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Reload Design and Core Management in Operating Nuclear Power Plants

Experiences and Lessons Learned

IAEA

International Atomic Energy Agency

RELOAD DESIGN AND CORE MANAGEMENT IN OPERATING NUCLEAR POWER PLANTS

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EXPERIENCES AND LESSONS LEARNED

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FOREWORD

Among owner/operating organizations in IAEA Member States there is increased interest in making nuclear power plants more efficient and economical. All aspects of the nuclear fuel cycle present challenges with respect to the costs of fuel, operations and maintenance. Proper, structured reactor core design and management can help to address these challenges.

The reactor core is the heart of a nuclear power plant. It is configured to produce the maximum safe level of thermal power from the available energy in nuclear fuel. Optimizing reactor core operation and management requires know-how in many different technical and economic fields. All aspects need to be managed in an integrated manner.

The term ‘core design’ is used to describe activities concerning loading patterns at existing reactors with known core geometry, a fixed number of fuel assemblies and related management activities. Typical core design objectives are to maximize core power density, to maximize attainable fuel burnup and to minimize the cost of electricity. These activities are generally divided into two phases: in-core and out-of-core management.

In-core management consists of determining energy needs and operating goals for a given operating cycle; calculating fuel performance parameters, core physics and thermohydraulics in order to demonstrate the safety case; and maximizing neutron economics and minimizing the waste of the remaining available energy in the fuel. These evaluations and calculations are performed from the first core load to maintain safety margins and are planned for the lifetime of the fuel and core. Economically, one important element of core design is deciding on reloading schemes in order to optimize the cost of producing energy. Cost optimization requires different calculation techniques to determine the fuel costs from the beginning to the end of the fuel cycle. It needs to be demonstrated that the new cycle’s core meets all safety requirements and the expectations and requirements of the owner/operating organization in terms of costs and maximizing the flexibility of operation in order to generate electricity safely, reliably and efficiently.

Out-of-core management concerns the supply of materials and required services in different stages of the nuclear fuel cycle, such as ordering, manufacturing, transportation, acceptance, fuel storage and spent fuel disposition, for both open and closed fuel cycles.

This publication is a result of consultancy and technical meetings, and presents a general consensus of the participating experts on the best common practices that can be used at nuclear power plants in reload design and core management.

The IAEA wishes to thank all of the experts and Member States involved for their contributions. The IAEA officer responsible for this publication was H. Varjonen of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

There is increased interest among owner/operating organizations of IAEA Member States in making electricity generation by nuclear power plants (NPPs) more efficient and economical.

This publication outlines the main issues to be considered when developing and improving reload design and core management in NPPs. It also provides good practices on reload design and core management in operating NPPs and describes challenges which may be encountered.

This publication reflects the advice of experts with experience of reactors of various types on how the operating organization may achieve its goals.

The term 'core design' is used to describe activities concerning loading patterns at existing reactors with known core geometry, a fixed number of fuel assemblies (FAs), and related management activities. Issues related to the design of new cores and new FAs are discussed in NS–G–1.12. *Design of the Reactor Core for Nuclear Power Plants* [1]. The term 'core designer' is used for an individual or an organization in charge of the core design.

1.2. OBJECTIVE

This publication provides information regarding good practices and recommendations on reload design and core management.

The objectives of this publication are:

- To collect recent information on fuel and core design in the management and operation of nuclear power plants;
- To identify and address important issues to optimize fuel parameters and the operating cycle;
- To discuss non–routine core design, e.g., redesigning the core during operation due to unforeseen issues including damaged fuel, changes in nominal operating power and cycle length;
- To highlight up to date best practices related to core management, operating experiences and lessons learned, as collected from Member States;
- To provide recommendations for core reload design and core management.

1.3. SCOPE

This publication is intended to describe general features of NPPs' nuclear fuel cycles and core management, taking into consideration safety aspects and fuel cycle economy. Guidance provided here also covers practices for different reload strategies and how to optimize reactor reload design and core management during the lifetime.

1.4. STRUCTURE

Section 2 gives an overview of general features of the nuclear fuel cycle of nuclear power plants and basic information on core design. Section 3 focuses on reload safety and its requirements, as well as on operational efficiency, regulatory requirements and international recommendations concerning the safety of nuclear reactors and nuclear power plants, requirements for the safe storage and transportation of spent FAs as well as reload design limits. Section 4 deals with fuel cycle economy and different concepts of optimization of the fuel cycle, and different cost elements of nuclear fuel production, comparing fresh fuel reload cost per kWh with the cost of enrichment in different nominal power and cycle lengths. Section 5 is focused on practices for reload strategies and prerequisites for determining core design. Section 6 discusses fuel reliability and how leaking fuel impacts on fuel core and its economy and fuel cycle and back end strategy. Section 7 considers fuel development and new materials to mitigate hydrogen release from zirconium. Section 8 presents operating experiences from some Member States. Each chapter contains a conclusion summarising the topic.

2. PRINCIPLES OF IN-CORE FUEL MANAGEMENT

2.1. GENERAL DESCRIPTION OF THE REACTOR CORE AND FUEL MANAGEMENT

Fuel utilization in nuclear power plants differs greatly from those at other base-load power generating facilities. While fuels utilized in said facilities — i.e., natural gas, coal and various oil-derivatives — are continuously fed and combusted to produce power, each nuclear fuel assembly must be individually tailors to fit an overall core design scheme.

The majority of operating nuclear power plants, as well as those that had been in operation earlier or are under construction at the moment, are using fuel consisting of uranium dioxide (UO_2) (ceramic fuel pellets arranged in fuel columns) placed into zirconium or stainless steel alloy cladding tubes forming a fuel pin. Fuel pins loaded with UO_2 fuel rods are arranged in structure of FAs of varying geometry and size, thus the fuel assembly forms a single movable and replaceable fuel assembly.

Figure 1 illustrates a pressurized water reactor (PWR) fuel assembly. The FAs are used in boiling water reactor (BWR) and water cooled water moderated power reactor (WWER) type reactors have similar designs in the geometrical configuration, most notably differing in the total number and dimensions of the individual fuel pins. The fuel bundles used in heavy water moderated Canada deuterium–uranium reactor (CANDU), gas cooled MAGNOX reactors and graphite moderated water cooled high power channel type reactor (RBMK) reactors — though having a differing geometry, are designed based on the same design basis concepts.

FIG. 1. PWR Fuel assembly.

The main components of a nuclear reactor core are: fuel (including fuel pins and assembly structure), the coolant/moderator, and reactivity control mechanisms (chemical shim, burnable poisons, and control rods). The main components are supported by ancillary components such as the reactor pressure vessel internals i.e. coolant inlet plenum, fuel pin spacers, core support plates, and the lower and upper internal structures in light water reactors. The fuel is specifically designed for each given reactor type and fabricated to meet the needs of the overall core design. Figure 2 shows the relative size of a PWR reactor core.

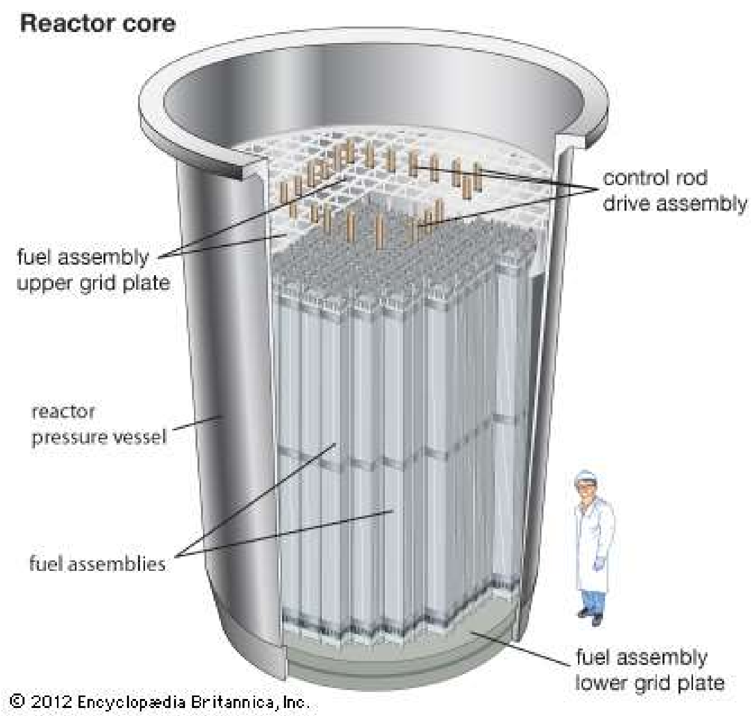

FIG. 2. PWR reactor core (By courtesy of Encyclopaedia Britannica, Inc., copyright 2012; used with permission).

Fissile materials in the fuel are the source of the fission chain reaction leading to excess heat being transferred to the coolant and excess neutrons. The ratio of neutrons generated from subsequent neutron generations, i.e. reactivity, is compensated for by different control means (control rods, burnable poison integrated to the fuel and absorber material dissolved in the coolant) in order to keep the reactor in a stable critical condition- even at maximum nominal power level. Later during the course of reactor operation, fissile material depletion in the fuel assembly will require either it be re-located within the core or replaced entirely. The replaced irradiated FAs are then discharged and stored in the spent fuel pool, followed by either reprocessing or final disposal depending on the fuel cycle scheme (closed vs open, respectively).

The scheme for a theoretical fuel reload can be seen in Figure 3. The appropriate rearrangement of the FAs remaining in the core may contribute to maintaining the excess reactivity and can be considered a special condition for the fuel replacement. It should be noted that just because a fuel assembly has been removed from the reactor core, it does not mean that it cannot be utilized in a future fuel reload scheme.

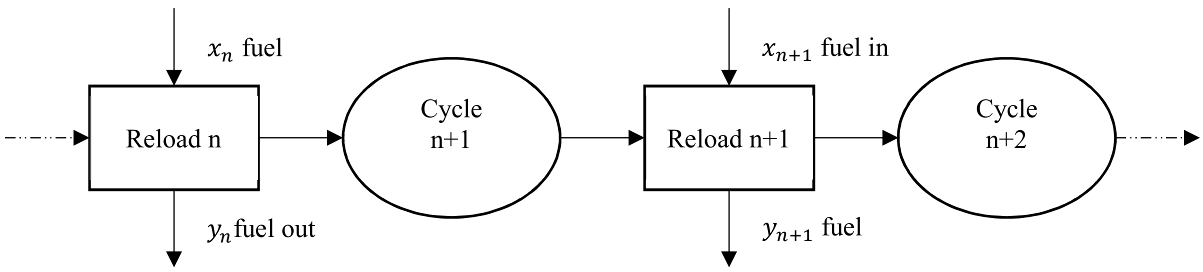

FIG. 3. Typical reloading scheme.

Refuelling strategies for nuclear power plants can be divided into two cases: during reactor operations and during a reactor outage.

- In the first case, the refuelling and rearrangement of fuel can be done during reactor operation. This group includes CANDU and RBMK reactor types. One of the operating advantages offered by these reactors consists in the fact the there is no need for high installed excess reactivity which may cause difficulties related to reactivity control under certain accident conditions;
- In the second case, the refuelling and rearrangement of fuel can be done during reactor off-power (during the outage) Including LWR type reactors.

2.2. PURPOSE OF RELOAD DESIGN

The primary purpose of the nuclear power reactor operation is electricity generation. The operation of power units should serve this purpose, taking into consideration the specific features of the units and their design limitations. Generally, well–designed operations optimised for maximum reactor operation times contributes to the most economical method of electricity generation. Nevertheless, the reality of the operating conditions and management needs (both internal and external to the reactor core) can never be predicted precisely; as a rule, there are always uncertainties. One of the possible operation modes of the nuclear power units is the equilibrium cycle, though strictly speaking, the conditions thereof are never fulfilled exactly as they have to be. In other cases, the equilibrium is not the purpose; the required cycle is determined by other conditions. Consequently, it can be concluded that in case of LWR reactors, the fuel reload is more or less unique and the core after reloading is characterized, correspondingly, by unique reactor physics characteristics. These characteristics significantly change later during the operation cycle. These reactor physics characteristics have a direct impact on reactor safety and operational efficiency.

The goal of proper reload design is to develop — with the help of adequate modelling — such fuel reload schemes as to assure that the designed core:

- Meets the requirements for electricity generation;
- Corresponds to the safety analysis conditions at the given power unit;
- Will be economically efficient in terms of fuel utilization during reactor operation.

These requirements can be fulfilled only if every fuel reload and cycle is designed individually.

2.3. LIGHT WATER REACTOR FUEL MANAGEMENT

The most frequently built light water reactors (LWR): PWR, WWER, BWR, operate with regular fuel reload schemes or ‘cycles’ in English and ‘campaigns’ in Russian and French. For refuelling, it is necessary that the reactor has been stopped to allow for the opening of the reactor vessel, a rather time–consuming process, which is why refuelling is done alongside corresponding maintenance outages. During the refuelling outage, a significant part of the fuel assemblies are replaced (i.e., one third, one quarter or one fifth, depending on the fuel cycle), after the refuelling the reactor is normally scheduled for continuous operation lasting between 12 and 24 months. An important specific feature of nuclear reactors consists in the fact that both following the refuelling and during the subsequent cycle the reactor neutron and thermal characteristics change and will be unique at every single moment of time, so they should be designed and controlled individually.

During core reloads conducted during maintenance outages, a portion of the irradiated FAs from the reactor core is replaced by fresh fuel. The new core is characterized by a high excess reactivity, which is moderated with the help of neutron absorbers in the coolant, control rods, or within the fuel itself via burnable poisons. The excess reactivity decreases during operation and when it is impossible to operate at the nominal power, the cycle comes to an end. The reactor will shutdown and the cycle repeats.

The cycle length is governed by the excess reactivity present at the beginning of cycle (BOC). The cycle length therefore depends on the number of the loaded fresh FAs and their enrichment, on the isotopic content of the remaining irradiated fuels, as well as on the loading pattern and overall core geometry (it will be discussed herein below).

During normal operations of a nuclear power plant, it is normal to have a combination of fresh fuel and fuel that have been burning for multiple cycles.

The situation varies slightly during the first startup of the reactor- the first core is considered as a special case. In order to achieve the equilibrium immediately after the reactor startup, and to assure the proper power density distribution in the reactor, the initial core is always made up of the FAs with different enrichments.

It is expedient to adapt the cycle length whenever possible to the calendar, and to design the reactor fuel and core load schedules, correspondingly. Thus, there are 12 to 24–month cycles and reactors operating with the corresponding cycle lengths.

In principle and in practice, the cycle length may vary within a very wide range, so 12 to 24-month cycles can be applied at one and the same reactor. However, this does not mean that the lengths of consequent cycles may be selected freely; design and planning of significantly different lengths may lead to serious challenge. It is general to strive for the uniform cycle length.

By means of design calculations or a simple modelling, one can see that if during each cycle the fuel is replaced with the same quantity and enrichment in the reactor, the fuel is arranged always in one and the same manner and the reactor is then operated until the physical end of the cycle, then due to the negative feedback effect of the burnup the equilibrium condition comes to exist; this phenomenon is called equilibrium cycle. The length of equilibrium cycles is always identical, the power density and temperature distribution will be identical at each cycle and the burnup of FA discharged at the end of cycle will be nearly identical as well.

The NPP operators typically plan for 12 to 24–month equilibrium cycles adapted to the calendar year and design the nuclear fuel and core load scheme correspondingly. However, it is very difficult to maintain the cycle schedule precisely as slight deviations from equilibrium, leading to potential reactor shut downs, are caused by operational and maintenance uncertainties or other anomalous events. Significant deviations from one reload strategy to another may affect how an equilibrium cycleis achieved. Such deviations can be caused, for example, due to a change of nuclear fuel strategy (e.g., transition from a 12–, to 18–month cycle). After such deviations, the operators need to achieve a new state for the reactor core, close to the equilibrium.

General features of the light water reactor fuel cycle in the equilibrium state or a state close to equilibrium are shown in Figure 4. Figure 4 is an example for WWER–440 case.

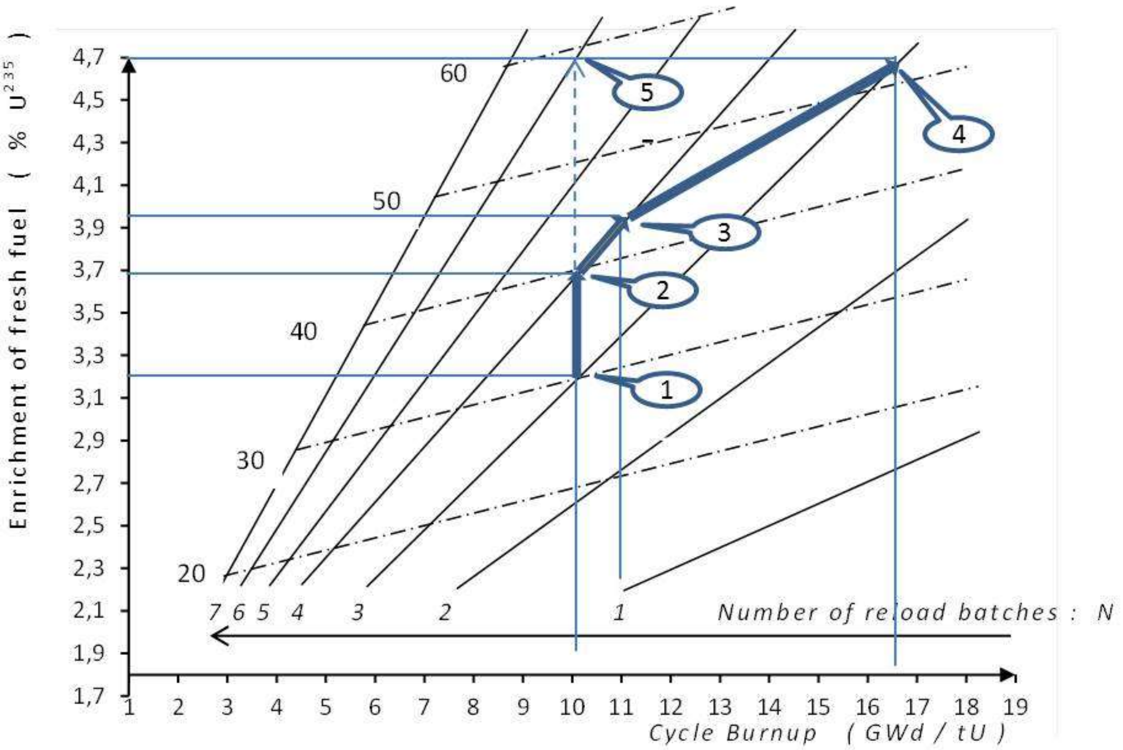

FIG. 4. Relation between cycle burnup, fresh fuel enrichment and reload batch rate.

In the simplest case, the general composition of material, geometry and efficiency of the FAs are governed by the U–235 (or other fissile material such as plutonium in the case of mixed-oxide (MOX) fuel) content of the fuel. In the X–axis of Figure 4 the average burnup is shown per cycle, which is proportional to the cycle length with the given nominal power. The Y–axis shows the required enrichment of the fresh nuclear fuel needed to maintain an equilibrium cycle. The solid lines in Figure 4 show the fraction of the load replaced during one refuelling, i.e., the number of cycles spend by a fuel element in the reactor. The iso–curves marked by dotted lines shows the average burnup of the fuel discharged at the end of its lifetime.

This graph is of a general explanatory nature and, in the case of other reactors, the axes on the graph may be shifted as to those demonstrated above, however, the trend will be identical:

- At the beginning of the operation, the units usually operate based on an annual cycle and require a 1/3 reload (3–batch reload strategy): During one reload carried out yearly, 1/3 of the fuel is replaced (Point 1 in the Figure 4). Typically, to do so the fresh fuel with the 3.2 wt % enrichment was required, and the final burnup of the discharged fuel made about 30 GWd/tU. In order to increase the fuel utilization efficiency, the transition to the 1/4 reload strategy was done. For this purpose, the required enrichment was ca. 0,5 wt % higher and the burnup of the discharged fuel reached approximately 40 GWd/tU (Point 2 in the Figure 4);
- Later, the majority of LWR units have undergone the power uprate program. Considering a 10% power uprate in our figure, the burnup increases to the value of 11 GWd/tU per each cycle. In order to continue using the 1/4 reload strategy, the

enrichment should be increased by another 0.25 wt %. The burnup of the discharged fuel comes up to 44 GWd/tU (Point 3 in the Figure 4);
- In order to increase the Unit Capability Factor (UCF), many LWR units have changed their operating cycles to exceed one year. In case of the unit shown in our figure, an 18-month operation cycle can be implemented in a manner providing for the use of the 4.7 wt % enrichment along with the 1/3 reload strategy (Point 4 in the Figure 4). The burn–up of the discharged fuel will be higher than 50 GWd/tU;
- It should be noted that if we operated with the use of a yearly operation cycle and original nominal power at the demonstrated power unit, then we would be able to use the 1/6 reload strategy with the same fuel enrichment, achieving the burn–up of approximately 60 GWd/tU (Point 5 in the Figure 4).

2.4. HEAVY WATER REACTOR FUEL MANAGEMENT

The majority of PHWRs have horizontal fuel channels, analogous to assemblies, that utilize natural uranium fuel (i.e. CANDU reactors) bundles. A series of bundles are loaded into each channel. Refuelling is typically carried out during active, or on–power, reactor operations. As each fuel bundle reaches its final burnup, the bundle is pushed out of the channel as new fuel is loaded into the opposite end of the channel. This makes the in–core fuel management substantially different from those of LWR's.

The primary objective of CANDU in–core fuel management is to determine fuel–loading and fuel–replacement strategies in order to operate the reactor under safe and reliable condition while keeping the total unit energy cost at a minimum. Within this context, the specific objectives of CANDU in–core fuel management are as follows:

- Adjust the refuelling rate to maintain reactor criticality;
- Control the core power to meet safety and operational limits of the fuel and channel power;
- Maximize burnup within operational constraints, to minimize fuelling cost;
- Avoid fuel defects, to minimize replacement fuel costs and radiological occupational hazards.

The capability for on–power refuelling means that excess reactivity requirements are minimized: only a few hundred pcm are necessary for continuous and short–term reactivity control. This leads to excellent neutron economy and low fuelling costs but high costs for spent–fuel treatment.

To refuel a channel, a pair of fuel loading machines latch onto the ends of the channel. Four or eight fresh fuel bundles, depending on fuelling strategy, are inserted into the channel by the machine at one end, and the same number of irradiated fuel bundles are discharged into the fuelling machine at the other end of the channel. For symmetry, the refuelling direction is opposite for next–neighbour channels. In the CANDU–6 reactor, the refuelling direction is the same as that of coolant flow in the channel.

3. RELOAD SAFETY

3.1. RELOAD DESIGN CONSIDERATIONS

3.1.1. Safety goal

There exist international recommendations concerning the safety of nuclear reactors and nuclear power plants. The peaceful cooperation on nuclear energy between countries is outlined by various international agreements. The summary of these can be find in Ref. [4]. Relevant legal requirements are laid down at the level of legal acts of the certain states, in the form of nuclear safety rules. Tasks pertaining to the elaboration and enforcement of the respective rules and regulations belong to the scope of competence of the nuclear authority of a certain country which operates the given nuclear facility on its territory.

The national regulatory requirements are formulated on a general level.

These requirements are typically as follows:

(a) In case of anticipated operational occurrences, the radiation dose rate affecting a certain population group shall not exceed the specified limiting value;
(b) The probability of unanticipated operational occurrences shall be below a certain specified limit;
(c) In order to provide the fulfilment of the above, operating conditions and limits shall be elaborated in respect of the equipment systems and system components. The latter include also the reactor and nuclear fuel characteristics as well;
(d) Beyond the above, the general regulatory requirements put a special focus on the reactor and fuel safety, specifying that it should be:

- Controllable;
- Should be characterised by favourable self–control properties;
- Should assure the possibility of shutdown and cooling of reactor.

In addition to those described above, there could be more detailed legal requirements specified both at the general and lower regulatory levels as well, e.g., in the form of recommendations. However, said recommendations do not specifically to a reactor's reload design. At the same time, it is obvious that based on the general requirements listed previously, one can formulate, after the definition of normal operation and design basis accident conditions, the requirements for the reactor physics characteristics described in 3.1.1, as well as for the characteristics describing the physical condition of the nuclear fuel. The latter in a direct manner determine one of the purposes pursued by the reload design calculations.

3.1.2. Fuel loads and safety

As mentioned in Chapter 2, the nuclear characteristics of a reactor core vary for each reload; moreover, they significantly change throughout the cycle having a significant impact on the reactor safety and operational efficiency. Let us consider the most important characteristics. Firstly, we mean here such global characteristics of the reactor core as:

— Reactivity margin for a cycle;

- Criticality–related characteristics, critical boron concentration, and critical control rod position;
- Shutdown subcriticality and re–criticality (after fast cooldown) features;
- Efficiency of control tools (boric acid–, control rod efficiency);
- Feedback characteristics: moderator, coolant, fuel temperature coefficients of reactivity.

Secondly, there are also local core characteristics having a unique nature, like:

- Fuel assembly and fuel rod wise power, axial power distribution;
- Coolant characteristics (pressure and temperature) in certain volume, in the sub–channels between fuel rods.

All those above mentioned characteristics, coupled with the process and safety equipment of the given power unit, determine the safety of the reactor unit both under normal operating conditions and during accident scenarios. As long as the process characteristics are defined, the core characteristics will change as described above.

The primary purpose of the reload design calculations is to design core loads that would guarantee a proper safety level of the reactor operation- both under normal operating conditions as well in accident scenarios.

3.1.3. Considerations from thermal mechanical aspects

As a rule, the reactor designer and nuclear fuel manufacturer specify limits for the core and nuclear fuel conditions. These limits are usually built into the national regulations as description and details of paragraph (c) in 3.1.2.

During the operations of NPP units, main parameters such as power, temperature, pressure, vessel levels, etc. are regulated in detail along with conditions of safety systems, protections and operator actions. All these rules and constraints determine the safety cases taken into consideration during the safety analysis. The operator actions appear in the form of operating instructions and may be taken into consideration in the analyses.

In the sense of paragraph 3.1.2 (c) above the nuclear fuel manufacturers set up design requirements for the application of the nuclear fuel.

For example, these can be as follows:

- Strength criteria (stress–corrosion cracking (SCC) of the fuel cladding, cladding collapse, fatigue strength, plastic deformation);
- Deformation criteria (change of diameter, elongation);
- Thermal physics criteria (fuel temperature, gas pressure inside the cladding, linear power and maximum enthalpy);
- Oxidation, hydrating, fretting, catches, welding, bundle stability;
- Cladding embrittlement as the result of high temperature and oxidation;
- Burnup limits; etc.

At the reload design phase, the linear power (or fuel channel/bundle power limit for CANDU) and maximum burnup serve as the direct design limitation. The fulfilment of a greater part of the design requirements can be confirmed through the analysis and evaluation of the normal operation, transient and accident scenarios. For the purpose of these evaluations, the fulfilment of secondary criteria is required. For instance, the maximum fuel assembly —and fuel rod–power, maximum local coolant temperatures, speed of power change, etc.

3.1.4. Considerations from impact on the handling, storage and transport of spent fuel assemblies

From the safety point of view, the condition of nuclear fuel discharged from the reactor, stored in the spent fuel pool, interim or final spent fuel storage and/or transported in various containers is determined by its final isotopic composition which is presented by the characteristics that change in the course of time:

- K–effective value;
- Residual heat;
- Radiation exposure (dose rate: primarily, gamma and neutron);
- Appearance of hydrogen (under alpha irradiation).

Reload design affects spent fuel (storage and transport) safety trough the following parameters:

- Max. k_{eff} value of used fuel. Changes in the fuel design, e.g., enrichment, mass, material changes need re–assessment of criticality of transport and storage;
- Reload quantity is usually handled conservatively in safety analysis of transport and storage. However, limitations of spent fuel safety analysis (SA) should be known;
- Max fuel burnup, cycle length, core nominal power may have impact on short and long–term residual heat and radiation features of spent fuel. Limitations of existing conservative SA should be taken into consideration.

After a specified time period, the spent fuel is transferred to the interim storage, assuming the spend fuel meets the requirements outlined above. The fulfilment of the above requirements can be also assured with the cooling time during which the fuel is stored in the spent fuel pool (SFP) prior to the transportation.

It is important that all the interrelations of the entire process should be taken into consideration. The limits pertaining to the interim storage have impact on the required cooling time, temperature and dose conditions of the spent fuel pool, through the characteristics of the FA discharged after each cycle (enrichment, weight, burnup, quantity).

In case of the given fuel and refuelling strategy, spent fuel management can contribute to the establishment of the burnup limit for the core design and minimal holding time for storing in the spent fuel pool.

If the long–time management strategy involves reprocessing, the burnup of the discharged fuel and time before reprocessing begins may have a significant impact on the isotopic composition.

The basic requirements for CANDU spent fuel storage and transportation are nearly the same as mentioned above. Below are a few additional considerations for spent fuel storage for CANDU fuel, as outlined in [5]:

- Both the fuel and the materials used in storage (wet or dry) and structures and facilities have their respective temperature limits for safe operation. Therefore, the potential temperature increase caused by decay power must be controlled effectively to prevent overheating;
- Radiological contamination must be controlled to acceptably low levels to protect the workers and the public. Therefore, degradation and damage to bundles must also be limited to acceptably low levels;
- The fuelling machines must be designed and operated so that the irradiated bundles are not damaged;

- Even though CANDU fuel is made of natural uranium oxide and Zircaloy, it cannot be put into a configuration that will achieve criticality in ordinary water;
- The most demanding conditions exist when the fuel is handled during the initial wet and dry storage phases; special shielding must be designed to protect workers from radiation exposure;
- IAEA–approved monitoring by remote means is used.

3.1.5. Safety analyses influencing core and fuel design

The operating conditions for a nuclear power unit are defined by the relevant safety analyses prepared accordingly. One of the main tasks for core engineers managing the reload design scheme is to assure the fulfilment of the acceptance criteria established by the safety analysis.

When carrying out safety analyses, the examination of radioactive release is performed rather rarely since most examined events are not accompanied by a radioactive release. A more frequently used practice is to establish an acceptance criteria for the assessment of the analysis results.

Such criteria are established for the characteristics listed in the previous compliance that guarantees the fuel integrity during operation and in spent fuel storage.

Some acceptance criteria may be as follows:

- Minimum DNBR characterizing the coolant;
- Maximum quantity of the leaking fuel rods;
- Maximum primary pressure;
- Maximum primary temperature and pressure, with consideration to embrittlement;
- Containment pressure;
- Time sufficient for the operator intervention;
- Long–time coolability.

Finally, one of the tasks of the reload design is to assure the fulfilment of the acceptance criteria established by the safety analysis.

3.1.6. Reload design limits

The designed reload exerts significant influence on the final results of the safety analysis- the neutron and thermal physics characteristics of the reactor core differ for eachreload. The task of the reload design is to assure the fulfilment of the acceptance criteria established by the safety analysis for the given cycle.

The direct manner means that the safety cases considered in the safety analysis is referred during the reload design, with consideration to the actual values of the neutron and thermal physics characteristics for the given cycle. Some certain conservative considerations are necessary in any case since it does make a significant difference whether the given scenario is reviewed for the beginning, middle or end of the cycle.

This solution is used rather rarely and pertains only to the certain selected part of the analyses.

The indirect method is used more widely. In this method, safety analyses are carried out assuming conservative values for the neutron and thermal physics parameters, while the load design process should assure that the value of the given parameter are within the boundaries assumed in the safety analysis.

These parameters can be named the reload design limit. Tables 1–3 below provide the frequently used reload design limits and the list of events that are influenced by the values of these parameters.

TABLE 1. MAXIMUM LOCAL POWER AND BURNUP

Parameter	Event
Maximum linear thermal power, maximum fuel rod power, maximum fuel assembly power, Maximum Linear heat generation rate (LHGR)	Changes in the secondary heat removal. Unintended operation or ejection of the control assembly. Loop false startup. Dry out. Main circulation pump overspeed. Instability. Loss of primary coolant accidents. Fuel rod pressure at end of life.
Maximum power jump (ramp) of pins	As a result of jump in a linear power of certain elements of fuel rods after refuelling or during power transient
Maximum fuel rod burnup, pellet burnup	Describes the fuel condition at the end of the burnup cycle for the purpose of operational, transient and accident analyses.
Maximum fuel assembly burnup	Requirement connected with the residual heat production and radiation exposure parameters of the storage and transportation means.

TABLE 2. FEEDBACK PARAMETERS

Parameter	Event
Maximum Moderator Temperature Coefficient (MTC) of reactivity	Reduction of heat removal on the secondary side. Uncontrolled movement of the control rods. Loss of primary coolant accidents.
Minimum Moderator Temperature Coefficient (MTC) of reactivity	False startup of the low temperature non–operating loop. Increase of the secondary heat removal.
Maximum value of the fuel temperature coefficient of reactivity	Increase of the secondary heat removal. Uncontrolled movement of the control rods. Control rod drop Loop false startup.
Minimum value of the fuel temperature coefficient of reactivity	Reduction of heat removal on the secondary side. Loss of primary coolant accidents.
Maximum value of the boron coefficient of reactivity	Increase of the secondary heat removal. Loss of primary coolant accidents.
Minimum value of the boron coefficient of reactivity	Loop false startup. Failure of the feedwater and boron control systems.

TABLE 3. CONTROL ROD RELATED PARAMETERS

Parameter	Event
Maximum efficiency of one control rod under different reactor states	Control–rod bundle ejection. Control–rod drop (BWR–case).
Maximum and minimum efficiency of the control rod group, maximum (and minimum) value of the efficiency of the regulating rod group	Uncontrolled movement of the control rods. Failure of one control–rod group during the scram (BWR–case).
Shutdown reactivity	Change in the secondary heat removal. Uncontrolled movement of the control–rods. Loop false startup. Loss of primary coolant accidents.
Minimum subcriticality	During refuelling condition
Re–criticality temperature	Increase of the secondary heat removal.

The list of above mentioned parameters can be further extended with additional elements; the entire list can be called the frame parameters. According to the standard approach, whenever the frame parameters of the reload core fall within the values assumed at the safety analysis, then the validity of the safety analysis is considered as confirmed. In other aspects, correspondingly, the frame parameters are the reload design limits.

In such a way, the task of the core design, which is related to the safety, has been reduced to the confirmation of the fulfilment of the reload design limits.

3.1.7. IAEA Safety Standards

The IAEA has developed a set of Safety Standards that comprise the Fundamental Safety Principles, Safety Requirements and Safety Guides:

- Fundamental Safety Principles [6] defines the safety objectives for global safety and presents principles to be comply with;
- Safety Requirements describe what to be met to comply with the safety objectives and fundamental principles; and
- Safety Guides describe how to meet the related safety requirements.

Safety requirements associated with the reload design, core management and the fuel handling and storage systems are provided primarily in the *Specific Safety Requirements SSR–2/1 Rev. 1* [7], *Safety of Nuclear Power Plants: Design and partly in the Specific Safety Requirements SSR–2/2 Rev.1*, Safety of Nuclear Power Plants: Commissioning and Operation [8].

Requirements 43 to 46 established in SSR-2/1 (Rev. 1) [7] are the most relevant for the load and reload design to take into consideration by the reactor engineers and operators. It also addresses the interface with core management, which strongly influences the core design with regard to the performance of fuel rods and fuel assemblies. Requirements 43 to 46 of SSR-2/1 (Rev. 1) [7] are related to: the Performance of fuel elements and assemblies; Structural capability of the reactor core; Control of the reactor core and Reactor shutdown. Requirement 80 of SSR 2/1 (Rev. 1) [7] states that *"Fuel handling and storage systems shall be provided at the nuclear power plant to ensure that the integrity and properties of the fuel are maintained at all times during fuel handling and storage."*

The Safety Guide, Design of the Reactor Core for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.12, IAEA, Vienna (2005) [1] addresses the safety aspects of the core design including neutronic, thermohydraulic, thermomechanical, and structural mechanical aspects relating to reactor core control, shutdown and monitoring, and core management for the safe design of the reactor core for nuclear power plants. This safety guide provides recommendations on the reload design and core management in water–cooled power reactors.

The IAEA Safety Guide, Design of Fuel Handling and Storage Systems for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.4, IAEA, Vienna (2003) [9] provides recommendations for the design of fuel handling and storage systems in nuclear power plants, addressing the design aspects of handling and storage systems for fuel that remains part of the operational activities of a nuclear reactor.

IAEA Safety Standards Series No. NS-G-2.5, Core Management and Fuel Handling for Nuclear Power Plants [10] provides recommendations to meet the related operational safety requirements

Safety guide addressing safety requirements in the design of the reactor core for NPPs and guidance on the reload design and core management in water–cooled power reactors included in *Specific Safety Guide NS–G–1.12. Design of the Reactor Core for Nuclear Power* [1].

Consolidated description of these guidance is described as follows:

- The primary objective of core management is to ensure the safe, reliable and optimal use of the fuel in the reactor, while remaining within the operational limits and conditions. Therefore, each reloading cycle should be designed with appropriate means of controlling the core reactivity and the power distribution to address fuel design limits;

— Reload design objective is discussed in Chapters 2.3 and 2.4 of this technical publication; and
— Parameters associated with depletion of fuel and burnable absorber and other nuclear parameters are provided as input to safety analyses, plant monitoring and protection systems, and operator guidance. Therefore, these parameters should be analysed based on pre-determined plant operating objectives and resultant plans. These nuclear parameters include the reactor startup conditions (e.g., critical boron concentrations and control rod positions, reactor kinetics, fuel temperature coefficients, moderator temperature coefficients, control rod and control bank worth's, power peaking factors).

Connections between core parameters and safety analysis are widely discussed in Chapters 3.1.1. to 3.1.5. Reload design limits serving as inputs for safety analysis is listed in 3.1.6.:

— Unplanned power manoeuvring during flexible operation may alter the power and burnup profile across the core. As such, predictions of parameters associated with depletion of fuel and burnable absorber and other reactor physics parameters should be continuously or periodically examined and evaluated, using relevant monitoring parameters.

In–core monitoring features fulfilling these requirements is discussed in 3.2.5.:

— The design of the reactor core should include analyses to demonstrate that the fuel management strategy and the established limitations on operation are such that the nuclear design limits and hence the fuel design limits will be met during the whole reloading cycle;
— The reactor core analysis should be performed based on typical cases covering the whole reloading cycle for the following reactor core conditions:

- Full power, including representative power distributions;
- Load following (as applicable);
- Approach to criticality and power operation;
- Power cycling;
- Startup;
- Refuelling;
- Shutdown;
- Anticipated operational occurrences; and
- Operation at the thermohydraulic stability boundary (for boiling water reactors).

Requirements listed in previous two paragraphs are quite natural for experts performing reload design analysis. No specific discussion in this material required:

— Whenever the management of fuel in the core is changed or any characteristics of the fuel rods (such as the fuel enrichment, fuel rod dimensions, fuel rod configuration, or the fuel cladding material) are changed, a new reactor core analysis should be performed and documented.

Chapter 3.1.1. discusses the relation between fuel, core and safety analysis. Case of fuel and core changes are mentioned in several places of this document:

— The reactor core analysis should include analyses of the performance of the fuel rods based on average and local power levels and axial temperature distributions to demonstrate that the respective thermal and mechanical fuel design limits are met for all operational states. For light water reactors, the reactor core analysis should include analyses of peak channel power and peak linear power rates for normal full power operation and steady state radial power distribution at each fuel assembly location and axial power distributions in each fuel assembly. Allowance should be made considering the effects of changes in the geometry of the fuel assembly on its neutronic and thermohydraulic performance (e.g., changes in the moderator gap thickness due to bowing of the assembly). The reactor core analysis should also include the radial power distribution within a fuel assembly and the axial power distortion due to spacers, grids and other components in order to identify hot spots and to evaluate the local power levels.

Limits concerning power distribution are listed in Chapter 3.1.6. Changes of core geometry are usually topic of the safety analysis and consequences used as uncertainties to take into account as reserve to peak parameter limits:

— For on–power refuelling in pressurized heavy water reactors, the effects of the refuelling operation on the neutronic behaviour of the core should be demonstrated to remain within the control capability of the reactor control systems.

The requirements are discussed in 2.4.:

— The fuel loading sequence should be monitored using in–core (for boiling water reactors) or ex–core flux distribution measurements, or by means of special administrative measures. The fuel loading pattern after reloading should be validated through the measurements of the flux distribution;
— For light water reactors, the reactor core should be designed such that the consequences of the worst misloaded fuel assembly, if any, remain within nuclear design limits and fuel design limits. If a misloaded fuel assembly can be prevented by special measures and equipment, the effectiveness and reliability of these precautionary measures should be demonstrated. Computational analyses should be performed if it cannot be demonstrated that the specified precautionary measures are sufficient.

Misloaded assembly case usually investigated during safety analysis. Consequently, maximum asymmetry can be defined for reloaded core. This is out of topic of the present paper:

— The reactor core analysis should verify that the core fuel loading pattern will meet fuel design limits for all applicable plant states. For practical reasons and simplicity, for light water reactors, a system that develops and monitors the nuclear key safety parameters can be used to verify the suitability of the reload core design. The nuclear key safety parameters include but not limited to:

- Spatial distribution of the neutron flux and related power distribution peaking factors;
- Pressure of the reactor coolant system;
- Coolant temperature (e.g., inlet temperature, outlet temperature);
- Speed of the reactor coolant pump;
- Water level (for light water reactors);
- Radionuclide activity in the coolant;
- Insertion position of the control rods;
- Concentration of soluble boron or B–10 content when enriched boron is used (for a pressurized water reactor).

A more detailed and perfectly defined key parameter list is in Chapter 3.1.6. of this technical publication.

3.2. COMPUTER SIMULATION METHODS FOR THE CORE CALCULATIONS

3.2.1. Applied methods, uncertainty of the results

To accurately model both static and time–dependent processes taking place in the reactor, reactor physics, thermal hydraulics and fuel rod behaviour algorithms are necessary. Considering the special goals of the core design, as well as the characteristic dynamic nature of the associated processes to be modelled, different models are far from being identical in every case.

The time–dependent processes taking place in the reactor can be generally divided into the following groups:

(a) Reactor physics impact of the burnup: change in the isotopic composition due to the nuclear reactions, namely, the origination and transformation of the absorbing fission products and fissionable and absorbing actinides. There is a feedback loop between power and burnup which has to be considered. The local burnup and its increase is determined by the local power, which in its turn depends on the burnup and other characteristics of the fuel rod such as the enrichment influencing the multiplication factor.

(b) Origination and transformation of non–stable fission products, having a noticeable impact on the neutron balance:

- Iodine–135—Xe–135 fission product chain, with the characteristic change time: 1-2 days;
- Promethium149—Sm–149 fission product chain, with the characteristic change time: 1-2 weeks.

These latter processes play an essential role during the power changes, e.g., during the power ramp and power decrease, and similarly to the burnup have a significant impact on the power distribution and criticality parameters (critical boric acid concentration, absorber position).

(c) Rapid reactivity changes- primarily, the insertion of reactivity. The specific features of reactivity changes are as follows:

- The kinetic and dynamic processes are fundamentally accompanied and governed by the reactivity changes;
- The non–stable state originating from fission in an indirect manner through the beta–decay, with delay (so-called delayed neutrons) should be taken into consideration;
- In the case of reactivity transients, the time dependent power makes necessary the discussion of dependence of the fuel rod heat removal on its place and time. The change of the coolant temperature during a transient should be also taken into consideration;
- The characteristic time in these cases can range from 10ms up to minutes. In the latter case, the cause of the transient is usually related to a thermo hydraulic issue (adjustment of coolant flow or loss of coolant in the primary loop).

The reload design is primarily governed by factors in (a) and (b) above. In these cases, the reactor during operation at power is in a critical state, when it is sufficient to monitor thermal hydraulics and fuel rod behaviour–related processes with the heat-up models, simplified as compared to the reactor physics processes, necessary for the feedback. For such calculations, the 'criticality' corresponding to the stationary state is achieved by the proper adjustment of the boric acid concentration, or absorbent position.

According to the above said, the decisive role during the core designing process belongs to the neutron physics processes, which are in the most commonly used way described by the neutron transport equation.

As to the implemented practices, some computer algorithms and programs solve the transport equation through the application of approximations. As a rule, the following two calculation methods are applied:

- Solving the partial transport equation in the energy (e.g., multi–groups) not for the entire reactor. Based on the received results the region–wise (node) homogenization is carried out and the group constants concentrated in the energy are generated;
- Solving the few–group equation resulting from the diffusion approximation, usually for the entire reactor.

One of the methods used by the applied calculation systems consists in the 'mapping' of the reactor state with the help of the transport calculations and generating homogenized group constants for the diffusion calculations depending on the reactor parameters (temperatures, pressure, boric acid, burnup, concentrations of certain isotopes). The homogenized cross–sectional areas generated in such a way are tabulated and used in the form of interpolation of the lower level calculations or embedded function coefficients. In this way, the core design uses the library of pre–calculated cross–sections depending on the reactor parameters, for the given fuel type and enrichment.

Temperature, boron, etc., changes may cause spectral differences and affect the actual fuel parameters. However, it is possible to use pre–calculated few–group cross section data on more sophisticated way to consider specific burnup route of a given fuel assembly or node. Cross section libraries may have a kind of tree–structure and choosing the best route from this structure makes possible to follow more realistic burnup history.

Instead of the transport calculations there is a possibility of using the Monte–Carlo method. Basically, both the transport and Monte Carlo methods use the ENDF (less frequently, JEF) libraries of cross–sections. These libraries change and are further developed, new versions thereof become available, with the basic data being more precise. Nevertheless, one should be aware that due to both the uncertainty of the basic data and simplifications applied by the methods, the results of our calculations will be burdened with some defined inaccuracy. The inaccuracy with the above mentioned error sources are increased by the fact that fuel characteristics related to its material and geometry, as well as its actual state (power, temperature, pressure, etc.) are known only to a certain limit of accuracy. Consequently, it is very important to know the accuracy of the parameters defined by our apparatus used for the reload design calculations.

For CANDU reactor, RFSP–IST code for finite–reactor physics analysis and design, WIMS–IST code to generate cross–sections for a nodal node, and MULTICELL code for the cross–section data of reactivity devices has been developed in heavy–water–moderated condition. The method for calculating whole core power distribution along with reactivity feedback parameters (temperature, burnup, poison materials, etc.) and taking into account six delay neutron group combined with dynamic neutron behaviour is essentially the same as mentioned above. In particular, RFSP can handle additional 11 delay neutron groups resulted from gamma–ray interaction with D_2O moderator.

3.2.2. Verification and validation requirements

When applying the parameters assuring the connection between the core design and safety analysis, it is necessary to provide such safety bands (margins) which would take into consideration the calculation uncertainties as well. Due to this reason, the validation of calculation systems and quantification of uncertainties is of a key importance. The following can be used for the multi–level validation:

- Parameters measured on critical systems with zero power (critically, distribution of reaction frequencies);
- Mathematical benchmarking tasks;
- Measured plant and operational data: critical boric acid concentrations, temperature and boron coefficient, efficiency of the control mechanisms, distribution of measured power and temperature, etc.

For validation, it is necessary that the accuracy of parameters received during the calculations should be determined and taken into consideration later during the use.

There are international recommendations in respect of the verification and validation of computer programs; such are outlined in the regulations of competent authorities of individual countries. However, the requirements specified in these regulations can vary: these can be the requirements of a general nature (e.g., in Finland, only V&V action is required, code author’s responsibility, how to do it), the performance of the given benchmark package (Germany) or the official certification of the applied programs (Russia, USA) can be required.

Independent on the solution specified by the requirement, one should not forget that even the validation may have a limited validity. Application of the modified fuel, other types of absorbers and materials differing from those used previously, modification of process parameters require that the accuracy of given computer code should be reviewed.

In the case of CANDU, all computer codes for safety analysis are managed by CANDU Owners Group. Utilities engaged in the COG R&D Industry Standard Toolset program supports the fund for developing and maintaining those computer codes and subcontractors controlling source codes has to do all quality assurance activities according to *Canadian Standard CSA–N286.7–99* [11] and their own quality assurance manual. Every year COG prepares code–wise development plans reflecting utility requests including validation and verification, assigns them to appropriate subcontractors and monitors the progress of those projects. V&V of computer code requires extensive numerical comparisons based on various measurement data from commercial and research reactors along with a line–by–line QA tests requested by IEEE standards. Therefore, the validation process followed for all these codes is well consistent with the industry–wide practices as defined in the Technical Basis Document.

The code validation process starts with a validation plan which identifies and discuss the applications of the code for modelling each phenomenon and the experimental or analytical data sets for validation of such applications. A series of validation tasks are then carried out, each focused on testing the code against a specific data set. Note that in a realistic operational manoeuvre or postulated accident transient, very often several phenomena occur simultaneously. The same situation is also reflected in the measurement data used for code validation. One validation task often addresses more than one phenomenon. The results from the various validation exercises have been summarized and collected in the computer code validation summary. The uncertainty of RFSP code is quantified through V&V process and applied to safety analysis in terms of penalty.

3.2.3. Start–up measurements after refuelling

Over the appropriate reload design made by properly validated computer tools another important part of core safety is the neutronic and hydraulic measurements. The purpose of these measurements is dual:

- Measured parameters can be directly compared to limit values;
- Measured values can be compared to calculated ones.

Both cases important to take into account measurement and also calculation uncertainty.

During the startup phase there are some important measurements made. These also vary from country to country and from reactor type to reactor type, but some measurement regarding when first criticality is achieved with respect to pre–calculations are performed and also calibration of the in–core measurement system for neutron flux is performed. The SCRAM system is also tested.

Measurements provided during core start–up after refuelling are important from the reactor safety point of view. The first experiment is criticality achievement using boron dilution process or control rod withdrawal, which represents the changeover from neutrons subcritical to critical. Critical state parameters are basic core status parameters for computer codes direct precision evaluation.

Next selectively carried out experiments are concentrated to the reactivity feedback coefficient (coolant temperature, coolant pressure, boric acid concentration), control rod system worth (scram with one most effective rod stuck in upper position, full scram, control rod group, ejected control rod) and core power distribution symmetry (at zero power, at defined power levels) measurement. Each of provided experiments has their own acceptability criteria [8, 12].

In–core thermocouple measurement is calibrated at zero power level.

3.2.4. In–core/ex–core monitoring

Monitoring in–core and ex–core parameters of core during operation plays the same rule in safety as mentioned in previous subchapter.

All reactors have some way of monitoring in–core neutron flux and power fields. These may consist of measuring channels that contain thermocouples, transvers in–core probes or others. The results from these measurements are compared to pre–calculations in order to verify fuel and core behaviour.

The PWR and WWER core monitoring system is divided to in–core and ex–core monitoring ones.

In–core monitoring systems are based on different detectors, which measure core power distribution parameters by direct (neutron flux response) or indirect (coolant temperature) means. Different neutron (self–powered neutron (SPND), activation, etc.) and temperature (thermocouple, etc.) detectors are utilized. Both of used methods are used to validate the core power distribution modelled prior to the reload phase.

Ex–core monitoring systems are based on ionization chambers present on the outside of the reactor vessel, which are used for reactivity evaluation during startup and core power evaluation during power operation. Ionization chambers core power evaluation is calibrated on measured thermal power (e.g., from secondary circuit). This fast–evaluated core power is used as the source signal for reactor power trip system and reactor power control system.

In BWRs, there are only in–core detectors. Typical two types of detectors exist in the core, one for low power and outage monitoring and one for monitoring during operations (Local Power Range Monitor, LPRM). Depending on the reactor design, the low power detectors are either constantly positioned in the core or only inserted when the reactor enters a low power operation regime. These detectors are also used to monitor fuel movements in the core during the outage. The LPRM detectors are monitoring the neutron flux during operations and positions at several axial and radial positions in the core to get a good cover of the power in the entire core. Typically, the detectors are positions in strings of 4 axially equidistant levels connected to a guide tube for the transverse inverse probe (TIP) that is used for calibration. TIP measurements are performed monthly and results are used for detector calibration and used for validation of the online core simulator.

To monitor reactor flux shape in CANDU reactors, there are a hundred of in–core vanadium neutron flux detectors. The readings are manipulated with fundamental and several abnormal flux modes so that fuel channel power map is generated on-line. There are about 14 liquid zone controller platinum detectors to control regional power tilt. And to actuate two shutdown systems independently about 30 platinum detectors for shutdown system 1 (SDS1) and about 20 for SDS2 are installed in CANDU reactor appropriately. There are 6 ex–core detectors out of the reactor covered with pb window to monitor neutron power rate and give shutdown signal if the rate exceeds the limit.

3.2.5. Radio–analytic control

During operation there are samples taken from the primary water to be analysed in order to verify that it is within analysed scope and regulations for emission. Off gases is measured in order to detect early if there is leaker (damaged fuel) in the core.

Results from samples of primary water or off gas measurements should fulfil operational or safety limits, otherwise predetermine actions should be taken.

If results of these measurements indicate damaged fuel in the core, it may be used for early preparation for changing core reloading pattern for the next fuel cycle, especially if estimation of burnup of the damaged fuel rod is obtained.

In CANDU, the on-line radioactive monitoring system is installed to check fuel damage. If the concentration of specific materials increases during operation, the site can determine the expected location of damaged fuel and extract those fuel bundles using the on-line refuelling process which is more simple and safer than PWR damage fuel handling approach.

4. FUEL CYCLE ECONOMY

4.1. SEQUENCE OF FUEL CYCLES

Along with the reactor type and fuel design, the process related to the NPP fuel cycle can be characterized as shown in Figure 5 with a slight modification of Figure 3.

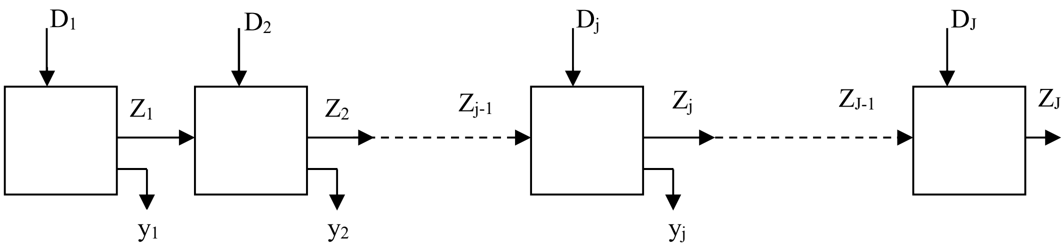

FIG. 5. Sequence of fuel cycles using decision vector symbol.

‘D’ operators in the figure refer to the ‘decision’ made in respect of the refuelling strategy, i.e., the number, enrichment and burnup of FA to be replaced and the position of FA within the core.

Thus, the characteristics of the cycle number ‘j’ depend on the following factors:

- End–of–cycle state of the previous cycle designated by the Z_{j-1}, visually. on the burnup of FA (j–1) at the end of cycle. If the previous cycle was long, e.g., the burnup levels were higher and vice versa, we should also take into consideration the quantity of FA from all the proceeding cycles of operation. Therefore, the $Z_{j–1}$ state is affected by all the preceding $D_{j–1}$, $D_{j–2}$, etc., decisions;
- Decision made at the cycle j, designated by the D_j; for example, we determine the quantity of FA to be replaced, the fresh fuel type and the load arrangement in the core.

Thus, in the cycle sequence, the any objective function (e.g., fuel cost) related cycle number *j* is reflected by the function of Z_{j-1} and D_j quantities (see Eq. (1)–(2)):

$$C_j = f\left(Z_j\right) = f\left(Z_{j-1}, D_j\right) \tag{1}$$

$$C^{opt} = \min C = \min \sum_{j=1}^{J} f\left(Z_{j-1}, D_j\right) = \sum_{j=1}^{J} f\left(Z_{j-1}^{opt}, D_j^{opt}\right) \tag{2}$$

The process described in Figure. 6 can be even more complicated if during the cycle j we utilize not only the fuel originating from the cycle (j–1) but also the fuel discharged earlier or interchange the fuel between two or more reactors or modify the fuel design. The latter will not be considered.

In practice, decisions concerned with reload plans are not completely free for reload planning person. Demonstrating this let us modify Figure 6 according to Figure 5. The ‘D’ decision vector here is broken down into the external and internal components (‘E’ and ‘I’, correspondingly). The external decision vector specifies the quantity and enrichment of the fresh FA to be replaced, while the internal one refers to their load arrangement.

External decision can be:

— Fixed cycle length. Annual, 18 months, etc., cycles for example, or even more precisely prescribed lengths. This case other parameters: enrichment, reload batch, core arrangement is free for reload design;
— In real life usually, more parameters are predefined:

- Enrichment;
- Reload batch (number of reloaded fuel);
- Arrangement (optional) for reload design work;

— Number of reloaded fuel, e.g., then the reload batch average enrichment can be varied using different number of differently enriched fuel.

Any case the maximal permitted burnup and maximum enrichment of applicable fuel are belong to external decision and determine all further internal decisions. Core arrangement is every time internal decision, but also has feedback to external, e.g., low leakage loading pattern provide longer cycles using the same set of assemblies.

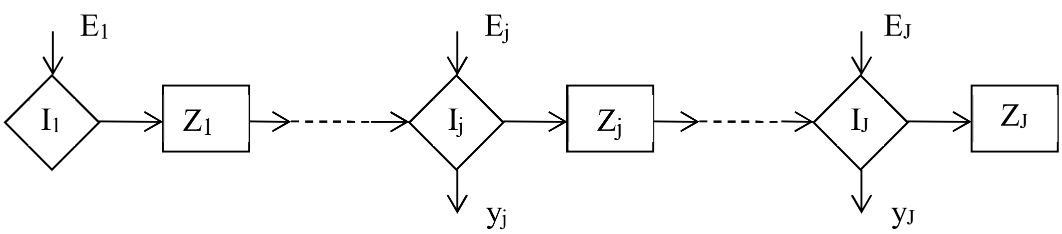

FIG. 6. Sequence of fuel cycle distinguishing inner and outer decisions.

Changes in conditions (required cycle lengths, max. allowed burnup, change in fuel properties) need a re–assessment of whole procedure, resulting completely different optimal solution.

4.2. CONCEPT OF OPTIMIZATION

4.2.1. General information about optimization

For all types of reactors core management and optimization may play significant role. A lot of parameters may be optimized in the area of core management and optimization are: physical characteristics of the core, economic performance of the fuel cycles, stock piles of the fresh fuel, length of the coast down period, date of fresh fuel delivery, etc.

Specific task for optimization may appear for NPPs, if it signs contracts not only for the fresh fuel delivery, but for enrichment of uranium as well. It is possible to optimize not only the number and enrichment of the fresh fuel assemblies, but concentration U–235 in tail, back–end and other cost.

Before starting any optimization, it is important to make clear what are the goals of optimization.

In the case of the optimization of the physical characteristics value of the physical parameter itself as a rule may be used to compare solutions. For more details see Appendix III. In the case of the optimization of the economic performance of the fuel cycle numeric indices related to performance of the cycle should be established. We will call these indices, as well as any optimizing value, the objective functions.

These objective functions of the fuel cycle should be minimized or maximized, at the same time its physical parameters have to remain within the established safety limits.

Examples of the objective functions:

- Primary cost of electricity needs to be minimized;
- Total profit before the decommissioning the plant, needs to be maximized;
- Maximal relative power of the fuel assembly needs to be minimized.

Optimization of the economic performance of fuel cycles and physical characteristics are closely related. Changes in physical characteristics may influence the economics of power production.

In generally most economical core loading is situated close to the reload design limits.

4.2.2. Fuel cycle reload strategy, equilibrium fuel cycle

The reload strategy should be optimized to achieve the best economic performance. Equilibrium fuel cycle may be used for optimization of the reload strategy.

The reload strategy and equilibrium fuel cycle are not synonyms. For the specific reload strategy many safe equilibrium fuel cycles may exist or may does not exist at least one. For a reload strategy having several equilibrium fuel cycles an optimal equilibrium fuel cycle can be found.

Fuel handling begins several months before commissioning. Design of the core and fuel cycle begins several years before first delivery of the fresh fuel to the NPP. Economic criteria must be established before development of the new reactor begins.

After commissioning and after three or four first fuel cycles are finished, the same reloading pattern will be mostly in use during next several fuel cycles (if there is no force–majeure). At the same time the nuclear fuel enrichment, the amount of the nuclear fuel purchased, and the lengths of the fuel cycle remain the same. Even fresh FA are placed mostly in the same places. The same thing is true for irradiated fuel.

An optimal equilibrium fuel cycles should be developed before the commissioning, at the stage of the new power unit design. Then the design for the first fuel cycle and transient fuel cycles should be developed. This equilibrium fuel cycle defines a paradigm or reload strategy.

The experience obtained from operating NPPs shows that this may last up to ten and more fuel cycles before the existing practice of refuelling will be changed.

It takes only three parameters to describe a reload strategy: enrichment, number of the fresh FA and length of the fuel cycle. Sometimes fuel assembly type may be added.

Example of a reload strategy:

- 12 months fuel cycle with uploading 54 fuel assemblies having enrichment within 4,4%.

We do not consider temporary decrease or increase of any of those three parameters as a change in the reload strategy, if there was a return to the previous fuel cycle.

Mentioned parameters are not absolutely independent. If any two of them are given, they set serious limits for the value of the third one and changing any of three leads to changing at list one of others.

If symbols for mentioned three parameters are as follows:

— X = enrichment;
— T = effective length of the fuel cycle;
— F = number of the fresh FA for refuelling.

Then the connection among these three parameters are:

$$\{X, T\} \rightarrow F \pm \delta_F \quad (3)$$

$$\{T, F\} \rightarrow X \pm \delta_x \quad (4)$$

$$\{F, X\} \rightarrow T \pm \delta_T \quad (5)$$

It is often possible to describe a connection among these three parameters using very simple equation. For example:

$$T = AFX + C \pm \delta_T \quad (6)$$

Here A and C are the constants which are specific for this type of the reactor and fuel type, δ_F, δ_x, and δ_T — a range of deviations containing the set of all acceptable values for F, x and T for the cores remaining within the established safety limits. A, C δ_F, δ_x, δ_T may be assessed empirically. And it is useful for core designer to estimate at list A, C and δ_T for own reactors.

Equilibrium fuel cycle (\ mentioned in 2.2) is a fuel cycle in which core discharge, reload and shuffle are carried out according to the same plan (or according to the same plans rotating after strict number of cycles). Equilibrium fuel cycles are very convenient for allowing for a consistent reload strategy. If within a specific reload strategy at least one safety equilibrium fuel cycle exists, then it is possible to work within that reload strategy for unlimited time. Another advantage of equilibrium fuel cycle is that it may be implemented at all the power units of this type.

Operational experience shows that a plant may strictly follow the equilibrium fuel cycle, or never load a core according the equilibrium loading pattern. But in that last case the plant works within chosen reload strategy (except force–majeure situations).

Another situation when following the equilibrium fuel cycle, or even following the reload strategy has no big sense because there is fast changing external circumstances (like prices for the fresh fuel, electricity and other).

4.2.3. Revising the reload strategy during lifetime of NPP

Due to the lifetime of the power unit may be as long as 30–80 years, most probably the optimal equilibrium fuel cycle and the reload strategy will be changed several times during a lifetime of the power unit because of changing internal or/and external conditions and circumstances.

Existing reload strategy for the operating NPPs should be revised on a periodical basis or after changing some of conditions.

Changing the internal or external conditions and circumstances may lead to a change in the value of the objective function for the fuel cycle in use, which in turn can mean that this optimal (in the past) fuel cycle is not an optimal one any more.

Change in the internal conditions and circumstances may lead to necessity of changing the fuel cycle or even the reload strategy:

- Changing the objective function itself (as a result of changing in the policy of operating organization, for example);
- Introduction of a new fuel type;
- Power uprate; etc.

Change in the next external conditions and circumstances may lead to necessity of changing the fuel cycle or even the reload strategy:

- Demands of the electrical grid was changed;
- Changing in the national regulations;
- Changing in the fuel limits (for example, increasing permitted burnup); etc.

4.2.4. Cycle length and flexibility

Developed and followed strategy of NPP reload design usually belongs to near equilibrium cycle. Nevertheless, the cycles almost never exactly equal, smaller–larges deviations from equilibrium state in fact every time exist. Requests from electricity grid often require cycle length differences from equilibrium state. If unit is in load–follow operation, effective cycle length can be different due to the needs from the grid. Unexpected events also may happen, thus influencing current and future cycles lengths. Reload designers and reactor operators should find the ways to regulate the cycle length following actual state of the core and outer needs.

Core designers and operators have several options to regulate the fuel cycle length:

- Coast down/stretch out operation:

  - If coast down/stretch–out operation is planned regularly, the actual length of it may cover deviations from equilibrium state. This method can be used, if deviation is small enough. In case of non–regular (much shorter or longer) stretch out, the EOC state of core may be so different, that cannot be handled any more with stretch out itself;

- Core arrangement:

  - Supposing the same content of core, low neutron leakage core arrangement provides longer cycles, other arrangement give shorter ones. Appropriate arrangement of the same content of core may help to fit the requested cycle length;

- Using fuel with various enrichments:

  - If the license of applicated fuel allows, cycle length can be effectively regulated by using differently enriched fresh FA. Different enrichment can be ordered if the deviation is foreseen in time. Another method, which is even more flexible is to use 2 (or more) differently enriched fuel, keep some reserve from each and use the appropriate mix to fit the designed cycle length to the required one;

— Using different numbers of fresh fuel:

- Another way of designing appropriate cycle length is to use different number of fresh fuel for reload: more for longer, less for shorter cycles. Following this strategy loading pattern optimization can help to reduce number of fresh fuel for the same cycle length.

Any method is followed from the listed ones. In practice flexibility is important part of a good reload design work and highly influences the real fuel cost and/or the profit from electricity production.

4.3. FURTHER EFFECTS INFLUENCING THE EFFICIENCY OF FUEL UTILISATION

Fuel–related costs appearing at a nuclear power plant are influenced by a great number of factors schematically presented herein below in Figure 7. Among those shown in the Figure 8 we have already reviewed the fuel enrichment, reload quantity and the effect of core arrangement. The rest of parameters are discussed as follows.

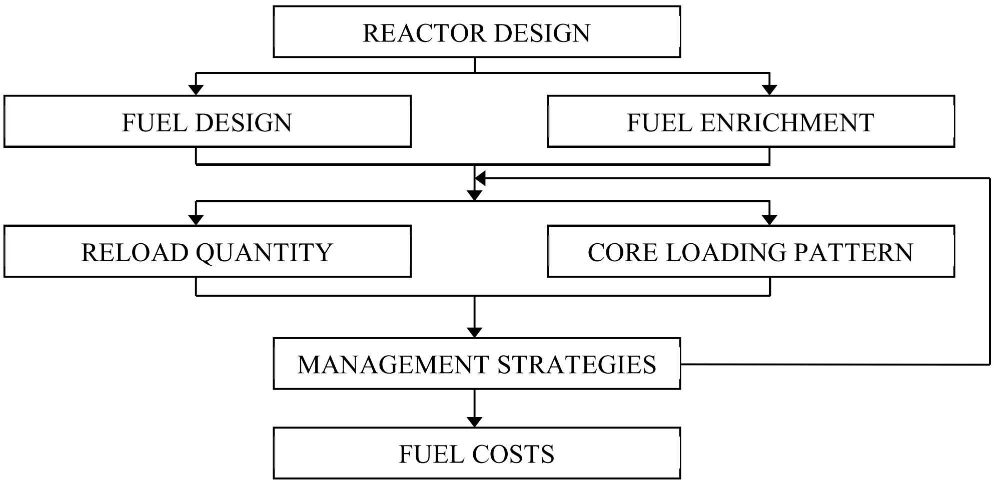

FIG. 7. Chart of effects influencing fuel utilization.

4.3.1. Power distribution control

In case of PWR reactors, the economic efficiency of fuel utilization is affected by, albeit to a lesser degree than a fuel's physical characteristics, the NPPs power distribution control process. The reactor power distribution control itself does not presume much variability in PWR reactors, the excess reactivity generated because of fuel reloading is absorbed by the boric acid diluted in the moderator and by burnable absorbers; control rods are used solely for the fine power adjustments and emergency shut–down of the power unit. Correspondingly, the method of power control is pre–determined and does not imply many possibilities for variations of any kind.

However, the situation is different in case of BWR type reactors. In this case, the power distribution control process is done with a combination of control rods and main circulation flow. The control rod position, as well as its time–wise behaviour has a significant impact on the reactor power shape and as such, on the economic efficiency of the fuel reload utilization.

The procedure of control rod movement provides for a higher level of freedom, even higher than in case of the fuel arrangement in terms of exerting influence on the reactor power distribution and, correspondingly, on the economic efficiency of fuel utilization. The process of optimal rod movement design shall take place simultaneously with the selection of a proper loading pattern. Such design methods are dealt with in the professional literature as well [13, 14].

4.3.2. Effect of reactor characteristics

In a first approximation, core dimensions and surface/volume ratio have impact on the neutron balance. A larger core characterized by a spherical shape and a smaller external surface provides for less neutron escape and more efficient fuel utilization.

Until now, we have reviewed the possibilities of cost effects from the viewpoint supposing that the reactor and fuel design remain unchanged. However, the historic background obtained during nuclear reactor operation attests to the fact that the characteristics of a reactor and, in numerous cases and almost constantly, of the nuclear fuel change significantly during the operation.

As far as the fuel economy is concerned, the most important and most frequently implemented modification is related to the power uprates which have been carried out at many power units since the 1990's.

In Chapter 2, it was demonstrated that changes connected with the fuel utilization which accompany the process of unit power uprate are: the reload quantity per a cycle increases with the unchanged enrichment; higher enrichment is required to achieve a new, more optimal work point.

Another modification of the core structure that has been implemented in case of several reactors is the replacement of peripheral FA with some reflector material in order to protect the pressure vessel material. In this case, the quantity of FA in the core is reduced that has an impact on the fuel utilization similar to that of the power uprate: along with the identical rated power and campaign length, more FA have to be replaced per each cycle.

4.3.3. Changes in fuel characteristics

A few decades of operating experience available for the LWR power units attest to the fact that numerous bigger and/or smaller modifications to the nuclear fuel have been made over these years. Since several years are required for the analysis and introduction of a modified design, the modification process is of a constant nature: during reloading at the power units, the previous fuel type is replaced for a new one and at the same time, the next modification is usually under preparation.

For BWRs the implementation of new fuel types containing different pellet size and water rod positions play an important role in fuel economy.

As far as fuel characteristics are concerned, the most important and most frequently implemented modifications are aimed at increasing the fuel burnp. A few modifications have been made both at PWR and BWR power units in respect of the fuel cladding material, ensuring the possibility for reaching higher burnup values. Along with that, some other modifications were necessary as well: design and material of the spacer grids, end–plugs and applied welding technologies had to keep up the pace with the more demanding requirements. The effect of these modifications implies that the fuel enrichment can be increased in case of a higher burnup and the quantity of fresh fuel assemblies used per campaign can be reduced.

Another important type of modification have resulted in the excess reactivity increase. For instance, one of such modifications consisted in the replacement of steel structural materials (e.g., spacers) for the ones made of zirconium alloy, which led to the reduction of parasitic neutron absorption. During the evolution of LWR fuel the cladding tube diameter has been gradually reduced. Fuel assembly that contained 15x15 tubes in the beginning today includes 17x17 or even 18x18 cladding tubes. Reducing pin diameter water uranium ratio can be optimized which leads to increase the excess reactivity. By means of reducing the cladding wall thickness and the fuel–cladding gap, it has been possible to increase the mass of uranium, if the pin outer diameter is the same. Keeping outer dimensions of fuel assembly, there is also evaluated elongation of fuel stack resulting higher uranium mass.

The fuel economy may be improved using so–called axial blankets at top and bottom of the fuel column. This fuel design implies the use of a few pellets containing unenriched uranium at the top and bottom of the enriched fuel column; the fuel pellet manufactured at a cheaper unit price reduces the possibility of the undesired neutron escape.

The implementation of the so–called mixing spacer grids — improving the coolant mixing and providing for a uniform cooling — has resulted in the improvement of hydraulic properties and an indirect increase of the economic efficiency.

4.3.4. Use of burnable poisons

Through the use of burnable poisons placed inside the fuel assembly with the use of various technologies, it is possible to significantly modify not only the global reactor parameters but also the local parameters of the fuel assembly and its internal characteristics.

The burnable poison can be either placed into a separate cladding, added to the pellet and cladding surface or integrated into the pellet. Solution to be selected for implementation depends on the neutron–capture cross–section of the given burnable poison and other technical conditions. Normally, the burnable poison is not distributed uniformly along the fuel; it is contained only in the radially and/or axially selected positions. On the one hand, the localization of burnable poisons has a significant impact on the power distribution within the fuel assembly; on the other hand, the flux and power of the volume/rod containing burnable poison change relatively fast in time.

Burnable poisons based on different technical solutions demonstrate different behaviour. In case of the most frequently used technical solutions implying the poison integration into the fuel pellet, the absorbed reactivity is proportional to the quantity of used burnable poison (similar to the boric regulation); its time–wise behaviour is determined by the burnable poison concentration. The burnup process is slower in case of a higher concentration, mostly due to significant self–shielding caused by a huge absorption.

The requirement produced in respect of the burnable poison behaviour is to ensure that its major part will burnup during the first cycle following the load; otherwise, the cycle is shortened and becomes economically inefficient. Knowing the data on the neutron flux prevailing in the reactor and campaign length, one can determine the optimal concentration of the burnable poison.

In the beginning, the k_{eff} multiplication factor of the fuel assembly containing burnable poison grows and then decreases depending on the burnup level.

The use of burnable poisons significantly modifies the application and applicability of core design and optimization methods. The amount of power gained from the given fuel assembly or fuel load will not be proportional to the excess reactivity at BOC; optimization done in respect of the beginning of cycle condition will cease to be effective along with the poison burnup, unevenness will appear, etc. Only tools taking into consideration the middle– and end–of–cycle condition of the cycle shall be used.

In case of using burnable poisons the application of practical methods and expert system are getting useful determining the loading patterns. Obviously, whenever we possess sufficient computer capacities (for example, to produce target function values for each examined configuration with the entire cycle calculations) the stochastic methods can be applied at any time.

4.3.5. Plutonium in the fuel

Certain countries use different technical solutions for the spent fuel management. One of these solutions consists of reprocessing the spent nuclear fuel. At the first stage of reprocessing structural materials are removed from the spent fuel and then uranium and plutonium are extracted from the fuel with the use of chemical processes.

Re–cycled plutonium is used for manufacturing mixed oxide fuel (MOX) operated in water reactors. In case of MOX fuel, remaining uranium with the 0,23% of U–235 originated during the enrichment processed are enriched by plutonium extracted from the spent nuclear fuel, plutonium concentration in the fuel is typically 3–12%. Plutonium contains a number of isotopes, the major part of which is presented by Pu–239 (>60%) and Pu–240 (>20%) isotopes.

One kilogram of spent fuel with the average burnup contains 955 g of remaining uranium with 0.9% of U–235, 10g of plutonium and 35g of fission products. In terms of plutonium, 7g are fissionable material, while the remaining quantity is the neutron absorber. 1kg of MOX fuel contains 55–60g of plutonium.

The use of mixed oxide fuels raised a range of problems not experienced before, including the following [15–17]:

- Neutron absorption of plutonium neutron in the thermal range is higher than that of uranium;
- Lower delayed neutron fraction;
- Reactor parameters change along with the use of MOX fuel, boric acid and control rod efficiency decreases, the temperature coefficient of reactivity become more negative.

The above phenomena demand special treatment. A higher neutron absorption may cause a power peaking in the neighbouring fuel rods, which is mitigated by means of radial profiling of the bundle in accordance with the MOX fuel enrichment. For the purposes of MOX fuel application, the power control system in some pressurized water reactors was re–designed, more control rods were applied, and it was necessary that the emergency positions should be re–evaluated.

Economic efficiency of MOX fuel utilization depends on the costs related to the traditional fuel; its application with the prices that existed at the turn of the century was extremely unprofitable; at the same time, other interests stand for the use of plutonium originated during the reprocessing. There are some serious research activities underway in order to return to the reactor trans–uranium elements with a higher mass number gained from the spent nuclear fuel. In such a case, it would be possible to store the remaining fission products under much more favourable conditions.

4.4. COST ELEMENTS OF THE NUCLEAR FUEL PRODUCTION PROCESS

4.4.1. General description of fuel

In general, LWR fuel assemblies are mechanically quite similar. There is a number of metal tubes, called fuel rods, bundled together, which contains uranium dioxide pellets. The outer geometry of the assemblies varies between square (PWR and BWR) and hexagonal (WWER). But there is a larger difference in the enrichment design of assemblies which depends on if the reactor has a direct steam production (BWR) or indirect with steam generators (PWR and WWER). In a PWR or WWER usually all the rods in an assembly have the same enrichment profile and there are small differences between the reloads from cycle to cycle. In BWR assemblies there are fewer rods but quite a lot of them have different enrichments of U–235 varying from 0.71 w–% to 4.95 w–%. Also, every reload can be specially designed for the upcoming fuel cycle. Different BWR fuel manufacturers have different number of rods in an assembly, different materials, different rod lattice, different max burnup of the assemblies, etc.

So, for BWR there are many options regarding fuel to consider and BWR NPPs often change fuel vendor and fuel type which make it hard to reach an equilibrium cycle and is an event that influences the core design heavily. The reason for changing is of course to improve and BWR have a great flexibility and growth potential due to its fuel design with fuel bundle in a channel. This means that the channel is the outer boundary and how the inside looks can be changed from 8 by 8 rod geometry to 9 by 9, 10 by 10, etc., allowing for drastically improved fuel in new generations within the same channel.

CANDU uses natural uranium only. Therefore, only one fuel bundle type is installed in several hundred fuel channels. Here are the licensed fuel bundles in CANDU industry at 2018: 28-element, 37–element, and modified 37–element. In the case of CANDU–6, there are 380 fuel channels and 12, bundles per fuel channel are loaded. As pressure tube diameters of CANDU and CANDU–6 are nearly the same, 37–element bundle's fuel rod diameter is smaller than that of the 28–element fuel bundle. The modified 37–element bundle is the same as the 37–element bundle except for the centre element's diameter.

4.4.2. Cost elements of nuclear fuel

In general, there are four main parts that make up the cost of a fuel assembly, uranium (U_3O_8), conversion from U_3O_8 to uranium hexafluoride (UF_6), and enrichment and fuel fabrication.

Back end cost can depend on each country's regulations be either small and not needs to be considered or the cost can be so large that it will be the single most important thing performing core design to reduce number of FA used. Back end cost will not be discussed further in this paper.

Developing a procurement strategy that takes all these components into account is essential in order to reduce fuel cost. Optimizing on only one of the components is not recommended. For instance, buying the fuel that is cheapest to fabricate might be more expensive in the end, if the specific fuel type is requiring higher enrichments or is prone to fuel failures.

The strategy will be very different depending on reactor type. Most obvious examples are CANDU fuel which is not enriched and PWR fuel which often have fixed enrichments.

Procurement of fuel is often managed by a single body within the operator's central functions. However, if this is strategy is developed together with core physicist and local outage planners, with continuous follow up of events and planning, it is easier to either call of the right amount uranium, enrichment and number of FA to be delivered. This would reduce the need for ordering fuel as insurance for the uncertainties in operations and therefor also reducing overall cost for the fuel.

A common strategy for LWR is to secure a large part (50–80%) of the required uranium ore with long term contracts to have a predictable and hopefully low cost. If needed the rest is procured on the spot market. For enrichment the part secured via long term contract is generally higher. Contracts with high flexibility are more expensive and the procuring body needs to analyse the risk of more fixed (and cheaper) contracts together with the operating body before signing.

Other costs, for instance transportation of fuel and mandatory inspections is small by comparison but when negotiating long contracts this have to be considered.

Cost of money, or what we usually call interest, is a significant cost since the two most expensive components, U_3O_8 and enrichment, are procured one to two years in advance of the FA being delivered to the NPP. A step by step planning for the procurement from ordering U_3O_8 up to the delivery of FA to minimizing led times is a good way of reducing interest costs.

Costs related to the nuclear fuel production can be divided into three elements: costs related to the raw uranium plus conversion, costs of enrichment and fabrication. To have a better understanding of the before mentioned cost elements, the nuclear fuel production steps are briefly described herein below.

4.4.3. Refining and conversion

The extracted uranium–bearing rock is ground to powder with the use of special mills and is then dissolved in the sulfuric acid. The rock material is filtered to precipitate uranium from the solution in the form of uranium oxide (U_3O_8). The yellow powder is pressed into a special briquette which is called the 'yellow cake'.

Prior to the isotopic enrichment, the uranium oxide is converted into uranium hexafluoride (UF_6) using the gas conversion technology. The natural fluorine has only one isotope (F–19), so the mass of UF_6 molecules may differ only due to the presence of either U–235 or U–238 isotope. The conversion process consists of the three steps: firstly, the uranium oxide is converted into uranium dioxide — usually with the help of H_2, and then by means of adding HF and F_2 gases the final product is obtained.

The raw uranium demand of the nuclear fuel containing unit of weight uranium can be calculated as to the following equation:

$$S = \frac{e_d - e_{dm}}{e_t - e_{dm}} * V \qquad (7)$$

Where:

- e_d = is the enrichment of fuel loaded into the fuel element;
- e_{dm} = is the U–235 concentration of the depleted uranium originated as tails;
- e_t = 0,00711–is the U–235 concentration of natural uranium; and
- V = is the loss factor (its value lies usually in the range of 1.01–1.02).

Based on the above equation, the raw uranium demand for 1kg of the nuclear fuel containing 4% enriched uranium is 9,18kg, taking into consideration 0,3% tails and 2% loss. The enrichment of natural uranium is supposed to be 0,711%.

4.4.4. Enrichment

During the enrichment process, the U–235/U–238 ratio is increased by means of separation. The natural uranium is decisively the mixture of two isotopes, i.e., 0.71% of U–235 and 99.3% of U–238; the ratio of all other isotopes is less than 0,1%. The Separative Work Unit (SWU) refers to the work carried out during the technology application and serves as the basis for the account of costs. The SWU value depends on the enrichment level of both enriched and depleted uranium. The so–called *enrichment work* required for manufacturing the unit of enriched uranium weight (visually *specific enrichment work)* can be calculated as follows:

$$SWU = \frac{SW}{M_d} = V(e_d) + m_{dm}V(e_{dm}) - m_t V(e_t) \qquad \frac{enrichment\ work}{kg\ enr.U} \tag{8}$$

Where:

- SW = the enrichment work required for the manufacture of enriched uranium with the Md weight, kg;
- e_d = the enrichment of fuel loaded into the fuel element;
- e_{dm} = the U–235 concentration of the depleted uranium originated as tails;
- e_t = 0,00711–is the U–235 concentration of natural uranium;
- V = the loss factor (its value lies usually in the range of 1.01–1.02).

$$V(e_i) = (1 - 2e_i)\ln\left(\frac{1 - e_i}{e_i}\right) \tag{9}$$

Where:

- e_i = the so-called value function applicable for the U–235 concentration;
- m_{dm} = the mass of depleted uranium (depleted tailings) originated simultaneously with the manufacture of 1kg enriched uranium;
- m_t = the mass of natural uranium and used for the manufacture of 1kg enriched uranium;
- V = the loss factor (its value lies usually in the range of 1.01–1.02).

Using definitions listed the equation pertaining to the enrichment work required for the manufacture of 1kg enriched uranium will look like this:

$$SWU = (1 - 2e_d)ln\left(\frac{1 - e_d}{e_d}\right) + \frac{e_d - e_t}{e_t - e_{dm}}(1 - 2d_{dm})ln\left(\frac{1 - e_{dm}}{e_{dm}}\right) - \frac{e_d - e_{dm}}{e_t - e_{dm}}(1 - 2e_{t)}ln\left(\frac{1 - e_t}{e_t}\right) \tag{10}$$

Based on the above equation, the SWU demand for 1kg of the nuclear fuel containing 4% enriched uranium is 5,276kg, taking into consideration 0.3% tails.

4.4.5. Fabrication

During the fabrication process the enriched UF_6 is converted into UO_2, followed by the manufacture of fuel pellets, fuel rods and assemblies. Costs related to this operation are given depending on the quantity of the already enriched uranium for PWR and WWER.

4.4.5.1. Finished fuel assembly

Correspondingly, the price of the finished fuel assembly is composed of the above three elements:

$$c_{fe} = m_t c_t + m_t c_k + SWU\ c_{swu} + c_f m_d \tag{11}$$

Where:

- m_t = the mass of natural uranium used for the manufacture of 1kg enriched uranium;
- c_t = the unit price of natural uranium (procurement price of 1kg natural uranium);
- c_k = the cost of uranium conversion referring to 1kg natural uranium;
- SWU = the total demand for SWU;
- c_{swu} = the cost of invested enrichment work per unit price;
- m_d = the enriched uranium weight in the bundle;
- c_f = fabrication price of kg/uranium.

For BWR fuel manufacturing varies a lot between different manufacturers and different fuel types since most manufacturers offer more than one type of fuel. Depending on fuel type the price of manufacturing can be between 10 and 25% of total fuel cost. New types are more expensive but offers savings in enrichment so in absolute cost the difference between different fuel types can be even greater. For BWR fuel there are usually some choices to be made regarding material, filter, number of different enrichments in the bundle, split design of the reload, etc., which also influences the final manufacturing price. Contracts also often contains steps where ordering a large amount of FA at the same time reduces unit cost with some extent.

4.4.6. Cost related finished fuel

Based on the above said, the price of a fuel assembly is an almost linear function of the enrichment, as it is shown in Figure 8.

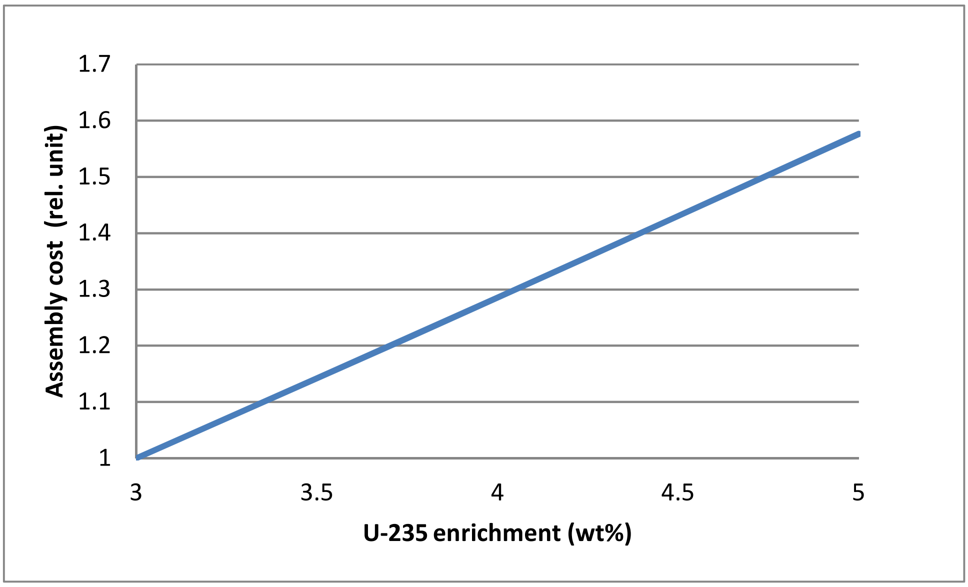

FIG. 8. Cost function of the enrichment.

As it has already been mentioned in Chapter 2, and LWR reactor can be operated using different fuel strategies. In case of using fuel with a lower enrichment, more fresh FA will be necessary for the make–up, in case of a higher enrichment — less FA will be necessary for the assurance of the operation time. With consideration to the above, fuel costs of a single cycle depending on the applied enrichment can be characterized as shown in Figure 9 (see Appendix I. Section 1.1).

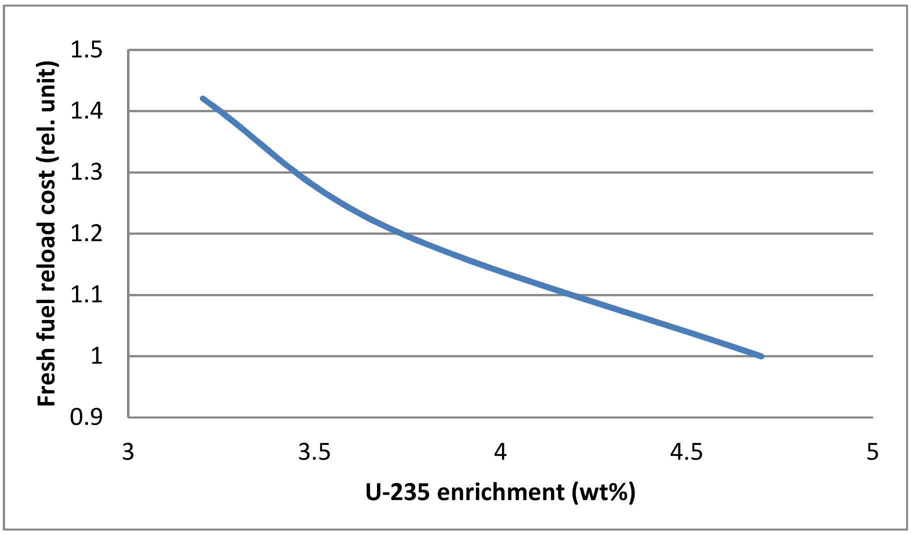

FIG. 9. Fresh fuel reload cost/kWh for equilibrium cycles as function of enrichment.

Typically, the above Figure 9 refers to the annual fuel cycle. At the same time, Figure 4 in Chapter 2 shows the correlation between the enrichment and quantity of fresh FA in case of uprated power and longer cycles. Using the data contained in Figure 4 for these cases, the correlation between the enrichment and fuel costs will be modified. The curves in Figure 10 show the relations as follows:

— At a certain nominal power and (equilibrium) cycle length fresh fuel cost is decreasing with increasing fuel enrichment (because of decreased number of fresh fuel per cycle);

- Using the same enrichment of fuel, at higher nominal power in general we have higher relative fuel cost;
- The same is true for extended (equilibrium) cycle length: using the same enrichment longer cycles have higher relative fuel cost;
- Correspondingly: using a certain enrichment of fuel, longer (equilibrium) cycles and uprated nominal powers are less economic from the standpoints of fuel application (Later will be discussed: more electricity production of higher nominal power and long operation cycle over–compensates the increase of fuel cost);
- Application of other technics, e.g., starting to use regular coast down, modified fuel, etc. also have significant impact on fuel costs, which is to be considered in given specific case;
- Beyond the general features described in Figure 10, actual optimum (enrichment, nominal power, cycle length, regular use of coast down) for an NPP unit is the function of its specific circumstances (e.g., outage time for a reload, electricity grid requirements, etc.) and constraints (e.g., burnup limits, maximum local power, MTC limit, etc.).

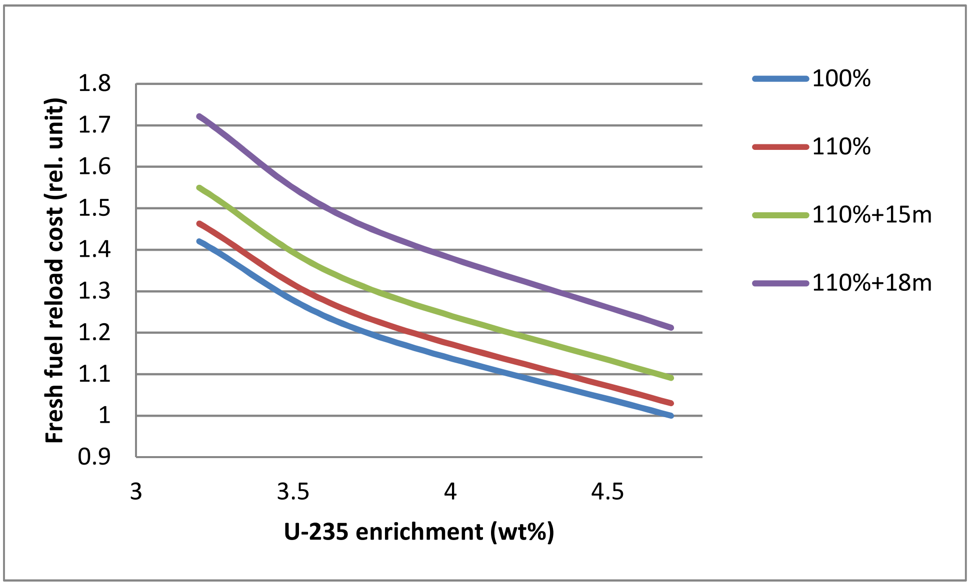

FIG. 10. Fresh fuel reload cost/kWh vs enrichment in different nominal power and cycle length.

It can be noted that the increase of the burnup/cycle either due to the power uprate or via longer cycles usually leads to the increase of fuel cost per the generated electricity unit. It is another question that the profit received from the increased electricity production in the above cases is significantly higher than the increase of fuel costs.

4.4.7. Costs related to the spent fuel management

Further management of spent FA discharged from the reactor is accompanied by the origination of considerable costs. The discharged fuel has to be firstly placed for cooling into the spent fuel pool, which requires the assurance of appropriate storage capacity by means of an adequate technology. Then, the fuel is transferred for the subsequent intermediate storage during 50-70 years and/or to the reprocessing facility, which also implies significant expenses. Though the cost ratio is not necessarily true, the costs related to the spent fuel management are usually given proportionally to the spent fuel and the weight of heavy metal (HM) in the fuel or number of spent FA.

Similarly, taking into consideration the quantities connected with the applied enrichment, the change of costs related to the spent fuel is shown in Figure 11. With the enrichment increase, the spent fuel costs demonstrate a steeper reduction than the fresh fuel costs.

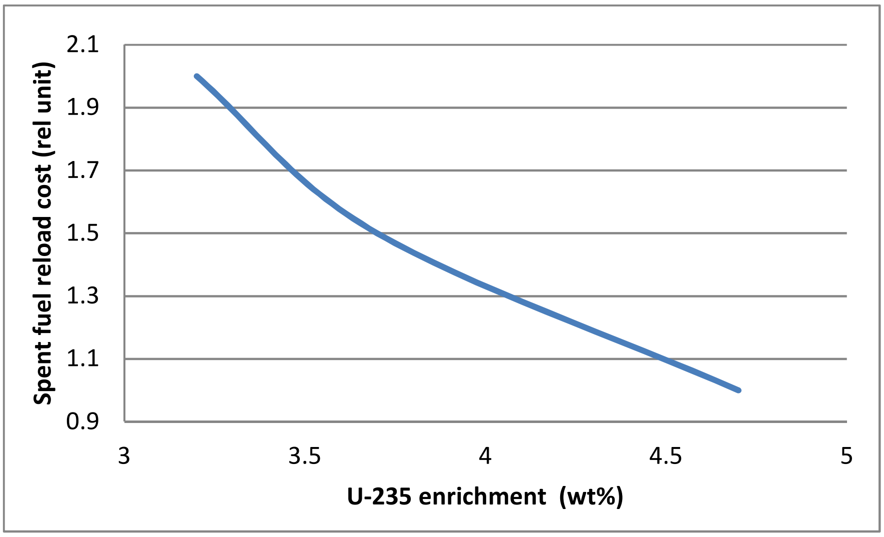

FIG. 11. Spent fuel reload cost/kWh vs fuel enrichment.

4.5. OPTIMIZING THE ELECTRICITY PRODUCTION COSTS

So far, it has only been discussed that the cost of fuel used to generate electricity and not the price of electricity delivered from an NPP. Regardless of the market the NPP is delivering electricity to be de–regulated or not there is always possible to reduce fuel cost.

In LWR normally between 20 and 40% of the FA in core are burnt up and replaced during refuelling outages. Exceptions are when there is an extra outage due to some unforeseen reason, for example a fuel failure, when only a small part (i.e., the damaged FA) of the core is replaced. This also means that the cost for a FA has to be divided into parts related to the core cycles the FA is in core to accurate compare the fuel cycle cost to the revenue incurred during the cycle.

A common relationship is that roughly 10–20% of the total cost of operation is fuel cost.

Optimizing core design will often save somewhere between 0.5–5% fuel assemblies depending on reactor type and rated power. However, this is conditioned to having an optimized core design (neutronics and fuel pin enrichment), typically more important for BWR than PWR and not at all applicable for CANDU reactors.

If the operator is considering more complex changes to operations, like a power uprate, more detailed analysis is required. Note that fuel cost is likely to increase if the power rating is increased or cycle length extended. However, in such cases, the expected increase in electricity revenue more than offsets any potential increase in the fuel cost.

A good strategy is to have some flexibility in the contracts, especially the fuel fabrication contract to allow for late changes in number of FAs delivered to the NPP to reduce cost. Flexibility allows for the possibility to change the reload strategy from utilizing few bundles with high enrichment to many bundles with low enrichment, depending on the cost relationship between U_3O_8 and enrichment. Also, optimal tail for the enrichment needs to be flexible in the contract (and check before every order). As mentioned before back end cost can have a strong influence on this decision:

- To achieve the best result, i.e., lowest cost for the fuel procured there are several crucial steps to attend to;
- Perform long time studies (several equilibrium cycles or specific cycles with realistic assumptions are needed) to determine the annual need for fuel and asses what risks can be accepted;
- Perform market surveys to find trends and try to predict the price for U_3O_8 and enrichment the coming years;
- Decide on long term contracts or spot market;
- Set up a detailed step by step plan to minimize led time for the delivery of FA to the NPP.

Set up a close cooperation between the department ordering the U_3O_8, enrichment and fuel fabrication in order to be able to late in the process adjust fuel neutronic, FA and core design with respect to the latest plans in order to save fuel during the upcoming outage.

4.6. ECONOMICAL OPTIMIZATION OF THE FUEL CYCLE

It is necessary to have an outage plan that plans ahead for several years in order optimize the fuel and core design. For instance, if the core is loaded with more reactivity than is necessary for the cycle a big part (some studies show 20%) of the excess reactivity is wasted since the fuel will not be possible to use in an optimum way during the next cycle.

It is good practice to understand the economics fuel contracts during the fuel and core design work. For BWRs, in general, it is more optimal to use similar enrichments in most rods and for LWRs it is more optimal to order one type of FA design per reload batch.

In order to capitalize on opportunities that emerges on the fuel market a close cooperation between the reactor physicists who design the fuel and core and the department procuring the uranium, enrichment and fuel fabrication is necessary.

Minimizing lead time in the fuel procurement chain is essential in order to keep the fuel cost down since the interest, the cost of money, is significant. This in turn requires close interaction with the operating department and the maintenance department in order to have common planning for the refuelling of the reactor.

There are mainly two types of trade–offs that can be made during fuel and core design. First investigate optimum numbers of FA in a reload. Increasing enrichment is expensive and it might be more economical to buy many FA with low enrichment.

Second the fuel can be designed to have good performance with respect to one thermal margin, for instance dry–out, sacrificing performance with respect to LOCA depending on what is the limiting factor for the operations.

4.6.1. Fuel chain/cycle optimization

Most important during the fuel and core cycle cost optimization is to understand what the risks are and what risks the operator is willing to take. No risk is impossible, but a very low risk is more costly than high risk. For instance, a low risk approach may include delivery of a reload batch one year in advance. This is costly (interest and storage cost) but the risk of not operating is very low, at least when regarding securing the fuel deliveries.

A high–risk approach may include buying most of the material on the spot market, having no reserve fuel on site and eventually having the fuel to be inserted delivered close to the outage.

Information flow between reactor physicists, outage planning and procurement department with regular updates is a good way of monitoring the risks but also to take opportunities that arise. Understanding this information flow is most important at the start of the fuel design and core design process.

After a company strategy is put in place it can be used when procuring material and services to ensure that the signed contracts are complementing each other.

In general, the majority of the material (Uranium ore) is procured on long fixed contract to get a low and predictable material cost. The rest of the need are procured on the spot market in order to take advantage of market opportunities arising on the market and also not be looked in to buy material not needed if productions do not reach planed volumes. For instance, low price for uranium ore due to market fluctuations is such a kind of opportunity.

Enrichment is generally procured to a higher degree than Uranium Ore. Manufacturing of fuel should have some flexibility in numbers of assemblies delivered per reload to accommodate for fluctuations in need.

One way of decreasing the fuel cost is to optimize (e.g., minimize) lead times in the fuel procurement chain.

4.6.2. Optimization of a specific fuel cycle

In situations where the deviations between each loaded cycle are large enough to make it impossible to follow an equilibrium core loading strategy one has to start looking at either optimizing the cores one at the time or doing multi cycle optimizing as mentioned in Section 5.4.1. If the future beyond the coming cycle is highly uncertain due to external factor (highly changing power demand from year to year, maintenance planning, etc.) one option is to optimize each cycle as an individual cycle.

For a single cycle the optimization target is simply to produce the planned electricity with the lowest fuel cost, remain within safety limits. The cheapest cycle is not necessarily optimal in the long term.

If the knowledge about the coming cycles are more certain but it is still not possible to do an equilibrium cycle strategy it is worth doing multi cycle optimization. This is different compared to optimizing the specific fuel cycle in that it looks at what is the cheapest option for several cycles. So, it can be worth loading a more expensive next cycle if it saves you money over several cycles, see Section 5.4.1.

4.7. FUEL MANAGEMENT

There are a lot of definitions of a strategy exist, and most of them include goal which have to be achieved. As an example:

— A strategy is a high level plan to achieve one or more goals under conditions of uncertainty. Hence, to discuss the strategy we have to start from discussing the goals.

Goals usually are proclaimed by operating organization in policies or in other documents.

An example of the goal, or even a mission proclaimed by the operating organization:

- “Rosenergoatom Concern, OJSC sees its mission as providing consumers with electrical and thermal power produced by the Concern NPPs, with guaranteed safety as its top business priority.

  The Concern’s main values are energy security and economic development of Russia, protection and safety of people, and environmental protection. The Concern implements the following principles during its main activity which is the operation of NPPs: economic efficiency of production of electrical and thermal energy at NPPs” [18].

There are two aspects in the mission which are interesting for us: economic efficiency and safety.

The economic efficiency establishes task, safety sets limits on the ways by which this task can be achieved.

If economic efficiency is declared as a goal, economic criteria must be established as clear as safety limits. Economic criteria must be established in advance.

Good practice should be to always design a core that is in agreement with all safety analysis, with the desired power output and with a minimum numbers of fuel assemblies used in order to minimize cost and environmental impact.

4.7.1. The strategy of the fuel usage

If one of the goals proclaimed by operating organization is economic efficiency, the strategy may consist in following some of the principals mentioned below, at the same time staying within safety limits established for the core and reactor:

- For fuel cycles having equal length increasing enrichment leads to reducing fuel component of the price of electricity;
- The longer fuel cycle the higher fuel component of the price of electricity;
- For fuel cycles having equal enrichment of the fresh fuel the longer length of the fuel cycles the less primary cost of electricity.

It depends of the reload strategy which principle(s) to follow.

The strategy should include the objective function for economic criteria and the reload strategy which has the best value of the objective function and fit the economic performance in the best way.

In more wide sense the strategy may include a set of the conditions for revising the reload strategy.

Let’s call the staff in charge of the core design and nuclear fuel usage reactor physicist (RP). It may be staff working at NPP, in the central office of the operating organization or in the external organization.

The position of core designers is unique because they have a possibility to influence economic efficiency and at the same time to ensure compliance with safety limits established for the core and reactor.

Hence, the mission of a core designers is to design the cores syncretizing (matching) safety and economic efficiency in the best way possible.

4.8. CANDU CORE DESIGN

After the initial period following first reactor startup, on–power refuelling is the primary means of maintaining criticality in a CANDU reactor. Thus, a number of channels are refuelled every day, on the average. Note that some stations prefer to concentrate all refuelling operations to 2 or 3 days within each week.

Replacing irradiated fuel with fresh fuel has immediate consequences on the local power distribution and on the subsequent period of operation of the reactor. Therefore, a unique power distribution called a 'Time–averaged Model' determines the reference three–dimensional power distribution, the expected refuelling frequency of each channel (or its inverse, the channel dwell time), and the expected value of discharge burnup for the various channels. These factors are key to optimizing the reload strategy for CANDU reactors.

When a channel is refuelled, its local reactivity is high, and powers including neighbouring channel will be several percent higher than its time–average power. For 40 ~ 50 full power days after refuelled, the local reactivity and power increase because of plutonium generation and then power drops slowly as burnup of fissile materials. Eventually, the channel becomes a net sink or absorber of neutrons, and channel power drops to 10% or more below its time–average power during every cycle. The cycle length depends on the location of fuel channel. Therefore, the power of each channel goes through an 'oscillation' about the time average power. And at any given time, there are several channels in the core which are at or near the maximum power in their cycle. Therefore, the maximum instantaneous channel power is always higher than the maximum time–average channel power. To design reload core or reactor protection system, one has to reflect instantaneous power distribution peaks. The power peak (or CPPF, channel–power picking factor) behaviour depends on refuelling scheme; 4–bundle–shift or 8–bundle–shift. Minimizing the CPPF is very important in the normal core design and regional overpower protection system operation. That is why site reactor engineers make a strategy or rule to select proper channels to be refuelled. To get information, they use physics code, the on-line flux mapping system, in–core detector signals, and liquid zone water fills. They need also proper cross–section set for fuel bundle and reactivity devices, effective Xenon isotope cross–section, and core–follow results including bundle–wise power, burnup, and flux distribution. If they determine proper refuelling channels showing good maximum value over all fuel channels of the ratio between instantaneous channel power and the corresponding time–average channel power (CPPF) based on the instantaneous reactor calculation, the reload design is finished normally.

If refuelling were to stop because of refuelling machine unavailability or other reasons, core reactivity would continuously decrease. In that case, the reactor regulating system (RRS) would attempt to maintain criticality: 1) lowering the level of water in the liquid zone–control compartments; 2) withdrawal of the adjuster rods. All these sequences should be reviewed by using RFSP code [5, 19].

5. NUCLEAR POWER PLANTS PRACTICES FOR RELOAD STRATEGIES

5.1. CORE LOADING PATTERNS

In early days core loading patterns were optimized to have the flattest power and temperature profile. This goal was defined because of the limited accuracy of core simulation models at the time. Later, the so–called low neutron leakage core loading patterns became general because of two reasons: better fuel cycle economy and proper preservation of the reactor vessel from high neutron fluence. Burnable absorber technology helped to place fresh FA not at outer but rather inner core positions.

In PWR reactors most of FA are operating in core during several fuel cycles, changing its position several times during refuelling.

Core loading patterns having all peripheral cells filled with the fresh FA usually called ‘out–in’ type core pattern. Irradiated FA are placed in the inner part of the core, perhaps, together with some fresh FA.

In the ‘in–out’ pattern fresh fuel placed in the inner part of the core, then during refuelling these, already irradiated FA, are moved to the peripheral cells, so as all peripheral cells filled with the irradiated FA.

In the case of the ‘partial in–out’ the peripheral cells are filled with both fresh and irradiated fuel assemblies.

Every type of core loading pattern has its own advantages and disadvantages. Placing FA having low multiplication factor in the peripheral area reduces neutron leakage and increases length of the fuel cycle. At the same time this solution increases critical boron concentration and shifts neutron flux from periphery to the central part of the core. As a result that may shift temperature coefficient reactivity to the positive area. Due to high neutron flux in the central part of the core it is more difficult to meet demands for linear power for this pattern. Thus, to reduce linear power in the central part of the core, FA having higher multiplication factor better to put in the peripheral area. That is one of the most common technical contradictions in core design.

As a rule, fresh fuel has higher multiplication factor than irradiated fuel. For FA having integrated burnable absorber it should be considered that after first fuel cycle irradiated fuel assembly may have multiplication factor close to that of the fresh fuel assembly of the same type. Multiplication factor of FA after two and more fuel cycles is significantly lower. For the very first fuel cycle of a new reactor fresh FA having low enrichment play a role of irradiated FA.

Usually the task is to design a core having minimal neutron leakage and at the same time to meet all the safety requirements. For the core developing for the first time, if there are no well–known patterns, the next practical approach may be used. Initially to design any safety ‘out–in’ type core. Then step by step to reduce number of fresh FA in the peripheral cells, checking safety requirements.

Using integrated burnable absorber makes it more difficult to design a core due to characteristics of such FA are changing in nonlinear way during the fuel cycle. Necessity to check safety during whole the cycle increases time spent for calculations.

Some examples include the following practices that apply to WWER–1000 core designs:

- To increase worth of the scram and control rods it is useful to place fresh fuel assemblies having high $K\infty$ in the cells having scram’s or control rods.
- To shift moderator temperature coefficient of reactivity to negative direction it is useful to shift neutron flux to the periphery (example from WWER–1000).
- In the case of unplanned long outage, especially if reactor is open, it may be useful to try to reload the core without loading fresh FA.

5.2. OPTIMIZATION OF THE CORE ARRANGEMENT

For the core arrangement optimization several methods and technics have been developed such as:

- Linear programming;
- Method of the most efficient moves;
- Annealing algorithms;
- Genetic algorithms;
- Other stochastic methods;
- Expert system, neural network; etc.

All the listed methods are discussed in detailed in Appendix III.

Goal of optimization is usually to find arrangement within constraint having the maximum cycle length.

Objective function values can be defined various way, for example:

- Neutron flux of the most peripheral assemblies, if the objective is the reactor pressure vessel protection;
- Burnup of the FA to be discharged;
- Other, a little bit 'artificially' formulated objective function; etc.

Generally, the calculation of the objective function value is carried out through the criticality (and/or burn up) calculation of the loading having the actual fuel arrangement (possibly, through the cycle burnup calculation). Thus, the objective function definition, in fact, may be arbitrary; it should be the unambiguous function of the calculated parameters of the state (reactivity, neutron flux).

The optimal value as to the selected objective function should be always searched among the determining constraints. These are the maximum burnup, local power and temperature values, as well as their respective non–uniformity coefficients. Some other reactor physics parameters can be constrained as well, for instance, the rod effectivity, moderator temperature coefficient, etc. The given reactor and fuel characteristics determine those parameters, which are sensitive to the core arrangement.

For CANDU, there is no need to worry about optimum fuel arrangement. Since the time–averaged power distribution is the most economical when designing the reactor, on–power refuelling may be considered the optimum fuel arrangement itself.

5.3. FACTORS TAKEN INTO ACCOUNT FOR RELOAD STRATEGIES

5.3.1. Demand for electricity

In the case the demands of electricity depend of the season it may constrain length of the fuel cycle over the time permitted for maintenance and refuelling. Price and demand of electricity can also influence the use of coast down, i.e., the excess reactivity loaded in the reactor during the outage is not enough to sustain operation at full power all the way up to the next outage but will start to go down with for example 0.5% per day. This is done deliberate in LWR to save fuel (and therefor cost). Typically, a couple of weeks with coast down is planned and saves between 2 and 8 assemblies depending on length and reactor type.

5.3.2. Maintenance

Maintenance schedule should be considered as a part of planning of the fuel cycles. Certain safety classified equipment's needs regular maintenance which can be done only during the refuelling, or maintenance outages, and it that way as well influences for fuel cycle design.

Demands to periodical inspection of the equipment and other restrictions established in the license or in the technical specifications should be taken into consideration as well. Some requirements for fuel cycle planning based on ISI programmes and national regulations and technical standards which are applied to the design and operation of their nuclear power plants. Normally these requirements are defined and summarized in different codes and standards like ASME section XI, RSE–M, CSA, etc.

5.3.3. Licensed fuel

Licensed types of the FA should be only used for the loading to the core. If any type of fuel is promising to be more effective in achieving declared goals, this new type of the fuel assembly should be licensed according to the acting procedure.

5.3.4. Replacement of defective fuel

Possible ways: to use irradiated FA from the spent fuel storage (or planned for unload), or to use spare fresh FA having suitable enrichment. Replacing partially burned fuel assembly with another one having different k_{inf} may cause non–desired asymmetry in core power distribution.

In the case of damaged fuel assembly unexpected unload, it has to be replaced in the core.

Possible options are:

- To replace it with the fresh fuel assembly;
- To replace it with the irradiated fuel assembly initially planned to be removed from the core;
- To replace it with the irradiated fuel assembly stored in the pool since previous fuel cycle(s);
- To repair damaged fuel assembly during the outage.

As for repairing there may be not enough time during the outage to repair damaged FA.

A fresh fuel assembly having low enrichment may be more suitable for replacing damaged fuel assembly after first cycle. Number of such spare FA has to be determined in advance, based on operational experience, statistic, fuel price, etc.

Loss of symmetry is almost inevitable in all the cases, except if the central fuel assembly fails and need to be replaced. In the case of repairing symmetry may be affected not as seriously.

CANDU operates normally an on-line monitor system to scan Xe and Cs intensities. If the system shows those intensity increase, it means that there is a possibility of fuel failure. In that case, the site nuclear engineer identifies the location of the damaged fuel element or bundle, and then releases all damaged fuel through on–power refuelling and loads the new fuel bundles. Damaged fuel is stored separately in the spent fuel pool. Compared to light water reactors, the damaged fuel treatment is very easy.

5.4. FUEL CYCLE OPTIMIZATION

Developing of a new fuel cycle includes its optimization. The word ‘optimization’ may mean optimization of physical characteristic or economics.

5.4.1. Multi cycle optimization

In multi cycle optimization even defining the target function is challenging in an environment where the production plan for the future is changing. In situation where everything is fixed the goal is to reach the target burnup with as low enrichment with as few bundles as possible. In a moving environment the target function needs to weight the relationship between a profit today that might cause a loss in the long run vs a cost today that will lead to a profit in the future as we know it today:

- Define objective function F;
- Define number of fuel cycles for optimization;
- Set weight for each fuel cycle.

Target function for multi cycle optimization may look like:

$$F_m = \frac{sum(W_i * F_i)}{Sum\ (W_i)} \tag{12}$$

Where:

- W_i = weight for fuel cycle i;
- F_i = value of objective function for fuel cycle i.

5.4.2. Equilibrium cycle optimization

5.4.2.1. Why equilibrium

Equilibrium fuel cycle may be useful to develop at least in the following cases:

- For the new type of a reactor;
- If requirements of operation may be changed remarkably (power uprate, long fuel cycles implementation, etc.);
- If economic circumstances are changed remarkably and it is expected that they are changed for long time;
- If real characteristics of using fuel cycle do not meet expectations. In that case it may have sense to start developing a new equilibrium fuel cycle, but only after the reason of deviation is ascertained;
- For a new fuel type.

As a rule, a designer of the reactor development is in charge for the design of the equilibrium fuel cycle. But if it is possible to improve the equilibrium fuel cycle remaining within licensed fuel types and within design limits, the plant’s staff may develop an improved cycle, adapting it to changed requirements or circumstances.

5.4.2.2. Types of equilibrium fuel cycles

Equilibrium fuel cycle — is a fuel cycle in which core discharge, reload and shuffle are carried out, according to the same plan, or according to the same plans rotating after strict number of cycles.

(1) Last part of the definition mostly is suitable in the case a reactor has a central cell, and FA in the central cell remains in the core for two or more fuel cycles in a row. Number of fresh FA is odd for the first core and even for the following fuel cycles, until a new fresh FA will be loaded into the central cell and the cycle is repeated.

Number of the fresh FA may be loaded, for example, according to the chain

67 – 66 – 67 – 66 …

or

55 – 54 – 54 – 55 – 54 – 54 …

In the first example a central FA remains in the core for two fuel cycle, in the second example for three fuel cycles. Both examples are suitable for WWER–1000.

(2) Another example of equilibrium fuel cycle having central FA replacing every campaign but having two different patterns for refuelling: core patterns of two campaigns in a row may be partially symmetrical so as some of FA may remain in the same place for the second campaign. It may be intentionally done to reduce the number of permutations during refuelling, for example.
(3) Number of fresh FA may in two or three campaign may vary for more than one central fuel assembly.

Example:

66 – 66 – 61 – 66 – 66 – 61…

79 – 84 – 79 – 84 …

5.4.2.3. Possible demands for the equilibrium fuel cycle

- Minimal and maximal length of the fuel cycle are established. Length of all the fuel cycles must be between minimal and maximal values;
- Possible length of the coast down period. It must be clear whether coast down is included in the full length declared in previous paragraph;
- To reduce primary cost of electricity as possible;
- Limit for maximal enrichment;
- Whether only existing types of fuel must be used, or it is possible to develop new types. The difference may be, for example, in number and/or concentration of burnable poison, mass of fuel, etc.

5.4.2.4. Additional demands need to be checked during design

- Filling of the spent fuel storage, number of free cells for all the time of operation of the unit;
- Whether or not a new safety report is needed (for example, if enrichment is increased);
- What kind of corrections in licensing and operational documents are needed if any;

— Whether some kind of modernization of the equipment is needed. For example, ne casks, racks in the spent fuel pool, cranes, etc.

5.4.2.5. Economic optimization of the equilibrium fuel cycle

Possible economic objective functions may be simple but reflect most important dependencies.

As an economic objective function may be used primary cost of electricity and average annual profit.

In the last case the price for electricity is constant.

Several parameters influence the value of the objective functions. First, these are the parameters directly connected with the fuel usage, such as cost of the fresh fuel upload, cost of irradiated fuel handling, length of the fuel cycle. Cost of the fresh fuel upload in turn depends on the number of FA and types of FA, prices of fuel assembly of every type.

Some parameters which are not directly connected with the fuel may influence the objective function as well. Among these parameters are salaries and other annual costs and cost for maintenance. Profit depends on the price of electricity (see Appendix I).

As a rule, the next principles are correct:

— For fuel cycles having equal length increasing enrichment leads to reducing fuel component of the price of electricity;
— The longer the fuel cycle the higher the fuel component of the price of electricity is;
— For fuel cycles having equal enrichment of the fresh fuel the longer the length of the fuel cycles the less the primary cost of electricity is.

These principles may be used as a lighthouse in developing new equilibrium fuel cycles.

5.4.2.6. Recommended steps for designing equilibrium cycle

Recommended steps if desirable T_{ef} (effective length of cycle) is established:

(1) To build a set of possible combinations of the fresh fuel assemblies to meet demands for T_{ef}.

From the experience of design of equilibrium fuel cycles for this type of reactor it may be known rough estimate for functional dependence:

$$T_{ef}\ vs.\ n * x \tag{13}$$

Where:

— n = number of the fresh fuel assemblies loaded into each fuel cycle;
— x = average enrichment of all fresh fuel assemblies, x = (x1+x2+x3+…)/(n1+n2+n3+…);
— Where ni is number of fresh fuel assemblies having enrichment xi, n1+n2+n3+…=n.

For this purpose, linear dependence is good enough:

$$T_{ef} = A + B * n * x \tag{14}$$

Where:

- A and B are coefficients.

If such a dependence is unknown, it may be useful to build it, by designing several simple equilibrium fuel cycles having (approximately) the same pattern, but different initial enrichment. It is preferable if the pattern is similar or close to the pattern to be used for designing the final equilibrium fuel cycle.

It is quite enough to have linear function.

Let us assume that for design of the core the FA belonging to the set of possible enrichments x_1, x_2, x_3, … and $x_1 > x_2 > x_3$ … may be used.

As a first approach n·x may be estimated:

$$n * x = \frac{(T_{ef} - A)}{B} = D \tag{15}$$

Knowing D, it is possible to build a set of combinations for fresh FA driving T_{ef} close to what is needed.

Number n must be rounded to the nearest natural divisible by 6 for the cores consisting of hexahedral FA, and divisible by 4 for the cores consisting of square FA.

Table 4 shows an example of some of possible combinations of number of fresh fuel assemblies and its enrichment for n*x = 3,2.

In fact, there may be several types of FA having different amount of burnable poisoning. But it slightly influences T_{ef}.

TABLE 4. SOME OF POSSIBLE COMBINATIONS OF NUMBER OF FRESH FUEL ASSEMBLIES AND ITS ENRICHMENT FOR N*X = 3,2

Number FA having enrichment 5,0%	Number FA having enrichment 4,4%	Rounded number FA having enrichment 4,4%	Number of fresh FA	Average enrichment of fresh FA
66	0	0	66	5,0%
60	4,5	6	66	4,9%
54	11,4	12	66	4,9%
48	18,2	18	66	4,8%
42	25,0	24	66	4,8%
36	31,8	30	66	4,7%
30	38,6	36	66	4,7%
24	45,5	48	72	4,6%
18	52,3	54	72	4,6%
12	59,1	60	72	4,5%

Number FA having enrichment 5,0%	Number FA having enrichment 4,4%	Rounded number FA having enrichment 4,4%	Number of fresh FA	Average enrichment of fresh FA
6	65,9	66	72	4,5%
0	72,7	72	72	4,4%

(2) According to the principle 3 to have minimal primary cost it has sense to try to use fuel having maximal possible enrichment. For the example given in the table 6 that means, if 5,0% is a maximal possible combination, it is better to start to design an equilibrium fuel cycle using 66 fuel assemblies having enrichment 5,0% without using 4,4% fuel assemblies.

To design equilibrium fuel cycle which fits safety limits and requirements.

(3) In the case of failure to create the core meets safety requirements compound of 66*5,0% may be changed to the next combination 60×5,0% + 6×4,4% and so on.
(4) To make economic estimations of designed fuel cycles efficiency. If possible, it is important to use real prices for used fresh fuel. Price may be different for the FA having the same enrichment due to, for example, different amount of burnable absorber.
(5) For each variant to make full check for safety and other demands, like number of empty cells in spent fuel pond, possibility to use existing casks for both fresh and irradiated fuel, etc.
(6) Try to use a fuel assembly having one of enrichments used in this cycle. If central fuel assembly has to be in the core for more than one campaign, it is necessary to develop additional cores.
(7) For the best solution conduct one more time optimization of neutron–physical characteristics, paying attention to the minimal gaps.
(8) To develop transient fuel cycles for existing unit, or first core and transient fuel cycles for a new unit.

5.4.2.7. About central fuel assembly

For equilibrium fuel cycles having central FA operating during more than one cycles it is necessary to design every cycle and to check safety requirements. On the stage of preliminary design, it may be not quite convenient and requires more time.

So, to save time fresh FA having lower enrichment or irradiated FA after several cycles instead of unload may be placed in the central cell. This FA is replaced after each cycle. At the final stage of design this FA has to be replaced with the FA having one of enrichments used in this cycle.

Placing in the centre fresh FA having 'normal' enrichment but unloaded after every cycle may reduce economics of cycle.

5.4.2.8. Filling of the spent fuel storage and number of empty space estimation

Usually limits for maximal number of irradiated FA in the spent fuel pond exists. This limit may be a result of a demand to unload whole core in the case of necessity, or to make empty one section of the spent fuel storage. In that case it is important for given equilibrium fuel cycle to estimate maximal number of FA that will be finally stored.

Residual heat of irradiated fuel grows faster than burnup. Depending of the initial residual heat the cooling time may be quite different for FA unloaded at the same time.

Burnup of every fuel assembly is calculated according to the core pattern of the considering equilibrium fuel cycle.

In the simplest case when the same number of fresh FA is loaded into the core every fuel cycle, burnup distribution among FA is the same for every fuel cycle.

For every FA minimal duration of the cooling time may be calculated based on the type of shipping cask and permitted burnup. This minimal duration may by calculated in years. For NPP's where shipment is possible only during the outage it may be more convenient to count the number of fuel cycles.

Suppose F_i — number of irradiated FA which may be transported of the site in 'i' years (or fuel cycles).

$$F = \sum_{i=1}^{G} F_i \tag{16}$$

Where:

- G is the longest cooling time among all of the F fuel assemblies.

For the equilibrium fuel cycle number **n** of irradiated FA in the storage is:

$$n = \sum_{i=1}^{G} i \cdot F_i \tag{17}$$

Equilibrium number of irradiated FA in the pool may be reached with the remarkable time shift after the equilibrium core is loaded into reactor for the first time. This time shift depends on the time needed to cool down 'the hottest' FA. Usually, but not necessarily, that is the FA having the highest burnup.

Average burnup of unloading FA depends on cycle length and number of fuel unloading assemblies, which is the same. If because of economic reasons it is undesirable to increaser number of fresh FA, it is recommended to reduce time of cooling to make burnup distribution among irradiated FA as uniform as possible.

MOX–fuel has increased residual heat rate and longer cool down time.

5.4.2.9. Optimization of transient fuel cycles

For optimization transient fuel cycles, the same approach may be used as for optimization several cycles in a row.

5.4.3. On power refuelling on CANDU

Except when a reactor is loaded with fresh fuel in all fuel channels, reactor refuelling is a continuous process, with about a dozen channels being refuelled each week. On–power refuelling is the primary means of maintaining a CANDU reactor criticality. During the transitional period which follows refuelling, the reactor gradually approaches equilibrium, with the refuelling rate rapidly approaching the time–average value (for example, approximately 15–16 bundles per FPD in the CANDU 6).

During the transitional period, core–follow calculation should be performed without refuelling at site. Site engineers check the transition core to comply with license limits such as maximum channel/bundle power limit, to ensure that there are no ‘hot spots’ or large power tilts. After transitional period, each run of the diffusion code in the core–follow requires inputting the instantaneous three–dimensional irradiation/burnup distribution. This of course comes from the output of the preceding run. In addition, each run of the core–follow requires modelling of all channel refuelling, which have occurred since the previous run (and their timing). This allows modelling bundle movements and the entry of fresh FA in appropriate locations. The instantaneous positions of reactivity devices are also input. From the power calculated and the time step used, the fuel irradiation in each individual bundle are updated, and the lattice properties updated. The output of the current run becomes the starting point for the calculation at the next time step (i.e., the next EFPD). The current three–dimensional power and fuel–irradiation distributions obtained serve then specifically as the basis on which intelligent selections of channels to refuel are made.

One of the significant outputs from core–follow calculation is the determination of the Channel Power Peaking Factor (CPPF), which is used in the calibration of in–core protection–system (Regional Overpower/Neutron Overpower) detectors. At any given time, there are several channels in the core which are at or near the maximum power in their power cycle. Therefore, the maximum instantaneous channel power is always higher than the maximum channel power in the time–average power distribution model. The CPPF quantifies the degree by which the instantaneous power distribution peaks above the time–average distribution so that it is a very important parameter for setting the core protection system’s trip penalty. Therefore, keeping the CPPF value as low by selecting refuelling channels is on–going duties of the fuelling engineer or reactor physicist at a CANDU nuclear generating station.

Therefore, based on the above mentioned, site engineers have to establish a list of channels to be refuelled during the following period of operation by using computer simulations with on-line flux mapping system information, in–core detector readings, zone–control–compartment water fills, and plant–specific rules, here are some examples of plant–specific channel selection rules:

- Channels for which the time interval since the last refuelling is approximately equal to the channel dwell time;
- Channels with high exit burnup;
- Channels with low power relative to their time–average power channels;
- Channels which, taken together, promote axial, radial and azimuthal symmetry and a power distribution close to the reference power shape, etc [5, 20].

5.4.4. First cycle design

5.4.4.1. PWR initial core design

As a rule, a designer of a reactor develops a design of the first fuel cycle together with the equilibrium fuel cycle and simultaneously with the developing design of a new reactor.

A new first core design may be developed for the serial reactor before commissioning in the case if operation of a headwork unit of this type eliminated some imperfection of the first core. In that last case the plant may be involved in the developing of a new first core.

Design of the first cores features:

- After first cycle is developed it is necessary to design all transient fuel cycles to reach equilibrium fuel cycle. That means, that equilibrium fuel cycle has to be developed first.

  Optimal consequence of the developing first core is:

  equilibrium cycle — first core — transient cycles

- To make field of power more plain FA having different enrichment are used. It may be, for example, three or four groups of FAs having enrichment ranges between 1.6–2.0% and 4.4-5.0%. If possible, it is recommended to use the same maximal enrichment as maximal enrichment used in equilibrium fuel cycle. It may allow to reach equilibrium fuel cycle faster;
- Neutronic properties are distributed symmetrically within the cross–section of FA, and they are the same for one group of FA. That means that it has no sense to change place of FA within a group. Number of possible permutations is reduced dramatically for the first core;
- At the beginning period of the first fuel cycle the increase of power may take up to several months before nominal power rate is reached. So, during the begin period of the cycle it is not necessary to have first core meeting the same requirements as for other period of the cycle in terms of power distribution. It is quite enough to meet all the requirements soon before the power is increased to the nominal level;
- If a demand exists to shutdown the reactor for the revision of the equipment within certain period of time after commissioning, it may limit length of the first cycle. In that case it is necessary to consider an expected scheduler of power increasing, because it influences the effective length of the first cycle;
- After first cycle is designed it is necessary to develop 3–5 transient fuel cycles until equilibrium is reached. For the first fuel cycle, as well as for transient ones, it is necessary to pay attention to temperature coefficient reactivity and local power distribution;
- When developing transient fuel cycles, it is useful to try to use the same set of the fresh FA and the same pattern as for equilibrium fuel cycle;
- It is appropriate to make economic optimization for the first cycle and for all transient fuel cycles simultaneously. If the price of the first core is included into the capital cost of the power unit it has to be taken into account.

5.4.4.2. CANDU initial core design

At the very start of reactor operation, the entire initial fuel load (0.7% natural uranium fuel bundles and several depleted uranium bundles) goes through the plutonium peak at about the same time (about 40–50 FPD) by controlling poison material (B or Gd) concentration to suppress excess core reactivity. At this juncture, the core reaches its global plutonium peak and the core reactivity is the highest it will ever be. Following the plutonium peak, the plutonium production can no longer compensate for the depletion of U–235 and the build–up of fission products, and the excess core reactivity decreases. Also, at this time moderator poison must be removed as the excess reactivity drops gradually to zero, at about FPD 120. During this entire first period in the reactor life, refuelling is not necessary since there is already excess reactivity.

The core–follow for this period is therefore rather simple. It is a matter only of simulating the core every few days with instantaneous reactivity devices and a poison concentration which yields a critical reactor. When the excess core reactivity has fallen to a small value about 10 or 20 FPD before it reaches 0 (i.e., typically around FPD 100), refuelling operations start. It is best not to wait until excess reactivity is exactly 0, because the initial refuelling rate would prove too high [5, 21].

To show for the initial reactor to satisfy the safety margin, core follow calculation for about 450 FPD has been performed by using RFSP computer code.

5.5. CHANGES IN CIRCUMSTANCES (EXCEPTIONS)

5.5.1. External and internal circumstances

As the NPPs changes over the course of its licence's lifetime, internal and external circumstances may influence the core design and associated reload strategy. Some of changes may be predictable and anticipated, others relatively or totally unexpected.

Depending on what has happened, a spectrum of possible consequences exist- from changing design of only one core for the nearest cycle in one unit, to changing reload strategy, equilibrium fuel cycle including developing all necessary transient cycles for several units or NPPs.

In some cases, fuel supplier may be influenced.

Examples of external circumstances:

- Changes in laws;
- Changes in regulations;
- Changes of the electricity market;
- Variation in electricity demands;
- Changes of the fuel price;
- Appearing new types of the fuel; etc.

Examples of internal circumstances:

- Unplanned outage;
- Multiple failures of FA;
- Changes in company policies;
- Outage length changes; etc.

Some of these events have a limited effect, only influencing current and one to two of the subsequent fuel cycles. Others may have long term effects, demanding a change to the core design, reload strategy or equilibrium fuel cycle. This last situation also may influence current fuel cycle and all the transient fuel cycles to the new equilibrium fuel cycle.

For CANDU, there was no explicit burnup limitation for a fuel bundle. However, Korean regulatory body recently requests the fuel bundle maximum burnup limit reflecting fuel performance modelling code uncertainties. When a fuel engineer designs a fuel bundle, he/she considers the conservative power history to reflect computer code uncertainties as well as power operating fluctuation itself. However, there is no explicit evidence the conservative power history can cover overall code modelling uncertainties. In 2017 ~ 2018, COG launched R&D

programs to identify model uncertainties and estimate their effects on the key output parameters such as fuel centreline temperature, gap pressure, fuel bundle inventories, and sheath stress.

6. CHALLENGES

6.1. FUEL RELIABILITY

6.1.1. Fuel defects

Fuel defects are a wide concept including several different issues such as defects in manufacturing, handling and operations. Since fuel cycle economy is discussed here focus will be on fuel that fails during operation. Fuel that experience defects during manufacturing and handling is off course a cost but has minor impact on operations and therefor also on fuel cycle cost.

Damaged fuel is generally understood to have occurred when one or more fuel rods in the core have experienced a cladding failure resulting in emissions of fission gases. The detection of these gasses in the coolant is the first indication that fuel damage has occurred. This is called the primary failure. With time the rod will develop a secondary degradation due to hydrogen in the cladding often resulting in a larger opening in the cladding (secondary failure) which in turn results in emission of uranium from the rod.

From a fuel cycle cost perspective, the most interesting questions is if the failure leads to a mid–cycle outage. Extra, unplanned outages change first of all the needed reactivity during the cycle. If the core is loaded with reactivity to reach cost down one week before the next planned outage than an unplanned stop for more than a week will cause the reactor not to reach cost down before shutdown which is not optimal with regards to fuel cycle cost.

Mainly there are two situations, with regards to damaged fuel, that will lead to an extra outage. First, emissions of fission gases and eventually uranium, reaches pre–determined action levels causing the NPP to shutdown. These action levels should be specified in the technical specifications originating from the safety analysis. Second, information from manufacturer or other NPPs leads to a suspicion of risk of fuel failure which is so severe that the NPP decides to shutdown before the fuel fails.

6.1.2. Fuel reliability trends

There are several kinds of reasons for FA to fail. Today the most common reason for fuel failure in core [22] is grid to rod fretting and debris fretting.

Pellet Cladding Interaction (PCI) which was the biggest problem some decades ago largely mitigated due to better mechanical design of FA using liner (a thin layer of zirconium alloy on the inside of the fuel rod that is soft and therefor reduces tensions that otherwise would result in a failure) and operating rules like maximum power increase per hour. Most failure modes due to thermal margins (LOCA, rod internal pressure, etc.) have been using more conservative operating thermal limits much in the same way as PCI is mitigated. However, since the strive for higher burnup and higher power output from each bundle is ongoing, these kinds of issues may appear.

The same is true for fuel material problems. At current burnups they work fine but NPPs might run into problems as the operating envelope for the fuel is expanded.

The general trend is that all types of failure mechanisms is going down, with some local exceptions, but in order to achieve a good fuel economy the failures that leads to extra outages needs to be continuously mitigated.

It is most important to understand the weaknesses of a certain fuel type before procurement and evaluate its impact on fuel cycle economy before procuring fuel manufacturing.

6.1.3. Leaking fuels, impact on fuel core economy, solutions

A primary fuel failure itself is not a problem for continued operation and a successful fuel failure management plan needs to focus on keeping the leaker primary and avoid degradation. The most widely used method today is to detect the leaker through flux–tilting and suppressing the leaker by inserting the control–rod in the leaker cell [23].

This methodology has shown successful in avoiding degradation of the leaker and avoiding a mid–cycle outage. From a core design perspective this methodology offers some challenges since the suppression causes a power tilt in the reactor which in many cases will the thermal margins and will require bigger design margins. Suppression will also cause an asymmetry and PCI restrictions that need to be facilitated in the coming core designs.

For PWR the practice is to continue operation till the scheduled outage, if activity of the coolant remains within operational limits, and then to find and replace damaged fuel assembly during refuelling.

6.2. FUEL CYCLE AND BACK END STRATEGY

Usually the back end cost is measured as a cost per spent FA or the amount of heavy metal (HM) in the assembly. The cost might be the cost of spent fuel cask for interim dry storage or part of constructing the final repository or the cost of reprocessing the fuel. Each NPP have different prerequisite for how to handle spent fuel.

As a rule, increasing enrichment and burnup in the FA in order to minimize the number of spent fuel FA let to decrease back end costs, because average annual weight and amount of the spent fuel transported from the NPP is decreasing.

In some cases, increasing enrichment and burnup in the FA in order to minimize the number of spent fuel FA can lead to higher back end costs, if it leads to longer periods in the spent fuel pool to cool down and that fewer FA can be loaded in a single spent fuel cask. Even if the optimization leads to fewer FA used, the result might cause more transports of spent fuel to the interim storage and more fuel in the pool since the FA has to spend much longer time in the pool before transport than before increasing the burnup.

A long term planning with regards to outages and transports is important so that there is sufficient place for the spent fuel to be places in the spent fuel pool during the outages.

6.2.1. Spent fuel handling as a part of reload strategy

When redesign of the equilibrium fuel cycle is planned, especially if the reload strategy itself is going to be revised, it is important, among other, to estimate consequences of that in terms of irradiated fuel handling.

Usually it is necessary to consider options which are considering keeping irradiated fuel in the spent fuel storage and possibility for transportation outside the plant.

It is necessary to estimate maximal burnup to check if its value fits the demands and limits of the shipping cask.

It is important to estimate the number of irradiated FA in the storage as well as the number of empty cells remained to meet the regulation demands. In the case when minimal duration of the cooling time depends on burnup it may influence the number of empty cells in the spent fuel storage.

6.2.2. Average irradiated fuel burnup

Suppose that T_{ef} is effective length of the fuel cycle:

- m = mass of uranium in one fuel assembly;
- W = heat power of the reactor, MW; then

Energy production during one fuel cycle is:

$$WT_{ef}, MW\ day \quad (18)$$

Suppose every fuel cycle the load has F fuel assemblies. Total mass of uranium in the core is:

$$Fm, kg \quad (19)$$

Average increase of burnup per one fuel cycle is:

$$B = WT_{ef}, MW\ day \quad (20)$$

Energy production during N fuel cycles is:

$$WT_{ef}N, MW\,day \quad (21)$$

During N fuel cycles F·N fuel assemblies were loaded into reactor. Average burnup of one fuel assembly is:

$$\frac{WT_{ef}N}{FN} = \frac{WT_{ef}}{F}, MW\ day/FA \quad (22)$$

Average burnup B_{av} per one kilogram is:

$$B_{av} = \frac{WT_{ef}}{Fm}, MW\ day/kgU \quad (23)$$

Maximum and minimum burnup of unloaded fuel assembly may vary depending on the irradiating history of every fuel assembly.

6.3. MOX–FUEL APPLICATION AND IMPACTS FOR THE CORE DESIGN

6.3.1. Impacts for the core design

While designing the core having MOX–fuel the following features should be taken into consideration:

- Harder neutron spectrum;
- Lost in worth of scram and control rods;
- Axial fuel rods power variations depending of the design of the MOX–fuel assembly;
- Smaller fraction of delayed neutrons;
- Limits in amount of MOX–fuel established by license.

6.3.2. Irradiated MOX–fuel handling

Necessary cooling time for the irradiated MOX–fuel should be calculated in advance depending of the initial amount of plutonium and burnup of the FA.

6.3.3. Economic of the MOX–fuel

Influence of using MOX–fuel on the economic performance of the fuel cycle should be estimated with an allowance for risk of possible uncertainty.

6.4. OTHER ASPECTS

6.4.1. Using coast down

To use coast down regime means to continue to operate a reactor after reserve of reactivity is finished because of burning fuel.

Due to negative power and temperature coefficient of reactivity, reducing power (and average temperature, as well, as a rule, temperature of the coolant in the inlet) adds positive reactivity in to the core.

That provide a physical possibility to continue operating of the reactor, albeit on gradually reducing power.

Idealized curve for the power during coast down period is shown in the Figure 12. Pace the power dropping depends of the core's physical characteristics. In real operation power is usually reducing by the small steps, ones a day, for example.

Simplified equation for connection between calendar and effective length of coast down period for the case when pace of reducing power during coast down period is constant and initial power is 100% or 1.0:

$$T = ln(1/(1–\Delta T_{ef} \cdot \alpha))/\alpha \quad (24)$$

Where:

- T–calendar length of coast down period, days;
- ΔT_{ef}–effective length of coast down period, eff days;
- α – pace of reducing power during coast down period, % of initial power/one effective day.

Value of ΔT_{ef} has a not achievable limit:

$$\Delta T_{ef} < 1/\alpha \quad (25)$$

If the length of the coast down period is short, simpler equation may be used for a quick estimation:

$$T = \Delta T_{ef}/N_{av} \quad (26)$$

Where:

- N_{av} is average power during coast down period.

$$N_{av} = 1–\alpha \cdot \Delta T_{ef}/2 \quad (27)$$

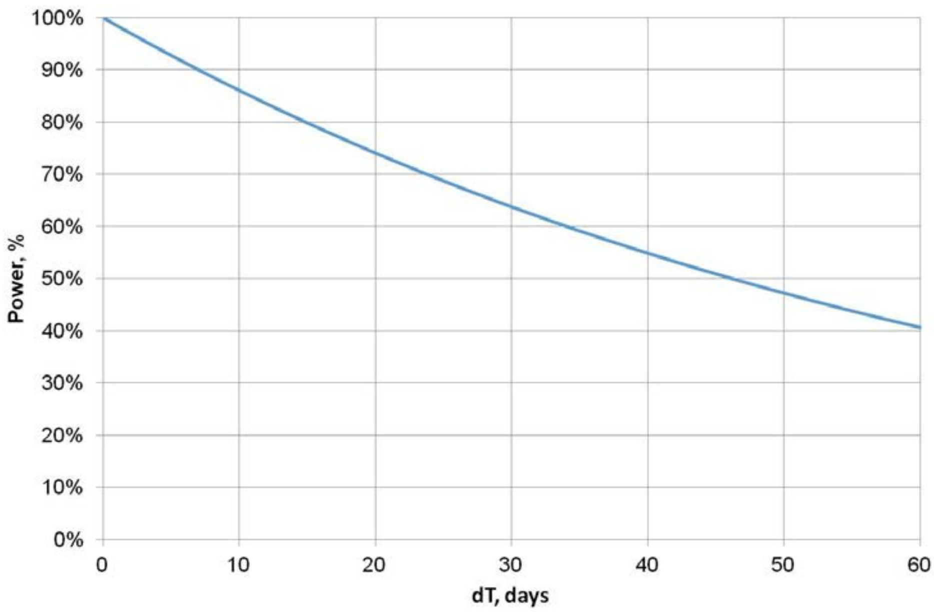

FIG. 12. Power decreasing example during the coast down period.

A limitation may be established by the fuel supplier or by the designer of the plant for the minimal power at the end of the coast dawn period. In that case that means a limit for the maximal length of the coast down period. In turn, it may be a reason for the limitation of the power increasing pace at the beginning of the next cycle after refuelling.

In terms of economy working using coast down let to reduce primary cost of electricity. Optimal length of the coast down regime mainly depends on fuel to total cost ratio and other parameters.

In the case if parameters vary from one cycle to another, optimal length of the coast down regime, Figure 13, may be calculated for each cycle if needed. Typically, its value is situated within 15–40 effective days. Reducing in primary cost is specific for the NPP and normally may be within 1.0% of the primary cost without coast down.

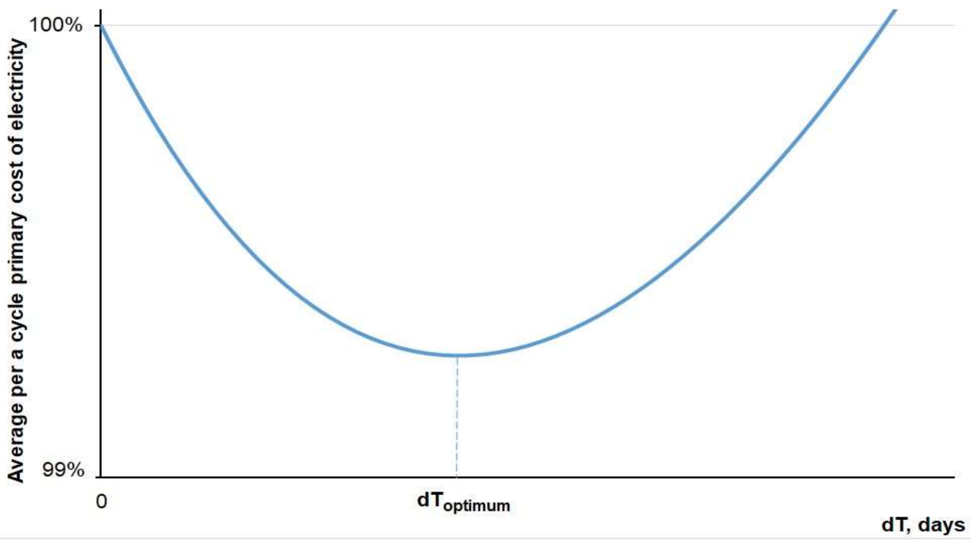

FIG. 13. Primary cost of electricity vs length of coast down period.

Advantages:

- It is economically reasonable to use coast down. Optimal length of coast down period exists, and it is specific for NPP;
- It gives possibility for changing length of the cycle, for shifting date of maintenance within certain limits;
- In some cases, it may help to make negative temperature coefficient for the next campaign.

Disadvantages:

- Work in coast down during more than certain period, depending on type of fuel rods, may demand special scheduler for increasing power at the begin of the next cycle;
- Too long coast–down may cause problems in turbine operation.

7. FUTURE PROSPECTS

7.1. DIRECTIONS FOR FUEL DEVELOPMENT

At the time of publication, fuel research and development trends to develop new fuel materials to mitigate the issues with hydrogen release from Zirconium alloys during accident scenarios, so called accident tolerant fuel (ATF). Depending on the fuel design, ATF might also be an effective remedy against debris fretting.

One possible way to reach the goal is to use coating on outer (and possibly on inner) surface of cladding. Different coating materials and technics are under investigation such as Cr, SiC, FeCrAl and others. Coated cladding can be thinner, increasing moderator volume, improving fuel cycle economics.

New ceramic cladding material can be even more beneficial in anticipated accident cases, but thickness of these should be 1–1.5mm, reducing place for fuel and moderator in the reactor core.

New pellet material such as U_3Si_2 can provide much better heat conductivity and higher uranium content related to actual UO_2 pellet.

This is matching with tendency trying to maximize the uranium mass in the assembly in order to gain economic benefits like reduced enrichment or higher k_{eff}.

General request from utilities is to have long cycles and high nominal power. One possible way to reach is to use U–235 enrichment beyond 5%, it is under investigation using special burnable poison such as Erbium in early phase of fuel fabrication.

The last year's pellets with additives have been introduced in order to gain uranium weight and increase performance with respect to PCI and other thermal margins.

For BWRs in general, more powerful computers and codes gives the opportunity to analyse more advanced lattices in the assembly. It is also possible, to a greater extent, use different length of fuel rods, water channels and other ways of allowing for more moderator in the assembly and keeping the optimum uranium/water ratio with increasing void in the upper parts of the assembly.

As for Accident Tolerant Fuel (ATF), Korea also has been run several R&D programs for about 10 years to develop fuel ingredients or sheath coating materials by several R&D labs. At the time of writing, KHNP, a nuclear utility in Korea, has planned to launch a new program to commercialize the previous R&D results, especially sheath coating materials having crud resistances. One of the main targets of the R&D program is to develop the coating technique from lab scale to commercial scale.

7.2. DEVELOPMENTS OF COMPUTER CODES

The development of existing computer codes meets the general legislation requirement to use state–of–the–arts methods. There are some possible ways in computer codes development which can support better utilization of nuclear fuel:

- Development of more accurate neutron physical codes system which models the core operation more reliably in acceptable time. This code system could contain more complicated and quite simple codes for different purposes (e.g., diffusion — differential vs. nodal vs. FEM, transport, Monte–Carlo approaches);
- Development of more accurate models in Multiphysics codes which are able to model in–core and ex–core detectors behaviour in different operational regimes. It can help better understanding of connection between calculated power distribution (from previous paragraph) and measured value (e.g., neutron transport calculations, CFD calculations);
- Development of more precise codes used in Core Monitoring System (e.g., based on pin–wise neutron physical model, subchannel thermal–hydraulic model);
- Development of more powerful loading pattern optimization codes using advanced mathematical optimization methods and parallel/quasi–parallel approach which allow more flexible core designer work.

First two approaches allow decrease the uncertainties used for design core limits derivation. The third one allows decrease the uncertainties for operational limits derivation. The fourth one provides to analyse the space of possible loading patterns more reliably.

The approach described above is used in NPP fuel cycle management in the Czech Republic.

The US Department of Energy (DOE) established the Consortium for Advanced Simulation of Light Water Reactors (CASL) in 2010 to confidently predict the performance of existing and next–generation commercial nuclear reactors through comprehensive, science–based modelling and simulation. It will be finished in 2020. According to USA interests, Korea also launched a similar but smaller project to develop its own Multiphysics code using in–house codes from cross–section generation code to chemical behaviour estimation of the primary heat transfer system. It will give us some information about the fuel behaviour including crud induced reactivity change during normal operation and about high burnup fuel dynamics during LOCA.

7.3. FUTURE CHALLENGES FOR CORE DESIGN

7.3.1. Knowledge and experience transfer for next generation

From the 1970s and 80s the 1st generation of engineers got involved to the core design works. They collected knowledge and experiences gradually, using more and more advanced tools and methods for the work. Often contributed to develop computer codes and introduced it into practice. Aging this generation, the transfer of this knowledge for the next generation required special techniques.

The CANDU computer codes have been managed by COG up to now. However, there is a small number of PHWR in the world and no new or upgraded PHWR has been revealed, so the demand for code development is low. Moreover, as developers get older, sustainability of code management is concerned. Therefore, it is necessary to solve the long–term problems by keeping the maintenance personnel of the computer code or analyzing the PHWR through the high–fidelity and multi-physics codes for safe operation of existing PHWR plants. There are already several joint projects for developing high–accuracy multi-physics code to solve LWR's safety issues and simulate LWR core behavior more accurately. If those codes use a more flexible geometry control module, they are able to simulate a PHWR normal operational condition in the future. In view of knowledge transfer from generation to generation, KHNP opened new educational courses in 2019 to those who have recently joined into the core design and management department in sites. And COG also opened additional computer code training courses through KHNP's request from 2017 to any site guys who want to manage their own PHWR efficiently

7.3.2. Reducing number of experimental facilities

There is a worldwide tendency to close small reactors built for experiments and trainings. Investigations concerning improvements of fuel component getting more problematic, reducing opportunities to use advanced fuel for reactors and improvement of fuel cycles.

7.3.3. Solutions for utility specific problems

As for AOA (Axial offset anomaly), KHNP made a response procedure in 2017 to reduce the effects of severe AOA condition to power plant based on the EPRI recommendation. However, the current AOA risk evaluation is needed to improve especially in a view point of crud distribution evaluation. KHNP is going to improve AOA prediction capability by using more detail information such as pin–wise thermalhydraulic condition, assembly–wise cross–section correction related to crud thickness, and water chemical prediction model, etc.

In the case of CANDU, the aging of pressure tubes affects the trip setpoint of the reactor protection system and leads the trip setpoint decrease with operation time. It means that the total reactor power is also going to decrease. At present, the aging condition of a pressure tube is within the expected range. However, the ageing status will go over the limit for some CANDU or CANDU–6 reactors before plant lifetime. if there is no plan for refurbishment, CANDU needs thermalhydraulic experiments to develop a new critical heat flux model and related correlations based on the expected aging status. And, one should perform the full scope of safety analysis to prove the operability and safety of the over–aged core.

8. OPERATING EXPERIENCES AND LESSONS LEARNED

8.1. FORSMARK, SWEDEN

To reduce the effects on fuel cycle economics there are two ways to handle big outage delays. The first is to move the next outage the same amount as the delay. This fixes the problem with the impact on fuel economy but will have implications in maintenance planning for the coming outage. It can also be impossible if the unit is operated in a market where the power demand varies a lot over the year and the moved outage would end up in the high–power demand season of the year.

If moving the outage is not possible another option is to change the core design or even reload the core if the core has already been loaded. One example of where this was done at unit 3 at Forsmark NPP in 2015. The outage was delayed for two months and the new core (12 months cycle) was loaded when the decision was made to reload the core and unload as much fresh fuel as possible. With a minimum number of shuffling operations about one fourth of the fresh batch could be unloaded and compensate for a large part of the fuel loss that would otherwise have been incurred from the delayed outage.

8.2. PAKS, HUNGARY

Starting from 90s, power uprate of existing LWR units as well as changing strategy from annual to longer fuel cycles became a worldwide tendency. In the beginning changes caused unexpected challenges: high power peaks and non–desired consequences appeared due to highly enriched fuel. Burnable absorber application in advanced fuel solved the problems and long fuel cycles became more or less general. Nevertheless, long cycles may not be proper solution on units with uprated power and older design.

In the last years PAKS NPP, Hungary, operating 4 units of WWER–440 on 500 MWe uprated power turned from 12 to 15m cycles. Preliminary estimations showed that using the existing fuel geometry, 18–month cycles cannot be evaluated with satisfactory results. The core is small, and the necessary fuel enrichment for a realistic and economic 18–month cycle would exceed 5 wt %. But the analysis also demonstrated that there was an opportunity for a 15–month cycle design.

The next task was to decide how to schedule the operation of the four PAKS units in order to make use of 15–month cycles. One can assume that the refuelling outages of the PAKS units are arranged as follows (see Figure 14). The first refuelling is in February, the second in May, the third in August and the fourth in November. Supposing that reloads take approximately one month, evaluating cycles with a cycle length of 415–420 full power days, the following will happen: outages of a given unit will move forward in each year, from February to May, from May to August, etc., and one of the units will jump over to the next year.

At present the actual order of outages within the year at PAKS is as follows: unit 1, unit 4, unit 2, unit 3. If they are operated with a 15–month cycle length, overall the PAKS plant will be operated around a 5–year period, and during four of these five years we may have only three outages per year.

In practice, reloads do not all take the same time. There are three short outages, typically of 25 days, followed by a long outage of 55 days. Taking this into consideration, operator can arrange maintenance in such a way that long outages are planned for those years, when only three units undergo stoppages, and in the fifth year we may have only short outages. As a result, PAKS has a consistent production–maintenance rate in each year.

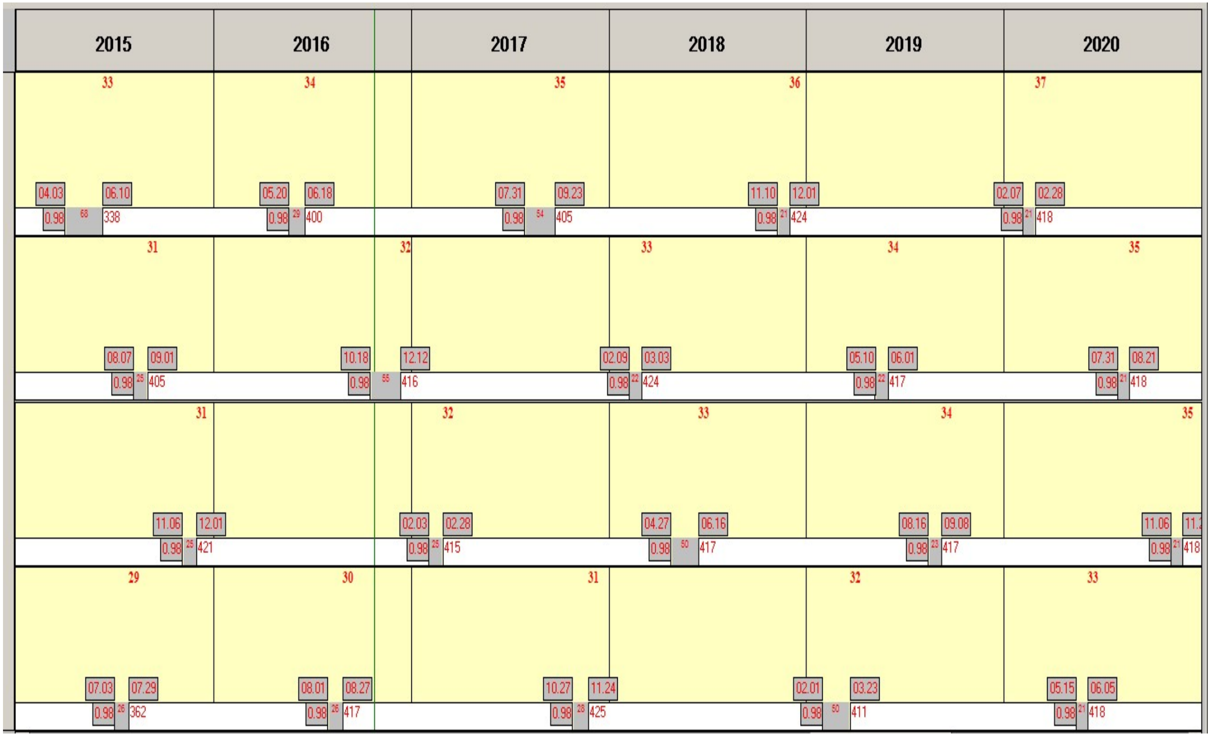

FIG. 14. PAKS NPP refuelling outages scheduled over five–year period (2015 is year no. 0).

8.3. METSAMOR, ARMENIA

The requirement to carry out an outage at the same time of the year, a sufficient number of transient loadings caused by the transfer of the core to a new fuel and the works associated with extending the service life of the unit are features of the reactor operation. The use of basic even and odd reactor overload schemes without considering the actual operation of previous loadings is not only economically disadvantageous, but also can lead to the release of the neutron–physical characteristics of the newly formed loading beyond the determined framework parameters and safety limits. Therefore, when carrying out prediction calculations to determine the nomenclature of fresh fuel loaded into the reactor, both the results of the operation of previous fuel loadings and the changes occurring in the planning durations of subsequent fuel cycles and outages are taken into account. The nomenclature and quantity of fresh fuel are determined from the need to ensure the duration of the newly formed loading and the unloading of irradiated fuel, whose neutron–physical characteristics may go beyond the safety criteria. As a result, this approach leads to the need to store a certain amount of fresh fuel at the station, it requires frequent adjustment of the fuel order and significantly increases the amount of calculations performed, however, it allows you to get economical fuel loadings, which meet all safety criteria.

As an example of the application of the above approach, the table below presents a comparison of the number of fresh fuel to be loaded during 10–years of the real operation of the WWER–440 reactor of the NPP Unit 2 versus the developed project during that period, as it's shown in Table 5.

TABLE 5. COMPARISON OF THE ACTUAL OPERATION OF THE WWER–440 REACTOR OF THE METSAMOR NPP UNIT 2 WITH THE DEVELOPED 10–YEAR BASE FUEL CYCLE

<table>
<tr><td>Loading №</td><td>21</td><td>22</td><td>23</td><td>24</td><td>25</td><td>26</td><td>27</td><td>28</td><td>29</td><td>30</td></tr>
<tr><td>Main features of the basic cycle</td><td colspan="8">Transient mixed loads for converting the core to enrichment fuel 3.82%</td><td colspan="2">Stationary overload mode</td></tr>
<tr><td>Quantity of fresh fuel loaded (base cycle)</td><td>78</td><td>78</td><td>78</td><td>72</td><td>78</td><td>72</td><td>78</td><td>72</td><td>78</td><td>72</td></tr>
<tr><td rowspan="2">The main features of the real cycle</td><td rowspan="2"></td><td rowspan="2">Work at reduced power. Holding of the outage in early autumn</td><td rowspan="2"></td><td rowspan="2">Holding of the outage in early autumn</td><td rowspan="2">Three months of work with one TG</td><td rowspan="2"></td><td rowspan="2"></td><td colspan="3">Start of work on extension of operation life</td></tr>
<tr><td>Transfer of the outage to spring</td><td></td><td></td></tr>
<tr><td>Quantity of fresh fuel loaded (real cycle)</td><td>78</td><td>60</td><td>78</td><td>66</td><td>66</td><td>66</td><td>78</td><td>66</td><td>54</td><td>54</td></tr>
</table>

8.4. KHNP, KOREA

In January 2019, KHNP is operating 23 units (20LWRs, 3PHWRs) and constructing 5 units and has two permanent shut down units. One of them is Advanced Power Reactor 1400 (APR1400) developed by a Korean national R&D project spanning ten years, which is already connected to the national power grid in 2016 and one unit may join within 2019. US NRC has reviewed all detail information of APR1400 for four years and will issue the design certification of APR1400 this year. In the design of APR1400, all nuclear knowledge developed, and lessons learned from 20 nuclear power plants operation for 30 years have been integrated so that APR1400 can meet more stringent safety standards including severe accident mitigation, improve public acceptance and economic competitiveness, and secure a means of stable energy supply and elevate nuclear power technology.

KHNP has designed a CANDU reactivity evaluation system in 2011 based on the LWR reactivity computer design technique and applied it to measure reactivity parameters such as control rod worth during initial core physics test. The CANDU reactivity computer system saved a total of 54 hours of critical pass time.

8.5. OSKARSHAMN 3, SWEDEN

Oskarshamn 3 has made a power uprate to 129% of rated power. This meant that the radial power distribution over the core became more or less rectangular instead of parabolic. This is caused by the problem of loading enough fresh fuel assemblies into the core to produce electricity at 129% during the entire 12–month cycle. Fresh fuel is loaded from the centre all the way out to the semi–periphery of the core. This has caused a re–work of all safety analysis and also presented totally new challenges to the reactor physicist both when it comes to core design and in–core fuel management.

The uprate had the consequence that the old reload patterns that has been derived by many years of experience had to be replaced with something entirely new. Extensive work was put in to finding new reload patterns and rules for finding an optimum core design. After performing several multi core analysis a new way of designing an optimum core was derived but we also found that some hardware changes had to be made in the NPP in order to facilitate a more optimal use of the fuel. After these hardware changes were made (for instance new indications for detecting turbine stop without the possibility of dumping steam into the condensation pool) new safety limits for dry-out in upset condition were implemented which in turn made it easier to utilize the fuel better and new even better reload patterns could be derived.

8.6. ČEZ, A.S., CZECH

ČEZ, a.s. is operated many thermal power plants and two NPP's — Dukovany NPP (4 units WWER–440) and Temelín NPP (2 units WWER–1000). It was decided to introduce new model of fuel cycle management (including nuclear fuel) after political changes in the Czech Republic and connected reorganization of anterior electric company in the nineties of 20th century.

The ČEZ's nuclear fuel management system (FMS) consists from three parts:

- Front–end part which provides nuclear fuel procurement (contracts with fuel suppliers) including fuel licensing, delivery organization and connected R&D support (cooperation with different Czech and international organizations);
- Middle part which controls the fuel utilization in reactor cores including all connected activities (technical support of front–end FMS part, loading patterns optimization including loading pattern safety evaluation, fuel assembly's manipulation plans, calculation of neutron–physical and thermalhydraulic parameters of the core including preparation of data for core monitoring and startup systems, startup experiments supervision);
- Back end part which provides the activities connected with the final discharged fuel storage.

The links among all FMS parts are defined clearly and precisely in company procedures.

Those activities are supported by cooperation with different Technical support organizations (TSO) on national or international levels (contracts with engineering organizations and research institutions, participation in different organizations connected with this topic).

An example of WWER–440 fuel assemblies and loading patterns improvements used in Dukovany NPP.

There were used many modernizations of fuel assembly construction during operation history from the 1st to the 2nd generation:

- Stainless steel spacer grids were replaced by Zr ones (1st);
- Shroud thickness was decreased;
- Radially profiled pin enrichment;
- Burnable absorber (Gd) introduction;
- Fuel pin pitch increasing (2nd);
- Fuel column length increasing;
- Fuel pellet central hole removing;
- Fuel pellet outer diameter increasing.

There is offered 3rd generation fuel assembly with so–called 'karkas' shroud (framed structure) instead of hexagonal shroud tube. It allows next pin pitch increasing.

There was used so–called out–in fuel strategy at the beginning of operation. Using more precise neutron–physical and core optimization codes it was possible to introduce so–called in–out loading pattern strategy (so–called Low Leakage Loading Pattern — L^3P) which allowed increased fuel utilisation.

8.6.1. Core loading pattern optimization process

This process is tightly connected with production planning process in the ČEZ. It means that the fuel cycle strategy is developed in connection with production and outages planning processes. Such kind–maintained fuel cycle strategy is able to reflect all company needs for electricity production.

The basic decision is how long fuel cycle has to be. Since the beginning of Dukovany NPP operation the annual fuel cycle has been used. The necessary FA amount and fuel enrichment were derived from the loading pattern type (out–in or in–out), average cycle length and allowed discharge FA burnup. The optimization of maintenance work during outages (outage length minimisation) led to prolongation of annual operation (cycle length), e.g., average outage length 26 days (3x20, 32, 3x20, 56) led to average cycle length 365-26=339 calendar days. The rule that different NPP units' outages in our company has not overlapping led to deviations in individual cycle lengths. Those deviations were solved using planning of necessary coast down operation for the case of small difference from average cycle length or by decreasing/increasing of fresh fuel amount (we use FA with the same enrichment and Gd content usually). In our case one fuel assembly's quantum represents 6 fuel assemblies, which is derived from 60° core symmetry. Those 6 fuel assemblies represent approximately 25 FPD. So, it could be stated that the length deviations from average one less than 20 days is solved using coast down operation. This operation mode serves us as a tool for solution of unexpected operation perturbations (like unplanned equipment breakdown, outage, load–follow operation, etc.). From other side the turbine producer recommended the coast down length less than 20 days which support our practice. It is clear that the situation has to be solved for each cycle individually.

Example of introduction of 5-year (quintuple annual) fuel cycle for WWER–440:

- Feasibility study (introduction of new fuel type, e.g., Gd–2M+) – core parameters definition (thermal power, reactor inlet temperature, average cycle length, etc.) led to new fuel design (radial fuel pin enrichment profiling) which was necessary for utilisation in quintuple annual fuel cycle at parameters specified above. WWER–440 core consists from 312 working fuel assemblies and 37 control fuel assemblies (including movable fuel parts). It means that one reload batch consists from 60 WFA+12 CFA or 66 WFA+6 CFA. This new fuel was licensed and introduced to the NPP. First loading pattern was developed in cooperation with fuel supplier according the Contract condition. Next loading patterns are optimized by the NPP's specialists;
- Whole licensing process including safety analyses and all NPP's procedures changes were done in cooperation with fuel supplier and other TSO's, which help to obtain regulatory authority license and introduce all conditions from licensing process into the NPP operation.

APPENDIX I.
RELATIONS BETWEEN ENRICHMENT, LENGTH OF THE CYCLE AND THE PRICE OF ELECTRICITY

I.1. FUNCTIONAL DEPENDENCY BETWEEN THE ENRICHMENT AND THE FUEL COMPONENT OF THE PRICE OF ELECTRICITY

For the PWR reactors fuel in narrow range of enrichment is in use, usually between 3% and 5%. In this range of enrichment price of the fuel may be described using linear equation:

$$c = ax + b \tag{28}$$

Where:

- x is enrichment, c is price of one fuel assembly having enrichment x;
- a and b are the coefficients depending on the reactor type, the type of the fuel assembly and the supplier:

- a > 0, because the higher enrichment is the higher price of the fuel assembly;
- b > 0, because fuel assembly without fuel has positive price.

Suppose in the given equilibrium fuel cycle all fuel assemblies have equal enrichment x.

Then cost S of one fresh fuel load:

$$S = F(ax + b) \tag{29}$$

Where:

- F is number of fresh fuel assemblies and having enrichment x.

Length of the equilibrium fuel cycle may be described using linear equation:

$$T = AFx + D \tag{30}$$

Where:

- A, D are coefficients, and D > 0.

For the practical purposes for any given reactor type coefficients A and D may be estimated empirically through comparing several equilibrium fuel cycles having different F and x.

Comparing costs of loads for two fuel cycles having equal length, but different enrichment:

$$T_1 = AF_1x_1 + D \tag{31}$$

$$T_2 = AF_2x_2 + D \quad (32)$$

By condition $T_1 = T_2$

$$AF_1x_1 + D = AF_2x_2 + D \quad (33)$$

$$F_1x_1 = F_2x_2 \rightarrow F_2 = F_1\frac{x_1}{x_2} \quad (34)$$

Suppose $x_2 > x_1$, then costs of two loads are:

$$S_1 = F_1c_1 = F_1(ax_1 + b) \quad (35)$$

$$S_2 = F_2c_2 = F_2(ax_2 + b) = \quad (36)$$

$$= F_1\left(\frac{x_1}{x_2}\right)(ax_2 + b) = F_1\left(ax_1 + b\frac{x_1}{x_2}\right) \quad (37)$$

Comparing equations for S_1 and S_2 are:

$$S_1 = F_1(ax_1 + b) = F_1ax_1 + F_1b \quad (38)$$

$$S_2 = F_1\left(ax_1 + b\frac{x_1}{x_2}\right) = F_1ax_1 + F_1b\left(\frac{x_1}{x_2}\right) \quad (39)$$

Since $x_2 > x_1$, then $x_1 / x_2 < 1$.

If $b > 0$, then $S_2 < S_1$, or the cost of the fresh fuel load is less for fuel cycle having higher enrichment.

Suppose that the fuel component of the price of electricity C_f is:

$$C_f = \frac{S}{W} \quad (40)$$

Where S is the cost of the fresh fuel load, W is the total electricity production in one fuel cycle.

Electricity production is proportional to effective length of the fuel cycle T_{ef}:

$$W = fT_{ef}, kWhr \quad (41)$$

Where:

— f = Ne*24, and Ne is electrical power, kW.

By condition T1 = T2, then electricity production is equal for both fuel cycles. But fuel component of the price of electricity for the second fuel cycle is less because S2 < S1.

Conclusion: for fuel cycles having equal length increasing enrichment leads to reducing fuel component of the price of electricity.

In other words, to reduce S it is necessary to increase enrichment and to decrease number of fresh FA in the load. It is necessary to consider that one of the consequences of that is increasing of the average burnup of unloaded fuel.

I.2. FUNCTIONAL DEPENDENCY BETWEEN THE LENGTH OF THE FUEL CYCLE AND THE FUEL COMPONENT OF THE PRICE OF ELECTRICITY

Now consider two equilibrium fuel cycles, and suppose effective length of the second fuel cycle longer then length of the first one:

$$T_2 > T_1 \quad (42)$$

As shown earlier it is profitable to use higher fuel enrichment. Suppose that in both fuel cycles there is fuel having maximum possible enrichment x.

Length and load's cost for the first fuel cycle are:

$$T_1 = AF_1x + D \quad (43)$$

$$S_1 = (ax + b)F_1 \quad (44)$$

Length and load's cost for the second fuel cycle are:

$$T_2 = AF_2x + D \quad (45)$$

$$S_2 = (ax + b)F_2 \quad (46)$$

Electricity production is proportional to effective length of the fuel cycle T_{ef}:

$$W = T_{ef}, kWhr \tag{47}$$

Where:

- f=Ne*24, and Ne is electrical power, kW.

Fuel component of the price of electricity C_f are:

$$C_{f1} = \frac{S_1}{W_1} = \frac{S_1}{fT_1} = F_1 \frac{(ax+b)}{f}(AF_1x+D) \tag{48}$$

$$C_{f2} = \frac{S_2}{W_2} = \frac{S_2}{fT_2} = F_2 \frac{(ax+b)}{f}(AF_2x+D) \tag{49}$$

$$\frac{C_{f2}}{C_{f1}} = \left[\frac{F_2(ax+b)}{(AF_2x+D)}\right] / \left[\frac{F_1(ax+b)}{AF_1x+D}\right] = (F_1F_2Ax + DF_2)/(F_1F_2Ax + DF_1) \tag{50}$$

First components in the numerator are equal, second component in the numerator is bigger, because D > 0 and $F_2 > F_1$ (length of the second fuel cycle is bigger, that means that having equal enrichment second fuel cycle has more fresh fuel assemblies), therefore CT2 / CT1 > 1, or CT2 > CT1.

Conclusion: The longer the fuel cycles the higher the fuel component of the price of electricity, if enrichments of the fresh fuel are equal.

I.3. FUNCTIONAL DEPENDENCY BETWEEN THE LENGTH OF THE FUEL CYCLE AND THE PRIMARY COST OF ELECTRICITY

Suppose that:

- E_a = annual expenses without the fresh fuel, 1/year;
- T_1 and T_2 = calendar length for two equilibrium fuel cycles, and suppose $T_1 < T_2$, and $T_1 = T_{ef1}$, $T_2 = T_{ef2}$;
- E_m = spending for the maintenance, 1 / fuel cycle;
- S_1 and S_2 = costs of the fresh fuel loads.

Full spending for the one fuel cycle are:

$$E_1 = \frac{E_aT_1}{365} + E_m + S_1 \tag{51}$$

$$E_2 = \frac{E_a T_2}{365} + E_m + S_2 \tag{52}$$

Average annual spending's are:

$$E_{1y} = \frac{E_1 * 365}{T_1} \tag{53}$$

$$E_{2y} = \frac{E_2 * 365}{T_2} \tag{54}$$

$$E_{1Y} = \left(\frac{E_a T_1}{365} + E_m + S_1\right) * \frac{365}{T_1} = E_a + (E_m + S_1) * \frac{365}{T_1} \tag{55}$$

$$E_{2Y} = \left(\frac{E_a T_2}{365} + E_m + S_2\right) * \frac{365}{T_2} = E_a + (E_m + S_2) * \frac{365}{T_2} \tag{56}$$

Usually maintenance spending is much less than spending on fresh fuel.
In that case:

$$E_{1Y} \approx \frac{E_a + S_1 * 365}{T_1} \tag{57}$$

$$E_{2Y} \approx \frac{E_a + S_2 * 365}{T_2} \tag{58}$$

$$E_{2Y} - E_{1Y} = \frac{S_2 * 365}{T_2} - \frac{S_1 * 365}{T_1} = \frac{365\,(T_1 S_2 - T_2 s_1)}{T_1 T_2} \tag{59}$$

$$T_1 = AF_1 x + B \tag{60}$$

$$T_2 = AF_2 x + B \tag{61}$$

$$F_2 > F_1 \tag{62}$$

$$S_1 = SF_1 \tag{63}$$

$$S_2 = SF_2 \tag{64}$$

After substitution and transformations:

$$E_{2Y} - E_{1Y} = \frac{BS * (F_1 - F_2)}{\big((AxF_1 + B)(AxF_2 + B)\big)} < 0 \tag{65}$$

Both multipliers in the denominator are positive, but the difference F_1–F_2 is negative because $F_1 < F_2$ because it was assumed that $T_1 < T_2$.

Hence, the difference $E_{2Y} - E_{1Y}$ is negative, and $E_{2Y} < E_{1Y}$.

Average annual electricity production is proportional to effective length of the fuel cycle T_{ef}:

$$W_Y = \frac{fT_{ef} * 365}{\left(T_{ef} + T_m\right)}, kWhr \tag{66}$$

Where T_m is average duration of the maintenance, days.

If $T = T_{ef}$ then:

$$W_y = \frac{fT * 365}{(T + T_m)}, kWhr \tag{67}$$

For two fuel cycles under consideration:

$$W_{Y1} = \frac{fT_1 * 365}{(T_1 + T_m)} \tag{68}$$

$$W_{Y2} = \frac{fT_2 * 365}{(T_2 + T_m)} \tag{69}$$

$W_{Y1} < W_{Y2}$ because it was assumed that $T_1 < T_2$.

So, we have $E_{1Y} > E_{2Y}$ and $W_{Y1} < W_{Y2}$. Hence:

$$\frac{E_{1Y}}{W_{Y1}} > \frac{E_{2Y}}{W_{Y2}} \qquad (70)$$

I.4. CONCLUSION

For fuel cycles having equal enrichment of the fresh fuel the longer length of the fuel cycles the less the average annual spending is, and therefore there is lower primary cost of electricity.

Due to the electricity production being higher for the longer cycle ($W_{Y1} < W_{Y2}$), average annual profit is higher for the longer fuel cycle.

In this case the fuel cycle is having lower primary cost and higher average annual profit. It is not necessarily true for the fuel cycles having different fuel enrichment.

APPENDIX II.
METHODS TO REDUCE POSSIBLE VARIATIONS OF CORE ARRANGEMENT

Let us consider a simplified task presuming that the reactor has the N number of positions for FA and we want to arrange in an optimal way the N number of fresh and irradiated FA. Let number of fresh fuel assemblies is F. If rotation of the FA is possible, in this case the number of unambiguous arrangements is $N!p^{(N-F)}$, where p is the number of possible rotations of the assemblies and bundles. Due to fresh FA has a symmetry, it does not have sense to rotate it.

Obviously, it is infeasible to examine in full all the potential states; so, to save computational time the number of states to be examined should be limited, based on some concepts.

To reduce the number of permutations the following tricks may be used:

- Relocation instead of rotation;
- Excluding central FA from permutation;
- Using symmetry;
- Specify sets of cells for specific groups of FA;
- Using good pattern for placing fresh and irradiated FA;
- Avoiding close placement of FA having high multiplication factor;
- Avoiding close placement of FA having low multiplication factor (except peripheral cells);
- Other.

Using all the principles mentioned above sometimes leads to the situation with the lack of possible permutations.

II.1. USING RELOCATION INSTEAD OF ROTATION

In some reactors rotations of FA may be forbidden or a refuelling machine cannot rotate the FA. Without rotation number of unambiguous arrangements is N!.

In fact, it happens quite seldom that the need to rotate a single FA appeared.

It may happen, for example, if damaged FA was unloaded from the core before the scheduler and has to be replaced with irradiated FA from the spent fuel storage. In that case it may be reasonable to try to rotate FA used for replacing, or, what is the same in terms of number of possible, to use symmetrical irradiated FA from the spent fuel storage.

In symmetrical core the symmetrical FA are placed in orbits of symmetry. Permutation within the orbit of symmetry to another part of the core may be more convenient then rotation. For example, transposition to the opposite part of the core is equal to 180–degree rotation. In the case of the orbit of symmetry having 12 fuel assemblies for hexahedral (or 8 for square) fuel assemblies, it may give additional opportunity to use ‘mirror–like’ FA.

Using rotation may be useful during the final stage of optimization to improve micro fields, to reduce neutron leakage or to reduce neutron flux on the reactor vessel.

But even without rotation number of possible permutations is huge.

For example, for WWER–1000:

$$163! = 2*10^{291} \tag{71}$$

Real number of permutations is much less, because it has no sense to make permutations among FA the same type. Yet another central FA may be excluded from permutations.

If the core has a cell for FA placed in the very centre, usually it does not make sense to move FA from the central place somewhere else, except certain situations with the asymmetric core (due to the damaged FA, for example).

Usually the number of options for placing in the central cell is very limited if exists at all. If central FA remains in the core for the next fuel cycle, there are no options at all.

II.2. USING SYMMETRY

If the core has reflectional symmetry and (or) rotational symmetry about a central point, it is quite enough to optimize just a part of the core. That reduces the number of possible permutations, and yet another reduces computational time for each permutation.

Cores with square FA may have 1, 2, 4 lines of reflection symmetry.

Cores with hexahedral FA may have 1, 2, 3, 6 or 12 lines of reflection symmetry. 1/12, 1/6, 1/3 or 1/2 of the core may be objects of reflection symmetry, and 1/6, 1/3 or 1/2 of the core may be objects of rotational symmetry with respect to a central point.

For example, the core of the WWER–1000 has only 27 cells (without a central one) in the 1/6 sector. Number of possible permutations is $27! = 1{,}1*10^{28}$ and this number is still too big.

II.3. SPECIFYING THE SET OF CELLS FOR FRESH FA

If the number of fresh fuel assemblies is F, and all fresh fuel assemblies are the same type, the number of possible sets for fresh fuel is C*(N-1,F).

Within each such set any permutations do not change property of the core, because all fresh FA are equal. Only permutations of the irradiated FA may change the properties of the core. Irradiated FA are placed in the remaining cells of the core.

Total number of permutations is:

$$C(N-1,F) * (N-F-1)! \tag{72}$$

For example, the core of the WWER–1000, if F = 66, in 1/6 of the core there are 11 fresh fuel assemblies.

The number of possible permutations is:

$$(28-11-1)! * C(28-1, 11) = 16! * C(27, 11) = 2.1*10^{13}*1.3*10^{7} = 2.7*10^{20} \tag{73}$$

Using 1/12 of the core allows to reduce that number more.

Specifying set of cells for irradiated FA (for example, in the case of full in–out FA with the highest burnup are placed in the peripheral area) leads to further reducing number of possible permutations.

II.4. SPECIFYING SETS OF CELLS FOR SPECIFIC GROUPS OF FUEL ASSEMBLIES

The core design seldom begins from the scratch. In the simplest case, core designer has a core which has to be a bit changed due to certain circumstances: initial burnup of irradiated fuel was changed because of changing length of the previous campaign, or he needs to replace damaged FA. In the cases like that it may be enough to check just several obvious rearrangements to find good enough solution, not necessarily the best one.

In more complicit case a designer may have a task to develop, let us say, a new equilibrium fuel cycle. Main steps of this process were described in the Section 5.4.2. But now let us discuss what technics may be used to reduce number of permutations to the reasonable level.

A designer knows main features of the new cycle, like number and enrichment of the fresh FA to reach a desirable average length of one campaign. Common sense and previous experience suggest trying to use in–out type of the core to reach maximal length.

Let us presume for more certainty that his core has 162 hexahedral fuel assemblies without a central one, and he needs to load 66 fresh fuel assemblies for every campaign. That means that composition of the core will be : 66 fresh fuel assemblies, 66 irradiated fuel assemblies after one campaign and 30 irradiated fuel assemblies after two campaigns, 66+66+30 = 162.

To organize in–out core all irradiated FA after two campaign has to be placed in peripheral cells. To reach maximal length this FA has to be placed as far from the centre of the core as possible and FA having higher burnup has to be placed farther from the centre of the core. In the 60–grad symmetrical core five FA are placed in each sector of symmetry. To keep symmetry one of this five has to be placed on the axe of symmetry. If among other four FA two couples of symmetrical FA exist, there is no choice, every FA has its own cell defined naturally.

If after that a pattern for the fresh fuel assemblies is established, that means that there are just 11 cells for the irradiated fuel assemblies after one campaign in the sector 60 degrees. If among them 5 fuel assemblies are placed on the axe of the symmetry, and other 6 consists of 3 couples of symmetrical fuel assemblies, number of all possible permutations is 5!*3! = 720. This number let to realize even full check–up if needed.

Using simple reasoning based on analysis of the appearing cores it is inevitable, that big part of this possible permutations does not deserve to be checked. At this point, if desirable solution still remains missing, core designer may start suffering from the lack of possible permutation. In that case he needs to make one step back and, for example, to change the pattern for the fresh fuel.

APPENDIX III.
MATHEMATICAL METHODS

III.1. MATHEMATICAL METHODS USED FOR THE RELOAD PATTERN OPTIMIZATION

III.1.1. Linear programming

In case of applying the mathematical methods (linear programming), the finding of the optimal value can be traced back to the classical, so called packing problem. Let's consider the following example: Place the N number of rods with different lengths into the cells with different depths in a manner, providing that the sum of protruding rod lengths will be minimal. The established constraint is that none of the rods should be shorter than the cell length and the maximal length of the protruding rod should not exceed X cm.

Mathematically speaking: X(i,j) marks the matrix (N*N), which gives the position of the (numbered) rods. X(i,j)=1, if the i–rod is in the j–position. Since one rod can be placed only into one place and vice versa (one position is occupied by one rod only), then only one piece of 1 can be found in each line and each column of the matrix, all other elements will be zero:

$$\sum_{i=1}^{N} X\,(i,j\,) = 1 \quad ; \; j = 1,2, \dots N \tag{74}$$

$$\sum_{j=1}^{N} X\,(i,j\,) = 1 \quad ; \; i = 1,2, \dots N \tag{75}$$

If C(i,j) marks the length difference between the rod and the cell, then the objective function value is:

$$c = \sum_{i,j=1}^{N} C(i,j)\, X(i,j) \tag{76}$$

Then, using the mathematical linear programming one can find the X(i,j) matrix, in case of which the objective function value will be minimal. Constraints formulated in the above example can be taken into consideration in a manner providing that in corresponding combinations the C(i,j) matrix elements give extremely high values, then the minimization procedure will not demonstrate the optimal value for the rod–cell combinations.

Whenever we wish to optimize the fuel assembly arrangement in the reactor by the above example, then at first sight the task seems to be rather similar. However, in reality it turns out to be more complicated: it is not possible to calculate or interpret the value of any of the objective functions for the case when the i–fuel is in the j–position. The result depends on the arrangement of the other fuel: the problem is not of a linear nature.

In order to be able to apply the linearization, we have to establish some conditions. First of all, as a starting point we should select some feasible arrangement and then by means of linear methods determine the effectiveness of modifications as compared to the initial one. Various attempts are described in the professional literature; however, some of their results can be doubtful.

The following method seems to be the most suitable for the case:

- By means of a perturbation model, we estimate the influence on the objective function if the reactor physics characteristics (k∞, burnup) of the j–bundle located in the i–position slightly change;
- Based on the above, we can also estimate the potential influence of the situation when k–bundle is placed into the i–position instead of the j–bundle. In this way, we can fill out the C (i,j) matrix;
- To preserve the validity of the linear approach, we can limit the movements: only such movements are allowed, which are accompanied by an insignificant change of reactor physics characteristics. A high C (i,j) value is assigned to the forbidden movements and to the states violating the constraints;
- By using the above steps in an iterative manner, we can minimize (or maximize) the objective function value.

Whenever the required computer capacities are available, the influence of the k→i movement can be calculated immediately, without the perturbation estimation, while the bundles being in other positions remain in their places. It is not possible however to exclude the influence of non–linearity on the result since the final result will change in case of the k→i movement, if other bundles remain in their places.

Should the conditions described above be fulfilled, we can hope that our method will be applicable in practice, as well.

Problems related to non–linearity and huge calculation demand, force us to continuously look for compromises in case of applying the other mathematical methods as well. We can try to mitigate the problems mentioned above using a simpler and faster procedure for the determination of criticality and neutron flux distribution. However, the accuracy of the above method is modest. Even though being very sophisticated, it cannot be used in practice, if after re–calculating our optimized arrangement by means of a more precise model we come to the conclusion that the constraints are not fulfilled. Application of burnable absorbers technology make the problem more non–linear reducing usefulness of linearization methods.

III.1.2. Method of the most efficient moves

The method implies the application of the so–called simplex procedure, used basically for the search of extreme values of multi–variate functions, for finding the optimal load arrangement [24].

Search of the maximum value of the multi–variate function with the use of the simplex method starts from a given point with the search of a maximum gradient (positive or negative), and then continues by making moves in its direction until the change of the function value corresponding to the direction of the objective function has been found. The length of the move made in the direction of maximum gradient can be varied and has a significant impact on the application efficiency. Usually and in this specific case as well, the function process is not analytically known, the search of optimum is done with the use of a computer algorithm. The appropriateness of the move made in the direction of the ‘most promising’ gradient depends on the linearity of the function in the given range.

During the application for the given purpose, the following aspects should be taken into consideration:

- The function has a lot of variables, even in the initial approximation the number of variables will be identical to the number of fuel assembly places (symmetry can reduce it);
- The set of function values is not continuous, meaning that not any but only FA with the specified enrichment, burnup and efficiency can be loaded into the certain positions, based on those available from the previous cycle and the fresh fuel batch.

Taking into consideration the above–said, the application of simplex methods for optimizing the fuel arrangement can be implemented as follows:

- Let us consider an initial fuel arrangement;
- Let us 'map' the influence of all reasonably possible pair replacements (prediction), as compared to the initial arrangement;
- Let us apply the suggestion of the prediction move (or the pair replacements), which contribute to the most efficient modification of the objective function value;
- Repeat steps 1–3 until the objective function reaches the desired value, along with the fulfilment of all constraints.

The prediction described in point 2 can be carried out similarly to the procedure described in connection with the linear programming in some linear approximation, for example:

- Efficiency of FA from the reload bath can be changed one by one by means of a perturbation model or direct method, as compared to the initial state, and the objective function should be examined in the linear approximation, viz. the following derivatives (d_i) are generated:

$$d_i = {}^{dC}/_{dk_i} \tag{77}$$

Where:

- C is the objective function value, k_i is the i – assembly efficiency (k_∞).

With the help of derivatives, we can estimate by linear assumption the influence of the i↔j assembly exchanges on the objective function value:

$$C_{ij} = (d_i - d_j) * (k_j - k_i) \tag{78}$$

After that we select the biggest (or the smallest) element of the C matrix, which will give the prediction 'corresponding to the maximum derivative', then the search is continued with the above iterative steps 3 and 4.

The method application can be interpreted similar to marking a random point on a 3–dimension relief map and then walking down to the valley using the steepest slope. The method is rather efficient for finding the deepest point of the valley; however, it cannot be guaranteed that the neighbouring valleys are not deeper. Starting procedure from other initial states may lead to finding another optimal value.

However, this method works in practice and can be efficient when supplemented with ad hoc elements. A modernized version of this method had been used for decades for optimizing the fuel arrangement in WWER–440 reactors at PAKS Nuclear Power Plant.

III.1.3. Annealing algorithms

The previously discussed, so-called deterministic algorithms mentioned earlier can easily stick into the local optimal places. Stochastic methods try to help solving this problem. These stochastic algorithms can be further divided into two main groups: genetic algorithms and simulated annealing procedures. The stochastic algorithms are efficient optimization procedures which may 'escape' from local optimums with a higher probability than the deterministic ones.

Two annealing algorithms: SA and PMA, use so called annealing analogy known from crystallography. If a high–temperature crystal is rapidly cooled down, there will be no sufficient time for the crystal particles to find the minimum of their potential energy and the crystal will 'stick' in the so–called local optimum. However, if the crystal temperature is decreasing slowly during a longer period of time, then some particles will be capable of finding the state characterized by the minimum potential energy and, correspondingly, the potential energy of this crystal will reach the global energy minimum [25–28].

III.1.4. The simulated annealing algorithm (SA) works as follows

We start with a randomly generated load and estimate the load objective function. After that two or three fuel assemblies selected based on the uniform distribution principle will be randomly exchanged and the objective function value will be calculated again. If the value has 'improved' along with the fulfilment of various constraints and limiting conditions, e.g., the maximum bundle power (PPF), then the new load is accepted as the reference, the calculation is continued, and the previously used reference load is 'forgotten'. Should the objective function value change in an unfavourable direction, the probability of accepting the new load as the reference one will look as follows:

$$a_i < exp[(z_i – z_{i-1})]/T_i \tag{79}$$

Where:

— a_i is the random number assigned in the [0,1] interval based on the uniform distribution principle, z_i and z_i–1 are the objective function values in the i– and (i–1) step of the algorithm, and T_i is the system temperature.

When applying the algorithm, the temperature value is reduced at every step:

$$T_i = \alpha * T_{i-1} \tag{80}$$

Where:

— α is the cooling parameter (0,995 in our case) and T_0 is equal to 1.

One can see from the equation above that there is rather a high probability in the algorithm beginning that a load having a worse objective function value than the reference load will be the reference load starting from the next step. At the same time, this probability significantly decreases at the end of algorithm, due to the high negative value in the exponent. When determining the maximum PP limit value during the application of the simulated annealing algorithm, we use the so–called limit annealing technology meaning that the PP_{max} limit value is continuously reduced based on the equation. Thus, the loads — which do not fulfil the 'strict' PPF limit values — have been accepted as the reference ones at the algorithm beginning, provided that the other conditions are fulfilled, and have been rejected at the algorithm end:

$$PPF \leq PPF_{lim} + \alpha * \beta * T_i \tag{81}$$

Where:

— PPF is the actual limit, PPF_{lim} is the 'strict' constraint, T_i is the temperature at the i–step, α is the cooling parameter and β is the deliberately selected value.

The population mutation annealing algorithm (PMA) is similar to the simulated annealing algorithm. Similar to the SA, we start with a randomly generated load and estimate the objective function. After that two or three fuel assemblies are exchanged, the decision is made based on the function (3 assemblies in the beginning and 2 later) and then the objective function is estimated again. Should both loads satisfy the actual PPF limit, again with the use of the limit annealing technology, both of them will belong to the population. The next step will consist in selecting the reference load from the two above loads based on the following equations:

$$P_m = \frac{\exp\left[\frac{z_m - z_{max}}{T_i} + \zeta_m\left(1 - \frac{T_0}{T_i}\right)\right]}{\sum_{j=1}^{pop}\left[\frac{z_j - z_{max}}{T_i} + \zeta_j\left(1 - \frac{T_0}{T_i}\right)\right]} \tag{82}$$

$$P_m = \frac{\exp\left[\frac{z_m - z_{\max}}{T_i} + \zeta_m\left(1 - \frac{T_0}{T_i}\right)\right]}{\sum_{j=1}^{pop}\exp\left[\frac{z_j - z_{\max}}{T_i} + \zeta_j\left(1 - \frac{T_0}{T_i}\right)\right]} \tag{83}$$

Where:

— z_m is the objective function value of the m–load in the population, z_{max} is the maximum objective function value in the population, T_i is the system temperature at the i–step, T_0 is the system temperature at the 0–step (equal to 1), ζ_m is the so–called penalty parameter (if the load does not fulfil the strict PPF limit, then it is equal to 1, otherwise 0). P_m is the probability of the m–load to be selected as the next reference load.

As long as we proceed with the algorithm, all the loads having the maximum PP value lower than the limit will belong to the population. If one member of the population does not fulfil the continuously reducing limit, then it will exclude from the population at the next step. As one can see, at the end of algorithm only such loads will belong to the population, which fulfil the 'strict' constraint.

The two described algorithms: the SA and the PMA, are very similar, however, the most significant difference consists in the fact that while the SA remembers only the last reference load, the PMA remembers all the loads belonging to the population.

III.1.5. Genetic algorithms

The PMA method discussed above brings us to the so–called genetic algorithms, where the SA can be considered as a special case thereof [29].

The method of genetic algorithms (genetic algorithm, GA) can be applied to a very wide range of problems occurring in computing; at the same time, it can be interpreted as the area–dependent knowledge, so it works even in the case if the task structure is hardly known. From this point of view, it belongs to the problem–independent group of metaheuristics, same as the above discussed simulated annealing (SA) and various hill climbers. This method is a kind of a global optimization which can be used in every and each such a case where the task should find the best decision among many possibilities, where the value is given by the estimating function, known as the fitness function. There exists the genetic algorithm solutions population, i.e., several solutions work simultaneously. At every step, a new population is generated from the actual population in a manner providing that various operators (e.g., mutation) are applied for the fit elements (parents) chosen by the selection operators.

The basic idea is that as a rule, every population contains elements which are fitted and compared to the previous one, there are several better solutions available in the course of search. In principle, every learning problem can be solved as an optimization task, so the genetic algorithms are widely used in the area of machine learning.

In case of the load arrangement optimization, the population members are the selected arrangement of the load and the fitness is the optimization objective function.

During the optimum search, there are hundreds of population elements, which can be stable or varying during the operations. Elements of the next population are received from the given population by means of mutation or crossover. In case of the fuel arrangement optimization, the individual is defined by the ordinal numbered FA assigned to the ordinal numbered positions. The mutation of the given individual can be achieved by the exchange of the positions of two or more bundles. During the crossover, one should take into consideration that only one bundle can be placed into one position, the crossover of any parents is not possible.

To obtain the next generation of the given population, the 'parent' individuals are chosen by means of various selection methods.

In case of the fitness–proportionate selection, the selection probability for each element of the population can be expressed by the following equation:

$$P(e) = \frac{f(e)}{n} * f(pop) \tag{84}$$

Where:

— f(e) is the fitness value, n is the element number in the population, f(pop) is the average fitness of the population members.

In case of the binary tournament selection, two (or more) individuals are randomly selected from the population and the fittest one will be selected as the parent. The disadvantage of this method consists in its excessive sensitivity to the fitness distribution in the given population. If, for instance, any individual is characterized by a considerably higher fitness value, then it can be too dominant, and the decision will 'stick' in its neighbourhood.

As follows from the above said, the described simulated annealing method is a special case of the genetic algorithms: the population consists of one element and the search operator is the mutation. However, the selection criteria differ from the above: if the new solution is worse than the old one, we also accept the value controlled by the temperature parameter, with some certain probability.

III.1.6. Other stochastic methods

During the tabu search we maintain a tabu list consisting of several previously examined solutions. The size of the tabu list is the algorithm parameter. Prior to creating a new generation, i.e., when selecting the new actual solution should firstly check whether the new element created by mutation is mentioned in the tabu list. If yes, then it is not accepted; if no, then it is accepted providing that it is not worse than the previous solution. The old solution is recorded to the tabu list and the oldest element of the tabu list is deleted.

Another special method is known as stochastic hill climbers. Here the population is also represented by one element and the search operator is the mutation. The new solution replaces the old one providing that it is as fit as the old one or even fitter. In its essence, the stochastic hill climber's method is the simulated annealing taking place at the fixed zero temperature or the tabu search using the zero–length tabu list.

At the same time, the stochastic hill climber's method is similar to the deterministic simplex procedure, the only difference is that the perturbation compared to the initial state is generated on a random basis.

III.1.7. Expert system, neural network

All those previously demonstrated methods used for the core arrangement optimization consider the optimized area as unknown and are combined with the other methods known from different mathematical applications. However, the efficient application of these methods requires the task–specific approximations and the finding of a solution is accelerated by the application of a favourable initial state or special constraints. It can be, for instance, the limitation of the possible positions of the fresh fuel assemblies [30, 31].

Our knowledge related to the appropriate initial state or positions of the fresh FA is accumulated from our previous experiences. The so–called expert system is methodically based specifically on the utilization of experiences gained and can be used for other applications as well.

The expert system is a framework program basing on the 'if, then' logic, which should be completed with the knowledge base. During the search of an optimal or almost optimal core arrangement the knowledge base consists of the below rules.

We specify or can specify:

- Potential positions of the fresh FA;
- That some certain neighbouring positions cannot be occupied by the fresh fuel;
- Positions of the FA having the highest burnup;

— That some certain neighbouring positions cannot be occupied by the fuel with high burnup.

Rules can be established in respect of the differently aged FA located next to each other, their arrangement as to each other, etc.

Fuel assemblies with very similar burnup levels can be practically exchanged, so such FA classified as to their ages or burnup levels have been grouped into arrangement families with similar properties. Through the formulation of detailed rules, the number of various solutions or solution groups can be reduced to a few hundred, and the solution close to optimum can be selected by the performance of respective calculations thereof. One should not forget that even those previously described methods, which seem to be mathematically more exact, are not capable of finding the absolute optimum!

Another heuristic method is the application of the neural network. The neural network operation (see Figure 15) includes two main phases: learning and recalling. The 'knowledge' of the neural network is stored in the weights between the neurons. Thus, the learning is the change of these weights. The method being the basis for this process is called the 'learning rule'.

The models applied for the reactor description use the so–called supervision–coupled learning. For the purpose of supervision coupled learning, input values and samples are 'shown' to the network and the sample is accompanied by attaching a desirable correct result. The network calculates its own result and the determination of new weights are based on the difference between the two results.

The most frequently used procedure is the back–propagation model, the name of which refers to the process providing that the modification of weights is done by sending back to the input the signal supplemented by the correction calculated based on an error appearing at the output.

The neural network applied for the estimation of the core arrangement characteristics is shown in Figure 15. As one can see in the Figure 15, the i_1, i_2,… is the activation of the neurons of the first layer, h_1 ,h_2,… of the second layer, w_{ij} are the weights. z_1, z… is the weight of the second layer neurons to the output value.

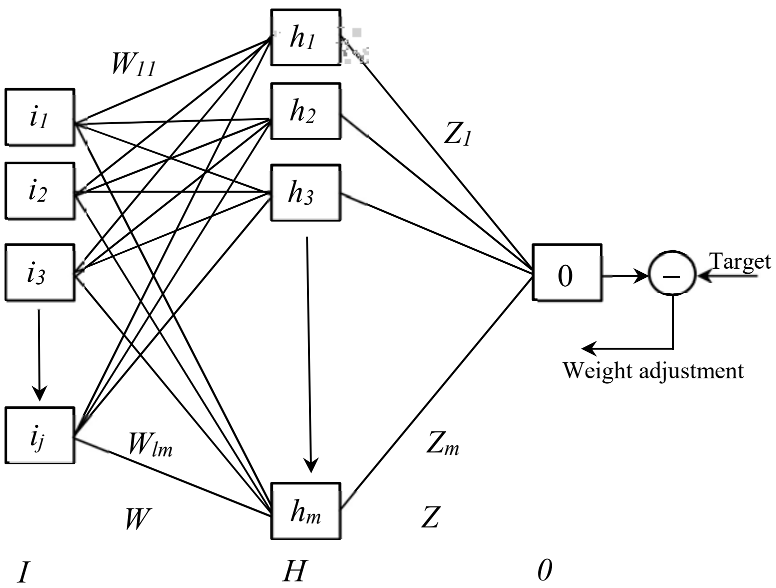

FIG. 15. Example of neural network.

During the learning process we get to know the activation values of the first layer and the objective function. During the start–up, the weights are selected randomly. By means of comparing the received and desirable output values, we apply the above mentioned back propagation method in an iterative way to find the weights and activation values of the hidden layer.

The number of elements in the hidden layer can be given from the experience, the method users recommend using the layer consisting of 400 elements in case of the k_{eff} objective function and 150 elements for estimating the maximum bundle power (P_{max}).

Based on the experience, with the help of less than 1000 (several hundred) of various arrangement samples, the output deviation from the sample can be minimized, the method accuracy is 0.25% for k_{eff} and approximately 5% for P_{max}.

The received results are estimated with the use of so-called fuzzy logic. It means that we do not qualify the arrangement as 'bad' if the estimated P_{max} value exceeds the limit or k_{eff} is lower than the desirable value but analyse the arrangement 'goodness' with the use of a continuous measuring number:

$$j = f\ (P_{max} - P_{lim}\ , k_{eff} - k_{eff}^{ref}) \tag{85}$$

After that the 'best' arrangements can be checked by detailed calculations.

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ABBREVIATIONS

AOA	axial offset anomaly
AOO	anticipated operational occurrences
ASME	American Society of Mechanical Engineers
ATF	accident tolerant fuel
B	boron
BDBA	beyond design basis accidents
BOC	beginning of cycle
BWR	boiling water reactor
CANDU	Canada Deuterium Uranium
CDF	core damaged frequency
CHF	critical heat flux
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CPPF	channel–power peaking factor
CSA	Canadian Standard Association
DBA	design basis accident
DNBR	departure from nucleate boiling ratio
ECC	emergency core cooling
ECCS	emergency core cooling system
EFPD	effective full power day
ENDF	Evaluated Nuclear Data File
FA	fuel assembly
FBD	full–power day
FC	fuel core
GA	genetic algorithm
Gd	gadolinium
GW	giga watt
HF	hexafluoride
HM	heavy metal
IEEE	Institute of Electrical and Electronics Engineers
k_{eff}	effective multiplication factor
kWh	kilo watt hour
LHGR	linear heat generation
LOCA	loss of coolant accident
LPRM	local power range monitoring
LWR	light water reactor
MOX	mixed oxide fuel
MTC	maximum moderator temperature coefficient
Nb	niobium
NPP	nuclear power plant
OJSC	open joint–stock company
OPG	Ontario Power Generation
PCI	pellet cladding interaction
pcm	per cent mille
PD	physicist designer
PHWR	pressurized heavy water reactor

PMA	population mutation annealing algorithm
P_{max}	maximum bundle power
PP	pin power
PPF	maximum bundle power
PSA	probabilistic safety assessment
PSR	periodic safety review
PWR	pressurized water reactor
R&D	research and development
RBMK	graphite moderated water–cooled high–power channel–type reactor
RP	reactor physicist
RSE–M	In–Service Inspection Rules for Mechanical Components of PWR Nuclear Islands
SA	simulated annealing
SA	safety analysis
SCC	stress corrosion cracking
SCRAM	A rapid emergency shutdown of a nuclear reactor
SDS	shutdown system
SFP	spent fuel pool
Sm	samarium
SPND	self–powered neutron detector
SWU	separative work unit
TECDOC	publication in the IAEA–TECDOC series
TIP	transverse inverse probe
U	Uranium
UCF	unit capability factor
UF_6	uranium hexafluoride
UO_2	uranium dioxide
WWER	water cooled water moderated power reactor
Xe	xenon
Zr	zirconium

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