study and presented in this report are deemed to be sufficient for the limited scope INPRO assessment to inform the developers of BN-1200 on the actions to be taken and criteria to be met to achieve the system sustainability. The assessment (self-assessment) is expected to be completed in the future along with the development of detailed design of the reactor.

4.3. UR1: ROBUSTNESS OF DESIGN DURING NORMAL OPERATION

INPRO methodology user requirement UR1 for sustainability assessment in the area of safety of nuclear reactor was formulated as follows [7]: "The nuclear reactor assessed is more robust than a reference design with regard to operation and systems, structures and components failures".

4.3.1. Criterion CR1.1: Design of normal operation systems

Indicator IN1.1: Robustness of design of normal operation systems.

Acceptance limit AL1.1: More robust than that in the reference design.

4.3.1.1. Evaluation parameter EP1.1.1: Margins of design

The BN-1200 and BN-800 are large pool type sodium cooled fast reactors. Both in the BN-1200 and BN-800 the reactor core, primary radiation shielding, primary coolant pumps and intermediate heat exchangers are located within the main reactor vessel. Main reactor vessel is enclosed in the guard vessel. Basic characteristics of BN-1200 and BN-800 reactors are provided in Table 10.

BN-800 reactor uses hexagonal fuel assemblies with 96 mm flat-to-flat outside duct size (fuel assembly pitch – 100 mm). The diameter of fuel pins in all assemblies is the same and amounts 6.9 mm. BN-1200 uses enlarged hexagonal fuel assemblies with 181 mm flat-to-flat outside duct size and fuel pins of different diameters – 9.3 mm and 10.5 mm [31]. Different fuel pins are used for designing required power distribution in the reactor core. Besides that, fuel modifications in the BN-1200 reactor involve enhancement of the gaseous plenum for accumulation of fission products at higher burnups.

Due to these modifications the volumetric power density in the BN-1200 reactor core has been reduced by a factor of 1.8 compared against BN-800. Ref. [32] estimates average power density in BN-800 reactor core at 400–450 MW/m^3 and in BN-1200 – at 237–255 MW/m^3. Maximum linear heat generation rate in BN-800 reactor core fuel is 48.5 kW/m [33] and in BN-1200 – 46.5 kW/m [34].

Melting point of the mixed oxide (MOX) fuel is 2750°C. At normal operation of the BN-800 reactor the fuel temperature may reach 2500°C with a corresponding safety margin of 250°C. The highest fuel temperature that the BN-1200 reaches is 2100°C and the fuel temperature safety margin of 650°C.

TABLE 10. BASIC CHARACTERISTICS OF BN-800 AND BN-1200 REACTORS [13, 15, 16]

Characteristics	Reactors	
	BN-800	BN-1200
Thermal power rating, MW(th)	2100	2800
Design lifetime of non-replaceable equipment, a	40	60
Specific metal content of the reactor, t/MW(th)	4.1	2.4
Number of coolant circuits	3	3
Primary coolant temperature, in/out, ºC	354/547	410/550
Secondary coolant temperature, in/out, ºC	309/505	355/527
Tertiary coolant temperature, in/out, ºC	210/490	275/510
Number of loops in every circuit	3	4
Steam generator design	modular	shell and tube type
SG outlet steam pressure, MPa	13.7	17
Steam reheating	steam	steam
Electric power rating, MW(e)	880	1220
NPP efficiency, gross/net, %	41.9/38.8	43.6 /40.9
Turbine type	K-800-130	K-1200-130
Generator type	TZV-800-2	TZV-1200-2
Estimated load factor, %	85	90

Maximum fuel cladding temperature at normal operation conditions in all BN family reactors is limited by 710°C. In different zones of the BN-800 reactor core the fuel claddings may reach 670 – 690°C providing a safety margin of 20 C. In BN-1200 at normal operation the claddings temperature does not exceed 680°C in any part of the reactor core and the safety margin amounts at least 30°C.

4.3.1.2. Evaluation parameter EP1.1.2: Design simplification

Unlike other members of the BN family of reactors, the BN-1200 design layout is symmetric and the layout of all four loops in every circuit is identical which simplifies the manufacturing and construction of reactor systems structures and components.

BN-800 secondary circuit scheme [35] including steam generators and EHRS is presented in Figure 10. Three loops are connected to the reactor vessel and 60 modules of steam generators. BN-1200 technology scheme [35] is presented in Figure 11. Four identical loops (located in identical premises at the same levels) and EHRS modules are connected to the reactor vessel in the centre.

Using bellows type compensators in secondary circuit of BN-1200 instead of compensating elbows as it was done in BN-800 resulted in reducing of the total length of secondary sodium pipelines from ~770 m to ~400 m [36]. The BN-1200 design reduced the number of fittings by about 930 [37]. Table 11 provides a comparison of fittings used in the BN-800 and BN-1200 reactors.

BN-1200 design includes 8 vertical steam generators of 'shell and tube type' instead of 60 modules of 'modular type' steam generators in BN-800 [35, 37]. Unlike BN-800 the level of sodium in BN-1200 reactor does not reach the conical part of the reactor vessel which eliminates the necessity of guard vessel and bellows in this part of reactor [32].

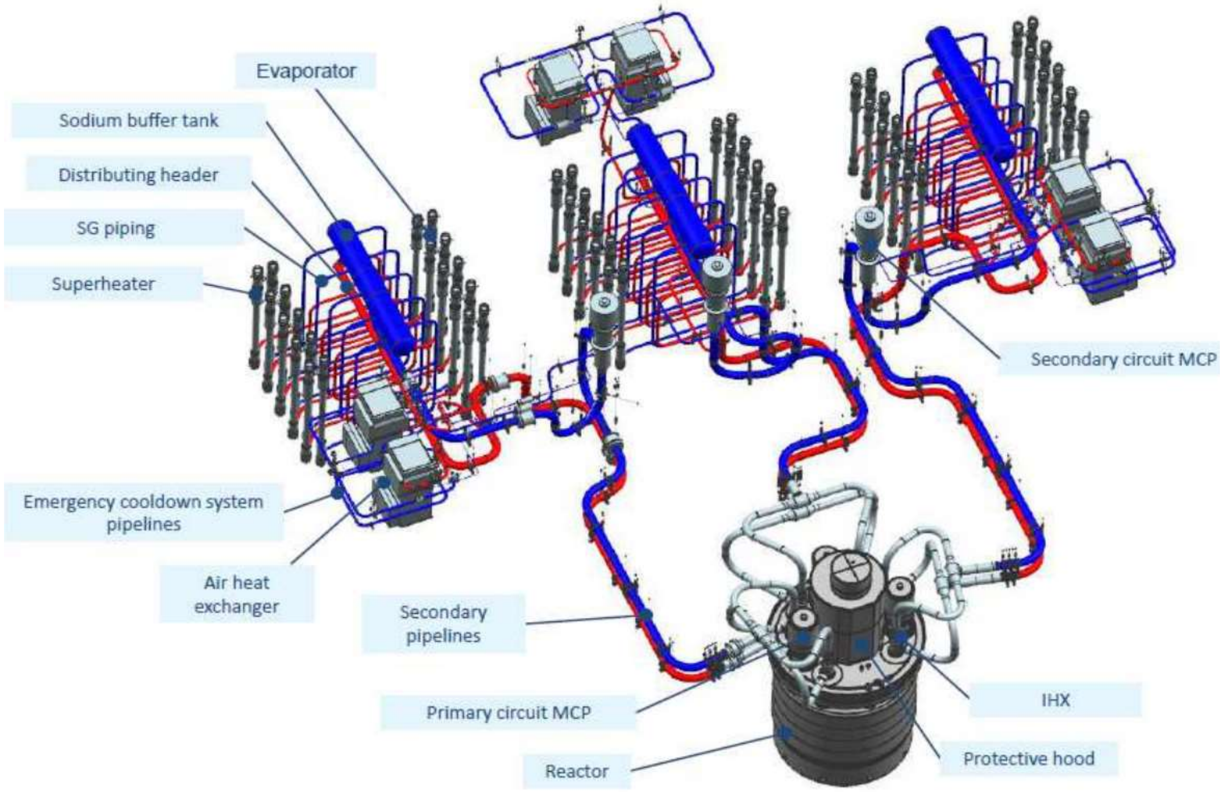

FIG. 10. BN-800 secondary circuit scheme [20].

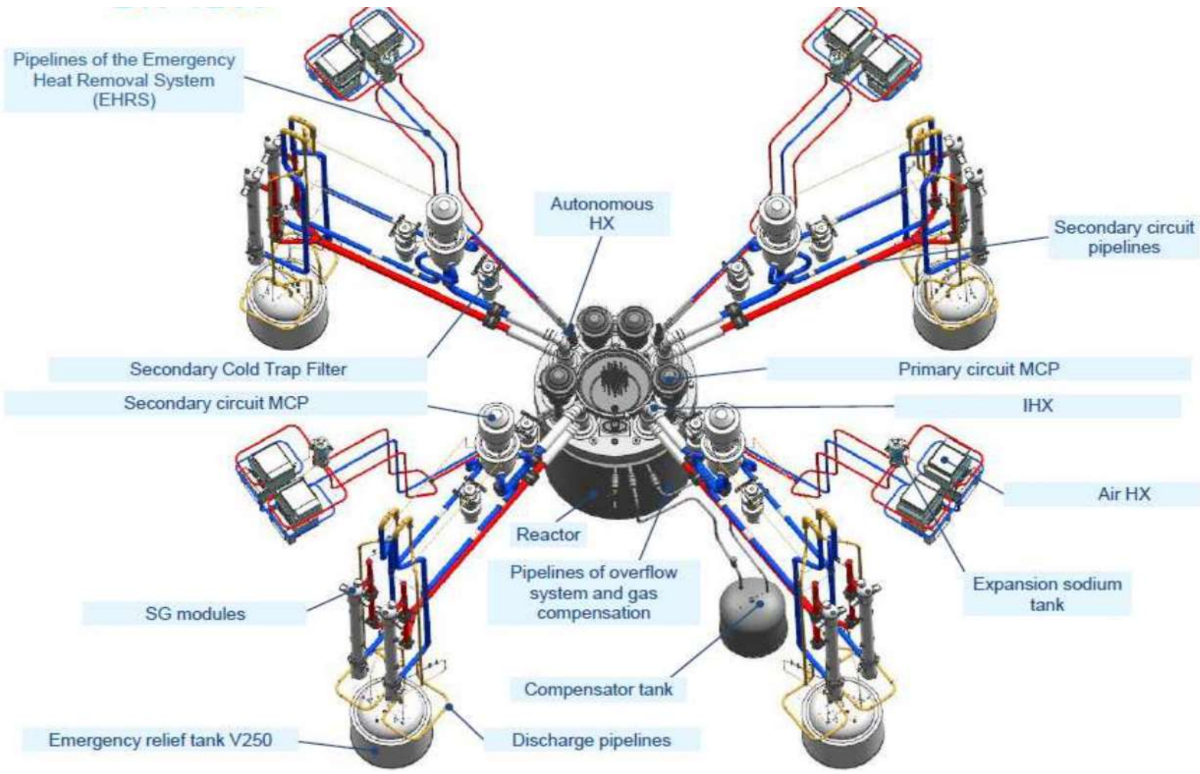

FIG. 11. BN-1200 technology scheme [20].

TABLE 11. FITTINGS USED IN BN-800 AND BN-1200 [37]

Fittings	BN-800	BN-1200
	Sodium fittings	
DN10-DN100	870	370
DN300	75	-
DN400	-	20
	Steam/ water fittings	
DN10-DN100	438	130
DN175-DN300	102	35

BN-800 reactor core contains 565 fuel assemblies having three different initial fuel composition (different content of fissile material). The 432 fuel assemblies of BN-1200 reactor core [35] have bigger cross-sections and initially have uniform fuel composition which simplifies the reloading procedures [34].

Increasing the capacity of in-vessel short term spent fuel storage allowed storing the spent fuel for 2 years and to reduce residual heat density from a spent fuel assembly to 2 W/cm^3 [32]. These low residual heat densities avoid needing external 'drum' storage (using sodium for cooling of spent fuel assemblies) [38] and a number of related auxiliary systems (for cooling, draining, sodium cleaning, protection from high pressure, oxides monitoring, cooling of oxides monitoring probes, etc). The BN-1200 fuel reloading system design includes the vertical elevator, combined reloading/washing section and other features differing from the BN-800 [38]. Using these features the BN-1200 reactor fuel reloading system was further simplified and the mass of steels and other materials (per kW of reactor power rating) used for the system manufacturing was reduced by a factor of 10 [39].

The BN-800 reactor was designed for using MOX fuel. The BN-1200 reactor design plans for use of MOX fuel only in early stages of the programme. It is expected that at the more advanced stages when nitride fuel technology is validated BN-1200 reactors will operate with nitride fuel [40]. The 174 blanket assemblies around the BN-1200 reactor core fuelled with nitride fuel will be replaced with steel/boron-carbide shielding that strongly reduces the neutron irradiation of in-vessel structures and components [41]. This allows elimination of in-vessel shielding which is used in the BN-800 reactor.

4.3.1.3. Evaluation parameter EP1.1.3: Improved fabrication and construction

BN-1200 reactor building is designed against new and more strict regulatory requirements [42]. External structures of the building are reinforced, and the expansion joint has been introduced along the confinement walls to prevent the transmission of harmful impact to the reactor systems structures and components in case of an accident. A new relatively light spherical dome is envisioned to be placed above the reactor. The dome is to be assembled using the modules fabricated from the modern high-strength reinforced concrete which increases the efficiency of construction process [42].

Other expected improvements in the construction of BN-1200 are mostly related to the reactor layout [43]. All four loops of secondary cooling circuit are identical and corresponding systems structures and components are located symmetrically at the same levels which allows to use an approach conceived for construction of a modular facility.

The Russian architect engineering company NIIAEP has developed an original 'Multi-D' digital technology which can be used at the design, construction and operation stages of the reactor lifecycle [44]. It creates a complete universal 3D (design + construction + technology) model of the object, controls potential discrepancies and collisions, manages the configuration at any details level, communicates among users and prepares working level documentation. This information system allows modelling of all basic construction processes, preparation of detailed graphs and involves several modes of in-field application. If this tool confirms its effectiveness it can theoretically help to reduce a potential negative impact of human factor during the construction of BN-1200.

The BN-1200 major equipment fabrication methods are not expected to differ from those used for BN-800 [43]. The reactor vessel assembly will be completed on site. The only expected difference is the necessity of on-site assembling of the large rotating plug of BN-1200, however this procedure cannot be considered as an improvement of fabrication process.

4.3.1.4. Evaluation parameter EP1.1.4: Improvement of materials

The BN-1200 design involves a number of modifications of structural material in fuel pins, reactor internals, steam generators, anti-cavitation coatings, etc. The designer used new high-strength stainless steel 07Cr12NiMoVNb instead of 10Cr2Mo in BN-800 [43, 45] to increase the durability and lifetime of the BN-1200 steam generators. The lifetime of the BN-800 steam generator module reaches 150 000 hours. BN-1200 steam generator's lifetime extends to 240 000 hours [45].

Several selected in-reactor systems, structures and components of BN-1200 will be fabricated using the new heat and radiation resistant steel Cr16Ni11Mo3. This material allows an increase of system's lifetime up to 60 years and improve safety characteristics of the reactor [38].

BN-600 and BN-800 reactors use austenitic steel EK-164 for fuel claddings. BN-1200 uses both austenitic steel EK-164 and ferritic-martensitic steels EK-181 and ChS-139 as fuel cladding materials which improves safety features (higher heat resistance, higher damaging dose) economics of fuel management (higher burnups) [46, 47].

As noted above, the BN-800 reactor can operate with both UOX and MOX fuel. The BN-1200 reactor will use MOX fuel only in early steps of the programme with nitride fuel expected for later use [40]. Nitride fuel provides higher density of fuel material and improves safety and economics characteristics of reactor [48, 49].

New effective steel/boron-carbide shielding surrounding the core strongly reduces the neutron irradiation of reactor vessel and in-vessel structures and components [41]. This allowed a reduction of the steel shielding located within the BN-1200 reactor from 830 t to 27 t.

4.3.1.5. Evaluation parameter EP1.1.5: Redundancy of operational systems

It is acknowledged that an increase of redundancy may increase the complexity of a system as discussed in evaluation parameter EP1.1.2 above. Thus, the design needs optimization with respect to system redundancy and complexity. Design simplification generally cannot be used as justification for reducing the redundancy of operational systems.

Data demonstrating that the redundancy of operational systems is greater than that in the reference design have not been available to the INPRO assessor. Assessment against this evaluation parameter is not complete.

4.3.2. Criterion CR1.2: Reactor performance

Indicator IN1.2: Reactor performance attributes.

Acceptance limit AL1.2: Superior to those of the reference design.

4.3.2.1. *Evaluation parameter EP1.2.1: Margins of operation*

Prior to the assessment of this evaluation parameter, it is important to note that prototype reactor BN-600 has relatively high performance characteristics compared to the currently operated water cooled reactors. Ref. [34] presents evaluation of scram frequency in BN-600 (0.2 per 7000 hour of operation) compared to global average value (0.5–0.7 per 7000 hours per power unit). There were no scrams in BN-600 in the period from 2000 through 2013.

Improvement of operational margins in BN-1200 reactor is mostly related to the reduction of the maximum power density of the core to 380 MW/m^3 which is lower than average power density in BN-800 and BN-600 reactors.

Ref. [32] provides the list of 12 normal operation states in BN-1200:

- 5 working states including reactor operation at different power levels, reactor start-up and shutdown from/to different states (hot or cold);
- 5 shutdown states including fuel reloading, steam generators chemical cleaning, etc;
- working in a mode providing energy for own needs only;
- manoeuvrable mode of operation.

Ref. [50] presents a list of normal operation levels and margins identified for BN-1200. Normal operation levels for primary coolant temperature at the outlet from the core and secondary coolant temperature at the reactor outlet can be compared with the nominal values of these parameters to estimate corresponding margins:

- Primary coolant temperature at the outlet from the core 550–600°C;
- Secondary coolant temperature at the reactor outlet 527–550°C;
- Pressure in the reactor gas cavity – 0.005–0.054 MPa;
- Gas pressure in the secondary circuit – 0.25–0.28 MPa.

Other normal operation levels provided in Ref. [50] are as follows:

- Reactor vessel temperature (60 years) ≤ 504°C;
- Reactor components temperature change rate ≤ 30°C/hour;
- Gas activity in the reactor gas cavity ≤ 500 MBq/l;
- Primary sodium activity ≤ 50 MBq/kg;
- Fuel pins damage of gas leak type – 0.05% of all fuel pins;
- Fuel pins damage of fuel-coolant contact type – 0.005% of all fuel pins.

The normal operation levels and margins of the BN-800 reactor are not available in the open domain to allow the assessor to finalize assessment.

4.3.2.2. *Evaluation parameter EP1.2.2: Reliability of control systems*

Reference design, BN-800, is one of the newest and most advanced currently operated reactor designs. Development of I&C system for BN-800 was based on the contemporary requirements [32]. Distributed control principle has been implemented in the system providing coordinated and independent processing and control functions to different system modules. This system fulfils the detailed diagnostics of reactor systems and self-diagnostics.

BN-800 reactor control and protection system is one of the I&C subsystems that combines control functions (e.g. reactor start-up, power control, shutdown etc), protective functions (emergency power level reduction, scram) and monitoring functions (core and neutron guide monitoring; systems structures and components monitoring; information exchange with other

I&C subsystems). The system was designed to maintain the following sequence of control priorities:

1. Reactor protection (scram);
2. Power level reduction;
3. Prevention withdrawal of control rods;
4. Manual control of reactor;
5. Automatic control of reactor.

Reactor control and protection system consists of several subsystems:

– Digital automatic reactor shutdown systems (scram);
– Analogue automatic reactor shutdown systems (scram);
– Digital control and monitoring subsystem;
– Driving gears control subsystem;
– Main control room safety panels;
– Emergency control room safety panels.

The control and monitoring subsystem design has a triple reservation and performs automated reactor control in start-up, manoeuvring and scheduled shutdown modes, carries out automatic reactor power control, emergency power reduction control, compensation of fuel burn-up effects, control of electric actuators of driving gears prior to and after reloading, monitoring and indication of the control rods position, of system structures and components status, automatic self-diagnostics of the system hardware and software in all states of the reactor.

The BN-1200 control systems have a basis in the BN-800 developments [32] and will accommodate a number of improvements related to the reactor optimization and simplification. For example, BN-800 neutron flux monitoring system contains a neutronic guide, two sets of in-reactor ionization chambers (2×4 KNT 54-2) and 24 near-reactor ionization chambers KNK 15-1. Use of four sets of in-vessel ionization chambers for the neutron flux monitoring and elimination of neutron guides improves the reliability of control systems in BN-1200.

4.3.2.3. Evaluation parameter EP1.2.3: Ageing management

In the operating prototype fast reactor BN-600 an ageing management programme was introduced more than two decades ago [51–54]. It is expected that in the BN-1200 reactor the ageing management programme will cover all steps of the project development. At the design stage, the BN-1200 designer determined the design life of items important to safety (e.g. see Refs [14, 35, 41]) and expected to provide appropriate design margins to take due account of age-related degradation and to provide methods and tools for assessing ageing during the NPP operation.

The detailed design of BN-1200 involves development of complex systems for diagnostics of damage and evaluation of residual lifetime of the reactor components and core structural elements [55].

4.3.2.4. Evaluation parameter EP1.2.4: Impact from incorrect human intervention

Data demonstrating that incorrect human intervention during normal operation has less impact on reactor operation than in the reference design have not been available to the INPRO assessment. Assessment against this evaluation parameter is not complete.

4.3.2.5. Evaluation parameter EP1.2.5: Sufficient technical documentation

The first BN-1200 reactor is planned to be constructed at Beloyarsk NPP site. There are four reactors currently located at this site including BN-600 and BN-800. Beloyarsk NPP is the oldest currently operating NPP in the Russian Federation and well-established enterprise having a very well developed documentation system. The general quality assurance programme of the Beloyarsk NPP POCAS (O), put into effect by order of national utility, Rosenergoatom JSC, in 2014. Detailed documented procedures for a quality management system have been developed and implemented at Beloyarsk NPP (titles of documents) [56]:

- Records Management at Beloyarsk NPP;
- Management of Technical Documentation for the Industrial Activities of Beloyarsk NPP;
- Paperwork Manual at Beloyarsk NPP;
- Planning and Monitoring of Occupational Activities at Beloyarsk NPP;
- Maintenance and Repair at Beloyarsk NPP;
- Metrological Support at Beloyarsk NPP;
- Procurement Procedures at Beloyarsk NPP;
- Human Resource Management at Beloyarsk NPP;
- Non-compliance. Corrective and Preventive Actions;
- Regulation on the Organization and Conduct of Internal Quality Audits at Beloyarsk NPP.

The quality management system of Beloyarsk NPP is subject to certification peer-reviews organized on a regular basis. In 2014, the ANO Atomcertifica (Russian independent certification body) conducted the peer-review which confirmed compliance of the quality management system of the Beloyarsk NPP to the national and international requirements.

Ref. [57] gives a brief overview of technical documentation related to the nuclear hazardous works including relevant testing. These documents include:

- Beloyarsk Nuclear Power Plant. Final Safety Assessment Report;
- Nuclear Safety Manual for Storage, Transportation and Reloading of Nuclear Fuel of AMB-100 and AMB-200 reactors (Units 1 and 2);
- Operating Manual of First Line of Beloyarsk NPP (Units 1 and 2);
- Operating Manual of Unit 3 of Beloyarsk NPP (BN-600);
- Operating Manual of Unit 4 of Beloyarsk NPP (BN-800);
- Nuclear Safety Manual for Storage, Transportation and Reloading of Nuclear Fuel of BN-600 reactor (Unit 3);
- Nuclear Safety Manual for Storage, Transportation and Reloading of Nuclear Fuel of BN-800 reactor (Unit 4).

BN-1200 reactor is reasonably expected to be provided with full set of necessary technical documentation.

4.3.2.6. Evaluation parameter EP1.2.6: Appropriate training programmes

The first BN-1200 reactor is planned to be constructed at Beloyarsk NPP site where BN-600 and BN-800 reactors are accommodated. Operating experience of these two reactors will be used in the training programmes to be developed for BN-1200 [58]. Well qualified personnel from BN-600 and BN-800 operating staff can be used for training of new staff and for commissioning and operation of BN-1200.

Training facilities of BN-800 reactor include a full-scale simulator created in 2015–2019 [59] and some of these facilities can be used for the BN-1200 personnel training.

4.3.2.7. Evaluation parameter EP1.2.7: Plant management organization

BN-800 reactor of Beloyarsk NPP was commissioned in 2016 (first criticality obtained in 2015). Russian companies that participated in the construction and commissioning of BN-800 including the designer organization (OKBM) and companies affiliated to the national utility (Rosenergoatom) have the necessary expertise for start-up and operation, and the competence of utility and Beloyarsk NPP is appropriate and meets national licensing requirements.

These organizations proved to have enough personnel, well defined structures, functions, lines of management and detailed qualification requirements/job descriptions. Beloyarsk NPP, like other Russian NPPs, has identified managers responsible for operation, maintenance, technical support, quality assurance, environmental protection, nuclear and industrial safety, and administration.

4.3.2.8. Evaluation parameter EP1.2.8: Use of worldwide operating experience

BN-1200 design is based on the experience obtained from the operation of BN-350, BN-600 and BN-800 reactors [18, 60]. BN-1200 designer participates in the information exchange programmes related to safety of LMFR and organized by the IAEA [61, 62] and other international organizations. JSC OKBM has developed several types of reactors different from BN-1200 (including several types of propulsion reactors for commercial fleet and navy) and design of fuel assemblies for commercial VVER-1000 (Russian type of large PWR).

However, the assessor had no direct information available on the accounting of operating experiences from foreign NPPs operating types of reactors other than fast reactors. Some experience from water cooled reactors may be applicable to the fast reactors either (e.g. the Forsmark NPP 2006 electrical event). Assessment against this evaluation parameter is not complete.

4.3.3. Criterion CR1.3: Inspection, testing and maintenance

Indicator IN1.3: Capabilities to inspect, test and maintain.

Acceptance limit AL1.3: Superior to those in the reference design.

The experience of operating the BN-600 power unit shows that leaks from sodium pipelines and in steam generators took place mainly in the early period of reactor operation. Leaks were mostly caused by deviations in the quality of equipment manufacturing. Pure sodium provides very low corrosive-erosive effects on the materials of systems, structures and components.

Total number of sodium leaks in BN-600 added up to 39 including 12 leaks in steam generators. The rest of the 27 leaks occurred in the auxiliary systems including 5 leaks from primary circuit systems. Six leaks exceeded 10 kg of sodium (largest mass of sodium leak obtained 1000 kg – level 1 of INES scale). One leak from the primary circuit caused a minor radioactive release to the environment (lower than regulatory limits for normal operation). Sodium fires occurred in 14 cases. The latest leakage of sodium in the system occurred in 1994 and in a steam generator in 1991 [32, 34, 36, 61].

Ref. [63] provides the detailed guidance on the inspection of systems, structures and components of BN-800 reactor. It determines the equipment to be controlled, control methods, scope of control and frequency. Because of the robust design features of the reactor (integral layout of the primary equipment of the primary circuit), primary and secondary coolant features (liquid sodium), double wall reactor vessel (main vessel and guard vessel) and the guard casings of primary auxiliary sodium systems pipelines, operational inspections of the base metal and welded joints are not provided for the in-vessel structures, parts of the reactor vessel and primary pipelines covered by guard vessel and guard casings.

For the rest of BN-800 systems, structures and components the inspections involve visual, capillary, ultrasonic, radiographic and magnetic particle methods.

Scope of inspections and their frequency depend on the system safety classification and on the knowledge of damage mechanisms, design specifics and operating conditions. Where the scope of a single inspection is less than 100%, different groups of welded joints should be inspected every next time. If defects are detected during the selective inspection, the scope of inspection must be doubled and the adjacent areas inspected. If further defects are detected, a 100% inspection of the system, structure or component has to occur.

Research studies on the improvement of inspection methods have been included in the R&D programme of BN-1200 [13]. Inspections in BN-1200 are expected to be more simplified and more effective due to the reactor design simplification.

Detailed information on BN-1200 design features facilitating the performance of testing and maintenance and potential improvements of their effectiveness and efficiency has not been available to the assessor. Assessment against this criterion is not complete.

4.3.4. Criterion CR1.4: Failures and deviations from normal operation

Indicator IN1.4: Expected frequency of failures and deviations from normal operation.

Acceptance limit AL1.4: Lower than that in the reference design.

Results of analysis of the number of failures and human errors in the prototype reactor BN-600 appears in Figure 12 (data taken from Ref. [60]). Based on this analysis, the frequency of AOO in first-of-a-kind reactors (including BN-600, BN-800 and BN-1200) can be a time-dependent characteristic. The ratio between the AOO frequencies in the beginning of operation and 30 years later can be higher than 10.

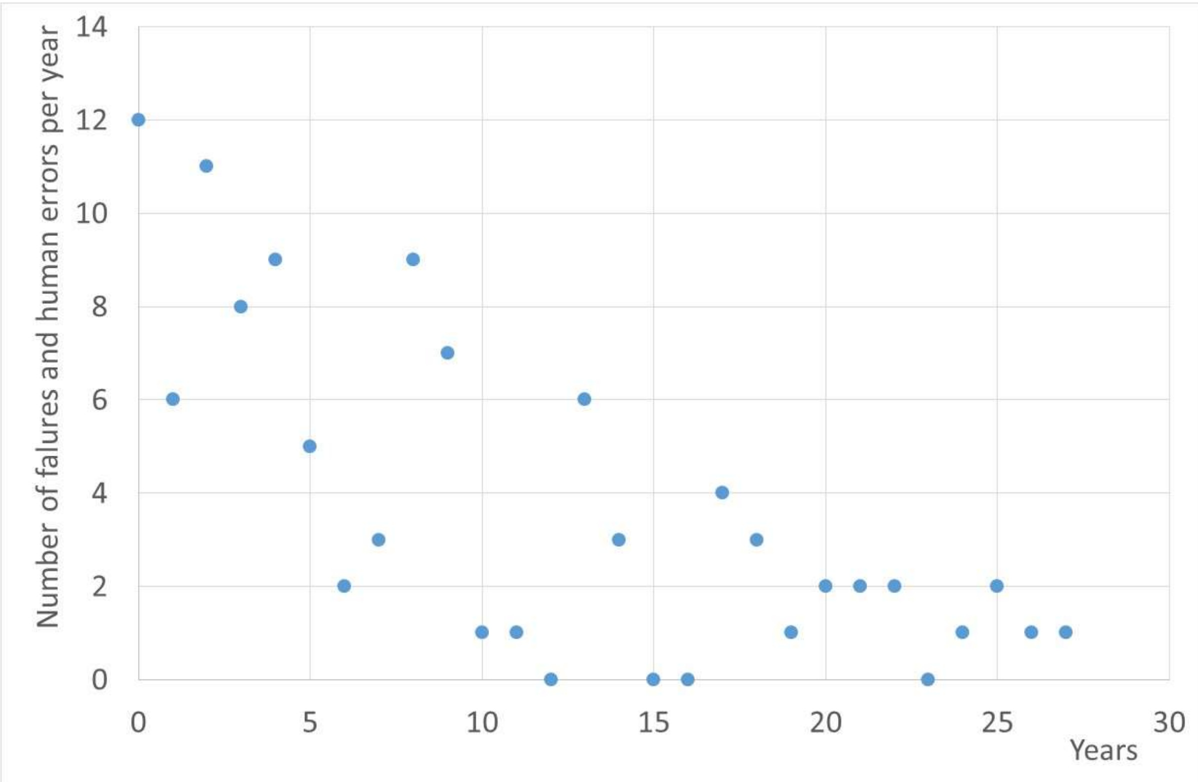

FIG. 12. Number of failures and human errors per calendar year from the BN-600 commissioning in 1982 [60].

Ref. [50] states that the BN-1200 design documentation considers 64 different AOOs. The most remarkable consequences (within acceptable limits) can be caused by the following events [64]:

– Uncontrolled withdrawal of a single control rod;
– Loss of power to a primary coolant pump;
– Loss of power to the own needs;
– Water to sodium leaks through the heat exchange tubes of steam generator.

The improvements implemented in the BN-1200 design are expected to reduce the frequency of AOOs. Most notable of them include:

– Design simplification discussed in EP1.1.2 reduced number of systems, structures and components along with pipelines length and is expected to reduce both the frequency of failures and frequency of deviations from normal operation caused by human errors;
– New layout of the entire primary circuit within the reactor vessel minimized the frequency of AOOs involving leaks of primary sodium;
– New materials improving reliability of systems structures and components will also reduce the frequency of failures.

Direct information on the frequencies of AOO in the BN-800 and BN-1200 was not available to the assessor. Assessment against this criterion is not complete.

4.3.5. Criterion CR1.5: Occupational dose

Indicator IN1.5: Occupational dose values during normal operation and AOOs.

Acceptance limit AL1.5: Lower than the dose constraints.

Ref. [36] provides collective occupational dose for period of 2002–2015 for operating prototype reactor BN-600 – 0.408 man×Sv/a. This value is lower than average values for operating PWRs, BWRs or PHWRs provided in the INPRO manual.

Ref. [32] provides average annual individual occupational doses for every Russian NPP between 1991 and 2003 (Table 12) and average dose for all NPPs in 2010 (1.4 mSv/a) which is lower than global average (2.4 mSv/a).

TABLE 12. AVERAGE ANNUAL INDIVIDUAL OCCUPATIONAL DOSES FOR RUSSIAN NPPS (mSv/a) [32]

NPP	1991	1995	1996	1997	1998	1999	2000	2001	2002	2003
Balakovo	0.8	1.4	1.0	1.0	1.2	1.0	0.8	0.7	0.7	0.7
Beloyarsk	1.8	1.6	1.6	1.3	2.2	1.4	1.8	1.7	1.6	1.0
Bilibino	9.7	8.8	11.5	6.0	6.9	5.8	4.9	5.3	5.2	4.4
Volgodonsk								0.02	0.07	0.1
Kalinin	2.3	1.8	1.5	1.4	1.2	1.2	1.2	1.0	0.7	0.6
Kolskiy	4.0	3.0	3.2	1.8	2.0	3.2	2.0	2.1	1.8	1.9
Kursk	16.0	11.0	9.8	7.9	6.2	6.9	5.9	4.3	4.4	3.6
Leningrad	5.2	7.1	6.6	5.8	4.9	3.5	3.9	4.0	3.5	3.5
Novovoronezh	4.8	4.9	2.9	2.8	2.3	3.5	2.3	3.1	2.7	2.6
Smolensk	4.6	4.3	3.8	4.6	5.4	5.2	4.8	4.6	4.6	2.3

For the BN-1200 project, the estimated occupational doses are lower than for BN-600 and BN-800 (due to the design simplification, optimization of fuel reloading procedures) and are reasonably expected to be lower than dose constraints.

The effects of potential releases of C-14 at normal operation from the BN-1200 reactor using nitride fuel are estimated in Ref. [65]. It was found that after 10 years of operation the activity of C-14 in the primary coolant will achieve ~10^{11} Bq and in the primary cover gas ~2×10^{10} Bq. Individual occupational dose from C-14 can amount ~0.02 mSv/a (compare to regulatory limit – 20 mSv/a). The normalized annual C-14 emission rate to the environment was estimated at ~0.02 TBq/GW(e)×a which is lower than typical values for water cooled reactors in Russian Federation.

4.4. UR2: DETECTION AND INTERCEPTION OF ANTICIPATED OPERATIONAL OCCURENCES

INPRO methodology user requirement UR2 for sustainability assessment in the area of safety of nuclear reactor was formulated as follows [7]: "The nuclear reactor assessed has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions".

4.4.1. Criterion CR2.1: I&C system and inherent characteristics

Indicator IN2.1: Capabilities of the I&C system to detect and intercept and/or capabilities of the reactor's inherent characteristics to compensate deviations from normal operational states.

Acceptance limit AL2.1: Superior to those in the reference design.

4.4.1.1. Evaluation parameter EP2.1.1: Continuous monitoring of plant health

Ref. [62] provides the list of basic monitoring provisions in BN-600 and BN-800 reactors:

- Primary vessel and internals – under-gas viewing (inside of primary vessel of reactor) / aerosol detection of primary vessel leak and electrical contact (outer surface of primary vessel);
- Secondary circuit pipes – leak detectors (electrical contact), smoke detectors;
- Intermediate heat exchangers – control Na level and Ar pressure;
- Steam generators – leak detectors, control of Na level and Ar pressure, regular visual inspection of tube-bores and structures.

The operational manuals provide more details on the monitoring systems of the BN-800. In-reactor monitoring system performs on-line measurement and processing of the following characteristics (operation at different power levels):

- Sodium temperature in the reactor vessel, at the inlets/outlets of intermediate heat exchangers, at the outlet from fuel assemblies (37×3 probes), etc. More than a hundred detectors in total;
- Temperature and deformation of reactor vessel (in 90 locations);
- Levels of sodium in the reactor vessel and in the primary coolant pumps tanks;
- Sodium flow rate through the core;
- Gas pressures in the reactor and between the main reactor vessel and guard vessel;
- Vibrations of reactor vessel, heat exchangers and primary coolant pumps;
- Neutron flux;
- Fuel leaks (through gas fission products activity and delayed neutrons);
- Coolant boiling;
- Reactor vessel leaks.

Different characteristics are monitored during the reactor reloading.

Refs [32, 66, 67] provide additional information on the following monitoring systems in BN-800:

- Loose parts monitoring;

– Monitoring of rotating machinery;
– Seismic monitoring;
– Radioactive releases and environmental impact monitoring;
– Monitoring of fires (as part of fire protection system).

Ref. [54] states that basic design principles of the BN-800 reactor I&C system are similar to those of water cooled reactors of Russian design. Peculiarities mostly occur because of using sodium as a coolant in fast reactors. Special tools for measurement and monitoring of sodium systems were developed together with sodium electric heating instrumentation and control system. In particular, the number of monitored heating areas was increased up to 3500, and temperature sensors (two thermocouples) were installed in each area in order to maintain coolant temperature within the range of 550 ± 30°C. For the purpose of sodium parameters monitoring, new sensors were developed and introduced to measure pressure, level and flow rate, monitor sodium leakage and hydrogen concentration in reference points. Changes were also introduced to I&C subsystems aimed at controlling the turbine hall parameters. In addition, I&C of Beloyarsk NPP Unit 4 performs online self-diagnostics and process equipment monitoring providing the operator with the full scope of necessary information.

These monitoring systems are expected to be incorporated in the BN-1200 design.

Direct information on the improvement of monitoring systems in BN-1200 was not available to the assessor. Assessment against this evaluation parameter is not complete.

4.4.1.2. Evaluation parameter EP2.1.2: Capability of the I&C system

Ref. [32] gives general overview of the BN type reactors features relevant to their controllability. It is stated that BN type reactors are easier to control and more stable than existing thermal reactors and can be operated at a given power level for many hours without interventions of the control system. Basic features of BN reactors involve:

– Highly stable relative power distribution in the core with a low sensitivity to the control rods position;
– No reactor poisoning with xenon and samarium;
– Low inertia of main constituents of negative feedbacks;
– Short lags in response to the changes of basic parameters;
– Low excess reactivity.

One improvement of I&C system capability in BN-800 has been achieved through introduction of the reactor diagnostics system [68]. This unique automated system was designed for comprehensive monitoring and projection of the scenarios and processes occurring in the reactor installation under normal operation conditions and AOO at different reactor power levels (from 0.1 to 120%). Reactor diagnostics system performs analysis of the neutronic (including neutronic noise), thermal hydraulic and other technological characteristics of the reactor, early detection of deviations that could result in the substantial damage of fuel including sodium boiling, formation of 'hot' spots, reduction of coolant flow rate in a fuel assembly, fuel rod melting, intermediate heat exchangers coolant flow rate deviations, uncontrolled withdrawal of control rods, in-vessel vibrations, reactor core coolant flow rate deviations, I&C systems failures.

Statistical estimates of temperature distributions, reactivity values, statistical and spectral characteristics of neutronic noise temperature channels during operation at different power levels were obtained. Reactor diagnostics system uses improved methods for the measurement and improved algorithms for complex analysis of data from the abnormal reactivity detection subsystem, neutronic noise diagnostics subsystem, reactor core temperature monitoring subsystem, fuel claddings leak-tightness gas/delayed neutrons monitoring systems, and from other I&C systems giving high efficiency of AOO detection [68]. A similar system is expected to be implemented in BN-1200 reactor.

The BN-1200 design provides several other improvements of the I&C systems. Four in-reactor sets of ionization chambers for the neutron flux measurements in BN-1200 provide better quality of measurements and improve reactor control. Ref. [66] provides a description of improved fire protection system developed for BN-1200.

4.4.1.3. Evaluation parameter EP2.1.3: Compensation of deviations from normal operation

Ref. [50] states that the BN-1200 design materials consider 64 different AOOs. An analysis of the BN-1200 reactor dynamic responses in transients show how the different events causing an AOO are compensated by the I&C system and inherent safety features. For example, in the case of loss of power to a primary coolant pump the following results were obtained. During the transient, the temperature of sodium at the top nozzles of the most stressed fuel assemblies rises by about 40°C and the temperature of fuel drops to about 400°C for 4–5 seconds. After that transient a stable operating mode of the reactor occurs with 3 loops. Safe operation criteria are met.

In the case of uncontrolled withdrawal of a single control rod a minor increase of the fuel cladding temperature (at ≤30°C) and the centre line temperature of the fuel rods occur within 30 seconds. The reactor shuts down by a scram initiated by the reactor power level alarm signal. After that event, all temperatures decrease in 1–2 seconds. Safe operation criteria are met.

Analysis of the fuel cladding leaks releasing gaseous fission product into sodium demonstrated that further ‘avalanche-like’ damage of claddings in neighbouring fuel rods is not possible.

The analysis of sodium fires based on the concept of ‘burning in a puddle’ was performed for the steam generator rooms, secondary pipelines, emergency heat removal system, and sodium drain tanks rooms.[46] By the implementation of measures excluding sodium drip combustion, fire zoning, fire proof structures, thermal insulation and steel cladding of premises, and passive fire protection structures and components (sodium catchers, trays and drain fire extinguishing) the safe operation criteria are met ($T_{gas} \leq 370$ ° C, $T_{concrete} \leq 100$ ° C, $\Delta P_{gas} = \pm 0.01$ atm) [50].

4.4.2. Criterion CR2.2: Grace periods after AOOs

Indicator IN2.2: Grace periods until human actions are required after AOOs.

Acceptance limit AL2.2: Longer than those in the reference design.

Direct information on the longer grace periods after AOO in BN-1200 design was not available to the assessor. Assessment against this criterion is not complete.

4.4.3. Criterion CR2.3: Inertia

Indicator IN2.3: Inertia to cope with transients.

Acceptance limit AL2.3: Larger than that in the reference design.

[46] New edition of Rules for the technology design of fire protection of premises with a sodium coolant of nuclear power plants with BN reactors was introduced in the Russian Federation in 2014. It requires elimination of the primary sodium fires and spray and drip type fires of secondary sodium (insulation and casings to be considered). Leak before break concept was introduced for sodium cooled reactors. Maintenance works on unprotected (without casings) areas of sodium equipment are allowed only during the reactor shut down. Analysis of the sodium fires needs to be carried out for the scenarios of ‘burning in a puddle’.

BN-1200 and BN-800 are both sodium cooled reactors. Thermal conductivity of sodium is ~300 times higher than that of liquid water. Both designs are the pool type reactors having all major[47] primary circuit components and primary coolant located within the reactor vessel. Such a reactor has high heat capacity. The BN-800 reactor vessel has diameter of 12.9 m and height of 16.41 m. Volume of the primary sodium in the BN-800 amounts approx. 1100 m^3. In BN-1200 these dimensions are larger – 16.9 m and 20.72 m [37], and the power rating / volume ratio for BN-1200 is lower than in BN-800 and corresponding characteristics of inertia can be expected to be higher.

The BN-1200 reactor possesses considerable thermal inertia. In the analysis of severe accidents scenario with station blackout, active shutdown and EHRS failures (ultimate loss of flow plus ultimate loss of heat sink type accident, ULOF+ULOHS) the process of reactor heat-up due to residual heating up to the sodium boiling temperature (900°C at a given pressure level) takes about 2 days (see Figure 13 taken with modifications from Ref. [50]).

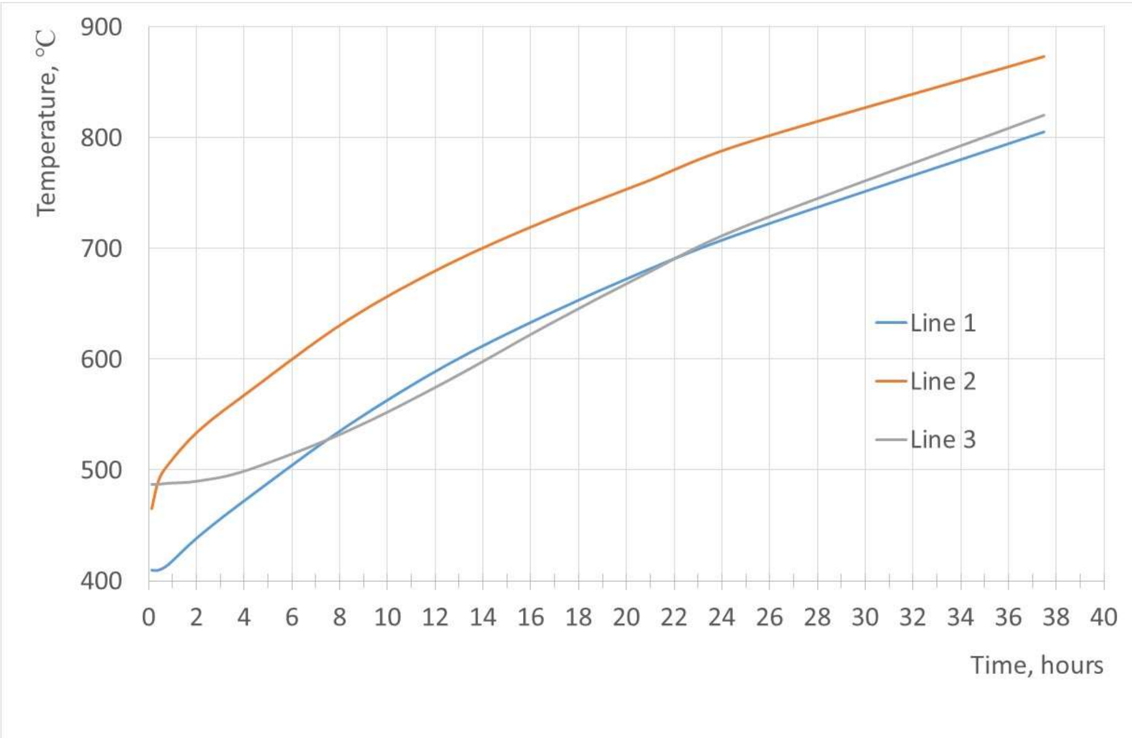

FIG. 13. Reactor coolant temperature in the BN-1200 accident with station blackout and EHRS failure [50]. Line 1 – sodium temperature at the bottom of reactor, Line 2 – sodium temperature at the intermediate heat exchangers inlet, Line 3 – argon temperature near the reactor lid.

Ref. [55] states that BN-1200 design uses a new type of primary coolant pumps providing longer coast-down in the case of power loss. The transient response seen in the analysis of station blackout scenario with the failure of two trains of EHRS out of total four [69] showed this behaviour. Loss of power results in disabling of primary and secondary coolant pumps and loss of feedwater supply to the steam generators. Reactor power drops down to the level of residual heat due scram provided by passive systems. During the primary coolant pumps coast-down (~60 s) the check valves of autonomous heat exchangers (EHRS) and outlet gates of air heat exchangers open providing natural circulation of coolants in the EHRS circuits and in

[47] In the case of BN-1200 all primary circuit components are placed within the reactor vessel.

reactor (for more details see below for the discussion of severe accidents). The fuel element cladding temperature reaches values close to the normal operation limit for a short period. The reactor vessel temperature remains close to the nominal value.

4.5. UR3: DESIGN BASIS ACCIDENTS

INPRO methodology user requirement UR3 for sustainability assessment in the area of safety of nuclear reactor was formulated as follows [7]: “The frequency of occurrence of DBAs in the nuclear reactor assessed is reduced. If an accident occurs, engineered safety features are able to restore the reactor to a controlled state, and subsequently to a safe shutdown state, and ensure the confinement of radioactive material. Reliance on human intervention is minimal, and only required after sufficient grace period.”

4.5.1. Criterion CR3.1: Frequency of DBAs

Indicator IN3.1: Calculated frequencies of occurrence of DBAs.

Acceptance limit AL3.1: Frequencies of DBAs that can cause plant damage are lower than those in the reference design.

BN-1200 design defines the set of requirements to the system parameters to be fulfilled to avoid potential accidents (safe operation limits). These requirements include the following [45, 50]:

- Reactor vessel temperature (72 hours) <750°C;
- Fuel claddings temperature <800°C;
- Fuel claddings leaks of gas-leak type ≤ 0.1% of all fuel rods in the core;
- Fuel claddings leaks of fuel-coolant direct contact type ≤ 0.01% of all fuel rods in the core;
- Activity of liquids and gases <1000 MBq/l;
- SG coolants temperatures change rates ≤ 5°C/s;
- Secondary sodium fires: $T_{gas} \leq 370°C$, $T_{concrete} \leq 100°C$, $\Delta P_{gas} = \pm 0.01$ atm;
- Secondary coolant pressure < (0.98−1.96) MPa depending on location.

Refs [45, 50, 64] provide a brief overview of the accidents considered for BN-1200 under the design basis conditions. These DBAs include:

- Partial clogging of a fuel assembly cross-section due the irradiation effects on assembly materials, coolant impurities or foreign objects;
- Leaks from primary circuit gas system;
- Delaying of spent fuel assembly in a gas filled room of reloading system;
- Large leaks in SG (rupture of a single tube).

Ref. [64] further explains that unlike BN-800 design, BN-1200 design does not consider leaks of primary sodium and associated effects (fires of radioactive sodium) under the DBA conditions due to the incorporation of entire primary circuit into the reactor vessel. This is clear indication of the reduced (to a zero level) frequency of a given class of DBA.

Results of deterministic analysis of DBA are available in public domain, e.g. Refs [69, 70].

Direct information on other frequencies of DBA caused by internal or external events or probable combinations thereof in BN-800 and BN-1200 was not available to the assessor[48]. Assessment against this criterion is not complete.

4.5.2. Criterion CR3.2: Grace period for DBAs

Indicator IN3.2: Grace periods for DBAs until human intervention is necessary.

Acceptance limit AL3.2: At least 8 hours and longer than those in the reference design.

In evolutionary water-cooled reactors a longer grace period results in extended design requirements ensuring longer fully automated system responses including longer emergency power supply (from on-site diesel-generators), residual heat removal, battery power for I&C, etc. Sufficient battery power is required for the I&C systems to identify and assess the plant state and initiate necessary actions. Usually battery power is used for many purposes (instrumentation, valves, lighting, etc) and the capacity of batteries in most operating reactors is usually designed to use this power for all purposes for about 2–4 hours.

BN-1200 reactor design extensively uses inherent safety characteristics and passive safety features reducing the need for emergency power supply (via diesels, turbines or batteries) for the reactor systems. Primary sodium leaks are not considered under DBA conditions due to the location of primary circuit within the reactor vessel (reactor vessel failure is not considered under the DBA conditions). Residual heat removal is provided by the passive EHRS. Even in the case of EHRS failure (which is not considered under the DBA conditions) the residual heating up to the level of sodium boiling takes about 2 days [50].

However, direct information on the grace periods after DBA in BN-1200 design was not available to the assessor. Assessment against this criterion is not complete.

4.5.3. Criterion CR3.3: Engineered safety features

Indicator IN3.3: Reliability and capability of engineered safety features.

Acceptance limit AL3.3: Superior to those in the reference design.

Ref. [62] provides the list of BN-800 major characteristics which are used for initiating automatic shutdown [62]:

- High primary coolant outlet temperatures;
- High neutron flux;
- High rate of neutron flux change (reactivity);
- Low coolant level in reactor vessel;
- Loss of electrical power;
- Low ratio of primary coolant flow to core flux;
- Earthquake;
- Failure of 2 loops;
- Delayed neutron detection signal.

[48] Considerations of frequencies of DBA caused by internal or external events or probable combinations thereof can provide a significant impact on the outcome of INPRO sustainability assessment. It is expected that in the future the assessment will be completed along with the development of detailed design of the reactor.

To make engineered safety features more reliable and capable the BN-1200 design introduces several new (compared to BN-800) passive systems, structures and components including the following:

- Control rods with passive actuation at high (~700°C) primary coolant outlet temperatures;[49]
- Emergency heat removal system using natural circulation of sodium and air (no power supply required after the system actuation);
- Confinement above reactor where radioactive gas and aerosols can be discharged from the gas system to condensate aerosols.

However, direct information from a BN-1200 probabilistic safety assessment demonstrating calculated increased reliability of the safety systems for all states of the nuclear reactor (full and reduced power operation, shutdown state) was not available to the assessor. Assessment against this criterion is not complete.

4.5.4. Criterion CR3.4: Barriers

Indicator IN3.4: Number of confinement barriers maintained (intact) after DBAs and DECs.

Acceptance limit AL3.4: At least one and consistent with regulatory requirements for the type of reactor and accident under consideration.

Table 13 provides a comparison of safety barriers in water cooled reactors and fast reactors of Russian design.

BN-1200 design incorporates several improvements targeted at strengthening of barriers. Seismic characteristics of confinement building had been improved as discussed in CR4.2. Robustness characteristics of the guard vessel have been improved and made the same as robustness of the main reactor vessel (unlike earlier BN designs). The in-vessel core catcher was introduced to protect reactor vessel from the molten fuel in the postulated accident with core melting.

TABLE 13. SAFETY BARRIERS IN WATER COOLED REACTORS (VVER-1000) AND FAST REACTORS (BN-1200) OF RUSSIAN DESIGN

Water cooled reactors	BN-1200
Fuel matrix (pellet)	Fuel matrix (pellet)
Fuel rod cladding	Fuel rod cladding
Reactor vessel and primary circuit boundary	Reactor vessel (and intermediate heat exchangers)
Containment	Guard vessel [a]
	Confinement

[a] - Guard vessel surrounds the main reactor vessel to protect from the release of radioactive sodium into the reactor shaft and its interaction with air. The tightness of the main reactor vessel and guard vessel is constantly monitored during the operation.

Ref. [34] states that the emergency protection zone[50] around BN-1200 reactor under DBA conditions remains within the NPP site meaning that no major release of radioactive materials from BN-1200 occurs after any DBA and at least one safety barrier is maintained intact.

[49] Hydraulically suspended control rods passively actuated at low flow rates of primary coolant are incorporated both in BN-800 and BN-1200.

[50] Area where protective measures are necessary.

Refs [50, 71] provide the list of beyond design basis accidents (BDBA) considered in the preliminary safety assessment report of BN-1200 design. These accidents include:

- Station blackout. Active shutdown system is not available. EHRS is available (ULOF);
- Station blackout. EHRS is not available. Active shutdown system is available (ULOHS);
- Station blackout. EHRS and active shutdown system are not available (ULOF+ULOHS);
- Unintentional sequential withdrawal of control rods from the core (false signal). Active shutdown system is not available (UTOP);
- Full instantaneous blockage of a fuel assembly (shutdown and heat removal systems are available);
- Fire with damage to control and power supply systems;
- Leaks of secondary pipelines in the areas where safety casing has not been provided;
- Rupture of a SG tube and additional damage of neighbouring tubes;
- Main reactor vessel leakage. Decrease of coolant level in the reactor caused by sodium leakage into the gap between the main reactor vessel and guard vessel;
- Earthquakes up to 7 MSK scale level.

Ref. [50] states that based on the safety analysis of BN-1200 design the most potentially dangerous scenario is initiated by station blackout at full power with unavailable EHRS and active shutdown system (ULOF+ULOHS). Reactor coolant temperature changes for this scenario are presented in Figure 13. To prevent heating of sodium over 900°C the residual heat removal needs to be restored within two days.

In this scenario the fuel rods claddings can lose tightness and start leaking releasing gaseous fission products into the primary coolant. Severe damage of fuel rods (melting) does not occur. Heating of reactor causes discharge of approx. 500 kg of argon (containing gaseous fission products from leaking fuel rods) from the reactor to the special ventilation system. Estimated public dose outside of NPP site does not exceed 8 mSv during the first year after the accident [50]. Ref. [64] provides more information on radiological consequences of different scenarios involving station blackout and combinations of accompanying failures. Reactor vessel, guard vessel and confinement are maintained intact.

Ref. [69] provides the results of analysis of the accident with total instantaneous blockage of a single fuel assembly bottom nozzle during the reactor full power operation. The initiating event causes quick heating and boiling of coolant and occasionally melting of the fuel rods. Gaseous fission products released from the damaged fuel rods move into the upper mixing chamber of the reactor. The cladding tightness monitoring system actuates a scram. Reactor power then drops to the level of residual heat which is removed through the secondary and tertiary circuits and turbine condenser. Parts of molten fuel and claddings in the clogged assembly move down, cool down and turn into solid form at the bottom of assembly. Part of the molten fuel burns through the wrapper pipe of the clogged fuel assembly and solidifies on the wrappers (outer surface) of neighbouring assemblies. Molten fuel and fuel melting do not propagate further. The accident results in release of gaseous fission products into coolant and reactor gas cavity. Limited release of radioactivity into the reactor premises and environment is possible. However, estimated public dose does not exceed 0.24 mSv. Three safety barriers remain intact in this accident.

Ref. [41] summarizes results of ULOF and UTOP scenarios for different types of fuel which can be used in BN-1200. Estimated radiological consequences in the case of nitride fuel are essentially lower than for MOX fuel. However, in both cases the reactor vessel, guard vessel and confinement remain intact.

4.5.5. Criterion CR3.5: Subcriticality margins

Indicator IN3.5: Subcriticality margins after reactor shutdown in accident conditions.

Acceptance limit AL3.5: Sufficient to cover uncertainties and to maintain shutdown conditions of the core.

Information on the subcriticality margins after reactor shutdown by the active system, by passive hydraulically suspended control rods system, or by passive high temperature actuated control rods system was not available to the assessor. Assessment against this criterion is not complete.

4.6. UR4: SEVERE PLANT CONDITIONS

INPRO methodology user requirement UR4 for sustainability assessment in the area of safety of nuclear reactor was formulated as follows [7]: "The frequency of an accidental release of radioactivity into the containment/confinement is reduced. If such a release occurs, the consequences are mitigated, preventing or reducing the frequency of occurrence of accidental release into the environment. The source term of the accidental release into the environment remains well within the envelope of the reference reactor source term and is so low that calculated consequences would not require evacuation of the public."

4.6.1. Criterion CR4.1: Frequency of release into the containment/ confinement

Indicator IN4.1: Calculated frequency of accidental release of radioactive materials into the containment / confinement.

Acceptance limit AL4.1: Lower than that in the reference design.

Calculated frequencies of severe core damage by internal events during the reactor power operation are presented in Table 14. National authorities in many countries including the Russian Federation introduce probabilistic safety goals or targets for the overall risk profile including internal and external hazards. External hazards may have higher risk contribution than the internal initiating events and the final frequencies of severe core damage can be potentially higher. It is expected that frequency of severe core damage by external events will be available at the advanced steps of BN-1200 development and deployment to finalise the INPRO assessment in this area.

TABLE 14. CALCULATED FREQUENCY OF SEVERE CORE DAMAGE BY INTERNAL EVENTS DURING POWER OPERATION [36, 41]

Type of reactor	Core damage frequency, event per reactor-year
BN-600	10^{-5}
BN-800	2×10^{-6}
BN-1200	5×10^{-7}

Information on frequencies of severe core damage during the reactor shutdown states was not available to the assessor. Information on uncertainties[51] associated with calculated frequencies of severe core damage was not available to the assessor. Assessment against this criterion is not complete.

[51] Given that statistical information on reliability of specific equipment used in BN-1200 is relatively limited compared to the thermal reactors the uncertainty of the results can be potentially higher than in thermal reactors.

4.6.2. Criterion CR4.2: Robustness of containment/confinement design

Indicator IN4.2: Containment loads covered by the design, and natural or engineered processes and equipment sufficient for controlling relevant system parameters and activity levels in containment/confinement.

Acceptance limit AL4.2: Larger than those in the reference design.

The BN-800 reactor building is a rectangular concrete building designed to withstand following seismic acceleration [62]:

- horizontal – 0.1g;
- vertical – 0.07g.

The BN-800 is designed to withstand the impact of 5.7 t aircraft falling at a speed of 100 m/s and shock wave caused by an external explosion with shock front pressure 10 kPa and duration of compression phase 1 s.

The BN-1200 reactor building is cylindrical [38]. The BN-1200 design basis earthquake is 6 points on the MSK 64 scale with a maximum considered earthquake equal to 7 points of MSK 64 scale [37, 55, 71, 72] which provides similar values of peak ground acceleration as the BN-800. However, the BN-1200 reactor building is designed to incorporate the following improvements:

- Enforced external walls and protective dome;
- Introduction of 'light' dome underneath of protective dome protecting the reactor from the potential danger of falling parts from the building structures [42];
- Expansion joints along the perimeter of confinement section of reactor building protecting from the transfer of potential harmful impact [42];
- Improvement of materials used in the protective structures.

The BN-1200 is designed to withstand the impact of 20 t aircraft and shock wave caused by an external explosion with shock front pressure 30 kPa and duration of compression phase 1 s [55, 71]. Besides that, the BDBA considerations of BN-1200 involve 400 t commercial aircraft crash on the reactor building and consequent fuel ignition. BDBA analysis considers earthquakes with horizontal peak ground acceleration 40% higher than in the maximum considered earthquake.

BN-1200 processes (new compared to BN-800) designed to maintain confinement function include:

- Leak-tight above-reactor premise (confinement of the radioactive products released from the reactor during the accident);
- Passive four trains EHRS connected to the primary circuit (reactor vessel) through autonomous heat exchanger with check valves;
- In-vessel structures and components for localization, cooling of molten fuel in the case of severe accident and effective protection of the reactor vessel from potential damage. [73].

4.6.3. Criterion CR4.3: Accident management

Indicator IN4.3: In-plant accident management.

Acceptance limit AL4.3: Accident management procedures and training sufficient to prevent an accidental release outside containment / confinement and regain control of the reactor.

Ref. [72] states that BN-1200 design defines the list of non-trivial BDBAs and defines the measures for accident management. The objectives of accident management are:

- Prevention (mitigation) of severe core damage;
- Prevention (mitigation) of release of radioactivity to the environment due to the maintenance of safety barriers;
- Achievement of the safe subcritical state, providing fuel cooling and confinement of radioactive materials.

To achieve these objectives the following tasks need to be resolved:

- Reactor has to be brought to a subcritical state and potential re-criticality has to be eliminated, including conditions involving core damage;
- Maintaining coolant level sufficient for covering the core and removing of the residual heat;
- Prevention of the reactor vessel and primary circuit damage;
- Prevention of the guard vessel damage.

Accident management has to be ensured even in the case of the failure of part of the safety systems, structures and components. The list of key safety parameters for BDBA is expected to be determined and measurements of the parameter values exceeding operational limits are to be envisaged. Accident management procedures are expected to be developed (required).

The BN-1200 design broadly uses passive safety systems. For example, in the case of loss of power for the spent fuel storage it can be cooled with the passive systems using water-filled single-phase thermal siphon [42]. However, the possibility of using of external mobile power sources also exists [55].

Considering the design extension conditions in the safety assessment report the BN-1200 designer provides recommendations on the accident management [50]. The BN-1200 accident management procedures and trainings are expected to be developed based on SAR recommendations and BN-600/800 experience.

4.6.4. Criterion CR4.4: Frequency of accidental release into environment

Indicator IN4.4: Calculated frequency of an accidental release of radioactive materials into the environment.

Acceptance limit AL4.4: Lower than that in the reference design. Large releases and early releases are practically eliminated.

Information on frequency of accidental release of radioactive materials into the environment and associated uncertainties was not available to the assessor. Information on frequency of large releases and early releases of radioactive materials into the environment or technically sound confirmation of elimination of such releases was not available to the assessor. Assessment against this criterion is not complete.

4.6.5. Criterion CR4.5: Source term of accidental release into environment

Indicator IN4.5: Calculated inventory and characteristics (release height, pressure, temperature, liquids/gas/aerosols, etc) of an accidental release.

Acceptance limit AL4.5: Remain well within the inventory and characteristics envelope of the reference reactor source term and are so low that calculated consequences would not require public evacuation.

One important feature of the BN type reactors is the effective retention of iodine and cesium isotopes by primary coolant (sodium) which was confirmed by the experience obtained during BN-600 operation. It essentially reduces the releases of radioactivity to the environment in the case of accident. Another feature is the absence of coolant phase transitions in the case of depressurization of primary circuit which reduces the pressure and inventory of radioactive materials in the case of release.

BN-1200 design considers two broad groups of BDBAs. The first group comprises combinations of initiating events with the failures of all active safety systems and normal operation systems relevant to safety, single failures of passive components having mechanically moving parts and operator's errors [69, 74]. Numerical analysis of these accidents demonstrated that they do not result in any severe damage to the reactor core or reactor vessel and can only cause a very limited release of radioactivity creating relatively low public doses [74].

In the second group of BDBAs total failures of the passive shutdown systems are postulated in addition to failures in the group one, to create conditions for potential severe damage of the reactor core and substantial release of radioactivity to the environment [74]. Data calculated for these scenarios are expected to be used for emergency planning and response measures.

The second group of BDBAs includes the following basic scenarios simulated using the code SOCRAT_BN and leading to the severe core damage [50, 74]:

- Accidents with positive reactivity insertion and total failures of all reactor shutdown systems;
- Accidents with blockage of a fuel assembly flow area and total failures of all reactor shutdown systems.

One of the most severe scenarios considered for BN-1200 includes unintentional insertion of positive reactivity (UTOP) along with total failure of all active and passive shutdown systems [50, 74]. Postulated initiating event was defined as simultaneous withdrawal of all control rods from the core which is normally prevented by 12 independent measures [50].

This scenario involves sodium boiling and insertion of positive reactivity due to sodium-void reactivity effect causing a rapid jump of the reactor power up to 13 full power levels for a short time. It causes damage (leakage) of all fuel assemblies in the reactor core and melting of fuel and claddings in approx. 39% of assemblies [74]. Reactor vessel and guard vessel remain intact, however approx. 1% of gaseous and approx. 10^{-4} % of volatile fission products accrued in the core by the moment of accident would be released to the environment through the systems protecting reactor vessel from overpressure. Estimated public dose equals 7.3 mSv/a [74].

Ref. [50] states that estimated public dose at the NPP site boundary remains lower than 16 mSv/a for the first year after accident in all BDBA scenarios considered in the BN-1200 design. National criterion for consideration of potential public evacuation / relocation is defined as a range 50–500 mSv/a in terms of dose expected in the first year after accident.

TABLE 15. PUBLIC DOSES IN CASE OF BDBA (ULOF), mSv/a

Distance	BN-600	BN-800	BN-1200
Protection zone boundary	70 (3 km)	11.3 (5 km)	< 10 (0.8 km)
Emergency planning zone boundary	50 (25 km)	3.5 (25 km)	< 1 (25 km)

Data confirming that radiological consequences of BDBAs (of ULOF type) in BN-1200 reactor are lower than in BN-800 reactor appear in Table 15.

Direct information on the source terms of accidental releases from the BN-1200 and BN-800 reactors was not available to the assessor. Information on the methods (deterministic or probabilistic) and assumptions (e.g. whether conditions) used by BN-1200 designer for transport calculations of radioactive materials in the environment was not available to the assessor. Assessment against this criterion is not complete.

4.8. UR5: INDEPENDENCE OF DID LEVELS, INHERENT SAFETY CHARACTERISTICS AND PASSIVE SAFETY SYSTEMS

INPRO methodology user requirement UR5 for sustainability assessment in the area of safety of nuclear reactor was formulated as follows [7]: “An assessment is performed to demonstrate that the DID levels are more independent from each other than in the reference design. To excel in safety and reliability, the nuclear reactor assessed strives for better elimination or minimization of hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics and/or passive systems, when appropriate.”

4.8.1. Criterion CR5.1: Independence of DID levels

Indicator IN5.1: Independence of different levels of DID.

Acceptance limit AL5.1: More independence of the DID levels than in the reference design, e.g. as demonstrated through deterministic and probabilistic means, hazards analysis, etc.

Implementation of DID in sodium cooled fast reactors differs essentially from DID in water-cooled reactors. In water cooled reactors for some events such as sudden reactor pressure vessel failure, where it is not feasible to have independent levels of DID, several levels of precautions need to be demonstrated in the design (e.g. selection of materials, periodic inspection, additional margins of safety, etc) to make this event practically eliminated. BN type reactors use a different strategy.

Pressure in the BN reactor vessel is normally nearly atmospheric and regulated by the passive system. In case of raising pressure, the gas from the upper part of reactor can be discharged into the confinement room (new feature introduced in BN-1200) for temporary storage and decay of short-lived isotopes and occasionally to the special filtered venting system. This approach reduces the probability of a sudden reactor vessel failure due to overpressure. Moreover, all BN reactors use a guard vessel as a secondary barrier in case of potential leaks from the main reactor vessel, however unlike earlier designs in BN-1200 reactor the robustness characteristics of the guard vessel are made equal to the main reactor vessel. These features (confinement room and more robust guard vessel) are the indications of more independent levels of protection.

The BN-1200 uses three independent shutdown systems including two passive systems using different actuation principles. The BN-800 uses two independent shutdown systems. Although all these systems are mechanical and use control rods for absorbing neutrons this modification can be considered as improvement of protection levels independence.

Improvements related to the protection of BN-1200 against such hazards as fire and external impacts (earthquakes, airplanes crash, etc) that could potentially impair several levels of DID (for example, they could bring about accident situations and, at the same time, inhibit the means of coping with such situations) are discussed in CR4.2 (earthquakes, airplanes crash, etc) and EP5.2.2 (fires). BN-1200 reactor building design involves the enforced external walls and protective dome, ‘light’ dome underneath of protective dome protecting the reactor from the potential danger of falling parts from the building structures, expansion joints along the perimeter of confinement section of reactor building protecting from the transfer of potential harmful impact. The BN-1200 protective structures use improved materials.

The BN-1200 design improvements protecting against fire involve the minimization of primary sodium burning frequency due to enclosing of entire primary circuit in the reactor vessel, elimination of sodium leaks from autonomous heat exchangers of EHRS located in the reactor vessel, reduction of the specific amount of secondary sodium and the leakage frequency by shortening the pipe length of secondary circuit, installation of guard casings on secondary circuit pipelines within the central hall of reactor building. These improvements are planned to be implemented along with other protective measures such as measures excluding sodium drip combustion, fire zoning, fire proof structures, thermal insulation and steel cladding of premises, and passive fire protection structures and components (sodium catchers, trays and drain fire extinguishing).

Direct information from probabilistic and deterministic analysis on the improvement of independence of DID levels in BN-1200 reactor was not available to the assessor. Assessment against this criterion is not complete.

4.8.2. Criterion CR5.2: Minimization of hazards

Indicator IN5.2: Characteristics of hazards.

Acceptance limit AL5.2: Hazards smaller than those in the reference design.

4.8.2.1. Evaluation parameter EP5.2.1: Stored energy

As discussed in CR2.3 (inertia), BN-1200 and BN-800 are the pool type reactors having all primary circuit components and primary coolant located within the reactor vessel. The reactor power rating to volume ratio for BN-1200 is slightly lower than in BN-800 and corresponding specific energy (per MW installed) stored in the reactor coolant and systems, structures and components can be expected to be lower. However, the average temperature of primary sodium in the BN-800 is lower than in BN-1200 and these two characteristics partly compensate each other.

Normal operation pressure in the primary circuit is maintained at approx. 0.15 MPa level (close to the normal atmospheric pressure). Unlike in water cooled reactors, BN type reactors do not have phase transitions of coolant in the case of depressurization of primary circuit.

Energy stored in a form of potential chemical reactions of sodium can be associated with both BN-1200 and BN-800 reactors. These effects are discussed in EP5.2.2 (flammability).

As discussed in EP1.1.1 (margins of design), the power density in the BN-1200 reactor core has been reduced by a factor of 1.8 compared against BN-800. Average power density in BN-800 reactor core amounts approx. 400–450 MW/m^3 and in BN-1200 – approx. 237–255 MW/m^3. Moreover, the BN-1200 reactor is supposed to be operated with nitride fuel and thermal conductivity of the nitride fuel (20.1 W/m×K at 2000 K) is essentially higher than that of MOX fuel (2.4 W/m×K at 2000 K) as estimated in Ref. [75]. Specific heat capacity of the nitride fuel (70 kJ/mol×K) is lower than that of MOX fuel (95 kJ/mol×K) [75] and Ref. [75] estimates that the fuel pin centreline temperature in nitride fuel can be approx. 800 °C lower than in MOX fuel. These estimations mean that specific energy (per MW installed) stored in the fuel of operating BN-1200 reactor can be reasonably expected lower than in BN-800.

4.8.2.2. Evaluation parameter EP5.2.2: Flammability

The possibility of a fire in a sodium cooled reactor represents a considerable hazard. The design of BN-1200 minimizes this hazard through implementation of several modifications.

– Entire primary circuit is located within the reactor vessel and guard vessel (no pipes or equipment containing primary sodium outside of reactor vessel);
– Autonomous heat exchangers of EHRS are located in the reactor vessel;
– Reduced the specific amount of secondary sodium (per MW installed compared against BN-800) and the leakage frequency by shortening the pipe length of secondary circuit;
– Secondary circuit pipelines within the central hall of reactor building are covered with guard casings [55, 69].

Refs [50, 66] provide a description of improved fire protection system developed for BN-1200 and result of analysis of sodium fires (see EP2.1.3 for a short summary).

4.8.2.3. Evaluation parameter EP5.2.3: Excess reactivity in the core

Introducing nitride fuel in the BN-1200 reactor core increases the conversion factor which reduces the excess of reactivity necessary for fuel burn-up [55, 76]. Maximum excess reactivity at full power in BN-1200 reactor operated with mixed nitride fuel, ~ 0.5% Δk/k, is lower than in BN-800 operated with mixed oxide fuel, which reduces calculated consequences of potential reactivity accidents. Maximum excess of reactivity in BN-1200 operated with mixed oxide fuel is ~2.0% Δk/k [31, 76].

Ref. [69] provides the analysis results of BDBA caused by reactor control system malfunction and sequential withdrawal of all control rods from the reactor core operated at full power. Postulated failures include the failure of active reactor shutdown system. Maximum power level achieved in this scenario amounts to 220% of full power. This scenario involves the leaks of fuel assemblies, potential melting of fuel in a few assemblies in the central part of the reactor core and limited release of radioactivity to the environment. Estimated public dose does not exceed 0.12 mSv/a.

4.8.2.4. Evaluation parameter EP5.2.4: Reactivity feedbacks

In the BN-1200 the reactor power reactivity coefficient is negative (-2.6×10^{-6} MW(th)$^{-1}$). The core material temperature reactivity coefficient is negative (-2.3×10^{-5} °C^{-1}).

The sodium void reactivity effect is normally negative in small SFRs having thermal power rating up to the level of approx. 100 MW(th). In large SFRs including BN-600, BN-800 and BN-1200 reactors this effect is positive. Ref. [55] gives an overview of potential methods and tools which can be used to reduce the sodium void reactivity coefficient / effect in a sodium cooled fast reactor including:

– Design of a 'sodium plenum' above the reactor core (in case of sodium boiling the bubbles appearing in the core rise into the sodium plenum and increase neutron leakage introducing negative reactivity);
– Reduced core height;
– Annular or modular core, zones with different concentrations of isotopes in fuel, etc;
– Moderator added to the reactor.

Ref. [77] provides a detailed analysis of links between the sodium void reactivity effect and the reactor core height or sodium plenum/axial blanket design in BN-1200. It also involves some economic consideration of potential designs. Although the combination of reduced height of the reactor core and sodium plenum implies some economic losses, it essentially cuts down the value of sodium void reactivity effect and therefore provides safety benefits. A sodium plenum in combination with an absorbing shield placed above it was introduced instead of an upper axial blanket to reduce the sodium void reactivity effect both in the BN-800 and BN-1200.

Height of the BN-800 reactor core (fuel column height) is 90 cm [78]. Height of the BN-1200 reactor core is 85 cm [43]. Sodium void reactivity effect in BN-1200 is lower than β_{eff} [32]. Ref. [55] also provides the result of detailed analysis of sodium void effect of reactivity in the ULOF type of BDBA confirming the effectiveness and efficiency of BN-1200 design measures. Moreover, the use of nitride fuel in BN-1200 provides very strong Doppler effect of reactivity as compared to MOX fuel. This further improves the reactor core inherent safety characteristics [79].

4.8.2.5. Evaluation parameter EP5.2.5: Criticality outside the core

Such an analysis is required by the national regulatory documents in the Russian Federation as part of licensing process. It was performed for BN-800 and all earlier designs. However direct information on the outcome of criticality analysis for BN-1200 fuel was not available to the assessor. Assessment against this evaluation parameter is not complete.

4.8.3. Criterion CR5.3: Passive safety systems

Indicator IN5.3: Reliability of passive safety systems.

Acceptance limit AL5.3: More reliable than the active safety systems in the reference plant.

Three groups of passive safety systems, structures and components can be distinguished in the BN-1200 design. First group comprises passive systems available in reference plant BN-800. Examples of such systems are the in-vessel core catcher and the control rods hydraulically suspended in the sodium flow (actuated when sodium flow rate through the core drops to 50% of the nominal level) [69].

The second group of BN-1200 passive safety features involves systems added to the design, i.e. systems which do not exist in BN-800 reactor. Examples involve leak-tight above-reactor premise for confinement of radioactive gases discharged from the reactor during the accident and additional passive control rods actuated when sodium temperature rises [69].

The third group comprises the passive systems, structures and components replacing other (active or passive) systems performing the same or similar function in BN-800. Examples of such systems are the EHRS discussed several times in this report and the passive scheme flushing the sodium-water interaction products in the case of SG leak (it replaces bursting disks in BN-800).

In BN-800 reactor the passive EHRS consists of three trains connected to the secondary circuit in parallel to the SGs [80].

Ref. [81] provides the description of EHRS of BN-1200 reactor. In BN-1200 the EHRS consists of four independent trains connected to the reactor vessel through the autonomous heat exchangers. Full capacity of the system is 80 MW(th) (approx. 2.9% of the reactor full power level). The reliability of emergency heat removal in BN-1200 is more than 10 times higher than in BN-800 [69, 82].

Information on reliability of other passive systems, structures and components of BN-1200 was not available to the assessor. Assessment against this criterion is not complete.

4.9. UR6: HUMAN FACTORS RELATED TO SAFETY

INPRO methodology user requirement UR6 for sustainability assessment in the area of safety of nuclear reactor was formulated as follows [7]: “Safe operation of the nuclear reactor assessed is supported by accounting for human factor requirements in the design and operation of the plant, and by establishing and maintaining a strong safety culture in all organizations involved”.

4.9.1. Criterion CR6.1: Human factors

Indicator IN6.1: Human factor considerations are addressed systematically throughout the life cycle of the reactor.

Acceptance limit AL6.1: Human factor assessment results are better than those for the reference design.

INPRO methodology [7] provides an exemplary list of design and operational features to be considered at the assessment of this criterion. These features involve: a probabilistic safety assessment taking human error into account; existence of a main control room, a supplementary control room (remote shutdown station) and a technical support centre with appropriate ambient conditions; using visualizations of plant equipment status, the dynamics of processes, the performance of automated processes and their relation with the state of the plant; monitoring by knowledge-based systems; appropriate organizational and administrative structure and appropriate plant operating procedures; control of human reliability; feedback of experience including a formal methodology; use of adequate models considering the causes of human error; existence of a verification of design implementation adequacy.

Human errors are considered in the BN-1200 design safety assessment report [69].

Main control rooms and supplementary control rooms (remote shutdown stations) exist both in BN-800 and in BN-1200 design and appropriate ambient conditions are normally maintained in the relevant premises. Technical support to the operator, in the case of emergency in Russian NPPs, is provided on two levels, from on-site emergency centres and from the technical support centres organised in 14 leading design and research organisations that participated in the development of every reactor currently operated in the Russian Federation [83]. On-site emergency centres are located in protected premises, provided with reliable energy supply (in the case of station blackout) and equipped with different types of communication tools (including videoconference and satellite communication tools).

Using visualizations of plant systems, structures and components status, the dynamics of processes, the performance of automated processes and their relation to the state of the plant to help guide operator actions are discussed in EP2.1.2. Monitoring by knowledge-based (expert) systems is discussed in EP2.1.1.

Availability and development of appropriate plant operating procedures for normal operation, incident and accident situations in BN-1200 is discussed in EP1.2.5 and CR4.3.

Organizational and administrative structure at Beloyarsk NPP is discussed in EP1.2.7. Control of human reliability including periodic trainings and personnel selection procedure at Beloyarsk NPP is discussed under EP1.2.5 and EP1.2.6.

Information on formal procedure for collecting and disseminating of feedback of experience in Russian utility Rosenergoatom is provided in Ref. [84]. The All-Russian Research Institute for Nuclear Power Plants Operation (VNIIAES) collects technical decisions on improvement of operating regimes, of repair, maintenance and modernizations procedures of NPP systems, structures and components. VNIIAES developed an information system (data bases) for efficient information exchange and dissemination of good practices among NPPs. NPPs provide their feedback of the improvements implemented (proposed) in other NPPs.

Information from BN-1200 design organization on using of adequate quantitative models considering the causes of human error, which may assist to find appropriate design measures to avoid the causes and thus minimize human errors has not been available to the assessor.

Information on existence of a design implementation adequacy verification process / procedures has not been available to the assessor. Assessment against this criterion is not complete.

4.9.2. Criterion CR6.2: Attitude to safety

Indicator IN6.2: Prevailing safety culture.

Acceptance limit AL6.2: Evidence is provided by periodic safety culture reviews.

National regulatory body (Rostechnadzor) hosted IRRS missions in 2009 and 2013 (follow up mission). Brief overview of the latest mission results (2013) are available at the IAEA web site [85]. No recommendations have been made about the safety culture.

Rosenergoatom hosted the IAEA OSART mission in November 2018. General results are positive (three good practices, 32 areas of good performance and 6 areas where potential improvements have been suggested).

WANO peer-review mission (partnership verification) was held in Beloyarsk NPP in August–September 2016 and follow up mission in 2018.

The IAEA plans to conduct the first OSART mission in Beloyarsk NPP in 2021. Preparation to the mission started in 2019.

Ref. [72] provides brief information on the safety culture requirements in the designer organizations. All organizations and experts involved in the development of BN-1200 design have to be aware of the relation between their specific activities and the NPP safety. They have to maintain safety culture through the selection and training of personnel in the areas related to safety, through definition, updates using the accumulated experience and monitoring of requirements in the corresponding guides and manuals. High level of safety culture can be supported by the appropriate implementation of quality management system.

4.10. UR7: NECESSARY RD&D FOR ADVANCED DESIGNS

INPRO methodology user requirement UR7 for sustainability assessment in the area of safety of nuclear reactor was formulated as follows [7]: “The development of innovative design features of the nuclear reactor assessed includes associated research, development and demonstration (RD&D) to bring the knowledge of plant characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for operating plants”.

4.10.1. Criterion CR7.1: Safety basis and safety issues

Indicator IN7.1: Safety basis and a clear process for addressing safety issues.

Acceptance limit AL7.1: The safety basis for advanced designs is defined and safety issues are addressed.

National nuclear rules and regulations comprise approx. hundred documents applicable to the fast reactor safety. The high-level requirements are defined in the following documents:

– General Safety Provisions for Nuclear Power Plants (OPB-88/97);
– Requirements to the Contents of Safety Assessment Report for NPPs with Fast Reactors (NP-018-05);
– Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants (NP-082-07);
– Radiation Safety Standards (NRB-99/2009);
– Basic Sanitary Regulations of the Radiation Safety Assurance (OSPORB-99/2010).

National regulations require using both deterministic and probabilistic approach for the reactor safety assessment. The scope of safety assessment is defined in the Requirements NP-018-05 and includes consideration of the abnormal situations, design basis accidents and beyond design basis accidents.

The BN-1200 reactor design target indicator on safety [17, 31, 36, 69] states that the necessity of public evacuation in the case of any technically credible accident including those having very low frequency and involving failures of all active safety systems and normal operation systems relevant to safety, single failures of passive components having mechanically moving parts and operator's errors has to be eliminated. The BN-1200 design is expected to prevent or mitigate the radioactive releases into the environment so that the radiation levels stay below regulatory criteria for public evacuation.

Safety criteria for BN-1200 have been defined in terms of the public doses and frequencies of accidents [71]:

– Public dose during normal operation at site boundary and beyond – 0.02 mSv/a;
– Public dose at AOO at site boundary and beyond – 0.1 mSv/a;
– Public dose at DBA at site boundary – 1 mSv/a for the first year after accident and no protective measures outside of NPP site;
– Public dose at BDBA beyond site boundary – 50 mGy in the first 10 days and 50 mSv/a for the first year after accident (no evacuation required);
– Frequency of BDBA with fuel damage exceeding design limits – 10^{-6};
– Frequency of release to the environment requiring protective measures – 10^{-7}.

The safety concept of BN-1200 is based on implementation of DID concept [71], using of proven technologies, techniques and experience from BN-600 and BN-800 to the extent possible [18, 19] and incorporation of new technologies developed as the result of extensive R&D studies on existing and newly created research facilities. It is focused on the development and enhancement of inherent safety (compared to BN-600 and BN-800) and improvement of passive safety ensuring the fulfilment of three basic safety functions (controlling the power, cooling the fuel and confining the radioactive material).

The BN-1200 design incorporates several inherent safety features and passive systems including the following [69]:

– High reactor heat capacity ensuring a relatively low temperature growth rate during the AOO and accidents;
– Negative reactivity feedbacks on the reactor power and core temperature, and design provisions (passive) for introduction of negative reactivity in case of sodium boiling minimizing effects from the positive sodium-void reactivity feedback;
– Placing of all primary sodium pipelines in the reactor vessel (including cold filters-traps) eliminating leaks of radioactive sodium;
– Emergency removal of heat directly from the reactor vessel through the passive system;

- Passive reactor shutdown system, actuated in case of the coolant temperature increases (in addition to the passive system of hydraulically suspended rods actuated in case of loss of flow);
- Confinement of the accidental gaseous discharges from reactor. After cool-down and decay of short-lived isotopes the gas is further discharged into a special ventilation system.

BN-1200 safety basis demonstrating and documenting that safety requirements are met is presented in Refs [86, 87]. These documents are the result of iterations among design, RD&D and safety analysis.

Documented results of the process addressing all safety issues including sensitivity and uncertainty analyses and independent reviews have not been available to the assessor. Assessment against this criterion is not complete.

4.10.2. Criterion CR7.2: RD&D

Indicator IN7.2: RD&D status.

Acceptance limit AL7.2: Necessary RD&D is defined and performed, and the database is developed.

Refs [13, 32] provide information on the structure of R&D works performed and planned for BN-1200 reactor design. This structure splits into six areas:

- Justification of technical and economic characteristics of BN-1200 power plant and associated nuclear fuel cycle;
- Development, modification and validation of computer codes;
- R&D on systems, structures and components of BN-1200 reactor installation;
- R&D on steam-generators;
- R&D on reactor core;
- R&D on safety.

R&D on justification of technical and economic characteristics of BN-1200 power plant and associated fuel cycle are out of the scope of this study. Development, modification and validation of computer codes is discussed in the next CR7.3. The next four groups of R&D involve the following research studies.

R&D on systems, structures and components of BN-1200 reactor installation:

- Studies on primary coolant pumps and secondary coolant pumps;
- Studies on control rods and drivers;
- Studies of fuel reloading systems;
- I&C studies;
- EHRS studies, EHRS heat exchangers valves studies and EHRS sodium / air heat exchangers studies;
- Primary sodium cleaning system studies;
- Bellows type compensators;
- Fuel claddings leak-tightness monitoring systems;
- Thermal insulation and electric heaters.

R&D on steam-generators:

- Materials study;
- Experimental justification of materials used;
- Computational and experimental justification of the steam generator characteristics;
- Tests of bellows compensators of the steam generator vessel;

- Steam generator throttle device study;
- Steam generator automatic protection system study and computational justification of protection system modifications;
- Development of a system for timely detection of small leaks in the steam generator;
- Steam generator fabrication technology development;
- Base metal and welds inspection methods.

R&D on reactor core;

- Core computational studies;
- Experiments on justification of neutronic characteristics, mechanical and hydraulic characteristics of fuel assemblies;
- Passive reactor protection systems (control rods);
- Reactor studies of fuel rods;
- Development of technology for fuel manufacturing;
- Development of quality management systems for fuel manufacturing.

R&D on safety:

- Studies on nuclear safety and radiation protection in BN-1200;
- Studies on release of fission products, tritium production and migration, contamination of sodium and systems, structures and components with actinides;
- Studies on radioactive waste disposal including scale and conditions;
- Studies of C-14 effects;
- Fuel management studies;
- Studies of AOOs, DBAs and BDBAs;
- Studies of inherent safety limits;
- Studies of hydrogen safety of BN-1200;
- Studies of fuel damage from boiling of coolant and loss of coolant flow;
- Fire protection studies.

Abundant information on the results of these R&D studies is available in the research publications. For example, Ref. [70] provides overview of the full-scale experiments on validation of the steam generator characteristics in the SPRUT installation. The new research installation TISEY has been created for testing of the EHRS design which is different from the EHRS installed in BN-350, BN-600 and BN-800 [31, 81]. Results of computational and experimental studies of sodium purification system of fast reactors are provided in Refs [88–91]. An overview of results of experimental studies of sodium leaks and fires under different conditions are provided in Ref. [92]. Ranges of reactor characteristics and processes parameters are covered either by data measured during BN-600 and BN-800 operation or by data measured in experiments. Scaling studies were performed using analytical methods and by conducting series of experiments with samples or systems having different sizes.

However, direct information demonstrating that all phenomena are understood, data uncertainties are quantified, and documented in reports has not been available to the assessor. Reliability data with uncertainty bands has not been available to the assessor. Assessment against this criterion is not complete.

4.10.3. Criterion CR7.3: Computer codes

Indicator IN7.3: Status of computer codes.

Acceptance limit AL7.3: Computer codes or analytical methods are developed and validated.

Ref. [13] presents the list of computer codes used in BN-1200 design process. It includes more than 30 tools developed in the Russian Federation and abroad for neutronic, thermal hydraulic, thermal mechanical calculations under normal operation, AOO and accidental conditions etc. Ref. [50] gives more details on the specific areas of application of these tools.

Ref. [93] states that by 2014 the national research organizations have created a methodology for the fast reactor safety analysis in all relevant areas. All safety related areas and parameter ranges of BN-1200 have been covered by the available computer tools and data libraries. Ref. [50] provides the list of experimental facilities created or used[52] as the part of BN-1200 development programme and used for the computer tools validation in the national research institutions:

- BFS (stands for 'fast physical installation') – complex zero-power research reactor;
- PLUTON – studies on fuel damage during loss of flow accidents;
- SGK – studies on fission products and actinides releases from irradiated fuel;
- AR-1 – studies on thermal hydraulics and heat exchange at coolant boiling in the core;
- B-200 and TESEY – studies on EHRS efficiency;
- STIZ and PUSCHM – studies on sodium burning in BN-1200;
- SPRUT and Acoustics – steam generator studies.

Further improvement of the computer tools and associated physical libraries [94] required deep revision of the existing algorithms and the development of new and more sophisticated approaches including integral codes [93]. New research programme called 'New generation computer codes' has been started in parallel to the development and optimization of BN-1200 design.

This new programme is managed through the System of congregate development of new generation computer codes (SURRK-ALM). This system provides the centralized storage of all development, testing and validation results. It plans and controls the development process, automatic tests and correction of errors, provides information support to the users and feedback between the users and developers [95, 96]. Seventeen new computer codes are planned to be developed and validated under the new generation computer codes programme and the programme is expected to be completed by 2020 [96, 97]. By 2016 approx. 60% of works have been fulfilled [38]. Tables 16 and 17 provide the list and brief description of the new generation codes developed for fast reactor programme in the Russian Federation.

TABLE 16. NEW GENERATION INTEGRAL COMPUTER CODES (modified from [97])

Code name	Brief description
SOKRAT-BN	Integrated design code for safety assessment of NPPs with sodium-cooled fast reactor
EVKLID	Integrated universal design code for safety assessment of NPPs with fast reactors. Accurate thermo-hydraulic, thermo-mechanical and neutronic calculations of fast reactor

[52] BFS-1 and BFS-2 installations were commissioned in 1961 and 1971.

TABLE 17. NEW GENERATION COMPUTER CODES FOR MODELLING OF INDIVIDUAL GROUPS OF PHYSICAL PROCESSES AND PHENOMENA (modified from [97])

Code name	Brief description
MCU-FR	Monte Carlo neutronics code for calculation of the fast reactor characteristics, radiation protection and fuel cycle installation parameters
ODETTA	Neutronic code based on the kinetic approximation (finite element and discrete ordinate methods) for calculation of the fast reactor characteristics, radiation protection and fuel cycle installation parameters
NDP-ACE	Software for processing nuclear data
HYDRA-IBRAE/LM	System thermo-hydraulic code modelling fast reactor coolant circulation loops, including the model of porous body and CFD-modelling of individual elements of the reactor facility
LOGOS	CFD code with RANS models of turbulence for detailed analysis of small-scale heat and mass transfer processes in fast reactors
CONV-3D	CFD code with DNS approximation for detailed analysis of small-scale heat and mass transfer processes in fast reactors
BERKUT	Code for analysis of fast reactors fuel elements behaviour under normal operation and accident conditions
	Thermo-mechanical module: behaviour of the core structures and components, core support lattice, pressure vessel, pipelines, fittings under the temperature and radiation conditions (change in the properties of structural materials, strength calculation etc)
KUPOL-BR	Modelling of the fission products propagation and heat / mass transfer through the reactor rooms
CRISS 5.3	Probabilistic safety analysis and risk assessment for fast reactors
ROM	Transport of radioactive materials through the environment (air pathways)
Sibilla	Transport of radioactive materials through the environment (water pathways)
ROUZ	Three-dimensional calculation of the radiation conditions in the reactor site
GeRa	Safety of management of radioactive waste from the closed nuclear fuel cycle
VIZART	Calculation of the material balance of materials and flows of isotopes in closed nuclear fuel cycle facilities
KOD TP	Code for simulating the operation of the technological scheme of the plant or individual production sections in real-time

Ref. [97] provides an overview of new generation codes status, areas of application, validation and certification progress. Certification of codes in Russian Federation involves confirmation of the region of code application, evaluation of uncertainties and sensitivities, and independent reviews. Detailed code manuals are necessary prerequisites for certification.

4.10.4. Criterion CR7.4: Novelty

Indicator IN7.4: Pilot or demonstration plant.

Acceptance limit AL7.4: In case of a high degree of novelty: a pilot or demonstration plant is specified, built and operated, lessons are learned and documented, and results are sufficient to be extrapolated to a full-size plant. In case of a low degree of novelty: a rationale is provided for bypassing a pilot or demonstration plant.

The BN-600 reactor (pool-type, sodium cooled reactor) has over 30 years of successful operational experience and is considered to be an adequate pilot plant for the BN-800 and BN-1200 reactors [32]. The results of peer reviews of the pilot facility (BN-600) operation are presented in numerous research articles, e.g. Refs [18, 98], and in a series of technical reports (access restricted). An overview of modifications of the major BN-1200 components compared to earlier designs (BN-600 and BN-800) is provided in Ref. [14].

4.10.5. Criterion CR7.5: Safety assessment

Indicator IN7.5: Adequate safety assessment involving a suitable combination of deterministic and probabilistic methods, and identification of uncertainties and sensitivities.

Acceptance limit AL7.5: Uncertainties and sensitivities are identified and appropriately dealt with, and the safety assessment is approved by a responsible regulatory authority.

Russian regulations require performing deterministic and probabilistic safety assessments of NPPs [32]. The scope of deterministic safety assessment is provided in Ref. [99] and involves AOO, DBA and BDBA analysis. Probabilistic safety assessment is required at probabilistic safety assessment level 1 (evaluation of the severe core damage frequency). Probabilistic analysis has to be performed for the accidental sequences (event trees) based on the defined in the design initiating events of AOOs and DBAs and failures of the systems important to safety.

Ref. [55] provides an explanation to the scope of fast reactor safety assessment in the Russian Federation including the list of examples of AOOs and DBAs. It also presents the list of postulated BDBAs which was defined by the national regulatory body ROSTEHNADZOR and which must be considered in the safety assessment report. BDBAs selected for the analysis should potentially be able to cause severe core damages and release of radioactivity from the primary circuit.

Preliminary results of the BN-1200 reactor safety assessment are presented in Refs [86, 87].

Direct information on analysis of uncertainties and sensitivity studies has not been available to the assessor. Information on the status of consideration of BN-1200 safety assessment report by the regulatory body has not been available to the assessor. Assessment against this criterion is not complete.

5. DISCUSSION AND CONCLUSIONS

Among several different fast reactor designs using different coolants which are being developed in the Russian Federation, the sodium cooled reactors are in the most advanced stage of maturity and closer than other designs to the commercial deployment stage. The sodium cooled fast reactors have successfully passed the experimental programme and prototypes tests, and closed fuel cycle technologies have been successfully developed and tested including the MOX fuel fabrication and spent fuel aqueous reprocessing. It is expected that sodium cooled fast reactors and closed fuel cycle technologies will be refined and validated in the next decade.

However, in the short and mid-term water cooled reactors most probably will remain the predominant technology for the nuclear power strategy of the Russian Federation. Fast reactors can contribute to the energy supply mix expanding significantly the resources of fissile material, minimizing the amount of spent nuclear fuel and radioactive waste.

The comparison of basic economic indicators of BN-1200 and those of advanced water cooled reactors can provide preliminary data for the estimation of the effects of the potential introduction of commercial fast reactors. Such estimations, if done correctly, should include effects from the production of plutonium for the reactors other than BN-1200 which have not been fully considered here. Benefits from the services of recycling minor actinides[53] which are currently under investigation and the economic effects from using nitride fuel have not been considered in this study[54].

In the INPRO area of reactor safety the sustainability assessment helps to understand the difference between implementation of the defence in depth concept in fast reactors and that of water cooled reactors. The absence of phase transients in a broad range of coolant parameters values affects the strategy of safety barriers protection and use. It is expected that INPRO assessment in this area can contribute to the development of safety assessment approaches for the sodium cooled fast reactors.

Data available for this limited scope INPRO sustainability assessment demonstrate that the fast reactor programme in Russian Federation develops, in general, along the lines of sustainable development. Conclusion on the sustainability of nuclear energy system with BN-1200 reactors cannot be complete in this study due to the limited scope of assessment and insufficient data available in the open domain for the assessment.

In this limited scope study in total 36 criteria and 24 evaluation parameters have been assessed including 8 criteria and 3 evaluation parameters in the INPRO area of economics and 28 criteria plus 21 evaluation parameters in the INPRO area of reactor safety. In 27 items (of total 60) the assessment is not complete due to the insufficiency of input data. When the missing data are compiled the assessment can be finalized and expanded to the rest of INPRO methodology areas to make the judgement on the system sustainability possible[55].

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[53] Reducing the amount of high level waste from the reactors other than BN and radiotoxicity.

[54] Recycling of minor actinides is considered in the INPRO collaborative project on Cooperative Approaches to the Back End of the Nuclear Fuel Cycle.

[55] Traditionally the developers of new reactors are aimed at the achievement of economics and safety characteristics necessary for the investment justification and for design authorization by the regulatory body. The number of characteristics can vary in different countries and they can be evaluated at different stages of the design development. When data available to the INPRO assessor are not sufficient for the assessment of specific criteria, the developer can be informed on the input information insufficiency and the assessment can be completed when the necessary data are provided.

Input data needed to finalize the limited scope INPRO sustainability assessment of BN-1200 will include the following:

- Information on status of licensing;
- Data on the accuracy of estimated construction time;
- Information on the BN-1200 design characteristic for the reactor operation in non-baseload regime;
- Data demonstrating that the redundancy of operational systems is greater than that in the reference design;
- Normal operation levels and margins of BN-800 reactor;
- Data demonstrating that incorrect human intervention during normal operation has less impact on reactor operation than in the reference design (BN-800);
- Information on the accounting of operating experience from foreign NPPs that are not fast reactors;
- Information on the BN-1200 design features facilitating the performance of testing and maintenance and potential improvements of their effectiveness and efficiency;
- Information on the frequencies of AOO in BN-800 and BN-1200;
- Information on the improvement of monitoring systems in BN-1200;
- Information on the longer grace periods after AOO in BN-1200 design;
- Information on frequencies of DBA caused by internal or external events or probable combinations thereof in BN-800 and BN-1200;
- Information on the grace periods after DBA in BN-1200 design;
- Information from a BN-1200 probabilistic safety assessment demonstrating calculated increased reliability of the safety systems for all states of the nuclear reactor (full and reduced power operation, shutdown state);
- Information on the subcriticality margins after reactor shut down by the active system, by passive hydraulically suspended control rods system, or by passive high temperature actuated control rods system;
- Information on frequencies of severe core damage during the reactor shutdown states;
- Information on uncertainties associated with calculated frequencies of severe core damage;
- Information on frequency of accidental release of radioactive materials into the environment and associated uncertainties;
- Information on frequency of large releases and early releases of radioactive materials into the environment or technically sound confirmation of elimination of such releases;
- Information on the source terms of accidental releases from the BN-1200 and BN-800 reactors;
- Information on the methods (deterministic or probabilistic) and assumptions (e.g. whether conditions) used by BN-1200 designer for transport calculations of radioactive materials in the environment;
- Information from probabilistic and deterministic analysis on the improvement of independence of DID levels in BN-1200 reactor;
- Information on the outcome of outside of the core criticality analysis for BN-1200 fuel;
- Information from the BN-1200 design organization on using adequate quantitative models considering the causes of human error, which may assist to find appropriate design measures to avoid the causes and thus minimize human errors;
- Information on existence of a design implementation adequacy verification process / procedures;
- Documented results of the process addressing all safety issues including sensitivity and uncertainty analyses and independent reviews;

- Information demonstrating that all phenomena are understood, data uncertainties are quantified, and documented in reports;
- Reliability data with uncertainty bands;
- Information on analysis of uncertainties and sensitivity studies;
- Information on the status of consideration of BN-1200 safety assessment report by the regulatory body.

The INPRO sustainability assessment of the nuclear energy system based on BN-1200 reactor is expected to be completed along with the development of detailed design of the reactor and deployment of the system including associated closed fuel cycle option[56]. The results of limited scope assessment presented in this report can inform the developers of BN-1200 on the actions to be taken and criteria to be met in the future to achieve the system sustainability.

[56] Closed nuclear fuel cycle with nitride or oxide fuel.

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LIST OF ABBREVIATIONS

AOO	anticipated operational occurrence
BN	sodium-cooled fast reactors designed in the Russian Federation
DBA	design basis accident
DID	defence in depth
EHRS	emergency heat removal system
FOAK	first-of-a-kind
I&C	instrumentation and control
IRR	internal rate of return
LUAC	levelized unit amortization cost
LUEC	levelized unit energy cost
mills	0.001 of the US dollar
MOX	mixed oxide fuel
MSK	Medvedev–Sponheuer–Karnik macroseismic intensity scale
NEST	nuclear energy system assessment economics support tool
NOAK	N-th-of-a-kind
NPP	nuclear power plant
ROI	return on investment
RD&D	research, development and demonstration
R&D	research and development work
ULOF	ultimate loss of flow
ULOHS	ultimate loss of heat sink
UTOP	unprotected transient over power
VVER	water cooled water moderated power reactor

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