Thermal and Reliability Criteria for Nuclear Fuel Safety

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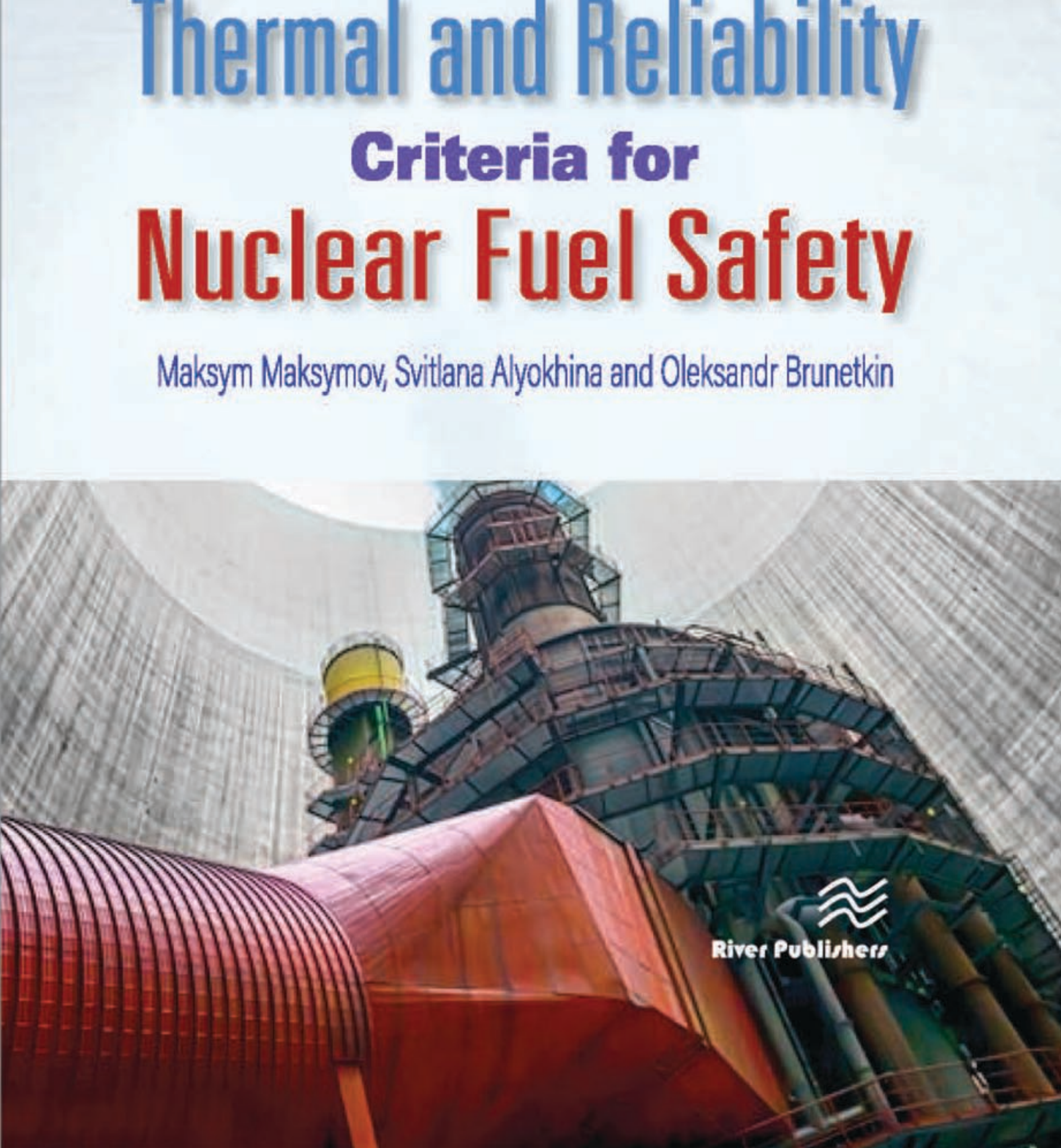

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Contents

Afterword

The trend in the development of commercial nuclear power indicates economic and technical attractiveness of the further application of light-water non-boiling pressure vessel reactors and the expansion of the application of such a technology for NPP power units. The expansion should be understood as two aspects: the extension of the plant life of operating units or the construction of new ones. One of the areas of commercialization is a significant reduction in the cost of technologies for SNF reprocessing or storing. The analysis has shown that even taking into consideration the variety of such technologies, only long-term dry storage of spent nuclear fuel from reactors of this class is common for fuel from different suppliers. This means that the issues of safe and reliable dry storage of spent nuclear fuel will be in the focus of attention of researchers for a long time.

The authors of the book are convinced proponents of the idea that there is no alternative to the development of nuclear energy. Within the framework of the book, the issue of how to develop nuclear power in the future was not discussed; the attention was focused on the issues on which the development of nuclear power depends, namely, the criteria for the safe operation of spent nuclear fuel were discussed.

The modern world is very dynamic in all its manifestations, but this dynamism is spasmodic; it is especially well manifested in the development of nuclear power, including the developed technologies for dry storage of fuel. The designers and manufacturers of this technology are changing, the geography of the introduction of new technology samples is expanding, but the basic technologies for removing energy from the irradiated fuel assembly and ensuring the strength of the cladding during its storage remain the same. Only the requirements for safety, reliability, and efficiency are constantly being reinforced.

These requirements are applied not only to newly created and designed technologies, but also to those dry storage facilities that are in operation. Scientific research in the field of materials science, extension of knowledge of the physics of processes associated with dry storage, new approaches and methods for modeling processes occurring during long-term storage of spent nuclear fuel, as well as improving information and measuring systems and information processing facilities are the most important components of ensuring safe and reliable SNF storage.

The potential for borrowing engineering ideas in the world practice of designing dry storage facilities is minimized with each project and has practically reached its limits. It is clear that the strategic search for innovations in order to improve dry storage technologies should become an important part of the complex task of innovation in nuclear energy. From the point of view of unsolved problems in the field of dry storage of spent nuclear fuel, the following direction presented in the book "Thermal and Reliability Criteria for Nuclear Fuel Safety" can be distinguished: the authors showed one of the possible ways of making a decision on the long-term storage of spent nuclear fuel in a dry storage facility and the possibility of its subsequent disposal.

On the basis on the material presented in the book, a number of the following important conclusions can be drawn. First, compliance with the strength criteria for making a decision on dry storage of spent nuclear fuel guarantees the absence of cladding failure of the fuel element. Secondly, the cladding can be destroyed due to the violation of the removal of residual energy release through it. Both in the first and the second cases, the ongoing processes depend on the properties of the cladding material.

Preface

The authors of this book are united in their research activities by the desire to ensure the European level of operational safety of nuclear power plants (NPP) in Ukraine, namely in terms how to save the resource of operating nuclear fuel and the possibility of making a decision on its long-term dry storage, which will undoubtedly result in a significant reduction in the risks of nuclear incidents and reputational losses at the final stages of the nuclear fuel cycle.

It should be noted that for a long time the authors have been cooperating with Studsvik, which is the moderator of a number of scientific and technical works as part of the Studsvik Cladding Integrity Project (SCIP). The cooperation is carried out on the basis of an international consortium, that includes Ukraine, which is represented by a number of research organizations. These are the results obtained in the framework of the SCIP-III and SCIP-IV projects that stimulated the desire to formalize the existing scientific groundwork on safety issues of the nuclear fuel for water-water energetic reactors (WWER) at the final stages of the nuclear fuel cycle in the form of a book. It is worth mentioning that it was the visual examination of irradiated nuclear fuel in the Studsvik laboratories, which the authors had the opportunity to observe; they influenced the final understanding of the paradigm that is presented in this book.

The object-matter of the research in the book is the safety of nuclear fuel for WWER-1000 reactors under normal operating conditions at the final stages of the nuclear fuel cycle.

The subject-matter of the research is the processes of thermal physics of nuclear fuel and the accumulation of failure to the cladding of nuclear fuel, which determine the model of its safe operation in the WWER-1000 reactor and in open dry container storage facilities for spent nuclear fuel under normal operation conditions.

First of all, the book is based on the scientific achievements of the authors of the book, M. Maksymov, S. Alyokhina, and O. Brunetkin, who are the Doctors of Engineering Science. A number of the propositions that are included in the book were obtained by the authors in collaboration with PhD students, where the authors were scientific supervisors.

There are six sections in the book. The first chapter “Physical Safety Basis of WWER Nuclear Fuel” is devoted to the safety of the SNF storage facility, which is necessary to create guaranteed conditions for thermal states throughout the entire operation time of the storage facility and it creates the possibility to control the sources of ionizing radiation.

The second chapter «Modern Approaches to the Heat Exchange Modelling in NPP Equipment» reveals the methods of the development of mathematical models of nonstationary heat transfer of the technical system, which provides the heat transfer related to any state of nuclear fuel. It is necessary to generalize the principles of physical modelling in order to fold information, which gives opportunity to check it by means of a mathematical model, evaluation of alternative variants of the physical model under consideration, and the choice of the best one.

The third chapter «Safety Criteria for WWER-1000 Fuel Assembly when Making a Decision about its Dry Storage» is devoted to the search for optimality criteria for the control of a NPP with WWER-1000 for which it is necessary to find efficiency criteria that would take into account the requirements of nuclear safety. This makes it possible to compare any methods of operating the reactor core, including power maneuvering.

The fourth chapter «Effect of Reactor Capacity Cyclic Changes on Energy Accumulation of Irreversible Creep Deformations in Fuel Claddings» presents the modelling of the operation of nuclear fuel in cyclic modes; this is necessary to ensure compensation for power changes within the daily or weekly production scheduling of the power system requirements, which makes it possible to compare the considered control programs with the inherent specifics of each change of technological parameters, which has a significant effect on the interaction “the fuel pellet and the cladding» and leads to leakage.

The fifth chapter «Analysis of WWER 1000 Fuel Cladding Failure» deals with the definition of the computed values of leakage probability of fuel element claddings, taking into the account the inhomogeneity of the distribution of energy release among fuel elements of fuel assemblies, which is necessary to control the properties of fuel elements; it also makes it possible to control the value of cladding failure and, therefore, at the same time, the predicted probability of depressurization of the cladding of fuel elements happens.

The sixth chapter “Thermal Safety Criteria for Dry Storage of Spent Nuclear Fuel” is devoted to the development of the basis for the analysis of thermal regimes of SNF dry storage, which is necessary for the safe long-term operation of an interim SNF storage, as a result of which it is possible to make an informed decision about the possibility of subsequent reprocessing or disposal of nuclear fuel.

The presented material in the chapters allowed the authors to formulate the following method, that ensures the safety of the nuclear fuel that is operated.

The conservatism which lies in the design of the fuel elements can be used not only for an increase in the capacity of a nuclear power unit in excess of the design one or for operation in maneuverable modes, but also for ensuring long-term dry storage of spent nuclear fuel.

It is generally accepted that mechanical failure of nuclear fuel according to the stress corrosion cracking model is completely excluded due to the limitation of the linear capacity and the rate of its increase, but this is not always the case, as it is possible to simultaneously impose technological operating conditions when such a previously excluded model starts affecting nuclear and thermal safety.

To prevent such a possibility, it is constantly necessary to minimize or practically eliminate the following four processes during the operation of nuclear fuel in the reactor:

— not to load any fuel assemblies with the first power leap immediately after refueling;

— alternate switching on of main circulation pumps when gaining power, especially in the first 40 effective days;

— if, after reloading of nuclear fuel, the unit operated at its nominal capacity for several days (there was no sufficient accumulation of cracks in the fuel pellets) and was unloaded or stopped, then its reloading must be carried out according to a special program, and not according to operating management recommendations;

— not to allow an opposite change in the coolant temperature with changes in the current power in the upper and lower parts of the reactor core.

If simultaneously any two of the stated processes occur, then they significantly reduce the resource and do not allow keeping the fuel in proper condition for its dry storage. Moreover, it will not be possible if at least three of any processes are superimposed in one time interval. It is very difficult to predict what will be the properties of the fuel if four processes coincide at the same time in the current time interval.

As a rule, a probabilistic safety analysis is carried out in order to prevent emergency design conditions of severe accident conditions. Such an analysis does not objectively evaluate the situation, but the analysis method allows ultimately estimating the state of the fuel, which turns out to be erroneous in principle under current operation conditions.

Due to the fact that all the options in the analysis were not estimated, during operation there is a high probability of making a wrong decision. As a result, the operations personnel choose "the best option from the worst one." The best one of those, which were considered in the probabilistic safety analysis.

If to apply this strategy when operating a power unit, then instead of objective assessing of the processes that occur, weighing all the pros and cons, the operator tries to evaluate the state of the nearest future. As a result, the decision made and the development of the situation keep the power unit from emergency modes, but at the same time the number of failures formed in the nuclear fuel increase. However, paradoxically, no one takes these failures into account in the future, and they usually begin to manifest themselves at the end of the period when the fuel is in the reactor core. The paradigm proposed by the authors is as it follows.

To identify the current state of the fuel and the ongoing processes that affect the safety of the fuel, and then operate it so that the subsequent states of the fuel ensure its long-term operation.

Anyone who cannot identify the current state of the nuclear fuel in the core simply does not know what is happening with the fuel at the moment.

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List of Abbreviations

BUC	Burn-up credit
BWR	boiling water reactor
CFD	Computational Fluid Dynamic
CRDM	Control Rod Drive Mechanism
IAEA	International Atomic Energy Agency
LOCA	loss-of-coolant accident
LWR	light water reactor
MM	mathematical model
NPP	nuclear power plant
PWR	pressurized water reactor
SNF	spent nuclear fuel
TDMA	TriDiagonal-Matrix-Algorithm
TS	technical system
WWER	water-water energetic reactor

1

Physical Safety Basis of WWER Nuclear Fuel

1.1 Fuel Burn-Up as a Nuclear Safety Criterion

The safety of spent nuclear fuel (SNF) management is based on the implementation of the following criteria [1, 2, 3]:

- non-exceedance of fuel element temperature limits due to residual energy release;
- non-exceedance of the level of ionizing radiation effect on staff and the environment;
- guaranteed subcriticality of the storage cask loading or transport cask of the spent nuclear fuel.

The issue of ensuring the fuel cladding integrity as one of the physical safety barriers is a topical matter in the process of the development, implementation, and operation of spent nuclear fuel interim storage [4].

The system of thermal and strength criteria of the cladding integrity support has been internationally adopted. Herewith, the thermal criteria are established keeping in mind the necessity to ensure the strength of the fuel cladding. Consequently, the predictive validity of the cladding failure detection under various storage conditions can have a significant impact on the set permissible storage temperatures and, as a result, it can influence the economic factor of spent nuclear fuel dry storage projects [5].

It should be mentioned that the residual heat of each spent fuel assembly under production-line conditions is not currently controlled by standard methods. Instead, computational methods based on experimental dependencies obtained by calorimetric measurements in laboratory conditions are used. Slow kinetic processes cause the residual energy release, while fast kinetic processes are accompanied by the release of gamma radiation, which is not absorbed or recorded. The results of research establishing the dependence of the ^{137}Cs gamma radiation intensity and the power of heat formation in the fuel assembly are known [6–10].

However, there may be a significant difference between fuel assemblies, depending on their burn-up and operating conditions in the reactor core. In this regard, the process of forming a container loading cannot entirely rely on computations. In addition, we can achieve significant financial savings if accurate measurements of the residual energy release are established since each container is expensive; that is why we should make the best of its usage. Therefore, the nuclear fuel burn-up should be considered as one of the safety criteria when loading into the storage system. For its effective identification, the method for the experimental determination of the heat of residual SNF energy release by means of fast measurements of gamma radiation has been developed.

When analyzing the safety of SNF management systems, the burn-up of specific fuel assemblies is not taken into account, that is, when making an estimate of nuclear safety parameters, all fuels are considered to operate under the same conditions and have some average characteristics. As a result, the calculated value of the subcriticality of the system is conservatively overestimated [11, 12].

This approach was initially due to the imperfection of the calculation programs for determining the reactivity of burned fuel systems and the eventuality of human errors.

The development and improvement of computational methods in recent years allow reducing the conservatism of the computational results at the cost of the burn-up account of a specific fuel assembly, without sacrificing the required subcriticality (coefficient k_{eff}) of a system with a given geometry that takes into consideration neutron leakage and does not reduce its nuclear safety.

Figure 1.1 shows an example of the k dependence (the subcriticality of the system with infinite geometric dimensions without taking into account neutron leakage) on the burn-up for a standard UO_2 fuel assembly. For burn-up of 40 MW-day/kg, the fission coefficient is approximately 30% less than that for fresh fuel [13].

When we consider burn-up as a nuclear safety parameter, the concept of nuclear safety maintenance can be used for all elements which provide the life cycle of spent nuclear fuel, spent fuel pool storage racks and central storage of the atomic nuclear plants, NSF dry storage cask, processing facilities, etc.

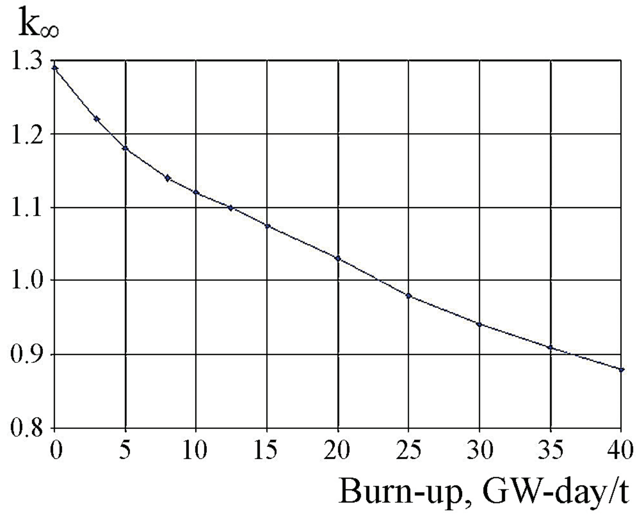

Figure 1.1 *k* dependence on the burn-up for a standard UO_2 nuclear fuel assembly.

Nowadays, the conditions for the promotion of nuclear power plant (NPP) competitiveness require bringing up average burn-up to 60–65 MW-day/kg, which, in its turn, puts a limit on the initial enrichment value of ^{235}U 4.8%–5.1% for reactors with the capacity of 1000 MW [14]. Under the specified enrichment values, the transportation of SNF in the current transportation cask without the account of burn-up is not possible. This problem has already been encountered when using fuel with enrichment of 4.4% in WWER-440 reactors. We can increase the enrichment value for the CASTOR-V/52 transportation cask from 4% up to 4.6% keeping track of burn-up. It is reported that a possible increase of transportation cask capacity is between 10% and 100% [15]. The allowable increase of capacity depends on the initial enrichment and the minimal guaranteed burn-up.

The researchers [13] found out that maximum permissible enrichment of 230 nuclear fuel assemblies, which are located simultaneously in the plant interim storage facility of La Hague, can be increased from 3.3% to 4% on condition that the burn-up is not less than 10 MW-day/kg.

The use of burn-up as a nuclear safety parameter faces quite complex problems; the main problems are [16]:

- which isotopes must be considered when determining the fission coefficient;

- what burn-up value should be taken into account since fuel assemblies have different burn-up profiles.

To solve the first problem, i.e., when choosing isotopes, three main approaches are used [17]:

- accounting only for the depletion of primary fissile material;
- additional accounting for actinoids with large atomic masses formed during the operation of the reactor;
- additional accounting for fission products, which have a high neutron-absorption cross section.

It is relatively simple to bring the first scheme into action because most of the calculation programs and evaluated nuclear data file are verified on a large amount of experimental data [18–20]. Any additional analysis of fuel compositions with respect to actinoids with large atomic masses formed during the operation of the reactor requires a higher level of experience with software codes and evaluated nuclear data file. Complete account of neutron absorption by fission products is one of the most complicated tasks, especially for high level of burn-up. This is largely due to abundance of isotopes that should be taken into account. Most of the corresponding computational programs and evaluated nuclear data file are currently being under implementation and validation. Figure 1.2 shows the dependence of the effective neutron fission coefficient for various calculation models [17].

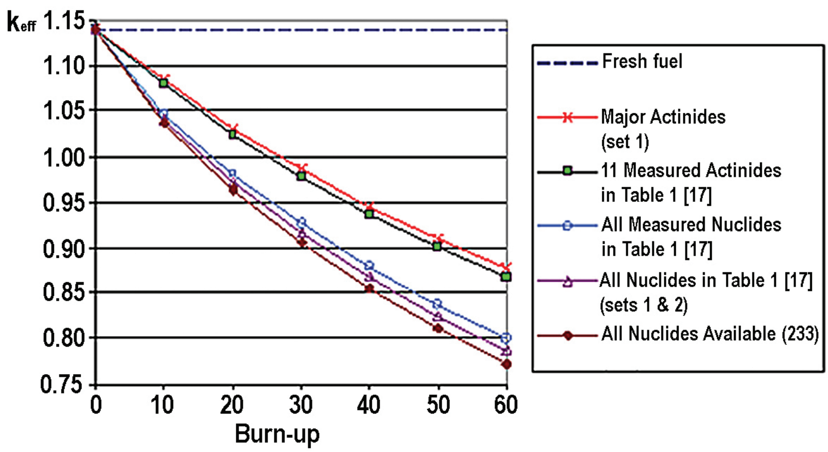

Figure 1.2 Dependence of the effective neutron fission coefficient for various calculation models [17].

It is worthwhile noting that the analysis of the use of burn-up as a safety criterion requires more calculations than the standard analysis of criticality, as it requires the calculation of the SNF isotopic composition.

As it is pointed out in [21], typical program codes (OECD/NEA) used for reactor computations may not be suitable to use Burn-Up Credit (BUC) as a safety criterion. This is due to the fact that complex models are used in the calculations of the reactor core and special requirements are imposed on the initial data. Therefore, the codes and data are closely related. The purpose of computations of the reactor is its efficiency. When we use codes for non-reactor zone facilities (for example, an SNF transport container, SNF dry storage container, etc.), the purpose of the computations is maximum safety of their operation. It should also be taken into account that these facilities can contain fuel with different production history and which was produced by different manufacturers.

It is noted in [22] that the use of BUC involves knowledge of the exposure time, burn-up, initial enrichment, and isotope distribution. For example, the practical application of this approach in France requires the fulfillment of the following criteria:

- burn-up value is based on the least irradiated 50 cm of the active length of the nuclear fuel assembly;
- actual burn-up value must be checked by measuring each nuclear fuel assembly.

The type of measurements, whether qualitative or quantitative, depends on specific conditions, expected burn-up, and initial enrichment. For example, if an expected burn-up is less than 5.6 MW-day/kg and enrichment is less than 3.3%, qualitative measurements are enough, while higher burn-up and enrichment values require quantitative measurements.

The concept of burn-up usage as a nuclear safety parameter is not a modification of the basic safety principles or an attempt to define new safety principles [23]. From this point of view, real-time burn-up definition allows ensuring the principle of safety priority directly in the process of SNF overload while improving the economic performance of nuclear power plants.

To provide the implementation of these alternatives, it is necessary to consider methods and means how to control nuclear materials, determine the nuclear fuel burn-up, as well as to find technical solutions that allow real-time burn-up measurement.

1.2 Influence of the Reactor Operating Mode on the Efficiency of WWER-1000 Fuel Cycles

The main issues of the economics of a fuel cycle are presented in the following works [24–28] and others. Even without taking into account macroeconomic aspects, they are extremely complex and belong to the class of optimization tasks. The traditional approach is based on two main principles:

- the power unit is to operate on rated capacity between refueling;
- the ratio of the plant unit downtime to its on-time is to be minimal.

Hence, we have the task of reducing the duration of preventive maintenance and developing a program to increase the duration of fuel lifetime. Due to the fact that a clear, comprehensive criterion for the efficiency of nuclear power plants is still unknown, many aspects are excluded from consideration. For example, in [29], the authors show that even the regular use of the operation mode of the WWER-1000 power unit with partial use of negative reactivity effects leads to a decrease in the average power level per a campaign. On the other hand, it allows increasing the yearly average power production or decreasing the cost value of the fuel component of supplied electricity without the reduction of the yearly average power production.

The operation time between refueling depends not only on the commercial efficiency of the cycle options but also on such conditions as ensuring preventative and predictive maintenance at any given time (for example, operation out of the bounds of the autumn and winter peak of electricity consumption, operation of other units of the given nuclear power plant, and the reliability of the equipment). Thus, the duration of the campaign can be significantly reduced, regardless of the type of the used fuel cycle.

The introduction of cycles with reduced neutron leakage, as it was implemented at the Khmelnitsky NPP (KhNPP), makes it possible to form a wide range of loadings within the framework of the limitations mentioned above. In this case, the fuel campaign becomes, on average, 10% shorter than the project provides. From the point of view of the traditional approach, these cycles are less efficient than the projects due to the proportional increase in the constant constituent of the nuclear generating cost. But in such a cycle, the neutron flux on the inner surface of the reactor vessel is reduced by 25%–40%, which creates the prerequisites to a proportional increase of the life of the reactor, and, thereby, it practically proportionally reduces the constant constituent of the cost value. In addition, such a cycle makes it possible to obtain significant savings in the fuel factor of the cost value and to reduce the specific amount of SNF per unit of supplied energy, as well as more frequent performance of preventative and predictive maintenance additionally increases the reliability of the nuclear power plant operation during the campaign. The influence of reactor core layout arrangements on the resource of the reactor vessel is so high that it actually makes it possible to operate and control it [30].

Under actual operating conditions, the average reactor capacity during the campaign is lower than the rated capacity. This can be caused by many factors: partial equipment malfunctioning, which requires a reduction in power, power line capacity, operation on the power reactivity effect, etc. When estimating the efficiency of such a unit, the balance of contributions from different criteria changes.

There are data about the possibility of operation of WWER-1000 power units at the so-called “daily and weekly” load schedule [28, 31, 32]. This mode of operation of power units is a promising one, as the market value of such energy increases by 1.5–2 times and requires a corresponding feasibility study. Computational and experimental researches [28, 31, 32] show rather stable behavior of the WWER-1000 reactor core in transient modes under the appropriate choice of control actions.

As a result, operating conditions of the power unit which can be constituents of estimation criteria of its operation (average capacity per a campaign, calendar and effective loading work time, duration of preventative and predictive maintenance, depth of burn-up of upload SNF fuel, average integral density of neutron flux on the reactor vessel, etc.) can be described by a system of dependencies. Constraint functions in these dependencies are non-linear and are determined starting with the characteristics of the reactor core that range from the simplest one (e.g., the number of reactor fuel assemblies) and ending with quite complex, empirically determined connections. The example can be a link between the fuel make-up nomenclature, the reduction of neutron leakage from the reactor core, and the value of the reactivity coefficient according to the coolant temperature at a minimum controlled power level. The type and amount of fuel assemblies depend on the reliability characteristics of the equipment as well as the relation between increasing the depth of fuel burn-up and severization of requirements for reactor control quality in transient modes [33–37].

In order to choose the type and amount of fuel assemblies, it is necessary to use a complex approach: at the first stage, based on the experience of steady-state fuel cycle formation, it is necessary to analyze the layouts of specific loads as perturbations of standard cycles. The second stage is to ensure that the phenomena manifested in the accumulated operating experience and subject to systematization are taken into account. At the third stage, the stochastic element must be taken into account, which can be done using the theory of optimal processes. After the second and third stages, it is necessary to adjust the results of the previous stages each time. After a series of iterations, the problem of optimal control of the entire fuel cycle of nuclear power plants is to be solved. Some fuel loading layout problems are discussed in more detail below.

The study of fuel cycles in "ideal" conditions of operation of the reactor excludes from consideration all the parameters except the characteristics of the reactor core and gives a schematic representation of fuel consumption effectiveness. Therefore, in the future, an analysis of the impact of the operation of the power unit in maneuverable mode on fuel consumption will be given.

It can be shown how the actual operating conditions influence the properties of irradiated fuel by taking into account the average power level of the reactor plant per campaign. The consideration of the average capacity level in a number of regarded parameters allows including two reactivity effects. Both of them are related to the fact that when operating at reduced power, the reactivity margin for fuel burn-up in the reactor is higher than when operating at the nominal level. The first effect is the most significant, and the power level at which the reactor operates at the end of the load plays a crucial role here. Reduced power provides the possibility of longer calendar work as well as longer effective work. Although this leads to a decrease in the average reactor capacity per campaign, the average annual energy production and plant capacity coefficient can increase. A detailed discussion of this phenomenon is given below. The second effect is not of such importance, but with an accurate evaluation of the operational efficiency of the reactor, it should be determined and taken into account. It is caused by the fact that with a decrease in the full capacity of the reactor, the redistribution of energy release between the fuel cells in the reactor core occurs. As a result, the neutron leakage value outside the reactor core changes, i.e., inefficient losses of reactivity margin as well as the distribution of nuclear fuel burn-up rate over the reactor core take place. The capacity level of operation and the duration of its operation per a campaign are of great importance for this effect. In this case, the advantage is in the effective duration of the campaign as well as the depth of fuel burn-up in the unloaded part of the reactor core.

The example of taking into account the decrease in power when analyzing the cycles is given in [38]; it demonstrates one of the transition methods to the consideration of the reactor within the conditions of a real operation. It is shown that for an additional evaluation of the effectiveness of fuel cycle options, it is possible to analyze the degree of their sensitiveness, from the point of view of reducing efficiency, and from reducing the reactor capacity during operation as well.

The control of axial offset is one of the tasks of reactor safety protection, the quality of its operation in case of the efficiency increase of fuel utilization, and the use of the established capacity level. In addition, the control of axial offset is one of the two main components of the problem, which is linked to the adaptation of WWER-1000 power units to operate in the maneuverable mode [39–41]. The studies undertaken allow identifying several main, conditionally independent, possible components of cost advantages:

- the effect of the operation in the maneuverable mode;
- the effect which is related to the nuclear fuel reliability growth;
- the increase of the installed capacity utilization factor;
- the effect of the operation in the mode of the higher burn-up.

The first component of the effect of maneuvering is determined by the fact that in the European energy market, the electricity generated by the power units participating in the regulation of the power system frequency is paid at a higher rate than the electricity generated by the power units, which provide the standard component of the capacity of the system. The effect can be estimated at the level of 50% of the cost of electricity generated at nuclear power plants, allowing for about 7% of losses from generation reduction during night unloading, as well as increased cost of advanced fuel and equipment, which will be determined by suppliers and, apparently, can be estimated at half of the expected effect. The total value of the effect from the entire complex of works can reach up to 20%–22% of the cost of the energy generated by the NPP.

If using advanced control algorithms of WWER-1000 control, the second part of the effect of maneuvering allows the nuclear power unit No. 1 of KhNPP to operate without preschedule reactor fuel assembly unload due to the leakage of fuel cladding, with coolant activity in the primary circuit that makes it possible to operate the reactor core without annual control of fuel assembly leak resistance. Thus, this component is close to the cost of all prematurely unloaded leaking fuel assemblies. The average number of prescheduled fuel assembly unload can be estimated at the level of two fuel assemblies for one power unit per year; the increase of fuel factor of the cost of energy caused by their replacement is about 1%–2% [42].

The third part of the effect of maneuvering allows reconsidering the approaches to the base-load operation condition. Here the effect of inefficient financial resources appears; in this case, the power unit has to operate at reduced parameters due to constraint violations set for the power distribution in the reactor core caused by the xenon transient process. This value can be evaluated by the 30-hour of load decrease of 25%.

The fourth component of the effect of maneuvering requires a deeper understanding, which is given below.

In addition to the studied components of the cost advantages, there are other aspects, the quantitative assessment of which has not been performed by the authors.

For example, the reliability growth of nuclear fuel leads to a decrease in reactor coolant activity, a decrease in the activity of gases in the ventilation system of a power unit, and a reduction of the personnel radiation doses during scheduled maintenance. It can also lead to a reduction in these maintenances and an additional increase in the installed capacity utilization factor. If handling with the fuel is on the critical path of the preventative and predictive maintenance conduction, the absence of necessity to change leaking reactor fuel assemblies or conduction of additional control of the leak resistance leads to the reduction of the maintenance period and to the additional increase of the installed capacity utilization factor [33–35].

1.3 Design Constraints and Engineer Suitable Coefficients When Designing and Operating WWER Fuel Loads

Operational limits or design limits under standard operation are values of parameters and characteristics of the system state and nuclear power plants as a whole, which are set by the project for normal operation [43].

The promotion of the nuclear energy competitiveness and nuclear fuel competitiveness in the global market requires the introduction of new, more efficient fuel cycles [44].

New fuel cycles include an increase in the burn-up depth, profiling of enrichment, introduction of burnable absorbers in fuel assemblies, and an increase of the capacity in a power unit

New fuel cycles make it necessary to review the existing set of operational limits, namely:

- introduction of new constraints;
- exclusion of duplication;
- physical "transparentness";
- exclusion of unreasonable conservatism.

Operational limits lie at the heart of the concept of safety and its constituents, from which the safety criteria follow. Necessary and sufficient conditions of safety criteria are given in Table 1.1.

Table 1.1 includes the requirements for the operation of nuclear fuel that comply with the IAEA recommendations. The principle is invariability, maintenance of safety criteria such as input data for operational limit development.

The accomplishment of operational limits is a purpose which can be achieved by controlling other parameters on the basis of the in-core control system.

Below, there are operational limits of the WWER-440 (B-213), which, under the operation of the reactor, are used when choosing loadings [45, 46]:

- reactor thermal capacity can exceed the nominal value of 1375 MW not more than 4%;
- coolant pressure at the output of fuel assemblies with six operating main circulation pumps (MCP) may differ from the nominal value (12.26 MPa) at the most 0.2 MPa.
- coolant inlet flow when there are six working MCPs is not less than 39,000 m^3/h.
- average coolant temperature at the inlet of the reactor core is to be in range of 265°C–270○.
- maximum capacity of the fuel element and fuel element with Gd is 54.5 kW if pin-to-pin spacing in an assembly is 12.2 mm and maximum capacity 56.6 if pin-to-pin spacing in an assembly is 12.3 mm.
- marginal linear load and changes (jumps) in the linear load of a fuel element and fuel element with Gd depending on the burn-up is in accordance with the graphs shown in Figures 1.3 and 1.4.
- subcriticality in case of shutdown is 1%, and in case of refueling it is 2%.

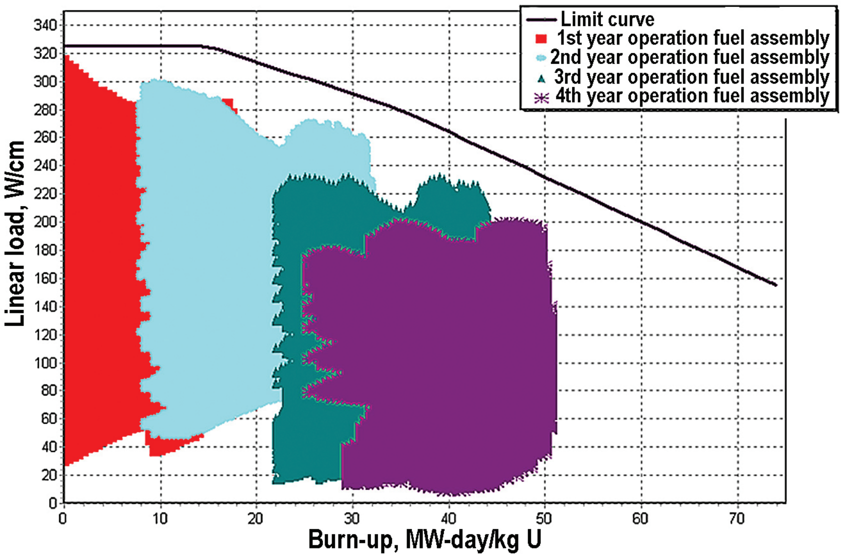

Figure 1.3 Field of local fuel element loads with respect to the factor of margin depending on the burn-up.

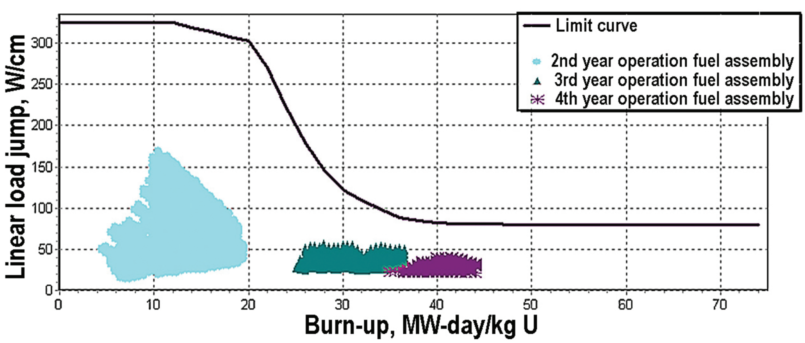

Figure 1.4 Fuel element linear load jumps as a result of refueling.

1.4 Criteria and Methods of Nuclear Fuel Safety Evaluation Under Operation

When performing a maneuver with a power pressurized water reactor, the operator faces the problem as to how to control the power density field because of xenon transient process occurrence and axial offset oscillations, which are the result of this occurrence. The axial offset is determined by a dependence

Table 1.1 Technological and radiation criteria, which provide safe operation of NPP with WWER.

Protection Level	Technological Criteria				Radiation Criteria
	Fuel Matrix	Fuel Cladding	Coolant Circulation Circuit	System of Leak-Tight Enclosure	
I Level – Normal operation (NO). **Purpose:** Provision of NPP safety due to the operation and maintenance reliability of barrier efficiency as well as provision of personnel with technical means and organizational measures necessary for NO mode.	No fuel melting. The yield of radiologically hazardous fission fractions from the fuel matrix is not more than 0.3% of the total amount.	Non-exceedance of the operational limit of fuel element failure: • defect of gas leakage is not more than 0.2% of fuel assemblies; • close contact of nuclear fuel with the coolant is not more than 0.02% of fuel assemblies;	Under NO, the amount of uncontrolled leakage from the primary coolant circuit is not more than 100 l/h. The pressure is less than or equal to the operating pressure. In case of emergency, there can be a pressure boost up to 1.15 from the operating pressure.	The amount of leakage from the containment is not more than 0.1% of the containment volume per day.	**Limits of occupational exposure:** 1. The limit of a personal dose of external and internal exposure is 20 mSv/yr. Design values of a dose rate corresponding to the given limit with respect to twofold design suitable factor according to the job characteristics are used to develop the protection against ionizing radiation. 2. A limit on the collective dose of personnel during routine maintenance is set. The routine maintenance is related to the dose consumption (preventative and predictive maintenance), refueling – 0.5 mSv/yr. **Dose limits of public exposure:** Individual exposure – 0.1 mSv/yr. The limit refers to the annual effective equivalent dose for the critical group of population caused by NPP operation, taking into account direct and indirect pathways of radiation exposure.

Table 1.1 *Continued*

Protection Level	Technological Criteria				Radiation Criteria
	Fuel Matrix	Fuel Cladding	Coolant Circulation Circuit	System of Leak-Tight Enclosure	
		• No boiling crisis, suitable factor before crisis is 1.2 ÷ 1.3 (1+2σ), where σ is a root-mean square error of the used correlation			The limit corresponds to 10% of basic dose limits established for population by the safety radiation level.
II Level – Abnormal operation (AO). **Purpose:** Provision of NPP safety due to the restriction of NO to the extent to its shutdown and maintenance of barrier efficiency as well as personnel activities with technical means including safety systems.	In a similar way to NO				

Table 1.1 *Continued*

Protection Level	Technological Criteria				Radiation Criteria
	Fuel Matrix	Fuel Cladding	Coolant Circulation Circuit	System of Leak-Tight Enclosure	
III Level – Design Basis Accident (DBA) **Purpose:** Provision of NPP safety due to the safe decommissioning by means of the safety system.	No fuel melting. Fuel enthalpy under the reactivity effect is not more than 145 cal/g.	Non-exceedance of maximum design damage limit for fuel elements: • temperature of fuel element claddings is not more than 1200○; • local oxidation rate of fuel element claddings is not more than 18% from initial wall thickness;	Loss of primary coolant can lead to short-term dryout of the core. Pressure does not exceed 1.15 P_{op}.	The amount of leakage from the containment is not more than 0.1% of the containment volume per day.	Planned special exposure of personnel. The limit of a personal dose of external exposure is 40 ÷ 80 mSv/yr. The limit of individual effective equivalent dose in the main control room and emergency control room is – 25 mSv/yr. It is installed in the design to reliably ensure the permanent stay of personnel in the main control room and emergency control room. The limits relate to the integral doses received by the personnel during the accident and the rectification of its consequences for external and internal exposure due to inhalation.

Table 1.1 *Continued*

Protection Level	Technological Criteria				Radiation Criteria
	Fuel Matrix	Fuel Cladding	Coolant Circulation Circuit	System of Leak-Tight Enclosure	
		• the proportion of reacted zirconium is not more than 1% of its mass in the claddings of fuel elements.			**Dose limits of public exposure:** Individual total body dose is 5 mSv/yr. Exposure effect on the public caused by accidental releases of radioactive substances into the environment during design accidents and/or external effects do not require the introduction of protective measures for the public.

$$\mathrm{AO} = \frac{N - N}{N + N} \times 100\ \%, \tag{1.1}$$

where N_B and N_H are the capacity of core in upper and lower packages, respectively.

Obviously, almost any change in the parameters of the WWER-1000 core (capacity, temperature, coolant flux rate, position of the regulating mechanisms, concentration of the integral absorber, etc.) can lead to xenon oscillations of the axial offset [47, 48].

As a rule, a power maneuver is planned and carried out as a certain sequence of relatively fast transitions between power levels at which the reactor operates for a rather long time. In this case, the task of controlling the power density field centers on maintaining the axial offset current value in proximity of the given value [49].

The analysis of the qualitative dependences of variations in axial offsets with respect to changes of the WWER-1000 main operational parameters shows that any exposure on the reactor installation leads to an ambiguous change in the axial offset (Table 1.2). In this regard, it is of interest to evaluate the thermal technological reliability of the WWER-1000 core during transient processes.

Since the existing WWER-1000 in-core control system does not allow measuring any local parameters of the power density field, a numerical simulation was used to obtain the power density field parameters in the WWER-1000 core. It was carried out for a reactor that was in the steady-state fueling mode with a three- and four-year campaign. The computer code BIPR-7A was used as a modeling tool [50]. For both campaigns, a

Table 1.2 Main axial offset perturbation actions.

Action	Direction	Result
Change of reactor thermal capacity	$N \Uparrow$ $N \Downarrow$	AO $\Uparrow$ AO $\Downarrow$
Change of *k*-group position of regulating mechanisms of control system and protection in the upper half of the core	$H_k \Uparrow$ $H_k \Downarrow$	AO $\Downarrow$ AO $\Uparrow$
Flow variation	$G \Uparrow$ $G \Downarrow$	AO $\Downarrow$ AO $\Uparrow$
Boric acids concentration change in a heat pump of the primary coolant circuit	$C_b \Uparrow$ $C_b \Downarrow$	AO $\Downarrow$ AO $\Uparrow$
Temperature change of a heat pump at the core inlet	$T_{in} \Uparrow$ $T_{in} \Downarrow$	AO $\Downarrow$ AO $\Uparrow$

hypothetical xenon transient process was considered both at the beginning and at the end of the campaign. A three-hour decrease of the thermal power in the reactor to 50% and its increase again up to 100% was simulated. This mode was chosen as the most logical and the most cost-effective one with a possible maneuverable operation cycle.

The analysis of the table data showed that the largest axial offset oscillations occur during a power maneuver at the end of the fuel campaign during a four-year fuel cycle (Figure 1.5). At the beginning of the campaign, with a large margin of reactivity, the arising axial offset oscillations tend to decay, but at the end of the fuel campaign, the opposite effect is observed.

Let us consider the external temperature of the fuel element claddings and the safety flux before the heat transfer crisis in the most loaded fuel assemblies at the upper and lower maximums of the power density. An unambiguous relation between axial offset oscillations and local values of the power density field is shown in the computation and experimental work [51].

The rate of heat transfer from the fuel elements to the coolant determines the temperature regime of the fuel element cladding. The value of the heat transfer coefficient varies significantly depending on the hydrodynamic structure of the coolant flow and its state of aggregation. When calculating nuclear power reactors with water coolant in a general case, it is possible to consider convective heat transfer, heat transfer with surface boiling, and heat transfer with developed volume boiling of coolant.

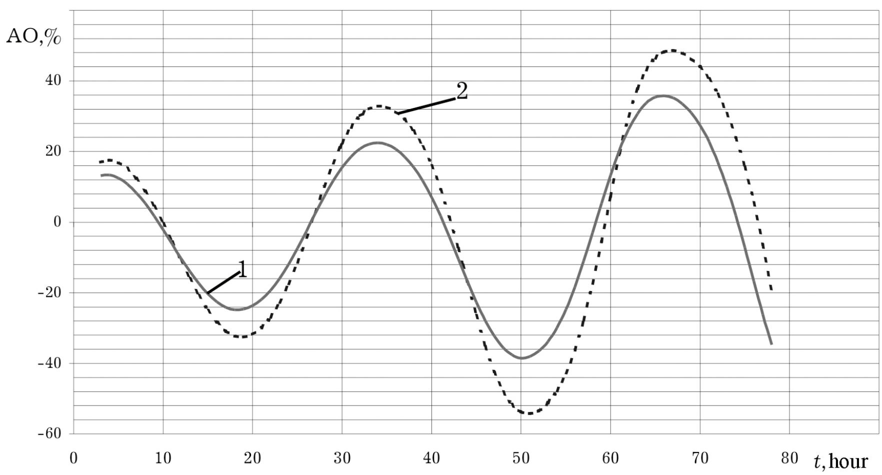

Figure 1.5 Axial offset after the transient process at the end of a (1) three- and (2) four-year campaign.

The heat transfer coefficient for convective heat transfer conditions, as is the case for WWER reactors, can be calculated using the well-known Mikheev empirical formula [52]

$$\alpha = 0,021\frac{\lambda}{d_\Gamma}\left(\frac{\rho\omega\ d_\Gamma}{\mu}\right)^{0,8}\ \mathrm{Pr}^{0,43} \tag{1.2}$$

where λ, μ, and Pr are correspondingly the heat conductivity coefficient, dynamic coefficient of viscosity, and Prandtl number for coolant in a design sector of the reactor fuel assembly; $\rho\omega$ and d_r are a mass flow of the coolant and a hydraulic diameter of a fuel element.

To determine the temperature of the fuel element wall, we shall apply the following equation:

$$T_{\mathrm{wall}} = \frac{q_s \cdot K_V}{\alpha} + T_{\mathrm{cool}} \tag{1.3}$$

where T_{cool} is the coolant temperature in the corresponding reactor core elementary volume; K_V is the radial power peaking coefficient; q_s is the graded density of fuel element thermal radiation.

The determination of thermal characteristics of the reactor core is possible according to the following scheme (Figure 1.6), taking into account the fact that the source data for it are the output files of the BIPR-7A program.

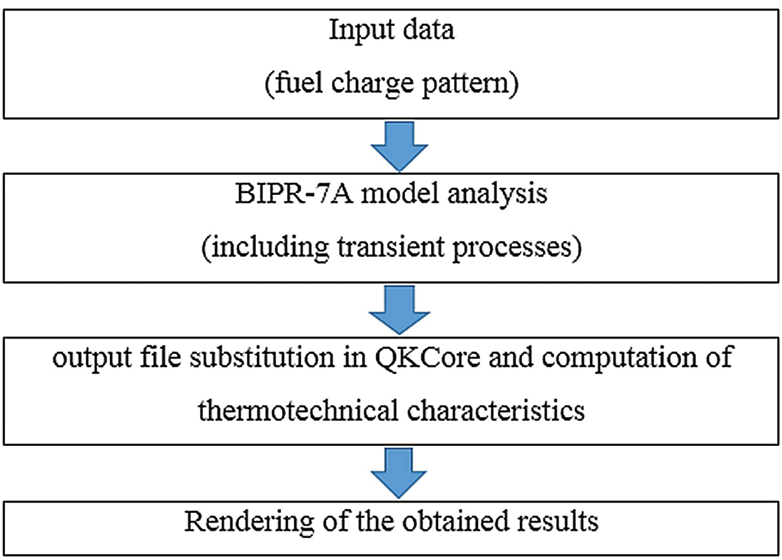

Figure 1.6 Computation scheme of thermotechnical reliability.

Table 1.3 Computational results of the fuel element walls during the first 7 hours after the completion of the power maneuver.

Time after the Beginning of the Transient Process, Hours	Maximum Loaded Lower Section of the Fuel Assembly			Maximum Loaded Upper Section of the Fuel Assembly		
	T_{cool}, °C	K_V	T_{wall}, °C	T_{cool}, °C	K_V	T_{wall}, °C
3	291.47	1.071	308.52	323.33	1.681	350.10
4	291.36	1.051	308.10	323.45	1.68	350.20
5	291.36	1.058	308.20	323.58	1.658	349.98
6	291.48	1.089	308.82	323.74	1.619	349.51
7	291.7	1.143	309.90	323.91	1.565	348.83

The definition of thermal characteristics of the core is possible according to the following scheme (Figure 1.6), taking into account the fact that the input data for it are the output files of the BIPR-7A program.

Such calculations allow running some versions of programs such as RELAP, DIN3D, as well as their combination.

The results of the calculations of T_{cool} and K_V, as well as the temperature of the fuel element walls for the upper, most loaded part of the fuel assembly and its lower section, which is symmetrical to it relatively to the horizontal plane of the core, are given in Table 1.3.

In order to calculate the margin of power before the heat transfer crisis (usually expressed as the ratio of the critical heat flux to the nominal $K = q_{\text{cr}}/q_S$), it is necessary to define the so-called heat flux critical density q_{cr}. of the fuel assemblies.

There are several empirical formulae to define fuel assembly q_{cr} which are based on the research of flow models for various types of reactors, various thermohydraulic parameters, and states [53]. For this analysis, we used the equation that allows evaluating q_{crI} in a fuel assembly with an error limit of 11%

$$q_{\text{crI}} = 0,0274 \cdot (\rho\omega)^{0,505} \cdot (1 - E)^{1,965} \cdot (1,3 - 9,4 \cdot 10^{-4} \cdot P) \qquad (1.4)$$

where *P* is the coolant pressure in the primary circuit; $\rho\omega$ is the coolant mass flux; and *x* is the flow quality.

This dependence is applicable to the pressure of WWER-1000 (*P* = 16.7 MPa) reactors under conditions of uniform heating of the rod bundles.

In addition, in order to calculate the heat transfer crisis in fuel assemblies of WWER-1000 reactors, experimental design bureau “Hydropress” recommends the formula obtained under conditions as close as possible to the operating conditions of this reactor [53]

Table 1.4 Data on how to calculate the safety coefficient before the heat transfer crisis in the maximum loaded fuel assemblies.

Time after the Beginning of the Transient Process, Hours	q_{crI}, $\mathrm{MW.m^{-2}}$	q_{crII}, $\mathrm{MW.m^{-2}}$	q_s, $\mathrm{MW.m^{-2}}$	C_{sf1}	C_{sf2}
3	3.163	5.139	0.940	3.37	5.470
4	3.160	5.126	0.939	3.37	5.459
5	3.155	5.111	0.927	3.40	5.516
6	3.150	5.094	0.905	3.48	5.629
7	3.145	5.075	0.875	3.60	5.802

$$q_{\mathrm{crII}} = 0,795 \cdot (1 - E)^n \cdot (\rho\omega)^m \cdot (1 - 0.0185 \cdot P) \tag{1.5}$$

where n = 0.105.P = 0.5, and m = 0.184 − 0.311.x; the other parameters are as in the formula for q_{crI}.

The behavior of the analyzed parameters allows concluding that during the first hours after the beginning of the transient process, there are favorable conditions to operate the reactor according to safety criteria (Table 1.4). We can observe an increase in the temperature of the fuel element cladding at the bottom of the reactor core due to a shift in the maximum power density exactly there, which, in fact, is not dangerous, because the temperatures at the bottom of the core are initially 20°C−25°C lower than at the top. Further, the temperature begins to fluctuate because of the redistribution of the neutron field due to the temperature effect of reactivity.

The temperatures in the lower and upper parts of the reactor core fluctuate in antiphase. If to analyze the temperature behavior in the upper part of the fuel assemblies, it can be noted that during the first 7 hours of the transient process, there is an opportunity to control the state of the reactor as, at this time, the lower harmonic of temperature fluctuations is observed. Fluctuations in the temperature of the containment reach dangerous values (350°C) (from the point of view of the storage before the beginning of surface boiling) which is especially evident at the end of the fuel campaign.

The methodology for evaluating the heat and technology reliability of the reactor core, its calculation, and experimental justification are given in [54] and [55], where the authors show that under such conditions, any temperature fluctuations (fluctuations in coolant or cladding of fuel elements) are practically absent, i.e., they are within the estimated error.

The formulae available in the literature and used here to calculate the critical heat flux q_{cr} and corresponding safety coefficient before the crisis approximately equally describe the behavior of the reactor core and its heat and technical reliability but differ significantly in the numerical values of the safety coefficient.

The WWER-1000 reactor is designed on a conservative approach, including a significant reserve of thermal and technical characteristics of fuel assemblies during operation in transient modes.

The presence of a significant reserve before the heat transfer crisis in fuel assemblies creates prerequisites for transferring the WWER-1000 reactor to a maneuvering mode of operation, which is extremely important for the current state of the energy industry in many countries. Problems before the heat transfer crisis in fuel assemblies belong to the class of unsteady-state heat transfer problems and are currently solved exclusively by numerical methods.

The problems of unsteady-state heat transfer in bodies that have the form of geometric primitives (an endless plate, a cylinder, and a ball) belong to classical problems. The results of their analytical solution are given in monographs and many textbooks on heat transfer. In well-known sources, the results of solving such problems are presented in a dimensionless form for generality. As a result, the functional dependence of the temperature of the bodies on time, for example, for the body central points, is presented in a two-criteria form: depending on the Fourier number (dimensionless time) and the Biot criterion. In a graphical form, it corresponds to a set of curves for each considered point. For each of the given bodies, its own solution and, accordingly, an individual set of curves are designed. But for the engineering application of such functional dependencies, the understanding of the adequacy of the mathematical model to a real unsteady-state heat transfer process is necessary.

One of the methods that allow adjusting the adequacy is the ability to bring the model to an automodeling form. The theory of similarity is closely connected with this method. In the given field, there is a basic π-theorem (in English literature − Buckingham theorem; in French – Vaschy theorem), fixing the possible number of dimensionless values in transformable models.

On the way to the further development of dimensionless methods to solve problems of unsteady-state heat conduction, a certain success has been achieved. But the methods used by researchers are the result of the intuition of their developers. As for the scientific approach, it is necessary to consider the method as a coherent logical system which provides the basis for further work in this direction and indicates the relevance of research. Therefore, in the future, it is advisable to consider new approaches for modeling unsteady-state heat transfer processes in nuclear power plant equipment as well as nuclear fuel.

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