

Analysis of UO₂+5% Cr Fuel Performance Under Normal and Accident Operating Reactor Conditions for VBER 300

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To Cite This Chapter:

Arat, M. (2025). Analysis of UO₂ + 5% Cr fuel performance under normal and accident operating reactor conditions for VBER-300. In H. Gokmese, S. Bulbul, & Y. Uzun (Eds.), *Innovative approaches in materials science and applications* (pp. 138–151). ISRES Book Series. ISRES Publishing.

Introduction

One of the most important sectors of the energy that determines the prosperity of a country is energy. The human equivalent of energy today is primarily electricity. The majority of global electricity is produced in various types of power plants, including thermal, natural gas, nuclear, solar, wind, geothermal, and hydroelectric facilities. Ensuring reliable, safe, and economical operation of the global nuclear reactor fleet has consistently been a fundamental objective of the nuclear industry. The continuous advancement of technology-particularly in the areas of innovative materials and nuclear fuels-remains essential to achieving this goal. Over decades, extensive research and operational experience have driven significant technological progress and generated a comprehensive body of data and knowledge regarding the behavior of light water reactors (LWRs) under both normal operating and accident conditions. In the aftermath of the 2011 Great East Japan Earthquake and the subsequent tsunami that led to the Fukushima Daiichi nuclear accident, improving the accident tolerance of light water reactors (LWRs) became a central theme in international research and policy discussions.

The current atomic power industry depends on innovation and has a phenomenal record of safety and functionality. Classic used UO₂ – zirconium combination framework (for example, fuel bar) meets all presentation or wellbeing necessities while keeping thermal power a financially serious clean-energy choice. Accident-tolerant fuel (ATF) advancement is to distinguish elective fuel framework innovations for additional improving the security, efficiency, and financial matters of business atomic power. Any new fuel idea might be thought about in contrast to the current plan, functional, financial, and security requirements to assess the administrative well-being consistency with functional and monetary imperatives. It is necessary to carry out an extensive assessment of the potential effects of ATF concepts across the whole nuclear fuel cycle.

The complex multi-physical science conduct of LWR atomic fuel makes characterizing explicit material or plan upgrades troublesome. Subsequently, building up beneficial execution credits is essential in directing the plan and advancement of fills and cladding

with improved accident resistance. ATF plans would persevere through a serious accident situation in the reactor center for a more drawn-out time of fuel framework while keeping up with or further developing fuel execution during typical activities. Key prerequisites for cutting-edge fills identify with atomic fuel execution, cladding execution, and adherence to framework imperatives.

This document outlines a unified set of global metrics, standardized tests, and representative severe accident scenarios for assessing ATF concepts across PWR, BWR, and VVER reactor types. Derived from a comprehensive U.S. assessment framework (Bragg-Sitton et al., 2016) and adapted for international use, the proposed methodology evaluates each concept's capability to achieve performance and safety objectives relative to the conventional UO₂–zirconium system. It also summarizes the current development status of fuel performance and severe accident analysis codes to inform future model enhancements, along with the internationally agreed accident scenarios for ATF evaluation (Agency and OECD, 2018; Bragg-Sitton et al., 2016).

Design Limitations and Performance Objectives of ATF Systems

Enhanced Accident-Tolerant Fuels (ATFs) refer to advanced nuclear fuels designed to endure severe reactor core accidents for a longer duration compared with the standard UO₂–zirconium alloy fuel system, while achieving comparable or superior performance under normal and transient operating conditions. In order to evaluate their compliance with regulatory safety requirements as well as operational and economic constraints, each new fuel must be thoroughly assessed compared with existing economic, safety, design, and operational criteria. The development and implementation of ATF concepts in commercial nuclear reactors are subject to several limitations and constraints, including those imposed by current nuclear fuel development and deployment practices.

- *Backward compatibility*: refers to the capacity of a new fuel design to be utilized within existing and future Light Water Reactors (LWRs) without requiring significant modifications. This includes compatibility with current fuel handling systems, fuel rod and assembly geometries, as well as the ability to operate alongside existing fuel types within the reactor core.
- *Plant operations*: refer to the requirement that new fuel designs should sustain or extend existing plant operating cycles, reactor control system configurations, and reactor power output. Any fuel system proposed for regulatory approval must demonstrate reliable performance not only under normal operating conditions but also during operational transients, i.e., anticipated operational occurrences.
- *Safety*: achieves performance comparable to or exceeding that of existing fuel systems under normal operation.
- *Front end of the nuclear fuel cycle*: complies with equipment, regulatory,

technical, and fuel performance requirements and rules.

- *Back end of the nuclear fuel cycle:* must not adversely affect the fuel's performance during transportation, interim wet or dry storage, or final disposal; should consider usage inside a closed fuel cycle.

To estimate the possibilities of the fuel system being accepted by the industry, the economics of a suggested ATF idea should be analyzed. However, because of high uncertainty in fabrication costs and materials during this research period, evaluating novel materials at early development stages is inherently challenging for ATFs. Therefore, separating economic analyses from technical and safety evaluations is essential for prioritizing fuel concepts. Technical performance results will serve as the basis for subsequent assessments of economic viability, which may differ across regional energy markets. It is important to emphasize that ensuring the economic viability of new fuel concepts is crucial in terms of additional costs (such as fabrication expenses) and potential cost savings achieved through enhanced performance, including higher burn-up capability, extended operating cycles, power uprates, reduced waste generation, or improved safety margins. Depending on the regulatory environment of the specific country, this safety margin and higher performance may be an economic advantage in terms of resource removal or plant equipment, risk reclassification (Lin et al., 2022).

Cladding Materials

Zirconium alloys, optimized through decades of LWR operation, remain the standard cladding materials. For ATF applications, alternative claddings-ranging from ceramics to metals-are being explored, each with unique material and neutronic characteristics. These candidates require systematic evaluation to determine feasibility and identify designs offering the best balance of properties and performance under both normal and accident conditions (Radford, 1979).

The nuclear fuel rod cladding serves two primary functions:

- The nuclear fuel cladding primarily serves to contain the fissile material and fission products within the fuel rod while maintaining compatibility with the nuclear process.
- allow for effective thermal transfer from fuel to coolant.

All ideal qualities and behaviors will not be present in an actual cladding material. As a result, a baseline for evaluating and assessing candidate materials that conforms to the criteria given below must be developed. These measures will assist researchers in making the optimal choice by analyzing the cladding material with the best combination of features and behaviors (Oguma, 1982).

Loss-of-coolant accidents (LOCAs) constitute a well-studied category of reactor incidents

and represent the scenarios for which protective systems and operational procedures are primarily designed. As the term suggests, a LOCA occurs when coolant is lost from the reactor system. Among these, a large-break LOCA (LBLOCA) represents the most severe scenario, characterized by a double-ended rupture (guillotine failure). LBLOCAs are generally defined for break areas exceeding 0.1 m² (Joyce, 2018).

In LOCA, the pressure within the primary circuit drives a rapid expulsion of coolant from the reactor vessel—a phenomenon known as coolant blowdown—accompanied by a swift depressurization of the system. This process may occur within a matter of seconds. In the absence of effective mitigating measures, the reactor core may be deprived of adequate cooling, rendering it susceptible to overheating, while a substantial quantity of vaporized coolant may accumulate within the containment structure. Most remedial strategies for mitigating the consequences of a large-break LOCA (LBLOCA) rely on rapid and continuous delivery of emergency coolant to maintain core coverage and prevent damage. These measures are typically designed to be implemented sequentially, depending on the core's pressurization status during blowdown.

Additional LOCA scenarios have been analyzed. For instance, if the rupture—and consequently the coolant loss rate—is significantly lower than in the LBLOCA case, the resulting events are classified as small-break LOCAs (SBLOCAs), generally associated with break areas smaller than 0.1 m². Although minor leaks capable of causing SBLOCAs are more probable than a large pipe rupture, their consequences can nonetheless be severe, even under slower coolant loss conditions.

Given the necessity of controlling decay heat, many LOCA scenarios are relevant not only to operating reactors but also to freshly shut-down reactors. This is an important consideration for LOCA countermeasures, as most LOCA events involve automatic reactor shutdown triggered by core overpressure, excessive temperature, or low coolant levels. Consequently, by the time LOCA mitigation measures are implemented, the reactor is likely to be safely shut down, and the primary focus shifts to decay heat management. Additional classifications, such as small LOCA (SLOCA) and medium-break LOCA (MBLOCA), are used to describe coolant losses that do not result from major pipe ruptures (Joyce, 2018).

Since light water serves as the primary coolant and moderator in most modern commercial reactors, as well as in the majority of existing operational nuclear plants, LOCAs have been extensively investigated in the context of Light Water Reactor (LWR) design. Nonetheless, LOCA events remain plausible in any reactor design if the fuel is vulnerable to overheating. Contemporary reactor designs often incorporate additional features to mitigate LOCA risks, including reduced piping (to lower the probability of pipe rupture) and enhanced countermeasure systems. Examples include passive natural

circulation cooling, isolation cooling systems capable of sustained operation, auxiliary make-up tanks, and high-pressure injection systems (Joyce, 2018).

Fuel Materials

UO₂, enriched to about 4.0-4.95% ²³⁵U, remains the standard fuel for water-cooled reactors (PWRs, BWRs, and VVERs). ATF development explores both incremental improvements to UO₂ and advanced alternatives with enhanced fission product retention. Fuel performance and accident tolerance are assessed through material properties-thermal, mechanical, chemical, and neutronic-which must be optimized considering their effects on existing fuel systems and irradiation behavior. Due to complex interdependencies among these variables, establishing quantitative target values for multiple attributes is challenging; thus, any advanced fuel design requires assessment of the combined effects of material characteristics to ensure reliable operation within the overall reactor system.

Thermal Properties of Light Water Reactor Fuel

The thermal properties of LWR fuel-such as melting point, conductivity, expansion, and specific heat-govern its overall thermal behavior, which is also affected by fuel density and porosity. These characteristics influence fission product release, fuel swelling, and rod integrity. Fuel design must control swelling and emissions while maintaining acceptable thermal performance under normal conditions before evaluating accident tolerance. Desirable safety-enhancing trends for these properties are outlined below.

- *Melting Point:* A higher melting point is desirable, as it defines fuel performance under AOOs, RIAs, and LOCA conditions. The temperature margin to melting depends primarily on the melting point and thermal conductivity for a given power level. Since alternative fuels may operate at different temperatures, the margin between operating and melting temperatures is more critical than the absolute melting temperature itself.
- *Thermal Conductivity:* High thermal conductivity is desirable, as it lowers the fuel centerline temperature for a given linear power and thermal barriers in the gap and cladding. Reduced centerline temperatures decrease diffusion-driven processes, including fission gas release. Furthermore, lower thermal energy storage at equivalent density and specific heat reduces fuel and cladding temperatures during coolant failures.
- *Specific Heat Capacity:* A lower specific heat capacity is advantageous, as it limits the energy retained in the fuel during accident conditions such as LOCAs or DNB-related transients, thereby helping to reduce peak fuel temperatures in rapid events.
- *Pellet Density and Porosity:* Pellet density influences the initial energy storage

during LOCA events and affects DNB behavior under loss-of-flow conditions. The type and distribution of porosity should be optimized to minimize fission gas release, reduce fuel-water interaction, and enhance overall fuel performance.

- *Coefficient of Thermal Expansion (CTE)*: A low thermal expansion coefficient is desirable, as expansion strongly affects pellet-cladding interaction (PCI) during overpower conditions, as excessive expansion can exacerbate mechanical stresses between the pellet and cladding (Agency, N.E. and OECD, 2018).

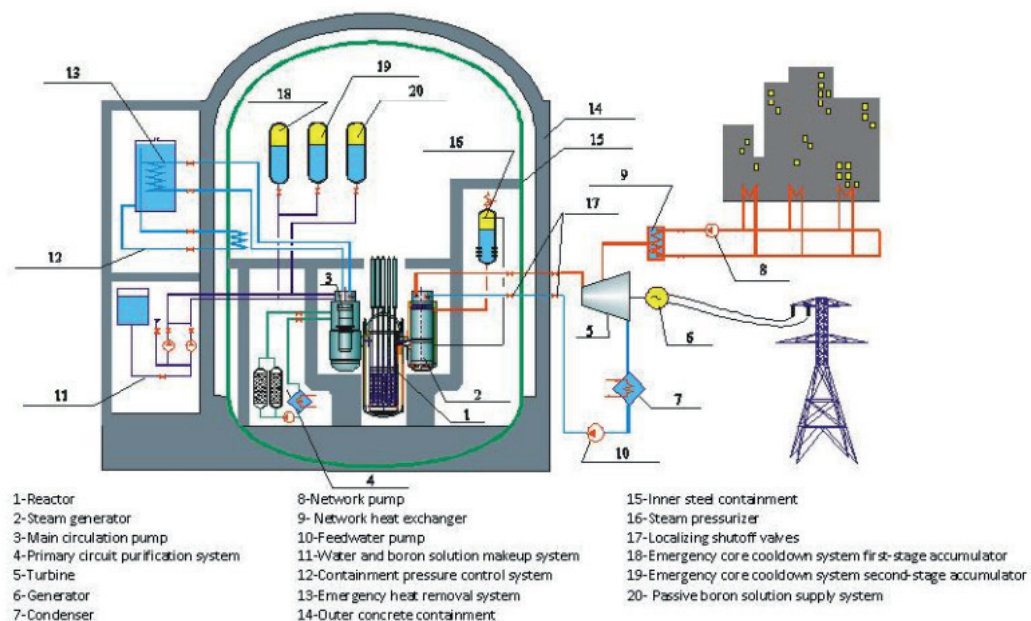
VBER-300

The VBER-300 is a Russian type of reactor with a proposed generating capacity of 325 MWe, intended for deployment in remote areas. Its external containment structure stands 16 meters tall, while the operational section, constructed from transportable modules, has a total weight of 1300 tonnes. Reactor plant of medium and low power VBER on thermal neutrons, designed based on block marine pressurized water reactors, is an improved setup, allowing it to be used as part of nuclear power plants for the production of heat and electricity. Such energy sources are designed to provide energy to regions that are not equipped with a centralized power supply. It's possible to use power units as energy sources as part of complexes for desalination of seawater. The VVER reactor plant implements the mastered technologies of ship block reactors, verified by many years of experience in operation in difficult navigation conditions, as well as the technology and experience of operating VVER-type reactors (Agency, I.A.E., 2006).

A general view of the VBER-300 reactor plant is shown.

Figure 1

Schematic diagram of VBER-300 (Agency, I.A.E., 2006)



The power unit of VBER includes a reactor and steam turbine installations. The generation and receipt of energy is carried out according to the traditional and proven scheme: reactor plant-turbine-generator-power transmission line. For heat transfer, the necessary thermal and mechanical equipment and thermal networks connecting the speaker with the consumer are provided.

The main technical solutions of the VBER project:

- Direct-flow SG (steam generator);
- Sealed MCPs (main circulation pump);
- Active zone of reduced energy intensity;
- An optimal integration of passive and active safety systems.

What sets this project apart is its implementation of diverse, well-established technical solutions derived from decades of experience in both naval reactor construction and VVER-type reactors.

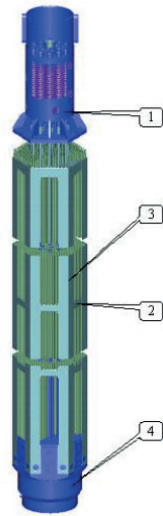
The main advantages of a VVER-type reactor plant include:

- The possibility of using the technology for nuclear power plants.
- Application of proven technologies of marine reactors, technical solutions of VVER reactors;
- Exclusion of the class of accidents of large and medium leaks, because there are no large-diameter pipelines;
- Increased strength and operability under;
- Operational loads, achieved due to the circulation of the primary circuit coolant through the connecting pipes of the main tract according to the “pipe in pipe” scheme;
- Minimum weight and size characteristics associated with the block execution of the reactor, allowing for the reduction of construction volumes, material consumption, and, as a result, achieving minimum capital investment, duration, and cost of installation work;
- The possibility of regulating the power range of the source using only standardized VBER equipment (Dimitriev et al., 2020).

VBER-300 core's the fuel assemblies (FAs) are shroudless and of the TVSA (Advanced shroudless fuel assembly for VVER reactors) type (Fig. 2) (Samojlov et al., 2010).

Figure 2

VBER 300 TVSA FA (Samojlov et al., 2010)



Thermal Calculation of the Reactor Core VBER 300

The input data for the calculation are presented in Table 1.

Table 1

Initial Data

Parameter		Value
Q_R, MWt		917
P_1, MPa		16.3
$T_{\text{inlet}}, ^\circ\text{C}$		292
$T_{\text{outlet}}, ^\circ\text{C}$		327.5
$\Delta T_s, ^\circ\text{C}$		21.4
$\Delta T_1, ^\circ\text{C}$		35.5
Fuel	$d_{\text{fuel}}, \text{mm}$	9.1
	Type of fuel	UO_2
	Cladding Material	Zr110
	$\delta_{\text{thickness of cladding mat.}}$	0.685
	δ_{gap}	0.065
Fuel Assembly	$n_{\text{number of fuel}} n_{\text{absorption element number}} n_{\text{center}}$	312 18 1
	Distance of fuel element, mm	12.75
	The distance between the faces of the fuel assembly, mm	4
	$m = \frac{H_{\text{high of core}}}{D_{\text{diameter of core}}}$	1.2

Table 2 shows the thermophysical parameters of the coolant at the outlet and inlet of the reactor.

Table 2

Thermophysical properties of the coolant

Parameter	Parameter Name, Dimension	Parameter Value
T_s	Saturation temperature, °C	348.9
T_{inlet}	Inlet Coolant Temperature, °C	292
T_{outlet}	Outlet Coolant Temperature, °C	327.5
$T_{average}$	Average water temperature in the core, °C	309.75
h_{inlet}	Enthalpy at the reactor inlet, $\frac{KJ}{kg}$	1294
h_{outlet}	Enthalpy at the reactor outlet, $\frac{KJ}{kg}$	1498
Δh	Enthalpy change, $\frac{KJ}{kg}$	204
ρ_{inlet}	The density of water at the reactor inlet, $\frac{kg}{m^3}$	751.9
ρ_{outlet}	The density of water at the outlet of the reactor, $\frac{kg}{m^3}$	659
$\rho_{average}$	Average density of water in the core, $\frac{kg}{m^3}$	705.45
μ_{inlet}	Dynamic viscosity of the inlet coolant, <i>Pa.s</i>	$9.37 \cdot 10^{-5}$
μ_{outlet}	Dynamic viscosity of the outlet coolant, <i>Pa.s</i>	$7.72 \cdot 10^{-5}$
$\mu_{average}$	Average dynamic viscosity of water in the core, <i>Pa.s</i>	$8.54 \cdot 10^{-5}$
$k_{average}$	Average thermal conductivity of water, $\frac{Watt}{m \cdot K}$	0.541
$Pr_{average}$	The average number of Prandtl of water	0.92

Given in Tables 1 and 2, the required calculations were carried out, and a nuclear fuel simulation was developed using the ANSYS software package.

During the thermal calculation, the core parameters were determined, which are in good agreement with the data (Table 3) for the VBER 300 reactor.

Table 3

Geometric characteristics of the active zone

Parameter	$D_{a,z}, m$	$H_{a,z}, m$	$N_{fuel\ assembly}$	$W_{coolant}, \frac{m}{s}$
Value	2.42	2.9	85	3

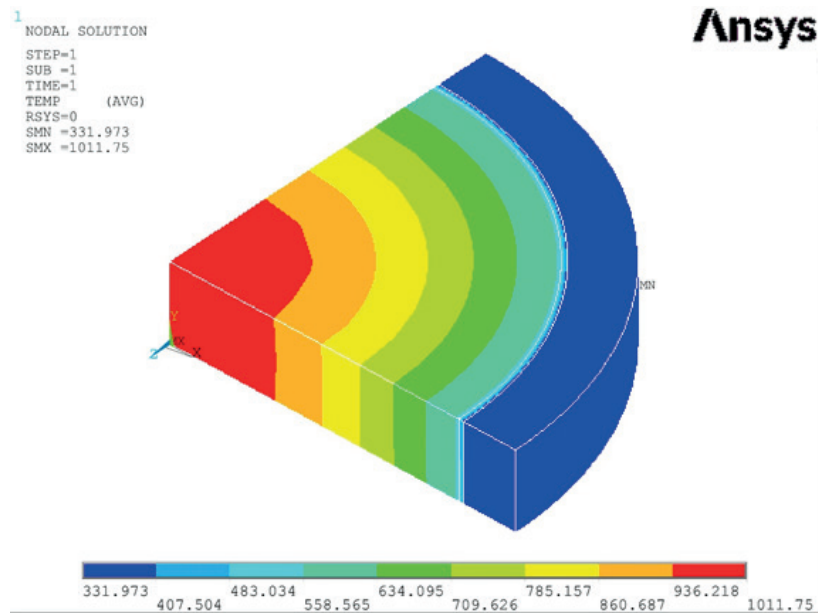
The distribution of temperatures along the height of the active zone makes it possible to find the volumetric heat flux in the fuel pellet $q_v = 570 \frac{MW}{m^3}$.

Using the volumetric heat release, the heat transfer coefficient from the outer surface of the fuel element $\alpha_{convection}^{fuel\ assembly} = 37055 \frac{W}{m^2 \cdot ^\circ C}$ and the coolant temperature near the fuel element wall (assumed to be average) $T_{avarage} = 309,75^\circ C$, the temperature distribution inside the fuel element in the center of the active zone ($z=0$) was obtained in Figure 3.

To reduce the volume of calculations, we neglect the calculations for emergency-resistant fuel and claddings, since the thermal calculation and obtaining the temperature distribution is similar for all types of fuel and claddings (only thermal conductivity changes).

Figure 3

Temperature distribution inside a fuel element with UO₂ and Zr+1%Nb cladding

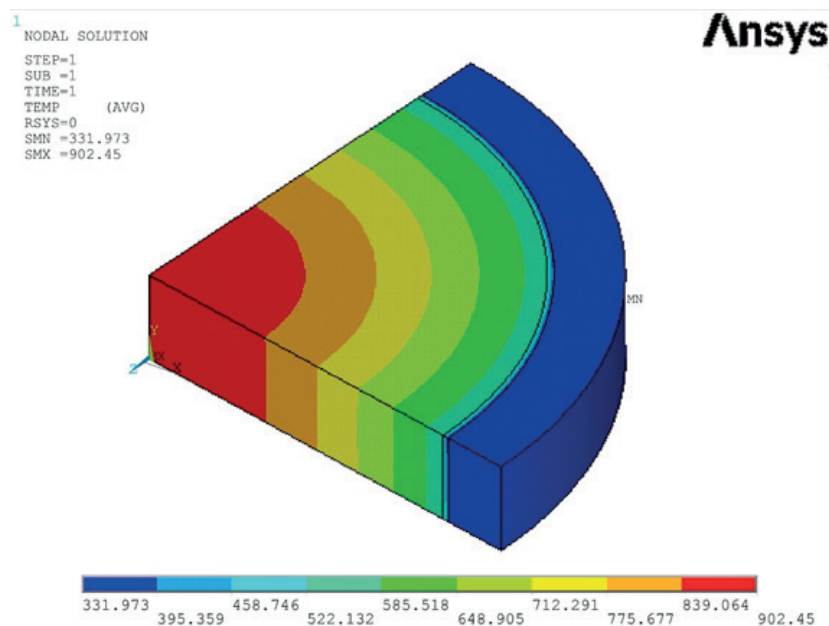


When using classic fuel, the temperature in the center of the pellet was obtained $T_{center} = 1011.75$ °C. On the surface $T_{surface} = 558.5$ °C, the temperature difference between the pellet and the cladding is approximately 150 °C (in the gas gap).

The following shows the temperature distribution with UO₂-5% Cr (CERMET) fuel and a Zr+1%Nb cladding of standard thickness, while maintaining the gas gap between the cladding and the pellet.

Figure 4

Temperature distribution inside a fuel element with UO₂ – 5%Cr (CERMET)



As illustrated in Figure 4, the temperature at the center of the fuel decreased by 109 °C and amounted to 902.45 °C.

At the surface of the fuel pellet, the temperature decreased and amounted to 585.5 °C.

$UO_2 + 5\%Cr$ fuel showed better results compared to UO_2 , which is attributed to its high thermal conductivity.

According to the literature, the melting temperature of UO_2 is approximately 2865 °C (Hausner, 1965). The maximum operating temperature for UO_2 in operation is 1011.75 °C. Since the operating temperature is lower than the melting temperature, we can say that the fuel works safely in the reactor.

The melting point of Zr+1%Nb typically ranges between 1726.85 °C and 1976.85 °C, depending on the oxygen content and alloy. Just above 1726.85 °C, the Zr+1%Nb cladding material can melt, in some cases, drain into the lower core regions (Pöml et al., 2012). The operating temperature of the cladding is 331.973 °C, and it is safe for operation as this operating temperature is lower than the melting temperature (1726.85 °C).

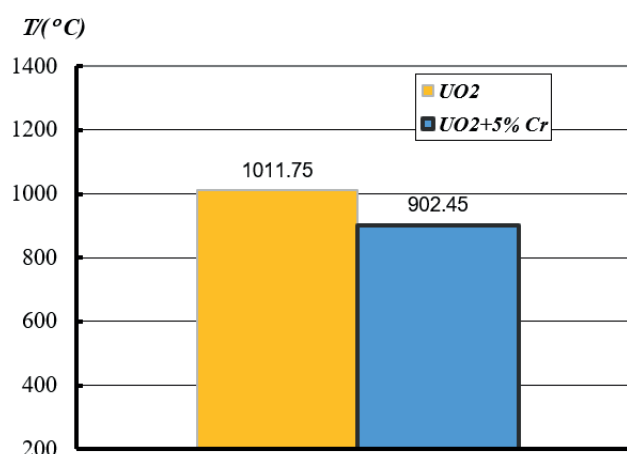
The operating temperature of fuel $UO_2 + 5\%Cr$ (CERMET) is 902.45 °C, which is safe as it is lower than the melting temperature of uranium dioxide.

For a fuel element with $UO_2 + 5\%Cr$ and a zirconium cladding, there are no studies on the diffusion contact between the cladding and fuel, although the swelling of this type of fuel is less than that of dioxide, so it is not advisable to carry out the calculation without a gas gap, as there is no evidence to suggest that Cr will not form intermetallic compounds with Zr.

Conclusion

Figure 5

The temperature at the center of the fuel element depends on the fuel system



As illustrated in Figure 5, the classical $UO_2|Zr$ fuel system has the highest temperature

compared with UO_2 -5% Cr fuel with Zr.

In the final qualifying work, the analysis of the ATF system was carried out, and thermal calculations for the core were performed.

Materials (fuel and cladding) are proposed that can potentially be used in the VBER 300 project reactor.

The advantages that the ATF system provides are clearly shown:

- providing lower operating temperature;
- ensuring security.

In this study, the main geometric characteristics of the VBER 300 reactor plant were considered, as well as the parameters of the fuel cycle, were indicated: the duration of the campaign, the number of fuel assemblies, enrichment, etc.

The application of a block reactor in various fields of energy is considered.

Also, the study was devoted to an overview of the concepts of an accident-tolerant fuel system. A review of various accident-resistant fissile materials and cladding materials was carried out (their thermophysical properties, behavior in normal operating conditions were described, and production technology was studied).

The second chapter contains the thermal calculation of the reactor core: determination of the coolant velocity (3 m/s) and core geometry ($D_{a,z} = 2.42\text{ m}$; $H_{a,z} = 2.42\text{ m}$), determination of the number of fuel assemblies, determination of fuel temperatures for different systems (902°C- 1011°C), and comparison of the obtained data with VVER 300.

The analysis and comparison of the temperature distribution configurations for UO_2 $UO_2 + 5\%Cr$ fuels have been performed.

It can be said that they are safe for both Uranium dioxide and cladding, as well as CERMET and cladding, because their operating temperatures are lower than their melting (decomposition) temperatures.

Further research can be directed to a more detailed study of the strength characteristics of ATF systems, as well as optimization of the process for their production.

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