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Pre-Conceptual Design of the Research High-Temperature Gas-Cooled Reactor TeResa for Non-Electrical Applications

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Abstract: In line with Polish national activities and research programs investigating non-electrical-reactor use, the national GOSPOSTRATEG-HTR project was launched, aiming at the development of a novel pre-conceptual design of a High-Temperature Gas-cooled Reactor (HTGR). The 40 MW_{th} research reactor would serve as a technology demonstrator for future industrial purposes. In the paper, the proposal of an established thermal-hydraulic and neutronic core design is presented as a result of the National Centre for Nuclear Research team studies, in the scope of the project, including the areas of fluid mechanics, heat exchange and reactor neutronic core design support analyses. The undertaken analyses were confirmed by the series of code investigations involving integral thermal-hydraulic (MELCOR (Sandia National Laboratories, USA), CATHARE (CEA, France)), neutronic (Serpent (VTT, Finland), MCB (AGH University's Department of Nuclear Energy, Poland)), Computational Fluid Dynamics (ANSYS Fluent (ANSYS, USA)) and others. The calculations performed within the preliminary safety analysis on the pre-concept showed its compliance with international safety standards for the normal operation and Design Basis Accident sequences.

Keywords: HTGR; core design; non-electrical nuclear applications; research reactor

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1. Introduction

As a consequence of the ratification of the Paris Agreement in 2015 by the European Union and other countries, responsible for 55% of global greenhouse gas emissions, a set of actions were undertaken in the EU member countries, including Poland, aiming at limiting an increase in the global average temperature by decreasing greenhouse gas emissions and slowing the speed of climate change [1]. Following the Paris Agreement, actions outlined by the European Commission, in the form of the European Green Deal formulated in 2019, provide a set of policy initiatives with the overarching aim of making Europe climate neutral by 2050 [2]. Furthermore, the Glasgow Climate Pact [3], which is an agreement reached at the United Nations Climate Change Conference (COP26) in November 2021, is a climate deal that presses for even more urgent emission cuts.

This direction of the transformation of the energy sources market is the main focus of the polish Ministry of Climate and Environment (MKiŚ) in establishing the policies, alongside supervised research and development (R&D) programs. In 2016, the Ministry established a departmental Committee, which was responsible for the elaboration on recommendations for the implementation of High Temperature Gas-cooled Reactors (HTGRs) in Poland. During several months of its work, the Team conducted a detailed analysis of the use of high-temperature reactors to cater to the domestic demand for industrial heat with a temperature of up to 700 °C. Among the analyzed technologies, the recommended helium-cooled reactors were found to be optimal [4]. The final report detailing the results of the Committee's works was published in 2017 [4]. The Minister accepted this report, further steps were taken, and as a result, HTGR technology was included in some strategic documents in Poland related to the economy, development and into energy policy framework. The Strategy for Responsible Development [5] is the governmental plan for Polish economy growth. This document contains the list of future action related to energy production, among which the preparation of HTGR deployment for industrial heat production, using both industrial and scientific potential of Poland, is considered. The high-temperature nuclear technology, as a possible future heat source for industry and cogeneration, is included in the in Polish Nuclear Power Program [6], Energy policy of Poland till 2040 [7], National Energy and Climate Plan for the years 2021–2030 [8], and National Smart Specializations [9,10]. The Polish Roadmap of Research Infrastructure [11] is a list of undertakings of strategic importance to the development of Polish science and its competitiveness. Here, the experimental, high-temperature reactor is considered so as to conduct scientific research and fulfil technical needs arising from the design and licensing process, as well as to build competences for the implementation of future industrial applications. It is worth emphasizing that in May 2021, the Government of the Republic of Poland and the Government of Japan signed an action plan for the strategic partnership [12]. In compliance with this document, the economic cooperation includes, inter alia, research and development cooperation in the field of HTGR between the National Centre for Nuclear Research of Poland and Japan Atomic Energy Agency. Additionally, in May, the National Centre for Nuclear Research and the Ministry of Education and Science signed a new contract for the implementation of another batch of design works for experimental HTGR.

Earlier, at the beginning of 2019, and following the work of the ministerial Committee, the R&D venture was proposed—a GOSPOSTRATEG-HTR project, which is an important vehicle for shaping the country's energy policy, allowing for the combination of the organizational potential of the state with the research and scientific capabilities of research institutions, supporting the coordination of preparation for the practical use of HTGR in the Polish economy [13]. The project was financed by the Polish National Center for Research and Development.

During the GOSPOSTRATEG-HTR project, a part of the work was focused on the broad issue of preparation of the licensing (certification) process of HTGR reactors with the example of a research reactor which could also serve a model of the technology for interested parties and end-users. The tasks involved a detailed iterative effort in the novel neutronic and thermal-hydraulic design development of the 40 MW_{th} reactor pre-concept. The design was characterized by core outlet temperature, which was estimated to be anticipated by the end-users based on the previous market studies of expected temperatures and pressure from the ministerial Committee in 2017.

The pre-conceptual core design was prepared on the basis of experiences from the European H2020/Euratom GEMINI Plus project [14]. The main idea was to have a research facility with the maximum number of similar features as a commercial plant, including core design, plant configuration, and safety systems.

The preliminary safety analysis for the developed plant design covered core neutronics and primary loop thermal-hydraulics and Reactor Cavity Cooling System (RCCS), which were investigated against typical prismatic HTGR Design Basis Accidents (DBA), such as the most challenging ones, i.e., the Depressurized Loss of Forced Cooling (DLOFC) accident, Pressurized Loss of Forced Cooling (PLOFC), along with Water and Air Ingress events. Those investigations were selected to be demonstrative of the pre-conceptual design compliance to the regulatory acceptance criteria and the paper will focus on summary of those of interest. The criteria that were considered from the neutronic and thermal-hydraulic perspective were:

- The core configuration being able to perform economically in fuel cycle length of around 3 years with sensible uniform radial power distribution;
- Low estimations of normal operational and accidental releases to the environment kept within the legal limits;

- A Core thermal-hydraulic design able to maintain the fuel temperature below 1600 °C during the most challenging selected DBA event sequences;
- An efficient pre-concept of the decay heat removal system able to perform in various operating conditions.

2. GOSPOSTRATEG-HTR Structure and Objectives

As was already mentioned in the Introduction, with regard to the decision pertaining to interest in the HTGR technology by Poland, on 30 January 2019, the Ministry of Energy signed a contract for the implementation and financing of the GOSPOSTRATEG-HTR project — "Preparation of legal, organizational, and technical instruments for the implementation of HTR reactors" concluded by the Ministry of Energy with the National Center for Research and Development in Poland.

The implementation period of the GOSPOSTRATEG-HTR project was 1 February 2019–31 March 2022. The aim of the project is to make a comprehensive legal, organizational and economic analysis in terms of the deployment of the HTGR in Poland (research and recommendations - Phase A, implementation Phase B). For these purposes, a consortium was established consisting of the Ministry of Climate and Environment (Project Leader), the National Centre for Nuclear Research (NCBJ) (financial leader) and the Institute of Nuclear Chemistry and Technology (IChTJ) (consortium member) [13].

The primary far-reaching aim is an increase in the fraction of clean energy sources in the Polish energy mix, leading to a decrease in greenhouse gas emissions. This would be possible to accomplish thanks to the implementation and use of HTGR technology, which would result in a reduction in the dependence on gas and oil imports, a reduction in CO_2 emissions, obtaining and developing new technologies, and increasing the technological level of Polish component suppliers, as well as exporting these components to other countries interested in HTGR technology. It was evaluated that launching the production of such reactors in Poland would contribute to the high-tech re-industrialization of the country and the creation of a new economy branch with export potential [4].

The GOSPOSTRATEG-HTR project consists of two phases, which are presented, respectively, in Tables 1 and 2. Phase A of the project includes research work aimed at the identification of the necessary changes of national legal acts, preparation of test procedures and the equipment necessary for their implementation, and an analysis of the potential social, economic and industrial benefits for the Polish economy. Phase B includes the implementation of the developed procedures and identification of the necessary changes in national legal acts by incorporating them into the system of approvals and permits, in particular in the field of the Atomic Law, the main source of nuclear-related regulation in Poland. In this way, the main goal of the project, which is to prepare legal and organizational instruments for the implementation of HTGR technology, will be achieved.

Table 1. GOSPOSTRATEG-HTR project Phase A.

Phase A—Research Phase (1 February 2019–31 July 2020)		
WP Number	Work Package Title	Involved Institutions
1	Development of methods for diagnostics of structural materials in the HTR construction	NCBJ
2	Development of methods for testing structural materials in a nuclear reactor, and equipment for the execution of tests in the core.	NCBJ
3	Research and analysis of selected chemical aspects of the production and use of TRISO fuel in the HTR nuclear reactor.	IChTJ
4	Comprehensive analysis of the necessary changes to the legal environment and the potential benefits of social, economic and industrial units for the Polish economy.	MKiŚ, NCBJ

Table 2. GOSPOSTRATEG-HTR project Phase B.

Phase B—Implementation Phase (1 August 2020–31 March 2022)		
WP Number	Work Package Title	Involved Institutions
5	Preparation-licensing process (certification) of HTGRs with the example of a research reactor.	NCBJ, MKiŚ, IChTJ
6	Preparation draft of legal regulations for the HTR investments implementation and developing a strategy in the social, economic and industrial aspects of the project.	NCBJ, MKiŚ, IChTJ
7	Piloting test procedures for the use of construction materials for the HTR design, including tests in the MARIA reactor core.	NCBJ
8	Preparation of technical and economic assumptions for the construction of a fuel production unit for high-temperature reactors.	IChTJ

One of the key work packages which contains the elements of all the others work packages and also results in the development of a pre-concept of the research reactor is WP Number 5: Preparation licensing process (certification) of HTGRs on the example of a research reactor.

The licensing framework for nuclear facilities in Poland is regulated by two main acts together with a few more detailed regulations included in secondary legislation, and several other acts to which they refer. The Act on Atomic Law [15] regulates the civil use of nuclear energy. The second is the Act on the Preparation and Implementation of Investments in Nuclear Power Facilities and Accompanying Facilities [16]. Nuclear law is continuously revised by the Regulator, taking into consideration EU regulations, including Western European Nuclear Regulators Association (WENRA) and International Atomic Energy Agency (IAEA) safety standards, and other binding acts of international law. However, the existing licensing framework is not technology-neutral. It is focused on current nuclear technologies, mainly on light water reactors (LWRs), and commercial nuclear units aimed at electricity generation only. The main design characteristics differentiating HTGRs from LWRs are the use of a helium coolant, the graphite moderator and the use of coated fuel particles. Additionally, the innovative features and safety characteristics of HTGRs provide safe and reliable operation. For this reason, many domestic provisions need to either be modified or created, taking into account the characteristics of HTGR technology. The three crucial issues, specific for HTGRs, have been identified to be included in the licensing framework, namely the possibility of industrial or district heat production, inherent safety characteristics, and a containment system.

The possibility of heat or combined heat and power (CHP, cogeneration) generation, as well as the possible influence of a nuclear site on end-user site, and vice versa, should be included in the regulations. The inherent safety characteristics of HTGRs, and thus the reactor design, which are not similar to that of LWRs, should be incorporated into the licensing requirements. On the other hand, not all existing LWRs-focused provisions are applicable for HTGRs. For HTGRs, the fuel design (multi-layer coated fuel particles) can be considered as a kind of micro-containment (also considered as a part of the so-called functional containment). The TRISO fuel is used as the dominant safety barrier to retain fission products [17]. Less importance is placed on the containment structure (reactor building). Thus, the main contributors to fulfil the confinement function are different than in LWRs, where the leak-tight containment structure is regarded as the final and very important radionuclide retention barrier. The key legal acts were detected, and the proposition of provisions modifications were presented. This action was preceded by recognizing the legislation frameworks in nuclear countries across the world. Especially those which already implement the non-electrical application of nuclear energy or have interests in HTGR technologies. Moreover, country-level legal frameworks as well as the IAEA and WENRA recommendations and different regulatory guidelines were taken into account, e.g., IAEA technical report entitled Applicability of Design Safety Requirements to Small Modular Reactor Technologies Intended for Near Term Deployment (TECDOC-1936) [17]. In this document, the applicability of the requirements for nuclear power plants established in IAEA Safety Standards on Safety of Nuclear Power Plants: Design (SSR-2/1, rev. 1) [18] to HTGRs is considered. The document SSR-2/1 is clearly reflected in Polish nuclear law, thus the TECDOC-1936 was very helpful to point out necessary modifications.

In the above context, some of the selected milestones of WP Number 5 were as follows:

- Development of the novel 40 MW_{th} research HTGR pre-conceptual design.
- Development of the analysis for the radioactive substance distribution in the HTGR circulation loop and radiation hazards under normal operating conditions. Core releases and effects of those hypothetical releases.
- Analysis of built-in safety features, safety systems, requirements for their operation in emergency situations and their classification and qualification for emergency conditions.
- Identification of initiating events and accident scenarios.
- Determination of the distribution and possible propagation processes of fission products in the HTGR reactor and their releases outside the HTGR reactor in situations of DBAs and severe accidents.
- Conclusions regarding the requirements for the barriers in the HTGR reactor as well as the restricted zone and proposals for appropriate changes in legal regulations.

In conclusion, in the scope of the GOSPOSTRATEG-HTR project, the pre-conceptual design of a research HTGR, named TeResa, was developed. Based on the TeResa pre-conceptual design, some basic analyses for the licensing process purposes were performed.

3. TeResa Reactor Design Pre-Concept and Calculations

The pre-concept of the multipurpose research and demonstration reactor TeResa emerged as a necessary step before the possible development of an industrial scale first of a kind unit—such as GEMINI+. It is crucial to develop technology and establish a supply chain, confirm safety, validate safety methods, gain proper experience and decrease the risk of implementation of a larger scale HTGR reactor. This idea progressed to proposition of the low power experimental prismatic reactor.

The TeResa reactor is a concept based on the GEMINI+ HTGR solution. The GEMINI+ reactor [14] is a prismatic block-type reactor HTGR with a gross thermal power of 180 MW_{th}, designed to supply process steam to end users. The thermal power output of TeResa reactor is reduced to 40 MW_{th}. In relation to GEMINI+ core configuration, the radial dimensions remain unchanged (identical horizontal cross section), however, in the axial direction, the active core height is reduced from 11 layers of fuel blocks to 6 layers. The reactor components, e.g., fuel and reflector blocks, fuel compacts, coated fuel particles, reactor pressure vessel (RPV), core barrel, etc., are almost the same as those proposed in the GEMINI+ concept [14,19,20]. GEMINI+ builds on the knowledge acquired in past European R&D projects as well as existing HTGR designs, such as GT-MHR, MHTGR, and SC-HTGR. In relation to GEMINI+ core configuration, some adjustments have been made. The key modifications are described in this chapter. The current (January 2022) state of the design is presented. It is termed the reference configuration, and it is a starting point for further core optimization. This configuration is a result of internal analysis and the external project participant input (AGH University of Science and Technology [21]).

3.1. Research Reactor Design Philosophy

The TeResa reactor is a pre-concept of a prismatic HTGR with 40 MW_{th} thermal power. It is a helium-cooled, graphite-moderated reactor with a thermal neutron spectrum. The TeResa design also anticipated the need to provide steam, which in the future can be used for a wide variety of applications at the NCBJ's site, including district heating and electricity generation, and in the next step, possibly for hydrogen production or other applications. The pre-conceptual configuration of the TeResa plant is presented in Figure 1. Some important design data of the TeResa plant are given in Table 3. The thermal energy generated in the core is removed by a downward flow of helium coolant, which is heated up from 325 °C to an average value of 750 °C. The hot coolant is transported from the reactor vessel outlet via the coaxial duct to the steam generator.

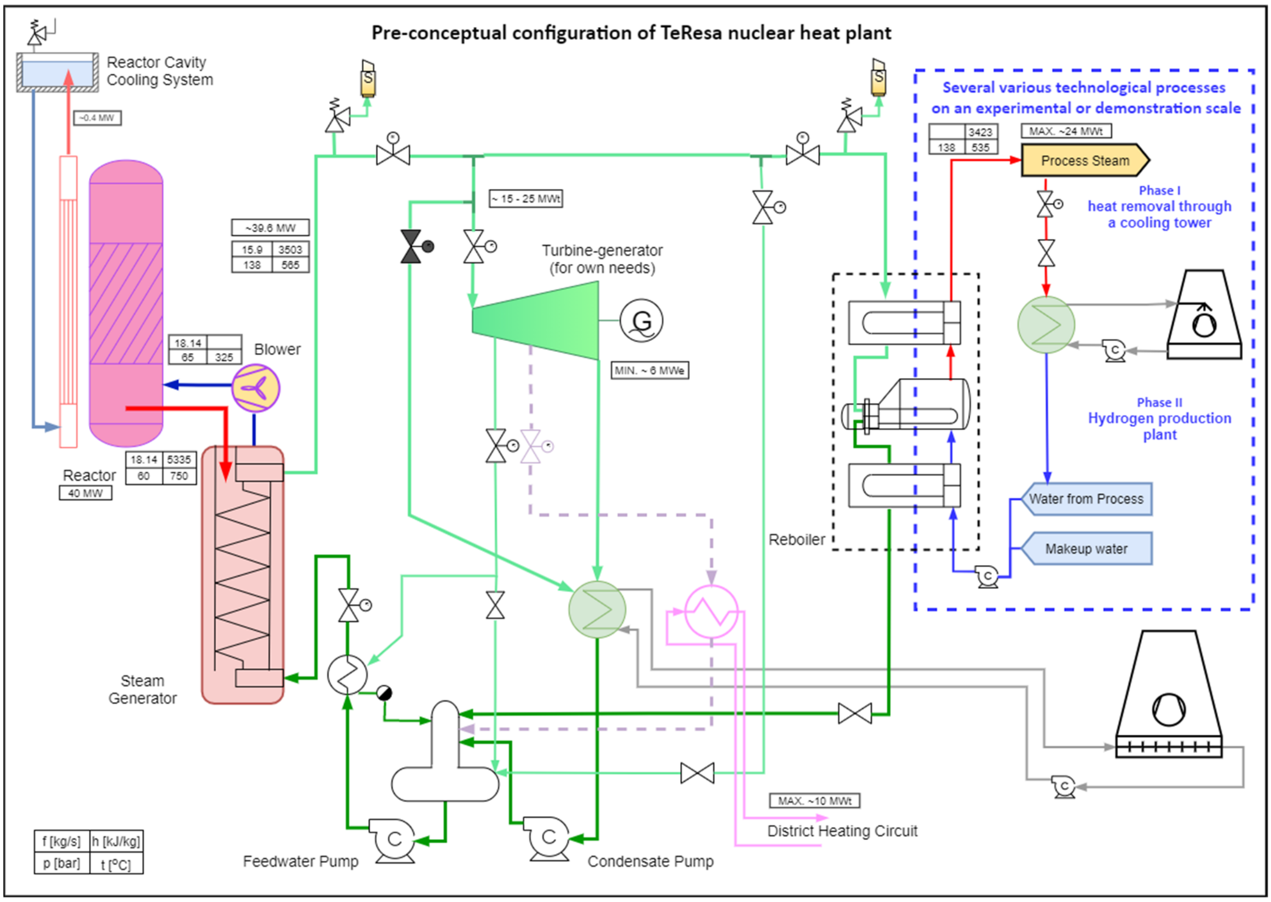


Figure 1. The TeResa facility pre-conceptual design [22].

The helium flow is induced by a helium circulator located at the top of the steam generator. The helium coolant pressure at circulator discharge is 6 MPa. In the steam generator, the heat is transferred to the water/steam cycle. The helium coolant parameters allow for the production of steam with a temperature of 540 °C and pressure of 13.8 MPa.

The TeResa reactor core uses TRISO fuel, i.e., a particulate fuel with ceramic multi-layer coatings surrounding a UO_2 kernel. A single TRISO kernel has a diameter of 500 µm and is covered by subsequent layers of a porous carbon buffer, inner pyrolytic carbon, silicon carbide, and outer pyrolytic carbon. The outer diameter of coated fuel particle is 920 µm. A detailed coated fuel particle specification is presented in Table 4. TRISO particles are randomly dispersed in a cylindrical graphite matrix (fuel compact, 1.245 cm in diameter and 5 cm in height) with a packing fraction of 15%. The fissile fuel kernels are enriched in 12% U-235. For the TeResa reference configuration, a uniform enrichment over the core (one-zone) and once-through fuel cycle is considered.

The active core consists of 31 fuel columns arranged on a uniform triangular pitch and assembled as three rings around central fuel column with a nominal 2 mm gap between each. The fuel column comprises a stack of six fuel blocks. A single fuel block is a hexagonal prism of 36 cm across the flats and 80 cm in height. There are two types of fuel blocks, standard blocks (fully fueled) and control blocks (with control rod channels). The active core is surrounded by two rings of the replaceable side reflector and the permanent side reflector. There are top and bottom replaceable reflector structures (graphite blocks) above and below the active core. A metallic core barrel surrounds the periphery of the side permanent reflector, the outermost structure is the reactor pressure vessel. The active core arrangement and reflector structures are shown in Figure 2.

Table 3. General TeResa plant data—reference configuration, design values.

General Information		
	Unit	**Value**
Reactor thermal output (gross thermal power)	MW_{th}	40
HTGR type	-	prismatic, block-type
Graphite block type	-	similar to GEMINI+
Graphite block height	cm	80
Graphite block hexagon flat-to-flat distance	cm	36
Graphite block material (fuel and reflector blocks)	-	NBG-17
Fuel	-	TRISO, 12% enriched UO_2
Active core height	cm	480
Active core effective diameter	cm	212
RPV outer radius	cm	224.4
RPV material	-	SA508
Primary side		
Coolant type	-	helium
Coolant flow direction	-	downward flow pattern
Helium mass flow rate (at 100% power)	kg/s	18.14
Primary system pressure	MPa	6.0
Reactor vessel inlet coolant temperature	°C	325
Reactor vessel outlet coolant temperature	°C	750
Number of cooling loops	-	1
Steam generator type		once-through, helically coiled bundles
Secondary side		
Secondary side coolant	-	water
Main steam pressure (at Steam Generator (SG) outlet)	MPa	13.8
Main steam temperature (at SG outlet)	°C	540
Main steam mass flow rate (at 100% power)	kg/s	15.9
Feed water pressure (at SG inlet)	MPa	13.97
Feed water inlet temperature (at SG inlet)	°C	210
Feed water mass flow rate (at 100% power)	kg/s	15.9

Table 4. Coated fuel particle (CFP) specification.

CFP Layer	**Material**	**Density [g/cm^3]**	**Outer Radius [μm]**
Fuel kernel	uranium dioxide	10.65	250
Buffer layer	porous carbon	1.05	345
Inner PyC layer	pyrolytic carbon	1.90	385
SiC layer	silicon carbide	3.18	420
Outer PyC layer	pyrolytic carbon	1.90	460

In general, fuel blocks contain a triangular array (pitch 1.88 cm) of fuel holes (drilled blind from the top face of block) and coolant channels (through the block). Standard fuel block contains 108 cooling channels (⌀1.6 cm) and 216 fuel holes (⌀1.27 cm), whereas control block has 89 and 174 cooling channels and fuel holes, respectively. A regular fuel hole is filled with a stack of 15 fuel compacts. Some of fuel holes are filed with burnable poison (BP) rods, depending on the fuel block position. For the current reference design, Europium oxide, Eu_2O_3, was selected as a burnable poison. The material composition of a BP rod is uniform over the core, however, the number of BP rods per fuel blocks varies. A regular block contains one or six BP rods (one for the peripheral core ring), whilst the control block has four BP rods. A control block contains a single channel of 13 cm in diameter for the control rod.

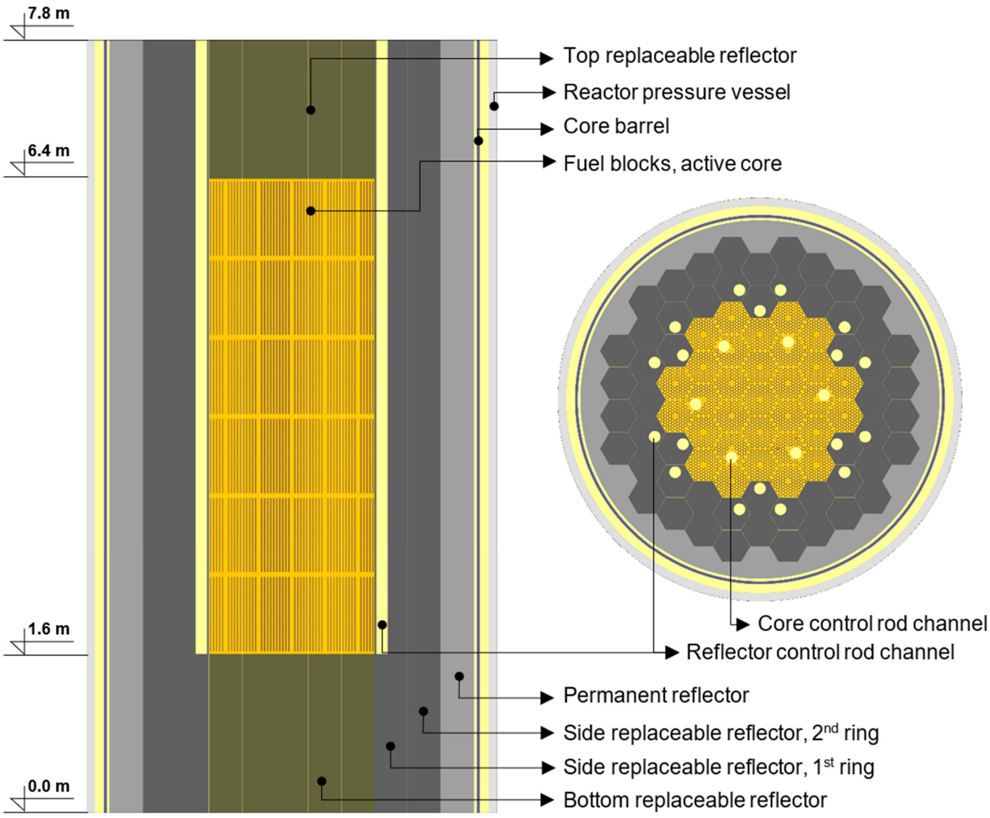


Figure 2. TeResa core configuration, Serpent model visualization.

The reactivity control system of the TeResa reactor pre-concept comprises a control rod system (CRS) and a reserve shutdown system (RSS). These are two independent and diverse means to control reactor power. Both reactivity control systems are safety-classified systems. At this stage of core design, the control rod position pattern was similar to that of the final GEMINI+ reactor configuration [14]. There are 18 rod channels in the first ring of the side replaceable reflector (with six clusters of three rods each) and six rod channels in the core. The pattern of CRS and RSS placed in the reflector area is designed in such a way that is compatible with the needs of fuel-handling operations. Furthermore, this configuration presents the possibility to use the central column as an irradiation column, i.e., for material test applications or production of radiopharmaceuticals. The proposed arrangement permits the use of the same penetrations at the reactor top vessel head for the fuel-handling machine operations. The control rods are inserted into the core and the replaceable reflector vertically, from the top. They are moved individually by dedicated control-rod drive mechanisms. As in GEMINI+, the control rod system (CRS) uses boron carbide (B_4C) absorbers dispersed in a graphite matrix of annular shape. Absorbers are enclosed in canisters for structural support. Canisters are assembled in a stack to form an active part of the control rod. A reserve shutdown system (RSS) is actuated if the CRS becomes inoperable. The RSS uses boron carbide pellets (small spheres or rounded cylinders) which can be dropped into channels in the core. The RSS B_4C pellets are housed in hoppers above the core. In the current core configuration, the RSS system uses the same channels as CRS in the active core (6 channels), as is the case in the final GEMINI+ concept [14].

3.2. Current Safety Evaluation Status

In the current state of the project, the main focus is completing a partial and preliminary safety analysis. The neutronic and thermal-hydraulic calculations are performed using computational tools, which are summarized in Table 5.

Table 5. Tools used in the partial safety analysis in thermal-hydraulic and neutronic calculations.

Field of Study	Code Name	Outline
Neutronics	Serpent 2 [23]	Serpent is a continuous-energy multi-purpose three-dimensional Monte Carlo particle transport code. It is in development at VTT Technical Research Centre of Finland since 2004.
	MVP [24]	MVP (JAEA, Japan) is a general-purpose Monte Carlo code that performs neutron and photon three-dimensional transport calculations using the continuous energy method.
	MCB [25]	MCB—Monte Carlo continuous energy burnup code is a general-purpose code used to calculate a nuclide density time evolution, including burnup and decay. Internally, MCB comprises MCNP code, which is used for transport calculations, and is coupled with thermal-hydraulic code POKE (thermohydraulic software)(Gulf General Atomic Incorporated, USA).
Thermal hydraulics	MELCOR 2.2 [26]	MELCOR is a fully integrated, engineering-level computer code developed by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, that models the progression of severe accidents in nuclear power plants.
	CATHARE2 [27]	CATHARE (Code for Analysis of Thermal hydraulics during an Accident of Reactor and safety Evaluation) is a two-phase thermal-hydraulic system used in pressurized water reactor safety analyses, the verification post-accidental operating procedures, and in research and development.
	ANSYS Fluent 2020 R1 [28]	Ansys FLUENT software contains the broad physical modeling capabilities needed to model flow, turbulence, heat transfer, and reactions for industrial applications.

This project was developed in close cooperation between the research fields (Figure 3). An iterative process of the reactor design was accomplished on the basis of data and links between the neutronic, the thermal-hydraulic deterministic studies, and the probabilistic safety analysis. All of the listed fields are necessary areas of analyses for the preliminary reactor safety considerations.

The neutronic design of the reactor core focused on the core optimization process [21]—with respect to the fuel cycle length, fuel enrichment, placement and composition of the burnable poison rods in the core. The results of the calculations performed in the Serpent 2 and MCB codes included the TeResa core power profiles during the fuel cycle, especially at the beginning of life (BOL) and at the end of life (EOL), its core inventory for specific fission products created during operation and other operational safety parameters (temperature reactivity coefficients and point kinetics related data—e.g., delayed neutrons fraction). All of these data served as input data for the thermal-hydraulic analyses of the TeResa reactor system—primary and secondary loops with the use of MELCOR 2.2 and CATHARE2 codes. The steady state (SS) and transient (typical DLOFC and PLOFC, water ingress incidents) thermal evaluations were performed and an anticipated plant response was assessed. The temperature distribution in the reactor core during steady state and the maximal fuel temperatures in the core during accidents were of prime interest, due to their influence on the neutronic core performance and possible release of fission products under DBA conditions, respectively. The heat removal by the RCCS system was analyzed both by system codes (MELCOR 2.2) during transient progression and by the ANSYS Fluent computational fluid dynamics (CFD) simulations in the stationary states of accident conditions for specific boundary conditions (BCs). During the project, an in depth analysis of the potential faults of safety and protection systems was performed by presenting the appropriate fault trees for the Reactor Protection System (RPS) and primary and secondary engineering measures (e.g., systems of valves), showing a decrease in the accident impact on the reactor system. A preliminary probabilistic studies (in terms of event trees) was performed for the selected Postulated Initiating Events (PIEs) which were chosen carefully as the most representative and the most challenging for the HTGR plant. SAPHIRE 8.1.6 software was applied [29]. The evaluated plant states for the event trees of the Probabilistic Safety Analysis (PSA) analysis derived from the thermal-hydraulic calculations performed in the MELCOR 2.2 code. The resulting calculations will be used for the demonstration of the safety evaluation methodology and developed process of information exchange for the specific analytical areas for the demonstration process of licensing for the future construction of the future demonstrator reactor.

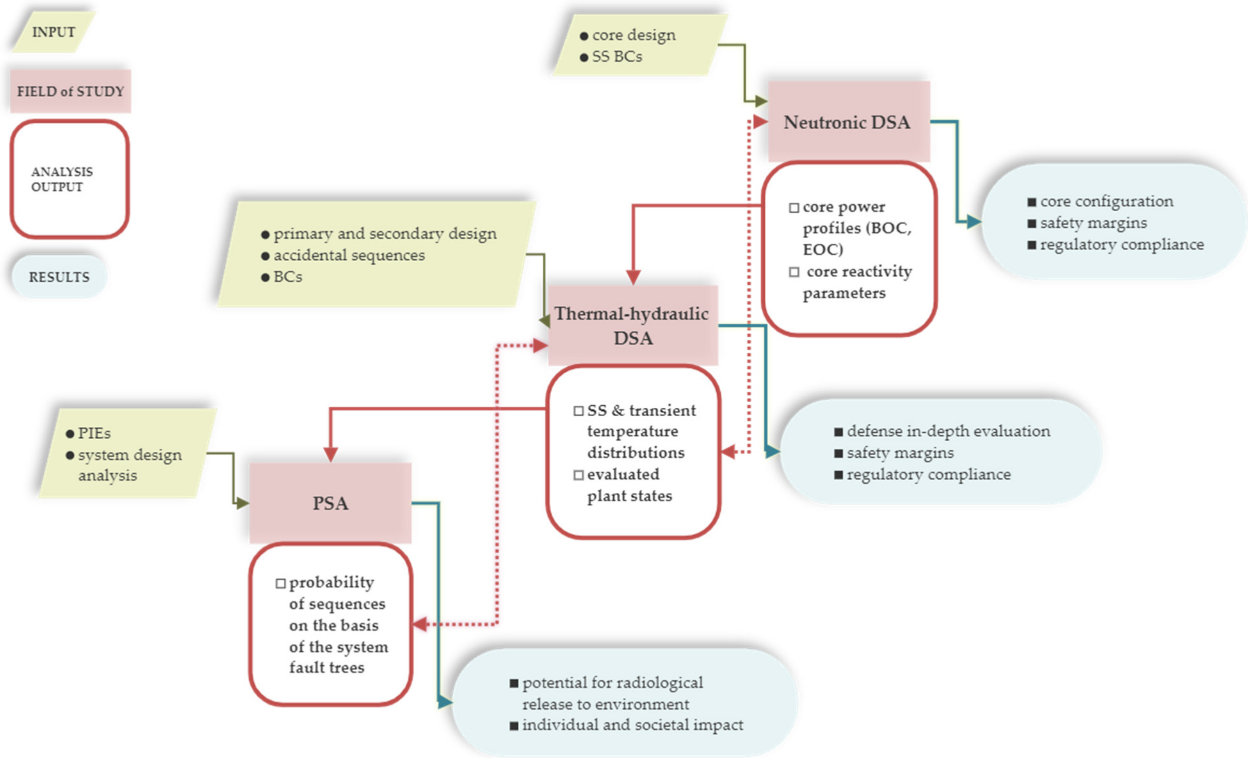


Figure 3. Flow diagram of the GOSPOSTRATEG-HTR safety analysis methodology in the Phase B Stage 5. DSA – Deterministic Safety Analysis. PSA – Probabilistic Safety Analysis.

Partial results coming from the analyses performed under the GOSPOSTRATEG-HTR project are presented in the next section.

3.3. Exemple Results of TeResa Analysis

The synthesis of the resulting calculations completed under the process methodology, presented in Figure 3, and the development of computational models created using conceptual design data from Tables 3 and 4, leads to a reference project of the TeResa research reactor. The presented results are sample determinants of the compliance of the developed TeResa design to the safety criteria of interest and should be treated as examples of the possible application of the developed methodology for the purposes of performing a full safety assessment for the HTGR.

In the field of coupled neutronic and thermal-hydraulic considerations, detailed cycle calculations with the MCB and POKE code [21] and the build model, on the reference fuel and BP design and control rods movement, were performed, which were the focus of the preliminary safety study. The calculations preconditioned reference fuel characteristics (Table 3) and led to cycle-optimization calculations, resulting in the estimation of a single fuel loading cycle of 3 years, which was one of the pre-established criteria to be met. As a result, in Figure 4, the comparison of the Beginning of Cycle (BOC) axial power profile normalized to the core height of the GEMINI+ and TeResa cores is presented (taking into account the thermal response in the neutronic calculations). Although the concept of the reactor design for the TeResa core was proposed to be similar to the GEMINI+, the additional cycle optimization calculations (burnable poison rods configuration able to ensure critical core state, with minimal control rod usage) influenced the shape of the power profile presented, which is more varied across its height. At the BOC, the core maximal power for the TeResa reactor was found to be closer to the bottom reflector component, due to the insertion of the control rod from the top by the Control Rod Drive Mechanism (CRDM). This reference BOC power axial profile was input data for the thermal hydraulic calculations performed for a variety of accident scenarios.

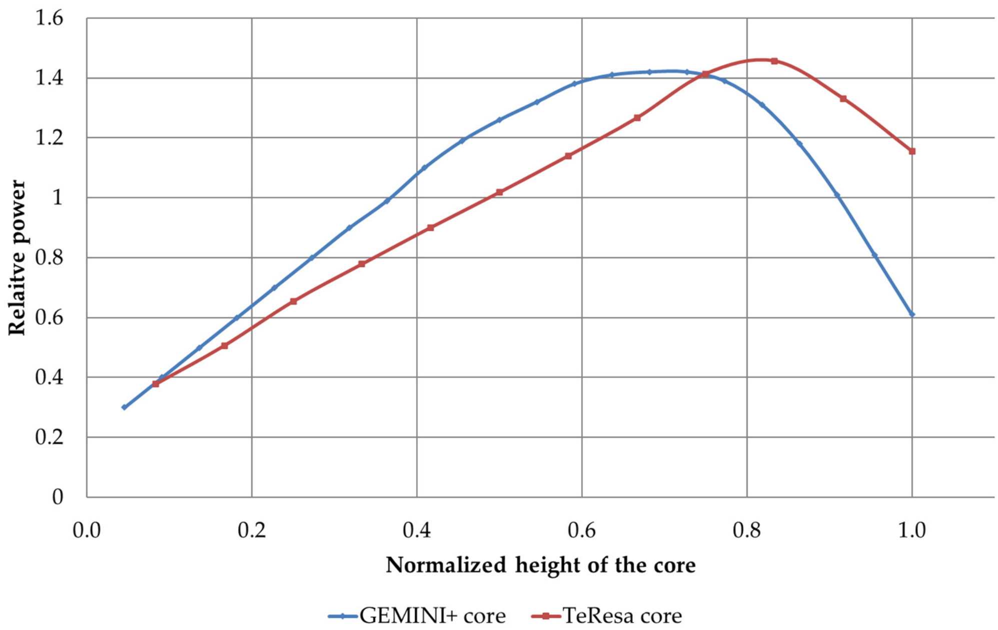


Figure 4. The comparison of the axial power distribution for GEMNI+ and TeResa core.

Part of the work was focused on the evaluation of the normal operation emissions and of the potential research reactor and their radiological consequences. The example methodology applied for the evaluation of the releases was based on [30] and was constructed upon the use of simple release coefficients and the consideration of possible release paths (for example. reactor building ventilation system). The fuel release rates and other factors related to the transport of radioactive isotopes in the graphite matrix and coolant were selected from [1,2,30,31]. A summary of the normal operation releases can be found in Table 6. Based on the operational and computational experience of other HTGR reactors (AVR, Peach Bottom and HTR-10), the values of the releases of tritium (H_3) [3,32], carbon (C_{14}) [33] and argon (Ar_{41}) [4,34] were estimated to be 4.09 x10^8 [Bq/y], 6.57 $\times$ 10^{12} [Bq/y] and 1.0 $\times 10^7$ [Bq/y], respectively.

The results presented are given for the ventilation system of the TeResa reactor with a filtration system adopted from the MARIA reactor [35]. The filtering system has a significant impact on the values of isotope releases from the iodine group and long-lived solid isotopes removed from the reactor building and helium purification system—the filtering coefficients were adopted to be 0.04 for iodine and 0.01 for other elements, except for noble gases and C_{14} carbon (for which no filtering was assumed). During normal operation, the isotopes contributing to the highest release values for the TeResa reactor were isotopes from the group of noble gases and iodine.

Table 6. Estimated normal operation releases for the TeResa reactor core ($A_{i,TOT}$—total core activity, $A_{i,TeResa,f}$—TeResa estimated releases to the environment).

Isotope	$A_{i,TOT}$ [Bq]	$A_{i,TeResa,f}$ [Bq/y]
H-3	2.88×10^{12}	4.09×10^{8}
Kr-83m	2.46×10^{16}	5.32×10^{10}
Kr-85	1.48×10^{15}	1.27×10^{6}
Kr-85m	6.12×10^{16}	1.32×10^{11}
Kr-87	1.12×10^{17}	2.42×10^{11}
Kr-88	1.50×10^{17}	3.23×10^{11}
Xe-131m	1.86×10^{15}	4.03×10^{9}
Xe-133	3.70×10^{17}	7.98×10^{11}
Xe-133m	1.11×10^{16}	2.39×10^{10}
Xe-135	7.73×10^{16}	1.68×10^{11}
Xe-135m	7.47×10^{16}	1.61×10^{11}
I-131	1.71×10^{17}	1.47×10^{8}
I-132	2.51×10^{17}	2.17×10^{8}
I-133	3.68×10^{17}	3.17×10^{8}
I-134	4.18×10^{17}	3.63×10^{8}
I-135	3.46×10^{17}	2.99×10^{8}
Rb-88	1.51×10^{17}	3.27×10^{8}
Sr-89	2.10×10^{17}	9.06×10^{4}
Sr-90	9.84×10^{15}	4.25×10^{3}
Cs-134	1.01×10^{16}	4.36×10^{2}
Cs-137	1.17×10^{16}	5.05×10^{2}
Ag-110m	1.52×10^{14}	6.55×10^{5}
Ar-41	-	6.57×10^{12}
C-14	-	1.02×10^{7}

The values of releases with the use of filtration systems on an annual basis are below the release limits used in the operation of the MARIA reactor [35]. They are as follows: 1.19×10^{12} [Bq/y] from noble gases, 6.57×10^{7} [Bq/y] from argon—Ar_{41}, that is 0.867% of the total release limit for noble gases and argon, and 1.39×10^{9} [Bq/y] for iodine isotopes, i.e., 27% of the release limit. The iodine release level is anticipated to be overestimated due to conservative assumptions about the release of iodine from the fuel during normal operation. The accidental releases are calculated by more sophisticated software—MELCOR 2.2 code.

The radionuclide releases from the TRISO fuel were simulated for the DLOFC with MELCOR 2.2. Exemplary results are discussed in this section. Most of the releases-related setup was based on the description of the Sandia National Laboratories efforts for Pebble Bed Modular Reactor (PBMR), PBMR-400 core design [36]. The initial failed fraction of particles was assumed to be 10^{-5} and a temperature-dependent fail curve for AVR reactor was used, which is conservative and default for MELCOR. Neither contamination, nor SiC failures were assumed. Five radionuclide groups were studied, namely XE (Xenon), CS (Cesium), BA (Barium/Strontium), I (Iodine), AG (Silver). In the model, they were represented by five radiologically important isotopes: Cs-137, I-131, Xe-135, Sr-90 and Ag-110.

Obtained releases expressed in terms of isotopic inventory fraction releases are presented in Figure 5. The TRISO fuel releases are driven by the fuel temperature transient which controls diffusion process in the microsphere layers and fuel failures. Results show that there is an initial rapid release for all studied radionuclide groups due to initial fuel failures. For iodine and xenon groups, some limited releases were observed after initial blowdown. It is because temperatures were too low to enhance diffusion. In the case of silver, cesium, and strontium, the releases increased with a rise in temperature, up to the peak temperature point (about 25,000 s, see Figure 6 a). After the peak, strontium and silver releases terminated, but cesium release continued at a decreasing rate till the end of the simulations at 250,000 sec (Figure 5).

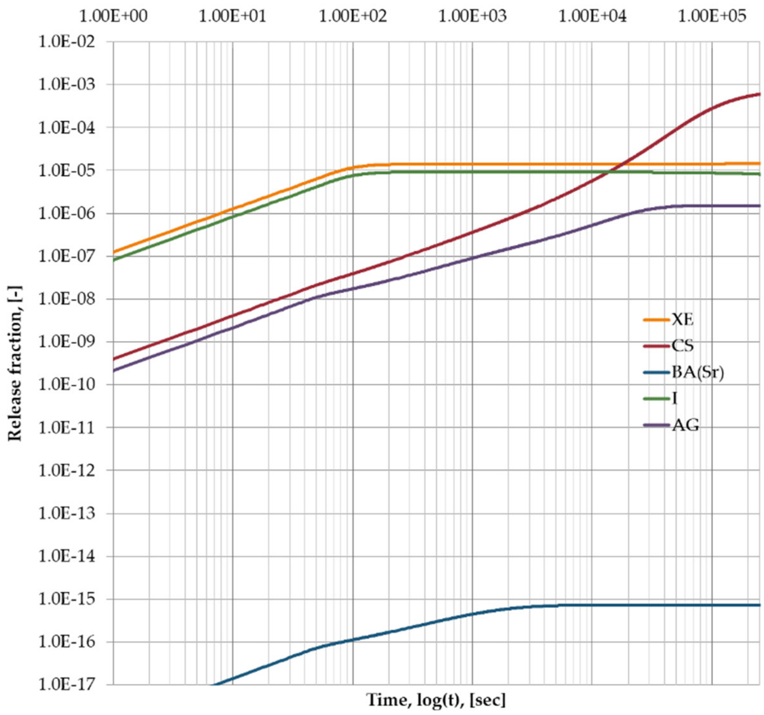
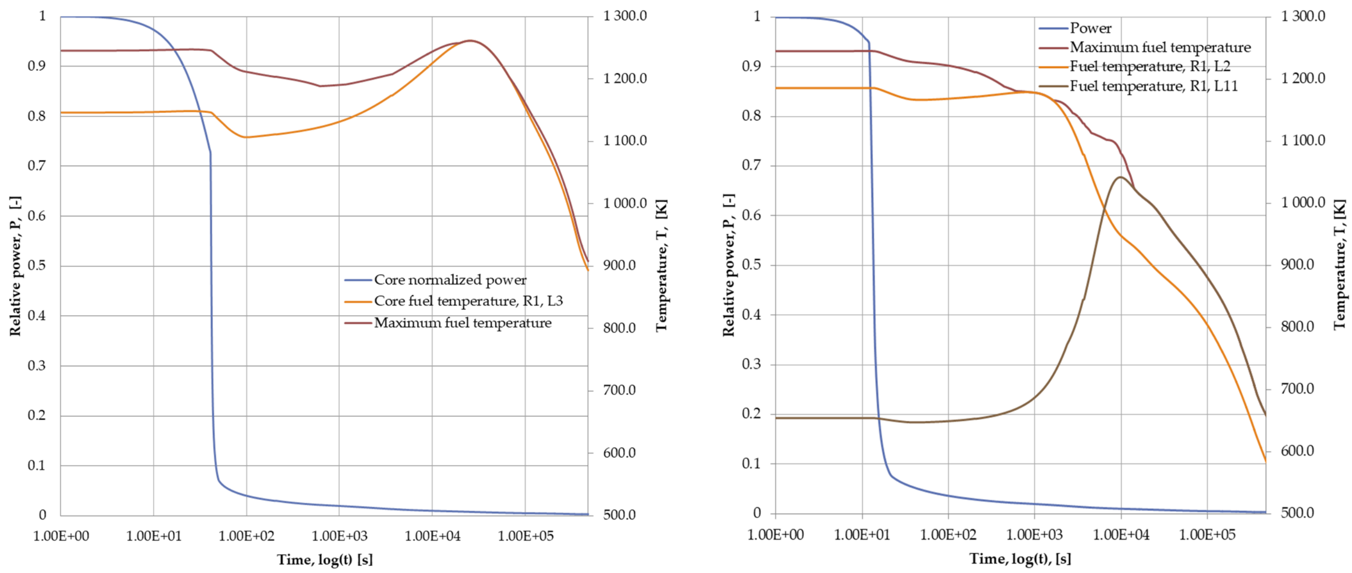


Figure 5. TRISO fuel release fraction of the core inventory for selected radionuclide groups. MELCOR2.2 results for DLOFC accident with inventory at EOC core state.



(**a**) (**b**)

Figure 6. DLOFC (**a**) and PLOFC (**b**) accident core normalized power (P, [−]) and fuel maximum temperatures (T_f, [K]).

For TeResa design, the thermal hydraulic response for DLOFC is studied in the subsequent paragraphs. The maximum fuel temperature is relatively low, and it is only slightly higher than nominal operation temperature (see Figure 6a). This is mainly due to the conservative design of the research reactor. However, releases are present because temperatures in some parts of the core are higher by as much as 200 K in comparison to the nominal operation temperatures. This difference drives the diffusion of cesium, silver, and strontium. The largest observed fractional release from the fuel was for cesium, and it was ~0.06% of the initial inventory.

The thermal-hydraulic calculations were organized in the set of analyses related to the specific scenarios. The prepared MELCOR 2.2 model of the primary and secondary circuits of the TeResa reactor system was qualified using the steady-state run (based on the methodology found in [37]) and the main system parameters were compared to the assumed values (partially presented in Table 3). The summary is presented in Table 7, showing very good agreement of designed and simulated values for the primary and secondary circuits parameters. The main discrepancies found in the results of the thermal-hydraulic simulations are related to the code modelling itself (active and bypass flow definition in the core region [26]), and imprecise primary–secondary heat exchanger preliminary design (scaled-down from the GEMINI Plus project [38]). An overall agreement of the steady state model's calculated and assumed operational parameters is acceptable well below 4% relative error value.

Table 7. MELCOR 2.2 code model qualification table at "steady-state" level.

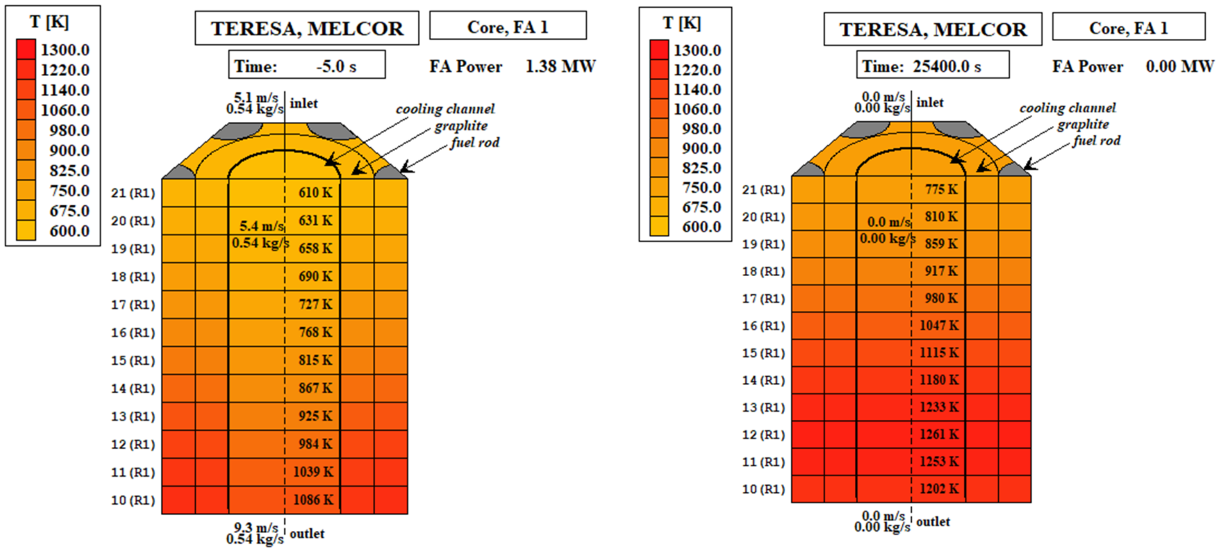
Parameters	Design Value	Simulation MELCOR 2.2	Relative Error
Reactor power (MW_{th})	40	40	0.000%
Helium pressure of primary loop (MPa)	6	6	0.000%
Helium mass flow rate (kg/s)	18.14368	18.1437	0.000%
Helium RPV Inlet temperature (K)	598	598	0.000%
Helium RPV Outlet temperature (K)	1023	1015	−0.782%
Main feed-water temperature (K)	483	482.5	0.000%
Main steam temperature (K)	813	794	−3.519%
Main steam pressure (MPa)	13.8	13.8	0.000%
Feed-water flow rate for steam generator (kg/s)	15.9	15.95	0.314%

The thermal-hydraulic analyses executed on the developed MELCOR 2.2 model covered investigations of the potential DBAs—the Depressurized Loss of Forced Cooling (DLOFC) accident and Pressurized Loss of Forced Cooling (PLOFC), which were performed with the Best Estimate (BE) assumptions. Following the GEMINI Plus project's thermal-hydraulic calculation campaign, the assumptions taken in the calculations are presented in [19]. In the case of TeResa pre-conceptual design, the results of the accidents simulations were well below the temperature limit for the TRISO fuel-accelerated fission product release rate [39]. Two analyzed cases are shown here as an example of thermal-hydraulic calculations. Both are summarized in Table 8, which shows the maximum fuel temperature reached during the accident course.

Table 8. Core maximal temperatures for the TeResa core at BOL state for DLOFC and PLOFC accident scenarios.

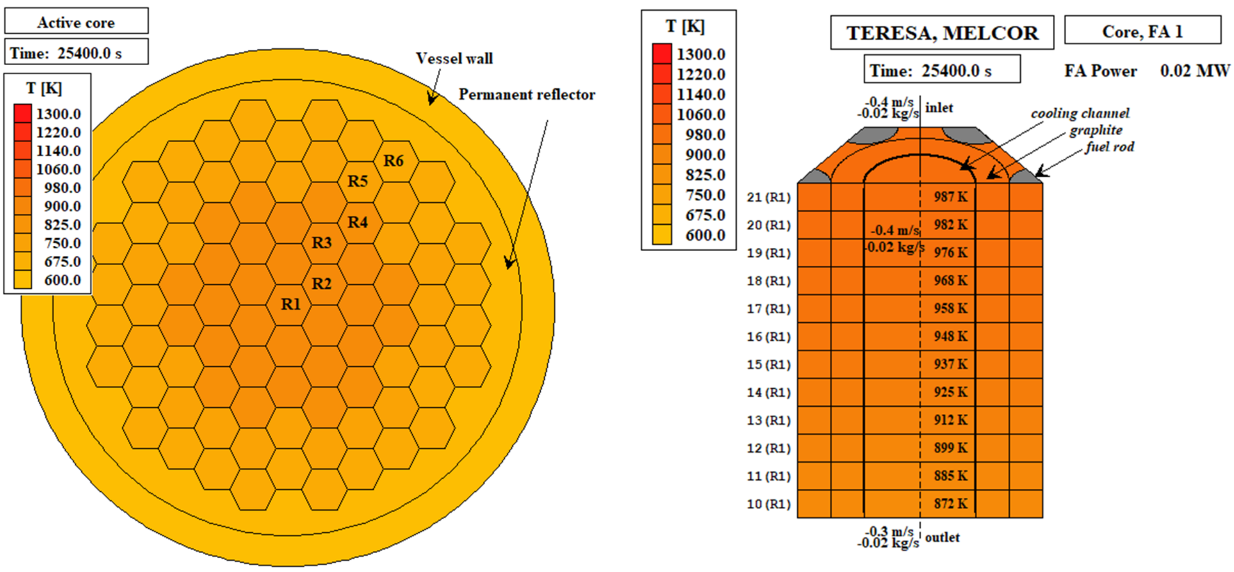
Accidental Scenario	Maximum Fuel Temperature	Active Core Axial Position from the Bottom	Active Core Radial Position	Time of Occurrence
-	[K]	[m]	[−]	[s]
DLOFC	1261.0	~1.0	central column	25,400
PLOFC	1050.0	~4.2	central column	10,000

The MELCOR 2.2 simulations demonstrate the expected response of the HTGR system to the specifics of the analyzed scenarios. As anticipated, the most challenging accident course is the DLOFC event, which exposes the core component to the temperatures, exceeding the temperatures of the steady state operation. The fuel blocks with the highest temperature elevations are subject to a 9.65% temperature rise in the DLOFC event and are achieved after the steady temperature increase at around the 7th hour after an accident initiation (Figures 6a and 7). Due to the neutronic optimization of the fuel cycle, the highest relative power at the BOL is found for the fourth core ring (Figure 8a), and it manifests itself in the highest steady state temperatures in those regions. This characteristics changes during the accident course (both for DLOFC and PLOFC events), when the fission power is dramatically decreased as a results of the Reactor Protection System (RPS) signal—Figure 6 (a and b blue curve). The SCRAM signals are shifted in time, due to the different times at which monitored parameters reached the setpoint level for SCRAM initiation.



(**a**) (**b**)

Figure 7. Temperature in the cooling channel of central block of the core: (**a**) steady state (5 s before accident initialization); (**b**) DLOFC conditions, T_{max} at t = 25,400 s (BOL case).



(**a**) (**b**)

Figure 8. Results of PLOFC scenario, at t = 25 400 s: (**a**) temperature of the fuel at the bottom of the core; (**b**) temperature of cooling channels in the central block (BOL case).

Comparing the two scenarios the PLOFC is less challenging from the point of view of the fuel exposure to unfavorable conditions, as expected. The discrepancy between the scenarios lies in the presence of the cooling fluid in the system, which for the pressurized scenario is directed towards the top of the Reactor Pressure Vessel. This behavior is mapped in the reversal of the temperatures, which migrates from the bottom of the core to the top in the PLOFC scenario (Figures 6b and 8—R1, L11 and R1, L2 fuel temperatures).

4. Final Remarks

The GOSPOSTRATEG-HTR is a project that aims to prepare a novel pre-conceptual design of the research HTGR and provide legal, organizational and technical implements for the utilization of HTGR technology. By conducting this project, Poland will progress towards more sustainable energy sources, with its transformation contingent to the well-developed implementation of HTGR technology. The advantages of the project are multiple, having significant outcomes in the areas of material science and technology, legal and economic analyses of the HTGR implementation, and an established methodology of the safety assessment preparation for the purpose of licensing process completion.

During the course of the GOSPOSTRATEG-HTR project, the main focus was on the development of a pre-conceptual design for a research reactor with a specific core configuration. All of the efforts in terms of the performance of the pre-concept assessment with the use of multi-disciplinary codes also concentrated on verification, given that the established acceptance criteria are fulfilled. Neutronic considerations and a series of performed simulations with Serpent 2 and MCB codes, allowed us to establish the core configuration with a fuel cycle of 3 years, moderate peaking factors, and an adequate excess of reactivity. The thermal-hydraulic calculations using MELCOR 2.2 code focused on the temperature distribution in the reactor core during steady state and the maximal fuel temperatures in the core during accidents. The steady state and transient (DLOFC, PLOFC, and water ingress incidents) thermal evaluations were performed and the plant response was assessed. The results of the accident simulations were promising and the maximal fuel temperatures were well below the acceptance criterium for the TRISO fuel, that is below 1600 °C. Following the thermal-hydraulic evaluation of the response of the GOSPOSTRATEG-HTR reactor system, release calculations for the DBA conditions were conducted. The calculations with the dedicated MELCOR 2.2 HTGR release packages illustrated a minimal increase in the releases of the selected isotope groups of interest (XE (Xenon), CS (Cesium), BA (Barium/Strontium), I (Iodine), AG (Silver)) during the DLOFC event, due to relatively low temperature changes during the accident. The investigations, in terms of the thermal-hydraulic design of the proposed system, reveal its robustness and compliance with legal regulations in the area of normal operational and DBA release.

The GOSPOSTRATEG-HTR project results in a few key accomplishments in both a scientific and legal framework. Examples include the already developed procedures, material investigations and the novel design concept of the TeResa reactor, which will be further explored in the next project. In May 2021, Poland's Ministry of Education and Science and the National Centre for Nuclear Research (NCBJ) signed an agreement focusing on the next round of high-temperature gas-cooled reactor design work, which will focus on the preparation of the conceptual and basic design of the research HTGR technology demonstrator that will be built at the NCBJ's institute site.

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