IAEA-TECDOC-1578

Computational Analysis of the Behaviour of Nuclear Fuel Under Steady State, Transient and Accident Conditions

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International Atomic Energy Agency

December 2007

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FOREWORD

Accident analysis is an important tool for ensuring the adequacy and efficiency of the provision in the defence in depth concept to cope with challenges to plant safety. Accident analysis is the milestone of the demonstration that the plant is capable of meeting any prescribed limits for radioactive releases and any other acceptable limits for the safe operation of the plant. It is used, by designers, utilities and regulators, in a number of applications such as: (a) licensing of new plants, (b) modification of existing plants, (c) analysis of operational events, (d) development, improvement or justification of the plant operational limits and conditions, and (e) safety cases.

According to the defence in depth concept, the fuel rod cladding constitutes the first containment barrier of the fission products. Therefore, related safety objectives and associated criteria are defined, in order to ensure, at least for normal operation and anticipated transients, the integrity of the cladding, and for accident conditions, acceptable radiological consequences with regard to the postulated frequency of the accident, as usually identified in the safety analysis reports. Therefore, computational analysis of fuel behaviour under steady state, transient and accident conditions constitutes a major link of the safety case in order to justify the design and the safety of the fuel assemblies, as far as all relevant phenomena are correctly addressed and modelled.

This publication complements the IAEA Safety Report on Accident Analysis for Nuclear Power Plants (Safety Report Series No. 23) that provides practical guidance for establishing a set of conceptual and formal methods and practices for performing accident analysis.

Computational analysis of the behaviour of nuclear fuel under transient and accident conditions, including normal operation (e.g. power ramp rates) is developed in this publication. For design basis accidents, depending on the type of influence on a fuel element, initiating events which may challenge fuel safety can, in general, be grouped into three basic categories: power excursion accident, power-cooling-mismatch accident and decrease of reactor coolant inventory.

This publication has been aided by two important trends. First, the methods of accident analysis have been developed significantly in recent years for a better understanding of physical phenomena, computing capabilities and the integration of research results into code development and application. Second, extensive studies have been carried out to investigate the transient behaviour for postulated initiating events sequences in order to establish that the subsequent fuel conditions do not exceed allowable limits.

More detailed information on available methods for analysis of fuel behaviour under accident conditions and provides practical guidance for use of the methods is provided in this publication. The publication is directed at analysts coordinating, performing or reviewing the analysis of fuel behaviour under accident conditions, both on the designer and utility as well as on the regulatory side.

The IAEA officer responsible for this publication was S. Lee of the Division of Nuclear Installation Safety.

CONTENTS

1. INTRODUCTION

1.1. Background

Accident analysis is an important tool for ensuring the adequacy and efficiency of the provision in the defence in depth concept to cope with challenges to plant safety. Accident analysis is the milestone of the demonstration that the plant is capable of meeting any prescribed limits for radioactive releases and any other acceptable limits for the safe operation of the plant. It is used, by designers, utilities and regulators, in a number of applications such as:

(a) Licensing of new plants:

(b) Modification of existing plants;

(c) Analysis of operational events;

(d) Development, improvement or justification of the plant operational limits and conditions;

(e) Safety cases.

Recently, the IAEA developed a set of guidance documents devoted to accident analysis of nuclear power plants, for different reactor designs and for more specific subjects related to accident analysis [1–4]. IAEA Safety Reports Series No. 23 [1] provides practical guidance for establishing a set of conceptual and formal methods and practices for performing accident analysis. These suggested methods and practices are based on current good practices around the world. This report covers all steps in performing the analysis, i.e. selection of initiating events and acceptance criteria, selection of computer codes and modelling assumptions, preparation of input data, and presentation of the results of calculations. Various aspects in ensuring an adequate quality of accident analysis are also discussed in this report. Requirements and guidelines for the scope and content of accident analysis have been established in several IAEA Safety Requirements [5, 6] and Safety Guides [7].

Detailed understanding of fuel behaviour under steady state, transient and accident conditions is an important part of the analyses of the safety of the nuclear power plant. Investigations of fuel behaviour are carried out in close connection with experimental research operation feedback and computational analyses.

This publication has been developed for computational analysis of the behaviour of nuclear fuel under design basis accident conditions, including normal operation (e.g. power ramp rates). Development of this TECDOC has been aided by two important trends. First, the methods of accident analysis have been developed significantly in recent years for a better understanding of physical phenomena, computing capabilities and the integration of research results into code development and application. Second, extensive studies have been carried out to investigate the transient behaviour for postulated initiating events sequences in order to establish that the subsequent fuel conditions do not exceed allowable limits.

One of the important underlying assumptions in this publication is that, depending on the purpose of the calculation (design improvement, safety demonstration), the analysis of fuel behaviour under steady state, transient and accident conditions will use either a conservative bounding approach for important parameters (still required by most of the regulatory bodies around the world for design basis accidents) or a realistic best estimate approach including evaluation of uncertainties. However, given the maturity of analysis methods and codes, best estimate approaches are now more commonly used, at least by fuel designers and utilities for fuel behaviour analysis and thus, mainly for normal operation. Best estimate approaches have the advantage that they provide a good view of existing margins or limits on nuclear power plant operation in relation to safety analyses. Of course, the use of a best estimate code is essential for a best estimate analysis. Such codes do not include models that are intentionally designed to be conservative. The disadvantage of the best estimate approach is that it is highly dependent upon an extensive experimental database to establish confidence in the best estimate codes and to define the uncertainties that have to be determined for the best estimate results.

Recently, the development of more accurate (best estimate or statistical) analysis methods as well as more advanced fuel and core designs (fuel management) have shown a potential for an increase of operational margins in the design and operation of many current reactor types. However, there is a concern that current safety criteria are directly applicable to the new advanced designs (e.g. new cladding materials) as well as to the full range of current designs. As a result, the IAEA and the OECD/NEA have developed additional documents [8, 9] related to fuel safety criteria in WWERs, PWRs and BWRs and other more advanced fuel and core designs.

The OECD developed a document [9] reviewing the results of existing fuel safety criteria, focusing on the new design elements (new fuel and core design, cladding materials, manufacturing processes, high burnup, mixed oxide fuel (MOX), etc.). This publication also identified whether additional (experimental, analytical) efforts may be required to ensure that the basis for fuel safety criteria is adequate to address the relevant safety issues. Extensive research programmes have also been initiated worldwide to investigate, especially for high burnup issues, the phenomena and mechanisms of fuel behaviour under transient and accident conditions, for example, the ANL test programme in the U.S.A., the Halden Research Project of the OECD, CABRI in France and NSRR in Japan. The ongoing and planned fuel safety research is summarized in Ref. [10].

1.2. Objectives and scope

The objective of this publication is to establish a set of conceptual and formal methods and practices for performing fuel behaviour analysis in water reactors under design basis accident (DBA) conditions. These suggested methods and practices are based on current good practices around the world. This publication applies to the analysis of the fuel condition both inside and outside the core and covers all steps in performing the analysis, i.e. selection of initiating events and acceptance criteria, selection of computer codes and modelling assumptions, preparation of input data, and presentation of the results of calculations. Physical parameters important under accident conditions are summarized in Annex I, while two simple examples of an integrated system thermohydraulic and detailed fuel behaviour analysis and of a neutronic calculation to establish an important set of initial and boundary conditions for a fuel behaviour analysis are provided in Annex II. This publication also summarizes previous experiments to investigate the phenomena and mechanisms of fuel behaviour and the current computer codes used in the accident analysis, methodology to be used and current safety issues.

This publication is intended primarily for code users or reviewers involved in the analysis of fuel behaviour for nuclear power plants. Therefore the initiating events and related phenomena, the methods how to select the analysis methods and how to develop or select appropriate computer codes for the analysis are addressed in this report.

This publication is consistent with the Safety Report [1] and can be considered as a complementary report specifically devoted to the computational analysis of fuel behaviour under normal, transient and accident conditions. Due to the large number of phenomena to be considered in severe accident conditions, the scope of the publication is currently limited to DBAs. Although the publication does not explicitly differentiate between various reactor types, it has been written essentially on the basis of available knowledge and databases developed for PWRs, BWRs and WWERs. However, it can be also used as a preliminary guidance for other types of reactor (RBMKs and CANDUs), with the most important potential differences seen in accident behaviour.

1.3. Structure

The structure of the present publication is as follows. After this introduction, it consists of nine main sections. Section 2 describes the fuel designs (fuel rods, bundles, assemblies) in the various reactors such as PWRs, BWRs, WWERs, RBMKs and CANDUs. Section 3 deals with the main initiating events important to fuel behaviour analysis. Section 4 describes the important fuel behaviour phenomena which can be observed. Section 5 summarizes the related safety criteria of fuel which have been presented in more detail in earlier IAEA and other publications. In order to select and use the computer codes for the analysis, selection of the methods, types of accident analysis, types of computer codes and the necessary code features are discussed in Section 6, including a general description of code validation and verification. Sources of the code user effects and ways to reduce the effects are also provided. In Section 7, a practical application of fuel behaviour codes is provided. Current safety issues and related ongoing or future experimental programmes are discussed and presented in Section 8. Section 9 provides a summary of the report and suggests useful recommendations for future work. Annex I describes important parameters under accident conditions, and Annex II provides examples of fuel behaviour calculations.

2. FUEL ASSEMBLY DESIGN AND CHARACTERISTICS

Design features important to the analysis of fuel behaviour during accident conditions include the design of the fuel and cladding, the arrangement of the fuel rods in the fuel assemblies and the arrangement of the fuel assemblies in the core. As shown in Table 1, the component of the fuel and cladding is approximately the same in all of the relevant power reactor designs. For example, UO_2 is generally used in most of the designs, although MOX or $UO_2Gd_2O_3$ pellets might be used. All of the fuel designs use fuel pellets, prepared from a sintered powder with a theoretical density above 90%. The fuel pellets typically are shaped to minimize pellet–cladding mechanical interactions and may include dished and/or tapered ends. Some of the designs use annular pellets (e.g. for WWERs) devoted, to some extent, to improving the performance of the fuel during normal and accident conditions (e.g. fission gas releases, internal pressure). The fuel cladding of the different reactor types is also more or less similar, using different alloys of zirconium. Some advanced designs may also include special coatings or different layers of cladding materials to further reduce corrosion and fuel rod failures during normal operation or accidental situations. Since these subtle differences in fuel and cladding design were intended to improve the economics of the operation of the fuel by reducing fuel failures or improving operational performance, much of the detailed design information may be proprietary to a particular fuel vendor and may not be modelled in the generally available fuel behaviour codes discussed in this report .

Therefore, despite this possible lack of detailed knowledge of the fuel and cladding properties, the most important physical phenomena involved in the considered initiating events can be calculated and assessed, at least by performing sensitivity studies. However, in general, it is important to include the specific details (properties) of the fuel and cladding design to the extent possible for any fuel behaviour analysis. For example, the relative gas volumes in the pellet dishes, fuel–cladding gaps and fuel plenum regions have a direct bearing on the calculation of internal fuel rod pressure and fuel rod failure. The initial fuel–cladding gaps also have a strong impact on the temperature distribution within the fuel because of the impact on the gap conductance (i.e. the ability of the heat to be removed from the fuel surface). As a result, the gap design information can also impact the release of fission products, the failure of the cladding through pellet–cladding mechanical interactions, and ultimately the peak temperatures reached in the fuel and cladding during accident conditions.

The arrangement of the fuel rods in the fuel assemblies is more varied and, like the fuel rods, may have proprietary features from each vendor. However, some of the general features of the designs are well documented and available for general use. For example, the CANDU design has relatively short horizontal fuel elements, stacked end-to-end within a single fuel channel. The RBMK also uses two vertically stacked fuel assemblies, but the length of each assembly (3.4 m) is approximately the same as that of the fuel assemblies in the other designs. The CANDU, RBMK and BWR assemblies also use control elements that are located outside the actual fuel assembly, while the PWR and WWER assemblies include control rods distributed within the fuel assembly. All the assemblies also include some type of grid spacers or spacer elements composed either of zirconium alloys or structural materials such as stainless steel or Inconel.

The arrangement of the fuel rods within the assembly, and the specific features of the assembly design, also have a direct and indirect impact on the analysis of the fuel behaviour. The spacing between the fuel rods, i.e. the rod pitch, determines the reduction in available flow area as fuel rod ballooning and rupture occurs. The spacing can also have an impact on the deformation itself, as temperature variations around the cladding radius can strongly influence the nature and timing of fuel rod deformation and failure. The orientation of the fuel assemblies also has a strong impact on the deformation and ultimate failure of the fuel rods, since horizontal fuel elements will sag and deform in a much different way from vertical fuel elements. The design features also have an indirect impact on the analysis, since the thermohydraulic and neutronic conditions within the core and assembly will depend on the design. For example, spacer grids can act to promote turbulence within the assembly and impact the temperature of the cladding, while the location of the control elements will determine the power distribution within the fuel assembly. The arrangement of the elements of a fuel assembly is even more important as accidents transition between DBA and severe accident conditions. In this case, chemical interactions between the cladding, spacer grids and, in some cases, control materials may result in the earliest liquefaction of the assembly structures. For detailed application cases, dedicated reports should be required from the fuel designers, including:

- Presentation of the design requirements and detailed description of the fuel assembly;
- Fuel assembly design justifications in the fields of mechanical, thermohydraulic, neutronic and thermomechanical behaviour;
- Impact of the introduction of the fuel assemblies on the safety demonstration.

A useful survey of the main characteristics of nuclear power plants in use in the EU and candidate countries, in 2001, is provided in Ref. [11]. This includes details of various types of PWR, BWR, WWER-440, WWER-1000, RBMK and the gas cooled reactors Magnox and AGR in the UK.

TABLE 1. DESIGN PARAMETERS FOR VARIOUS REACTORS

Design parameter	PWR	BWR	WWER	CANDU	RBMK
Fuel material	UO_2, MOX, UO_2,Gd_2O_3	UO_2	UO_2	UO_2	UO_2
Type	Pellet	Pellet	Annular pellet	Pellet	Annular pellet
Active length (m)	~3,6 for 3 loops ~4,2 for 4 loops	3.81	~2.4 (WWER 440) ~3.6 (WWER 1000)	~0.5 x 12/ element	~6.8 (channel) ~3.4 (fuel)
Cladding material	Alloys of Zr	Zircaloy-2	Zr4 and Zr+1% Nb	Zircaloy-4	Zr+1% Nb
Control element location	Internal	External	Internal for WWER-1000 Special assem-blies for WWER -440	External	External
Assembly	Rectangular fuel rod array including control rods, zircaloy spacer grids	Rectangular fuel rod array with water tubes, zircaloy spacer grids	Hexagonal fuel rod bundle, Zr–Nb spacer grids	Iincludes a Zr–Nb alloy pressure tube, a zirconium calandria tube, stainless steel (SS) end fittings at each end	Iincludes 18 fuel rods surrounding the Zr–Nb carrier rod, combination of SS & Zr–Nb spacer grids
Other features	Control rods are surrounded by zircaloy guide tubes, Ag/In/Cd absorber material, sometimes also B_4C	Fuel assembly surrounded by a zircaloy-2 channel. SS clad, B_4C control blades for each 4 assemblies	WWER-440 experimental research complex (ERC) assembly consists of two parts, the fuel assembly and the absorber, B_4C for WWER-1000, boron steel in WWER-440	Each pressure tube is thermally insulated from the cool, low pressure moderator by a CO_2-filled gas annulus formed between the pressure tube and the concentric calandria tube	Fuel assembly surrounded by a channel tube made of Zr–Nb in the central axial zone and SS in the upper and lower zone. Channels surrounded by graphite moderator

2.1. Western PWRs

A number of fuel vendors supply PWR fuels. In the following discussion, representative design features are described. However, analysts should refer to the detailed design information provided by the fuel vendor for specific design information.

Amongst others, the design example of the Framatome-ANP assembly has been chosen for illustration (Fig. 1). It consists of a 17 x 17 array of 264 fuel rods, 24 control rod guide tubes, one instrumentation tube, a bottom end piece, a top end piece and eight axially arranged spacer grids in the case of an active core height of 30.48 cm (12 ft). Optionally, the fuel assemblies are equipped with a debris filter and, for the increase of thermohydraulic margins, with three intermediate flow mixers.

The detailed features of this fuel assembly design are:

- Corrosion resistant Duplex cladding, capable of high burnup without loss of rod integrity. This cladding tube is proposed for rod burnups over 55 MW·d/kg U.

- Options are fuel rods with natural uranium axial blankets, which increase neutron economy by an enrichment saving of about 0.06 wt% ^{235}U.

- All-zircaloy high thermal performance spacers with integrated curved flow channels, utilized for all but the bottom spacer position, increase the coolant mixing and enhance departure from nucleate boiling (DNB) performance.

- The Inconel high thermal performance spacer at the lowermost position provides improved fuel rod support throughout service life at the bottom of the fuel rod region and minimizes the possibility of flow induced fretting failures.

- The debris resistant bottom end piece with curved blades provides almost complete protection against debris induced fretting failures.

- The readily removable top end piece allows quick and easy fuel assembly repair, reconstitution or surveillance from the topside.

- The demountable bottom end piece allows fuel assembly repair, reconstitution or surveillance also from the bottom side, should the need arise.

- The gadolinium burnable neutron absorber with optimized gadolinium absorber length provides operating and fuel cycle design flexibility. When incorporated in the UO_2 pellets of selected rods, gadoliniuim avoids the cost of separate encapsulation required for B_4C or borosilicate glass and its residual parasitic absorption. The integration of gadolinium-bearing fuel rods minimizes radial neutron leakage which, together with the reduced residual reactivity penalty, would decrease batch average enrichments.

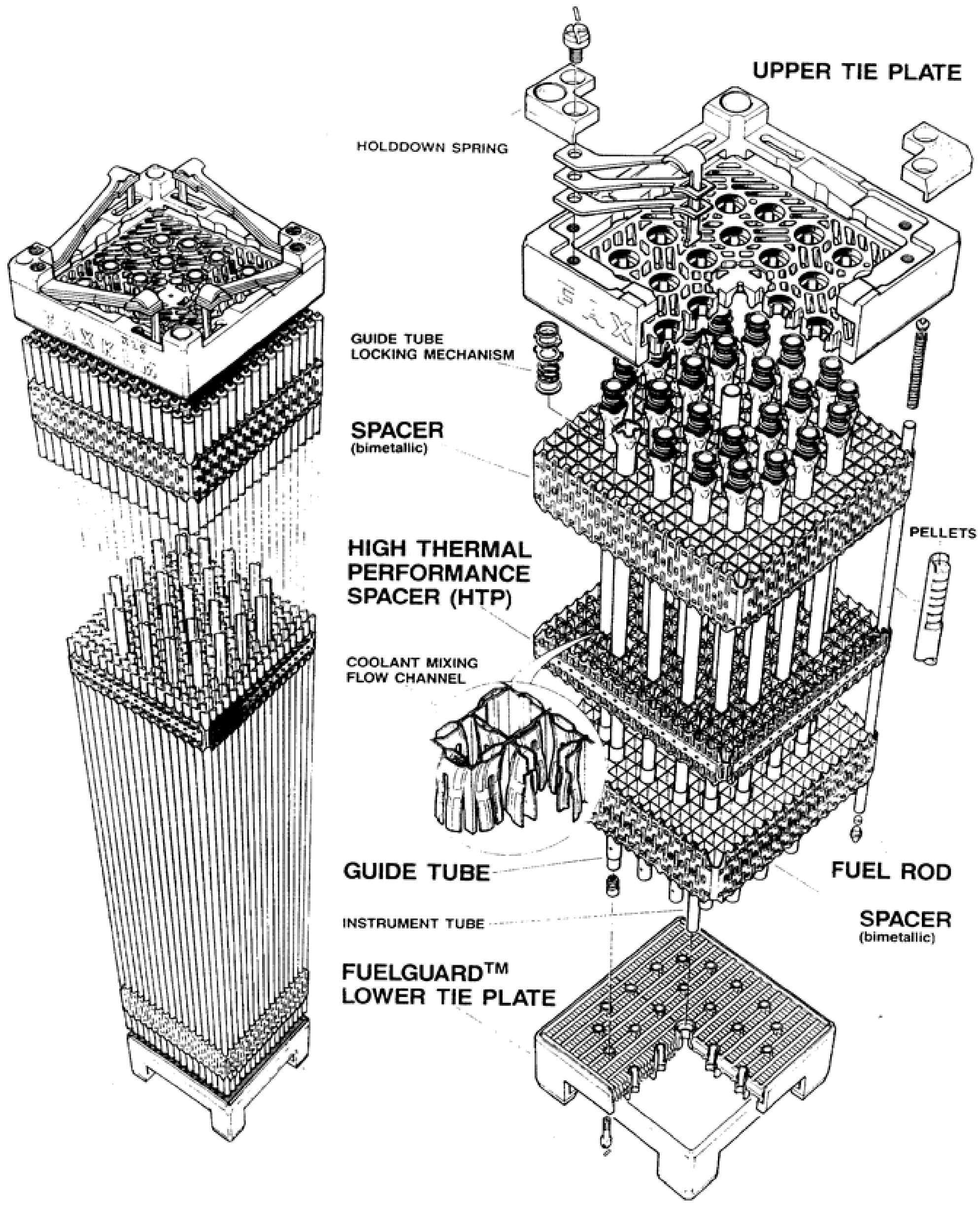


FIG. 1. PWR fuel assembly design.

2.2. BWRs

A number of fuel vendors supply BWR fuel assemblies. In the following discussion, representative design features are described. However, analysts should refer to the detailed design information provided by the fuel vendor for specific design information.

BWR assemblies have three general design features that are relevant for the analysis of fuel behaviour. Firstly, and most importantly from an assembly design viewpoint, the design effectively separates the control elements away from the fuel bundle, so that the fuel behaviour analyst does not have to consider the possible thermal or mechanical coupling between the fuel rods and control elements under typical design basis conditions. Figures 2 and 3 show representative BWR fuel assemblies and the relative location of control elements and fuel assemblies. Secondly, the thermohydraulic and neutronic boundary conditions may be somewhat more complicated than those in the more homogenous PWR core. For example, some fuel assembly loading schemes may result in different burnup and power levels in the four assemblies surrounding the control blade. Thirdly, because of the relatively large number of competitive BWR fuel assembly designs (relative to those of RMBKs or WWERs), it may be more difficult to apply generally available codes and models to vendor specific design features.

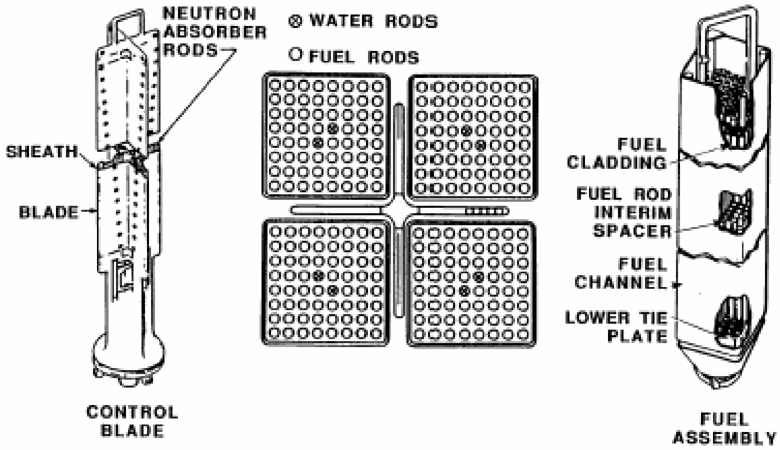


FIG. 2. Typical fuel assembly and control blade designs for BWRs [12].

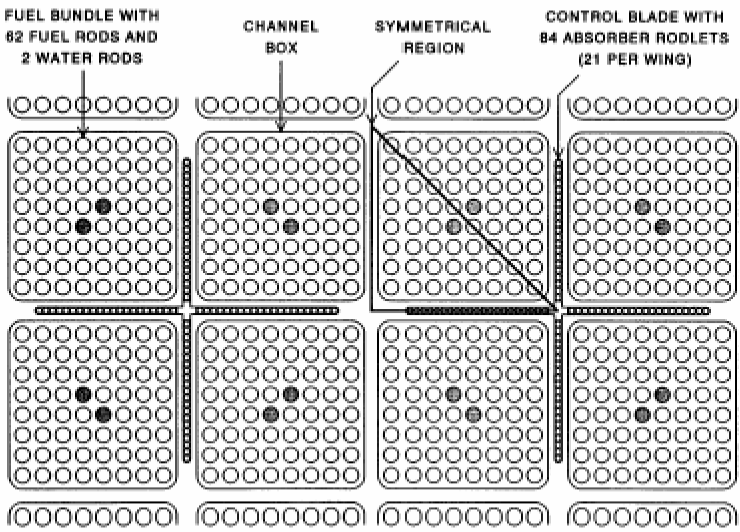


FIG. 3. Typical arrangement of fuel assemblies and control blades in a BWR core [12].

2.3. WWERs

2.3.1. WWER-440

The working assembly shown in Fig. 4 consists of the fuel rod bundle, cap, tailpiece and jacketed tube. The fuel rods within the bundle are arranged in a triangular array and are connected by the 'honeycomb type' spacer grids mechanically mounted on the central tube and by the lower support grid mounted on the tailpiece. The lower support grid is welded to the tailpiece intended to install the working assembly within the reactor basket bottom. The working assembly tailpiece is installed into the basket bottom seat resting by its ball surface upon the seat conical part.

The working assembly cap is rigidly attached (over the hexahedral surface) to the jacketed tube. In the working assembly cap there are two fingers for the transport grip of the fuel handling machine and six spring-loaded stops used to prevent the working assembly from floating and to compensate for thermal expansion and technological tolerances of the reactor internals. The bottom end of the cap is attached to the protective grid. The fuel rods are fixed in the support grid by the pin wire. To compensate for thermal expansion and radiation growth of the fuel rod bundle with respect to the support grid, the working assembly allows possible elongation of the fuel rods.

In the lower and upper parts of the working assembly jacket, in a region of the cap and tailpiece there are holes (two on each flat) intended for radial offloading of the jacketed tube from coolant pressure differential. The experimental research complex assembly consists of the fuel assembly (Fig. 5) and the absorber (Fig. 6), connected with each other by the intermediate mast. The fuel rods are triangle-arrayed in the fuel assembly. The absorber itself consists of a welded structure made of stainless steel, with hexahedral inserts of boron steel located inside.

The fuel assembly design is identical to the working assembly design except for the following features: special tailpiece, fuel assembly jacket without perforations, and fuel assembly cap without spring-loaded stops.

The fuel assembly cap is equipped with a grip device of bayonet type with a seat for a triangle catch used to provide engagement with the intermediate mast. The intermediate mast, which passes through the absorber centre, for its full height is engaged with the grip bayonet device located in the fuel assembly cap; in this case the fixing triangle rod of the intermediate mast enters the fuel assembly cap seat, thereby avoiding rotation and subsequent disengagement of the fuel assembly from the intermediate mast.

In the fuel assembly tailpiece there is a damper device (thimble) used to provide experimental research complex assembly damping during its movement (drop) by gravity under the accident condition related to a break of the intermediate mast. The damping principle consists of coolant (water) throttling through the gaps formed between the rod located in the reactor cavity bottom and the fuel assembly tailpiece thimble at the moment when the experimental research complex assembly drops and the thimble seats upon the rod.

In addition, water throttling occurs through two holes or more of 3 mm diameter located in the fuel assembly tailpiece thimble bottom.

2.3.2. WWER-1000

The WWER-1000 fuel assembly (Fig. 7) consists of the following components:

- Cap;
- Bundle of fuel rods;
- Tailpiece.

The cap consists of the following parts: upper shell, supporting plate, spring unit, lower shell, collets and components connecting the assembly units of the cap in the common structure.

The bundle of fuel rods is assembled of 312 fuel rods in a frame consisting of 15 spacer grids, a central tube, 18 guiding channels and the lower supporting grid.

The fuel rod consists of the following parts: upper plug, cladding, lower plug, fuel core made of UO_2 pellets, and a lock. The material of the fuel rod cladding and plugs is Zr–1%Nb alloy. The spacer grid is made to give an interference fit in pairs between a cell– fuel rod, and a cell–fuel assembly guiding channel.

The fuel assembly tailpiece is a supporting welded construction – the body, with a system of ribs. The ribs, welded to the shell, form the supporting grid, containing two parallel ribs crossing the third rib in a transverse direction. The ribs are enclosed in a hexahedron with transition to a cylinder. The inside of the lower part of the tailpiece body is made in the form of a diffuser, and from the outside has a supporting spherical part with transition to the cylinder. The lock is installed on the cylinder. The bundle of fuel rods through the lower supporting grid rests on the parallel ribs of the tailpiece.

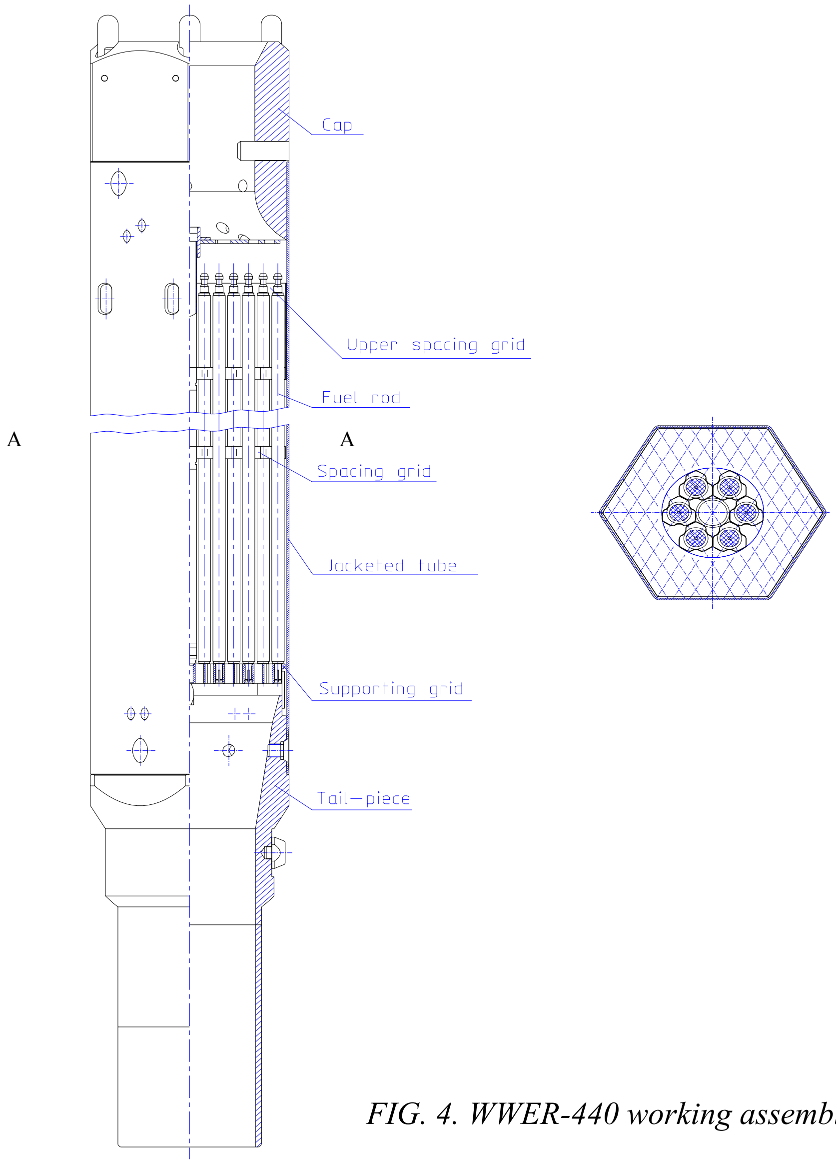
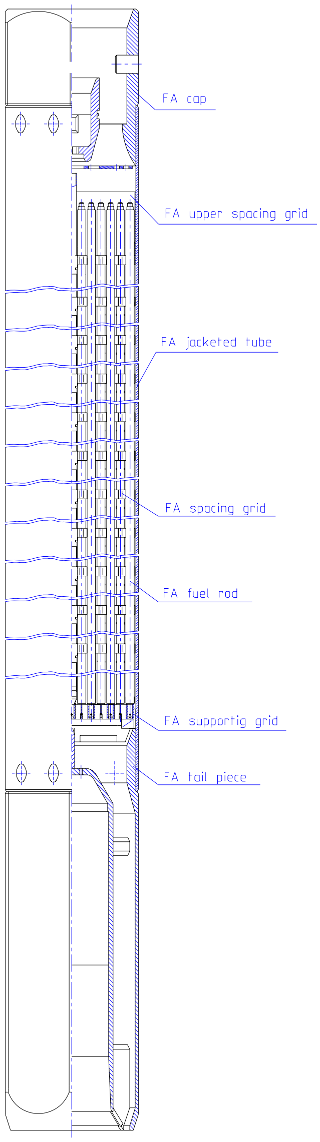


FIG. 4. WWER-440 working assembly.



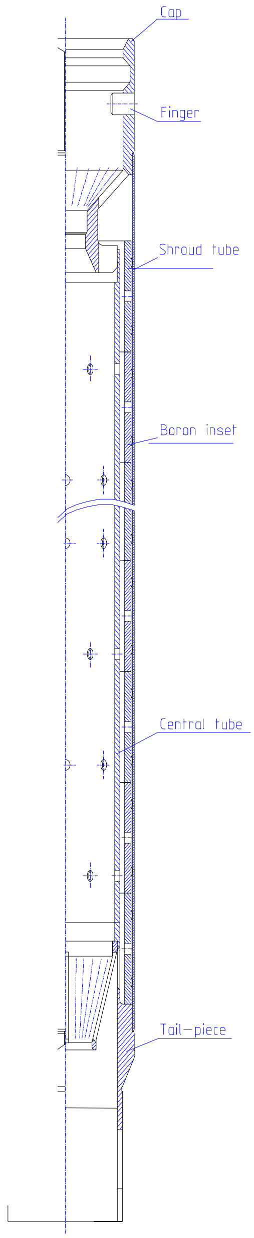


FIG. 5. Fuel part of ERC assembly. *FIG. 6. Absorber part of ERC assembly.*

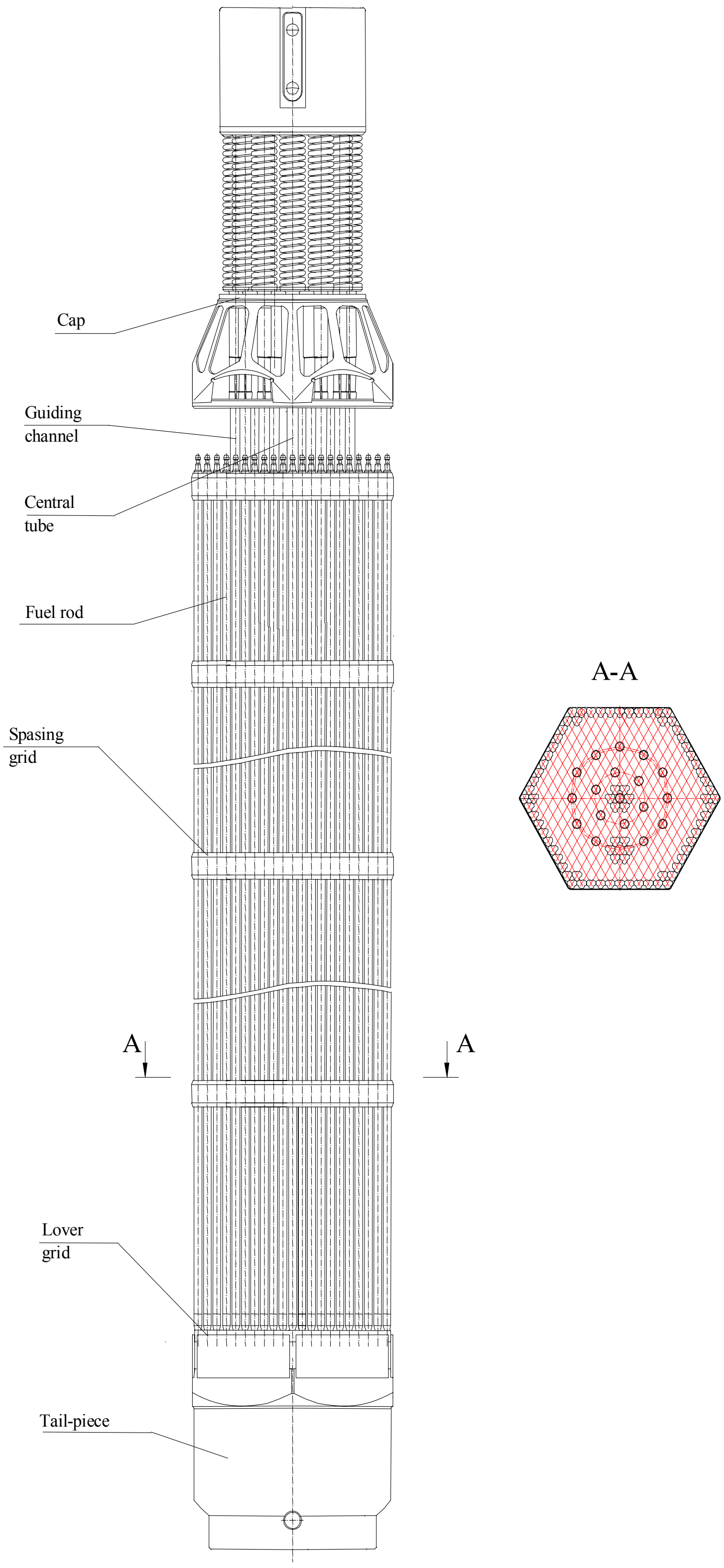


FIG. 7. WWER-1000 fuel assembly.

2.4. RBMKs

The RBMK fuel assemblies are unique in three general ways relative to the fuel assemblies in other reactor designs [13]. Firstly, the rod assemblies are contained in individual flow channels and effectively isolated from one another within a large graphite moderator core. Secondly, the active length of the fuel assemblies is much longer than those of other designs, with a total active length of nearly 7 m. As shown in Fig. 8, this height is obtained by vertically stacking two 3.4 m fuel assemblies within the single flow channel. Thirdly, as shown in the horizontal cross-section, the 18 fuel rods surround a central structural support or instrument rod. The RBMK fuel assemblies, like the CANDU and BWR fuel assemblies, do not include control elements within the assemblies. These control elements are contained in separate channels in the graphite moderator.

The fuel rods are composed of UO_2 fuel pellets surrounded by Zr–1% Nb cladding. The annular fuel pellets have dished pellet faces. The central hole reduces peak fuel temperatures as well as providing additional volume for fission gas release. The enrichment varies between 2 and 4%. The fuel rods are initially pressurized with He at 0.5 MPa. The initial pellet diameter is 11.5 mm with a central hole diameter of 2 mm. The initial cladding thickness is 0.9 mm. The initial gap thickness varies between 0.22 and 0.38 mm.

The 18 fuel rods are located in a circle surrounding the central support rod made of Zr–2.5% Nb. In some assemblies the central support rod is replaced by an instrument tube. The fuel assemblies also include a combination of stainless steel spacer grids located along the length of the assembly and Zr–Nb end fittings. The upper assembly also has grids designed to enhance the turbulence of the flow at selected positions along the assembly. The assemblies are supported in such a way that the axial expansion of fuel assemblies as they heat up elongates the assemblies towards the centre of the core; that is, the bottom assembly is supported from the bottom and the upper assembly is supported from the top.

The unique design features of the RBMK present challenges for the analyst, since the fuel behaviour, system thermohydraulics and core neutronics can be very tightly coupled. Indeed, the specific design of the reactor (size, combination of graphite moderator and water coolant, etc.) means that it is unstable at low power levels and calls for more accurate calculations for modelling all the coupled phenomena involved.

Coupled 3-D neutronic-system thermohydraulic calculations are normally required to determine the boundary conditions for a fuel behaviour calculation. In addition, since most system thermohydraulic codes lack detailed fuel behaviour models, an iterative procedure may be required to determine the actual response of the fuel and reactor system.

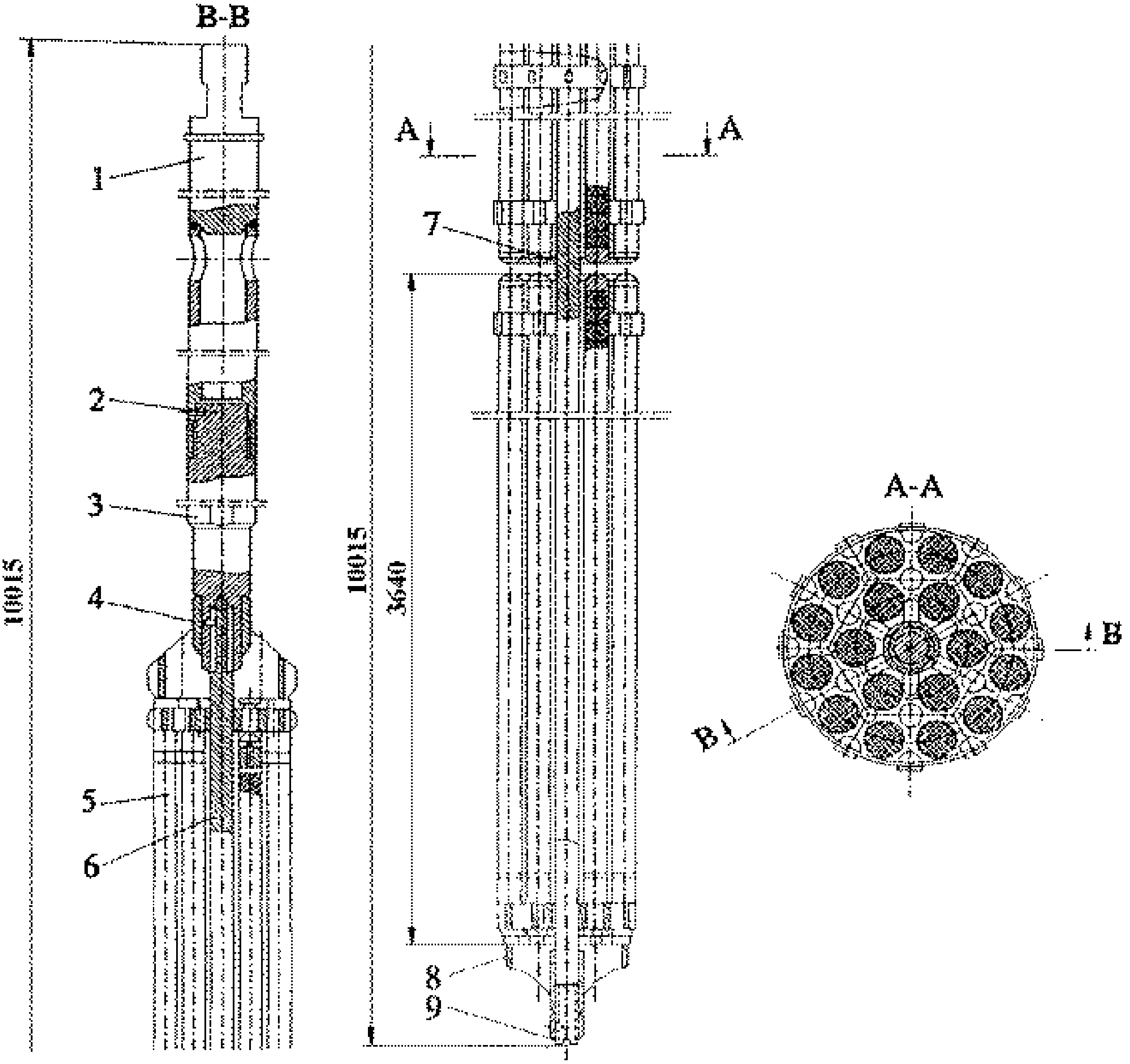


FIG. 8. RBMK fuel assembly [13]
(1 – suspension bracket,; 2 – top plug; 3 – adapter; 4 – connecting rod; 5 – fuel rod; 6 – carrier rod; 7 – end sleeve; 8 – end cap; 9 – nut; (dimensions in mm)).

2.5. CANDUs

The CANDU reactor assembly includes several hundred channels contained in and supported by a horizontal cylindrical tank known as a calandria. The calandria is closed and supported by end shields at each end. Each end shield comprises an inner and an outer tubesheet joined by lattice tubes at each fuel channel location and a peripheral shell. The inner spaces of the end shields are filled with steel balls and water, and are water cooled. The fuel channels, supported by the end shields, are located on a square lattice pitch. The calandria is filled with heavy water moderator at low temperature and pressure. The calandria is located in a light water filled shield tank.

Horizontal and vertical reactivity measurement and control devices are located between rows and columns of fuel channels, and are perpendicular to the fuel channels.

Each fuel channel locates and supports 12 fuel bundles in the reactor core. The fuel channel assembly includes a Zr–Nb alloy pressure tube, a Zr calandria tube, stainless steel end fittings at each end, and four spacers which maintain separation of the pressure tube and calandria tube. Each pressure tube is thermally insulated from the cool, low pressure moderator by the CO_2 filled gas annulus formed between the pressure tube and the concentric calandria tube.

The CANDU fuel bundle consists of 37 elements, arranged in circular rings as shown in Fig. 9. Each element consists of natural uranium in the form of cylindrical pellets of sintered uranium dioxide contained in a zircaloy-4 sheath closed at each end by an end cap. The 37 elements are held together by end plates at each end to form the fuel bundle. The required separation of the fuel element is maintained by spacers brazed to the fuel elements at the transverse midplane. The outer fuel elements have bearing pads to the outer surface to support the fuel bundle in the pressure tube.

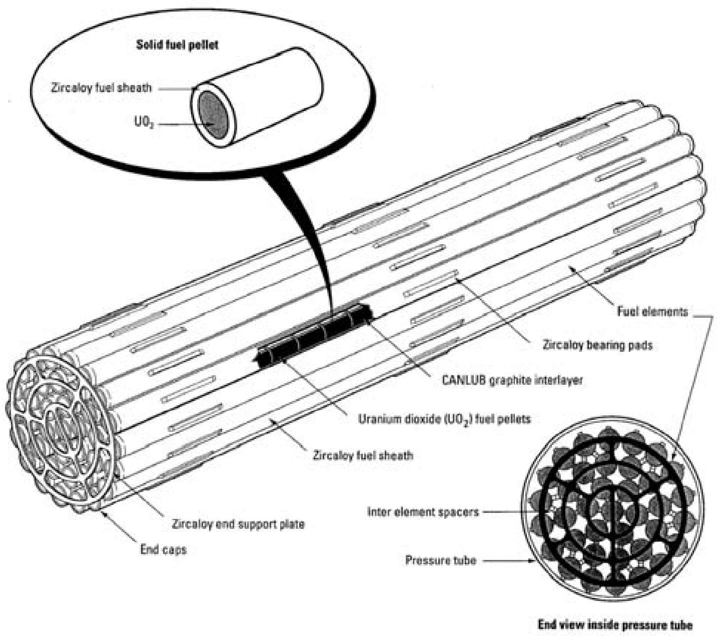


FIG. 9. CANDU fuel pellet and bundle [14].

3. INITIATING EVENTS

3.1. Categorization of initiating events

The categorization of initiating events has been discussed in Ref. [1], and the possible subdivision according to the frequency of occurrence is shown there. They can be considered as anticipated operational occurrences (DBAs), beyond design basis accidents and severe accidents. The present report will be limited to anticipated operational occurrences (including normal operation) and DBAs (anticipated transients and accidents).

The anticipated operational occurrences and the anticipated transients are expected to occur during the lifetime of the plant. The acceptance criteria for transients are determined to assure no additional fuel damage (integrity of the cladding) and reuse of fuel assemblies which experienced the transient.

For accidents, the acceptance criteria are determined to assure no radiological impact at all, or no radiological impact outside the exclusion area of the plant.

A large number of transient or accident scenarios can be derived from combinations of event categories. These are discussed in Refs [2–4] for various types of reactor. As an example, typical initiating events for anticipated transients and accidents of PWRs are shown in Table 2.

3.1.1. Normal operation and anticipated transients

For normal operation and anticipated transients, the integrity of the cladding must be ensured. Since the aim of the safety criteria for normal operation and transient conditions is to maintain fuel element integrity as summarized in Section 5, detailed fuel behaviour analysis is required.

For normal operation, the fuel designer must ensure integrity of the cladding during the time of irradiation of the fuel assemblies (e.g. corrosion thickness, internal pressure), including all the normal operation transients (e.g. power changes, power rates changes).

Due to the increased burnup policy, issues dealing with the evaluation of corrosion thickness, internal pressure, fretting wear and fretting corrosion should be addressed properly, notably by use of appropriate qualified and validated computer codes.

For anticipated transients, the cladding integrity is usually ensured by demonstrating that departure from nucleate boiling ratio (DNBR) and fuel melting temperature criteria are not exceeded. Nevertheless, in order to take into consideration the existing pellet–cladding mechanical interaction (PCMI) occurring during the transient, dedicated cladding stress thermomechanical calculations under transient conditions (following load follow operation) might be performed and compared to an adequate criterion issued from experimental ramp tests.

3.1.2. Accident conditions

For fuel behaviour computational calculations, accident conditions can be grouped into the three following conventional types of transients:

- Power excursion accident,
- Power cooling mismatch accident (PCMA),
- Decrease of reactor cooling inventory.

3.1.2.1. Power excursion accidents

Power excursion accidents include principally reactivity accidents (including reactivity initiated accidents (RIAs)) and cooling accidents (e.g. main steam line break). Since it does not call for in-depth fuel behaviour calculations, only the RIA case, corresponding to fast ejection of a rod control cluster assembly, is considered hereafter.

When a large reactivity spike is inserted by a very fast ejection of a rod control cluster assembly, a rapid power excursion occurs. Since the time duration of the accident is quite short, fuel behaviour is determined by how much energy is generated in the fuel during the short period of the accident in the local part of the core. Therefore, one of the acceptance criteria for this accident is usually defined as the maximum adiabatic enthalpy per unit mass added to the fuel element.

It is necessary to relate to this group also failures in which significant changes to thermophysical properties of the coolant lead to an increase in core power due to reactivity feedback effects. These failures can be accompanied by asymmetrical change of core power distribution and local impairment of heat removal conditions.

Phenomena which may be anticipated during the accident are as follows:

- Rapid reactor power excursion caused by excess reactivity insertion, resulting in a rapid fuel temperature rise.
- Rapid increase of fuel surface heat flux, resulting in a reduction of DNBR; then departure from nucleate boiling ratio (DNB) may occur, resulting in a reduction of the heat removal capability, then a cladding temperature rise.
- Contact of fuel pellet and cladding due to the thermal expansion of the pellets, resulting in enhancement of a pellet–cladding interaction.
- Cladding temperature rise, which may cause enhancement of cladding oxidation.
- Rapid fuel temperature rise, which may result in fuel melt; then the cladding deformation due to fuel volume expansion may be enhanced.
- Sudden decrease of power due to negative reactivity insertion results in the rewetting of cladding. A sudden cladding temperature drop may cause a thermal shock to the cladding.
- A large release of fission gases for fuel rods with high burnup. A large release of fission gas influences the cladding internal gas pressure, cladding ballooning, cladding temperature, cladding oxidation and fuel temperature.

The computational analysis of the fuel behaviour during such transient is particularly sensitive to the fuel design evolution (pellet and cladding) as well as to the operating conditions (burnup increase), and may call for further appropriate developments in order to demonstrate that the related criteria are not exceeded.

3.1.2.2. Power cooling mismatch accidents (PCMAs)

PCMAs include the transients of decrease of reactor coolant as well as decrease of heat removal by the secondary side. The example of seizure of one reactor coolant pump or a reactor coolant pump shaft break is described hereafter.

When heat removal capability is suddenly degraded by the previous initiating events, the generated heat in a fuel element may not be able to be removed to the coolant. This mismatch of heat generation and heat removal may cause DNB, and the cladding temperature may rise suddenly due to the small heat removal capability of transition boiling or film boiling. When control rods are inserted following a safety system actuation signal, the core power decreases, the heat transfer mode returns to nucleate boiling and the accident is terminated. The cladding temperature suddenly drops to the coolant temperature due to return to nucleate boiling or rewetting. Overheating of cladding in film boiling conditions or thermal shock at fast cooling may challenge the fuel integrity. Safety criteria for this type of accident are usually defined by the cladding temperature, the number of rods entering DNB and the amount of cladding oxidation.

Phenomena which may be anticipated during the accidents are as follows:

- Reduction of the primary flow, leading to a reduction of heat removal capability from the fuel, potentially exceeding core thermal limits, which means a DNB.
- When DNB occurs, the fuel and cladding temperatures will rise. A rapid fuel temperature rise can lead to an additional fission gas release to the space in the fuel rod and then cause the internal pressure to rise.
- At high cladding temperatures, collapse or swelling of the cladding may occur in accordance with the pressure difference across the cladding. In addition, cladding oxidation is enhanced.
- Sudden decrease of power followed after control rod insertion may result in rewetting of cladding. A sudden cladding temperature drop may cause thermal shock to the cladding.

3.1.2.3. Decrease of reactor coolant inventory

A loss of coolant accident (LOCA) is caused by the loss of integrity of the primary circuit or its associated pipes and devices. The direct cause of the accident is either a material defect or a material fatigue, either an external impact (internal missiles, heavy load), a seismic event or a device failure during operation of the plant. Inadvertent opening of the pressurizer valves or any other isolation valve on the primary system boundary (which belong to the anticipated transients category) can also be analysed as a LOCA. The spectrum of postulated leakage sizes within the reactor coolant pressure boundary has been divided in various ways depending on the selection of acceptance criteria. In this report, the subdivision is made into two groups, namely large break LOCAs and small break LOCAs.

Large break LOCAs mainly include a full or partial rupture of the main circulation line. Rupture of the major pipes connected to the primary circuit, such as the pressurizer surge line or the accumulator discharge lines, can also be considered as a large break LOCA. In the case of a large break LOCA, the loss of primary coolant cannot be compensated by the emergency core cooling system prior to substantial depressurization and loss of coolant inventory from the primary circuit.

A small break LOCA includes such breaks, smaller in size in comparison with large break LOCAs, which cannot, however, be compensated by the make-up system and thus require activation of the emergency core cooling system. The course of the accident is highly dependent on the break size and the position of the break.

The features of LOCAs relevant to fuel behaviour are extreme degradation of heat removal capability from fuel and a sudden fall in primary system pressure. The core becomes subcritical soon after the accident by control rod insertion or void formation in the core. However, cladding is gradually heated by the stored energy in the fuel and decay heat due to the extremely small heat transfer capability from the fuel rod exposed to the steam. In addition, at high temperatures, the exothermic zirconium–water reaction can also contribute to heat-up of fuel rods. This chemical reaction does not only make the cladding temperature rise but can accelerate oxidation and hydrogen absorption of the cladding. With the decrease of primary system pressure, the fuel internal pressure exceeds the system pressure. At high temperatures, large pressure differences across the cladding can cause cladding ballooning and/or burst. Ballooning of the cladding can lead to reduction of the flow cross-section of a rod bundle and may lead to a deterioration of heat removal conditions. Furthermore, additional heat generation due to inside surface oxidation of the burst clad makes the cladding temperature rise, and hydrogen by-produced may be absorbed by cladding and promote its embrittlement (in practice this inside surface oxidation is limited by hydrogen blanketing inside the clad and counter-current flow restrictions through the burst opening, so is restricted to a few centimetres from this site). The cladding temperature suddenly drops to the coolant temperature due to quench. This thermal shock may challenge the fuel integrity. Safety criteria for this type of accident are usually defined by cladding temperature and amount of cladding oxidation.

Phenomena which may be anticipated during the large break LOCA following a cold leg break are as follows:

- Sudden depressurization and flushing of coolant, degrading the heat removal capability from fuel rods; then the cladding temperature rapidly rises, with occurrence of DNB. The core becomes subcritical soon after the accident, resulting from the formation of a void.
- The cladding temperature once drops by reverse flow in the core formed by break flow at the break point.
- The core is dried out and then the fuel rods are exposed in the steam. Water injected by the emergency core cooling system fills the lower plenum, the bottom of the core and then the fuel assembly channels. During this period, the part of the fuel rods above the froth level gradually heats up.
- When the fuel rod is submerged, the cladding temperature suddenly drops to the water temperature. This phenomenon is called quenching. Quenching may cause a thermal shock to the cladding.
- At high temperatures, the cladding material reacts with the steam in an exothermic reaction, producing hydrogen. This reaction represents an additional heat source for the cladding and can cause further degradation of the cladding by positive feedback, and the potential for hydrogen burning or absorption.

In a CANDU reactor, the delay between initiation of the LOCA and insertion of the emergency control rods — a few seconds — can lead to a reactivity excursion and thus adds a RIA component to this kind of accident sequence.

3.1.3. Summary of categories of initiating events

Fuel characterization under normal operating conditions (steady state conditions) is fundamental, as input data, for studying fuel behaviour under anticipated and accidental conditions.

In this respect, the fuel behaviour transient analysis strongly depends on:

- Category of initiating event considered and associated criteria to be met;
- Kinetics and duration of the transient;
- Power level reached during the transient.

Therefore, all relevant related phenomena involved during the transient must be adequately addressed by the fuel behaviour computer codes.

TABLE 2. TYPICAL INITIATING EVENTS FOR PWR ANTICIPATED TRANSIENTS AND ACCIDENTS

Category	Transient	Accident
Sudden reactivity insertion	• Control rod withdrawal • Control rod malfunction • Incorrect connection of an isolated reactor coolant system loop • Boron dilution due to a chemical and volume control system malfunction • Inadvertent loading of a fuel assembly into an improper position	• Control rod ejection
Decrease of reactor coolant flow	• Inadvertent closure of a main isolation valve in a reactor coolant system loop • Partial (i.e. less than all pumps) loss of reactor coolant flow due to pump motor trip	• Seizure of one reactor coolant pump • Shaft break of one reactor coolant pump • Full (i.e. all pumps) loss of reactor coolant flow due to pump motor trip
Increase of reactor coolant inventory	• Inadvertent actuation of emergency core cooling system • Malfunction of chemical and volume control system leading to reactor coolant inventory increase	
Increase of heat removal by the secondary side	• Inadvertent opening of steam relief valve • Secondary pressure control malfunction with increase of steam flow • Feedwater system malfunction leading to increase of heat removal	• Steam line break • Main steam header rupture

TABLE 2. TYPICAL INITIATING EVENTS FOR PWR ANTICIPATED TRANSIENTS AND ACCIDENTS (continued)

Category	Transient	Accident
Decrease of heat removal by the secondary side	• Loss of off-site power • Loss of feedwater flow (pump trip, closure of valves) • Reduction of steam flow from the steam generators or turbine trip	• Feedwater line break
Decrease of reactor coolant inventory	• Malfunction of chemical and volume control system leading to reactor coolant inventory decrease	• Primary system pipe break (LOCA) • Leak from primary side to secondary side of the steam generator • Inadvertent opening of a pressurizer safety valve which is then stuck open

TABLE 3. IMPORTANT PHENOMENA TO BE CONSIDERED WITH RESPECT TO ACCIDENTS

Category of important phenomena	Power excursion accident (RIA)	PCMA	LOCA
Heat conduction	• Significant increase of gap conductance after fuel–cladding contact • Rapid cooling due to rewetting	• Significant degradation of heat transfer after DNB • Rapid cooling due to rewetting	• Very low heat transfer after blowdown • Onset of DNB • Sudden cool-down due to quench
Fission product behaviour	Increase of fission gas release rate with increase of fuel temperature	Increase of fission gas release rate with increase of fuel temperature	Release of fission products stored in the gap
Nuclear heat generation	• Sharp peaking of heat generation in a short period • Steep local power peaking		
Cladding behaviour	Cladding swelling, burst or creepdown depending on pressure difference across cladding and cladding temperature	Cladding swelling, burst or creepdown depending on pressure difference across cladding and claddingtemperature	Cladding swelling or burst depending on pressure difference across cladding and cladding temperature
Material interaction	• PCMI • Pellet–cladding channel interaction	• PCMI • Pellet–cladding channel interaction	• Metal–water reaction • Absorption of hydrogen into cladding
Thermal hydraulic effect		Flow blockage caused by multi-rod swelling or burst	Flow blockage caused by multi-rod swelling or burst

4. IMPORTANT PHENOMENA

4.1. Normal operating conditions and anticipated transients

For normal operating conditions and anticipated transients, for which fuel rod integrity has to be ensured, the mechanisms liable to cause loss of rod integrity include at least:

- Clad overheating,
- Fuel melting,
- Pellet–cladding interaction,
- Stresses and strains,
- Fatigue,
- Circumferential clad buckling,
- Clad burst,
- Fretting wear,
- Internal pressure,
- In-reactor rod bow,
- Irradiation growth,
- Corrosion (including cladding embrittlement due to oxidation),
- Hydriding,
- Stress corrosion.

Proof of proper fuel behaviour with regard to these phenomena shall be provided. Therefore, the analytical studies demonstrating the fuel design adequacy shall be based on widely accepted engineering methods or on computer programs based on physical models and experimental tests.

In particular, the models for evaluating fuel assembly behaviour shall cover, at least for normal operating conditions, the following phenomena:

4.1.1. Rods

- Temperature distribution inside the cladding,
- Heat transfer between pellet and cladding,
- Temperature distribution in the fuel,
- Fuel swelling and densification,
- Fuel restructuring,
- Fission gas release,
- Irradiation induced clad creepdown and elongation,

- Cladding stresses and strains,
- Pellet–cladding interaction,
- Cladding waterside corrosion.

4.1.2. Structure

- Holddown system mechanical strength under the effects of irradiation and pump overspeed,
- Mechanical strength of the assembly components and their interfaces,
- Rod holddown characteristics,
- In-reactor deformation (rod bow, assembly deformation),
- Axial and lateral compression behaviour,
- Vibration response,
- Impact characteristics.

The major phenomena from this list, involved as well during accident conditions, are described hereafter.

4.2. Accident conditions

Important phenomena involved in the accident transients strongly depend on the kinetics of the accident. Indeed, for very fast transients like RIAs, which are almost independent of plant performance, calculation should be focused on the fuel behaviour itself (for example, a sudden temperature rise in the fuel pellet may cause pellet–cladding mechanical and chemical interactions, and excess fuel internal pressure at high burnup fuel might cause cladding deformation). These phenomena would lead to a challenge to the fuel integrity. For other transients with slower kinetics than RIAs (PCMAs and LOCAs), the number and type of phenomena involved increase and might be more complex. For PCMAs, the increase of the cladding temperature may lead to cladding deformation and cladding oxidation. For LOCAs, phenomena such as cladding burst, cladding oxidation associated with the burst, and cladding hydrogen absorption associated with the hydrogen release resulting from cladding oxidation, may need to be considered, as well as fuel relocation.

Major phenomena to be taken into account in the accident analyses of RIAs, PCMAs and LOCAs are shown below and summarized in Table 3.

4.2.1. Heat conduction

The fuel behaviour during an accident is basically determined by solving the heat conduction equation. Parameters to be particularly taken into account when solving the heat conduction equation are as follows:

- *Thermal conductivity of the fuel element:* The thermal conductivity of the fuel element varies with material, density, burnup, etc. When pellet cracking occurs as the temperature during the transient increases, the effective thermal conductivity may change. The margin to fuel centreline melt will be decreased with increasing burnup, because the thermal conductivity of fuel is decreased and the fuel melting point is lowered. Fuel centreline melt may result in a detriment of fuel integrity, caused by volume expansion and fission gas release.

- *Power distribution of the fuel element*: The power distribution of the fuel element varies with material, density, burnup, etc. It is particularly important for rapid transients.

- *Gap conductance between fuel element and cladding*: The gap conductance significantly differs depending on the closure or opening of the gap between pellet and cladding and on whether the gap has been poisoned by fission gas release (see Section 4.2.2). In the very early stage of a RIA, the magnitude of the heat flow through the gap may affect the fuel behaviour during the accident. If the gap is open at the initial state, heat generated is accumulated in the pellet. This will cause a delay of occurrence of DNB. After closing of the gap due to thermal expansion of the fuel pellet, a large amount of heat accumulated in the pellet will be transferred to the cladding in a very short time. This may result in a cladding temperature that is higher than in the initially closed-gap case. This points out the importance of the initial state, depending on the objectives of the calculations to be performed (which criterion to verify).

- *Thermal resistance of the cladding:* This depends on the composition and thickness of the cladding, as well as on the oxidation layer thickness and crud buildup on the cladding surface. When fuel rod burst is anticipated to occur during the transient, the effect of inner cladding oxidation should be taken into account.

- *The heat transfer coefficient on the cladding surface*: This is mostly governed by the heat transfer regime. In normal operation, heat is transferred to the coolant by forced convection or nucleate boiling. In transients, with increasing heat flux, or with decreasing flow rate or pressure, boiling transition or DNB occurs and the heat transfer regime moves to transition boiling or film boiling. In the case of a large break LOCA, the core is firstly dried out, then cooled by water reflood from the bottom of core. The heat transfer during the reflood phase is so complicated that the heat transfer coefficient should be determined by precise thermohydraulic analysis determining heat transfer correlations on the basis of experimental data.

- *Quench or rewetting*: Sudden cooldown caused by quenching during a LOCA or by rewetting during a PCMA or RIA may cause a large thermal stress on the cladding, and this may lead to fuel failure.

4.2.2. Fission product behaviour

The effects of fission products on fuel behaviour during accidents are as follows:

- Thermal conductivity of the fuel element will be decreased;

- Melting temperature of the fuel element will be decreased;

- Gap conductance between fuel and cladding will be decreased, when fission gas is released to the gap.

In the course of the accident, due to the fuel temperature rise, some of the fission gas accumulated in the fuel matrix will be released to the gap and the internal pressure will be increased. On the other hand, since the coolant pressure might also be changed according to the transient of the primary systems (e.g. in a LOCA), the pressure difference across the cladding will be changed. Since the cladding temperature rise during the transient softens the cladding material, the change of pressure balance across the cladding may cause cladding creepdown or collapse when the outside pressure exceeds the internal pressure, or cladding swelling, ballooning or burst of the cladding, possibly associated to fuel relocation, when the inside pressure exceeds the outside pressure. These phenomena should be taken into account in the fuel behaviour analyses.

4.2.3. Nuclear heat generation

In cases of PCMAs and LOCAs, the nuclear heat generation reduces to a decay heat level just after control rod insertion. In the case of a RIA, the power of the fuel assembly where the control rod is ejected will be promptly increased. The magnitude of the power excursion depends on initial conditions such as power level, delayed neutron fraction and control rod worth. In addition, analysis conditions such as moderator temperature coefficient or Doppler coefficient depend on core and fuel specifications, operating conditions and fuel burnup.

4.2.4. Cladding behaviour

Phenomena to be expected to occur with the cladding as the cladding temperature during transients increases are as follows:

- At high temperatures, the cladding material softens; this may result in cladding deformation. The type of deformation depends on the cladding temperature and the pressure difference across the cladding. Cladding creepdown may cause a cladding–fuel reaction, which may lower the melting temperatures of the fuel and cladding. Cladding ballooning or geometrical distortions of fuel assemblies may degrade the long term coolability of the reactor core. In the case of a large pressure difference across the cladding, such as in a large break LOCA, cladding burst can occur.

- The cladding material reacts with steam at high temperatures. This reaction is an additional heat source to the cladding, and hydrogen produced by the reaction can be absorbed by the cladding and the mechanical properties of the cladding may be degraded.

4.2.5. Material interaction

Both mechanical and chemical interactions between cladding and fuel can be expected during transients, when fuel pellets touch the cladding. Since PCMI restricts the slip between pellets and cladding, axial strain of the cladding caused by PCMI may result in a cladding failure. At high temperatures, the pellet–cladding channel interaction can lower the melting points of fuel and cladding. This may cause an increase of fission gas release and degradation of the cladding.

4.2.6. Thermohydraulic effects

In general, system thermohydraulic behaviour such as core inlet temperature or core inlet flow rate are derived from system thermohydraulic analyses carried out prior to a simplified fuel behaviour analysis. Therefore, thermohydraulic effects to be taken into account in the fuel behaviour analysis are limited as follows:

- In the case where the power distribution in the hottest fuel assembly is relatively flat, most of the fuel rods will reach the fuel thermal criteria such as DNB at once. Therefore, once DNB occurs on one of these rods, this phenomenon can propagate to the surrounding fuel rods. Accordingly, the number of fuel rods which experience DNB could be increased.

- In the case of a high burnup fuel assembly, the internal pressure of most rods may exceed the system pressure during normal operation. When DNB occurs on these rods, cladding deformation such as swelling or burst may be caused. Flow blockages caused by cladding deformation may degrade the heat transfer performance downstream of the blockage.

- A similar effect of flow blockage could be observed during a large break LOCA. Since the cladding temperature of most rods in the hottest assembly should not be very different during the reflood phase of a large break LOCA, many fuel rods have a chance to reach the conditions of cladding deformation. Once one of these rods bursts, this may propagate to the surrounding fuel rods and may cause flow blockage. This may degrade the heat removal capability during a large break LOCA and impede the heat removal capabilities of the ECCS.

In general, fuel assemblies having the same structural configuration are loaded into the core in order to maintain neutronic, thermohydraulic and mechanical compatibility among fuel assemblies. In special cases, however, fuel assemblies having different structural configurations happen to be loaded into the core. These cases are, for example, installation of advanced fuel assemblies or installation of other vendors' fuel assemblies. Such core configurations are commonly called ' mixed cores'.

Since the thermohydraulic characteristics of these fuel assemblies are usually not the same, the flow distribution in the core will be affected. For example, the difference in the bottom nozzle structure of the fuel assembly will affect core inlet flow distribution, and the difference in the grid structure may cause flow redistribution at the grid position. In order to address mixed core effects precisely, initial and boundary conditions to fuel behaviour codes should be determined by calculations using system thermohydraulic codes where a multichannel model in the core is adopted. Subchannel codes may be used to investigate the thermohydraulic conditions in more detail.

5. ACCEPTANCE CRITERIA

5.1. General

Acceptance criteria (see Ref. [1]) are used to judge the acceptability of the results of safety analyses. They may set:

- Numerical limits on the values of predicted parameters;
- Conditions for plant states during and after accidents;
- Performance requirements on systems;
- Requirements on the need for, and the ability to credit, actions by the operator.

Acceptance criteria are most commonly applied to licensing calculations, both conservative and best estimate.

Basic (high level) acceptance criteria are usually defined as limits set by a regulatory body. They are aimed at achieving an adequate level of defence in depth.

Specific acceptance criteria, which may include additional margins, are often developed as well. These acceptance criteria are chosen to be sufficient but not necessarily to meet the basic acceptance criteria. Typically they are used to confirm that there are adequate safety margins beyond the authorized limits, to allow for uncertainties and to provide defence in depth. They may be developed by the designer and/or owner and approved by the regulatory body; or they may be set by the regulatory body itself. An example of the latter would be a limit on the cladding temperature during a LOCA in a PWR.

Example of basic acceptance criteria for DBAs are listed in Ref. [1] and can be summarized as follows:

(a) The dose to individuals and the public must be less than the values defined for that accident class by the regulatory body.

(b) An event must not generate a more serious plant condition without an additional independent failure.

(c) Systems necessary to mitigate the consequences of an accident must not be made ineffective because of conditions caused by the accident.

For fuel behaviour, some acceptance criteria are assessed in safety analyses; others may be the subject of specific design calculations. These acceptance criteria focused on fuel behaviour can be summarized as follows:

- For anticipated transients, the probability of failure of the fuel cladding, resulting from a heat transfer crisis or from some other cause, must be insignificant.
- For DBAs, the fuel damage must be limited for each type of accident, to ensure a coolable geometry. Energetic dispersal of fuel must be prevented in reactivity initiated accidents.

- In LOCAs with fuel uncovery and heat-up, coolable core geometry and structural integrity of the fuel rods upon cooling must be maintained.

Corresponding accident analysis needs to be continued to that point in time when the plant can be shown to have reached a safe and stable shutdown state, so that:

(a) Reactivity can be controlled normally, which means that the core is and remains subcritical.

(b) The core is in a coolable geometry and there is no further fuel failure.

(c) Heat is being removed by the appropriate heat removal systems.

(d) Releases of fission products from the containment have ceased, or an upper bound of further releases can be estimated.

A list of acceptance criteria ensuring the limits of safety and the design limits, and their detailed description, can be found in Refs [9, 15]. The cross-reference to these criteria is given in Ref. [8]. These criteria may not be applicable to all reactor types, and their implementation in different countries may also be slightly different. For example, the implementation of fuel safety criteria in NEA member countries has been surveyed by the OECD [16]. In addition, implementation in EU Member States has been reviewed [17] as part of a wider survey on the harmonization of nuclear safety criteria and requirements that took into account the situation worldwide. A companion report [11] surveyed the reactor types then in use in the EU and in the then candidate countries (2001), summariz ing their main technical details.

5.2. Selection of acceptance criteria

A selection of acceptance criteria for the analysis of transients and accidents is summarized below (design limits for normal operation in steady state conditions — e.g. oxidation or stress — were addressed in Section 4.1).

As noted above, these criteria may not be applicable to all reactor types and may be slightly different in each country.

5.2.1. Departure from nucleate boiling ratio (DNBR)

The value of this parameter is calculated by system thermohydraulic codes or by the reactor dynamic codes. Satisfaction of this criterion shows that there should be sufficient cooling of the core, and overheating of the fuel elements should not occur. The given criterion should be taken into account in the fuel-focused calculations, as the calculation of temperature of the cladding defines the fuel-focused boundary conditions.

5.2.2. Reactivity coefficients

The reactivity coefficients establish the relation between power produced in the fuel and the thermophysical parameters of the fuel and moderator. The value of a given parameter is calculated by the steady state reactor neutronics codes, it is not a parameter which is calculated in dynamic calculations. As a rule this parameter is used in the dynamic transient and accident calculations as input data.

5.2.3. Shutdown margin

The value of this parameter is usually calculated by steady state reactor neutronics codes. It is also checked by the system thermohydraulic codes or by the reactor dynamic codes; in both cases the codes should have a kinetics model. This parameter is important for all cases in which operation of the emergency protection system is supposed. It is taken into account in calculations in combination with the previous parameters and determines the dynamics of change of power in the reactor core.

5.2.4. Enrichment

This measure is an administrative restriction only, and is provided with the design. For the computing analyses this measure is not a criterion. The value of the enrichment is used in calculations only as input data.

5.2.5. Internal gas pressure

The value of this measure is calculated by the fuel behaviour codes. This parameter is used for estimation of the stress conditions of the fuel cladding and is mainly relevant in accidents with significant reduction of primary pressure (i.e. LOCAs). Nevertheless, non-reopening of the gap between pellet and cladding must be ensured during normal operation, whatever the final burnup reached. In some countries, the normal operation internal gas pressure must always remain less than the normal operation coolant pressure to ensure a small fuel–cladding gap.

5.2.6. Pellet – cladding mechanical interaction (PCMI)

This has no numerical value used in safety analysis, but should be taken into account for the stress analysis of cladding, as additional forces are imposed on the cladding by the expanding pellet, especially during a transient. The basic requirement is avoidance of damage to the cladding.

5.2.7. Fuel fragmentation

This parameter has no numerical values which are used in calculations. However, this phenomenon is determined by consideration of the pellet cross-section averaged maximum enthalpy. This parameter is used as a safety criterion for RIA safety analysis. Also, it can be checked/calculated by the system thermohydraulic codes or by the reactor dynamic codes; in both cases the codes should have a kinetics model.

5.2.8. No local fuel melting

The value of the maximum centreline pellet temperature, which should not reach the melting point, is used as the criterion for all transient and accident calculations and has most applicability to a power excursion accident, including RIAs. In some countries, no fuel melting is allowed in normal operation.

5.2.9. Non-LOCA runaway oxidation

This phenomenon limits significant increases of the cladding oxidation rate in a non-LOCA transient and takes place under supercritical heat transfer conditions. It can lead to cladding embritttlement and damage. This parameter has no numerical values which are used in calculations. As a rule, the peak cladding temperature is used for limiting this parameter.

The same criterion peak cladding temperature for LOCAs and non-LOCA accidents is used for WWERs.

5.2.10. LOCA PCT

This parameter is used as safety criterion for the LOCA safety analysis, and also for calculations of non-LOCA accidents. The value of this parameter is calculated by the system thermohydraulic codes or by the reactor dynamic codes, and is calculated by transient fuel behaviour codes.

5.2.11. LOCA oxidation

This parameter is used as safety criterion for the LOCA safety analysis. This criterion limits the oxidation thickness, to prevent embrittlement and damage of the cladding and loss of the geometry of the core. Usually, transient fuel behaviour codes are used to calculate this value, but the given parameter can be as well calculated by system thermohydraulic codes or by reactor dynamic codes, which include a fuel behaviour model.

5.2.12. LOCA hydrogen release

This parameter is used as safety criterion for the LOCA safety analysis. For both PWRs and BWRs, the LOCA limit on the amount of hydrogen generated from the chemical reaction between cladding and water/steam is generally 1% of the hypothetical amount that would be generated if all of the cladding in the core were oxidized.

5.2.13. LOCA long term cooling

This measure has no self-maintained numerical value and, actually, is the design requirement for the emergency core cooling system. The performance of this requirement is a checked peak cladding temperature criterion.

5.2.14. Blowdown/seismic loads

This parameter is taken into account by mechanical and hydraulic calculations.

5.2.15. Hold-down force

This criterion is defined to limit hydraulic vertical lift-off forces, in order to prevent a displacement (unseating) of the lower fuel assembly tie plate from the fuel support structure. This measure relates to the core structure analysis.

5.2.16. Criticality

This criterion is related to the procedures of fuel manufacturing, transportat and storage of fuel material and is not a subject of this report.

6. SELECTION AND USE OF COMPUTER CODES

The selection and use of computer codes to support the analysis of fuel behaviour during normal, transient and accident conditions will strongly depend on the objectives of the analysis. As shown in the examples provided in Annex II, in some cases a very large suite of codes may be required to support vendor design and licensing calculations. These codes may include specialized reactor physics codes, a system and a subchannel thermohydraulic code, and detailed fuel behaviour codes. These codes may include steady state fuel performance codes, which are needed to establish initial conditions for transient analysis, as well as transient fuel behaviour codes. In addition, there is an increasing trend to couple or integrate different types of codes in order to address complex phenomena and to provide a more complete picture of the system behaviour (e.g. bundle effect for high burnup LOCA calculations). Examples may include coupled system thermohydraulic/reactor physics codes or coupled system thermohydraulic/detailed fuel behaviour codes.

The selection process may include several important components: (i) selection of the type of analysis methods such as conservative or best estimate, (ii) identification of the type of analysis activities to be supported, such as audit calculations, and (iii) selection of the suite of codes, at minimum a set of steady state and transient fuel behaviour codes, to be used including such features as adequacy of the phenomena considered, modelling options, documentation, accuracy, and availability of user support and training.

The calculation of coolant behaviour is improved by simultaneous calculation of the coolant conditions, heat transfer regime and heat flux on the cladding surface. The possibility of an erroneous input of boundary conditions is reduced by a simultaneous calculation of coolant and fuel behaviour.

6.1. Selection of analysis methods

The selection of the appropriate analysis method may be determined by local regulatory requirements, the type of analysis required, and cost. However, the selection of the method will largely determine the type of computer codes or models that can be used for fuel rod behaviour analysis. Analysis methods can, as discussed in detail in Ref. [1], be broadly characterized as conservative and best estimate methods. Conservative methods may be adopted to ensure that the actual calculated response in relation to a selected criterion is bounded by a conservative value for that response. For example, a correlation that is known to overpredict the rate and extent of cladding oxidation may be selected to predict the oxidation of a fuel element during an accident. This result is then compared to the regulatory limits of equivalent cladding reacted. The best estimate methods are adopted to reduce unnecessary conservatisms in the calculated results. These methods have the added benefit that the margins between calculated and actual response may be estimated using uncertainty analysis techniques. In the case where the selected criterion was equivalent cladding reacted, the best estimate method might use a different oxidation correlation that was developed to predict the average oxidation rate. The result, in combination with quantifiable uncertainty bounds for the oxidation rate, would then be compared to the equivalent cladding reacted.

A combination of best estimate and conservative methods may also be selected to minimize cost and adjust the approach for the specific type of analysis required. For example, a BE code might be the best choice for training while a conservative code may be the least expensive to qualify for licensing calculations. Fortunately, many fuel behaviour codes offer both conservative and best estimate modelling options, although the trend is to focus on best estimate models. These options are typically selected through input, so the analysts can adjust the method used for different applications. A combination of approaches is also possible. For example, best estimate models may be used with conservative input to provide some degree of conservatism in the results.

The best estimate codes that allow some variation in the input for important models may also offer some advantages for applications to fuel designs or conditions that are still active research areas. For example, as discussed in Section 8, the operation of fuel rods to extended burnup levels is an active area of research. Thus, it is often necessary to adjust the input of important models to account for the uncertainties inherent in the results of ongoing research programmes. Although many of these codes are being adapted to incorporate the results of such research activities, these modified versions of the codes may not be widely available. In the interim, the use of the older, more widely available code versions may require the adjustment of their input as well as assessment of the accuracy of the codes using the available research results.

6.2. Identification of analysis activities to be supported

The type of accident analysis that is required will also be an important factor in the selection of the appropriate code or codes to be used. The IAEA’s general guidance for accident analysis [1] identifies several types of analysis that are relevant to fuel behaviour safety applications. These include design analysis, licensing analysis and regulatory audit analysis. Training and the support of ongoing research activities may be other important applications, although not directly related to safety analysis.

Design analysis is normally performed at the earlier stages of the design and has historically employed more conservative approaches to insure that there are adequate margins to satisfy any licensing requirement that might apply. However, best estimate codes or models can also be used to support design calculations. In the case of fuel design, the best estimate models allow the designers to establish the most likely performance of their fuel under a wide range of conditions.

Licensing and regulatory audit analysis may require more formalized approaches. Depending on the local licensing conditions, the use of conservative codes or models may be required. However, there is a trend to allow best estimate models, in combination with a formal uncertainty analysis, as an option for licensing calculations. For fuel rod analysis, this trend has resulted in the removal of conservative modelling options from some fuel rod behaviour codes. For example, the US Nuclear Regulatory Commission has removed conservative modelling options from their transient fuel behaviour code to be used in regulatory audit calculations by the USNRC [12].

The best estimate codes are typically selected to support training and research activities. In the case of fuel behaviour codes, BE codes describe the expected response of the fuel rods. Conservative codes or models very likely will provide distorted responses. For example, when conservative codes are used, the calculated temperatures or oxidation rates may be significantly overpredicted because of the use of conservative heat transfer or oxidation correlations.

As a result, the selection of conservative codes, or codes with the option for conservative models, may be necessary if the codes are to be used only for licensing or regulatory audit applications. However, if the codes are to be used for a variety of other applications as well, or if the local licensing regulations allow the use of best estimate codes associated to an uncertainties analysis, BE codes may be acceptable.

6.3. Selection of the suite of codes to be used

For convenience, the general types of computer codes that may be used to support fuel behaviour calculations may consist of the following four general types: (1) fuel behaviour, (2) reactor physics, (3) thermohydraulic and (4) structural analysis. In this context, fuel behaviour codes describe the behaviour of individual fuel rods in normal and DBA conditions, tend to be design specific, and may contain modelling options for both conservative and best estimate calculations. Reactor physics codes model the core neutronics in normal and accident conditions and, now, may more commonly describe the core in either two or three dimensions. Thermohydraulic codes are a very general class of codes and may be used to model the reactor coolant system, the core, individual fuel assemblies or special regions, using computational fluid dynamics methodologies. The structural analysis codes may be standard commercially available codes that have been adapted for nuclear applications through incorporation of the appropriate thermal and mechanical properties. These codes may be used to describe the structural response of fuel assemblies and other vessel structures during earthquakes or hydrodynamic events such as water hammer.

These codes may also be subdivided into steady state and transient codes. The steady state codes define the state of the core during normal operation and are important to establish the proper initial and boundary conditions for transient analysis. The transient codes describe the response of the fuel and system during the accident or transient.

Although the process may be somewhat iterative, the normal order of the calculations would be to perform a steady state calculation for a given reactor state or a set of reactor states. For example, steady state reactor physics calculations and core thermohydraulic calculations may be performed for several different stages of the reactor operating cycle to determine steady power distributions in the core. This information may then be used in a steady state system thermohydraulic calculation to determine the overall response of the plant at steady state or normal operating conditions. This information might also be used for a series of steady state fuel behaviour calculations to determine fuel temperatures, fission product distribution within the fuel, the buildup of ’crud‘ or corrosion products on cladding, fuel rod internal pressure, etc. The results of the fuel temperatures, reactor coolant system temperatures and other thermohydraulic conditions might then be used to adjust the parameters in the reactor physics calculations, and the process is then repeated until the conditions converge.

The results of these steady state calculations might then be used in the design process to improve the fuel design or operating conditions. For example, if fuel temperatures are higher or internal pressurization of the fuel rods is higher than desirable, the fuel assembly design may be altered appropriately. The results of the steady state calculations are also used in the transient calculations that are provided to establish the performance of the reactor and fuel during DBA conditions. The steady state calculations would be used as initial conditions for transient reactor physics calculations, core and system thermohydraulic calculations and fuel behaviour calculations. For example, the state of the fuel rods at the start of the transient, such as fill gas inventory, fission product inventory with the fuel, fuel–cladding gap, and other physical characteristics of the fuel rods, would then be used to calculate the performance of the fuel rods during the accident. In addition, the results from the transient reactor physics calculations would be used to determine the power distribution in the fuel during the transient, while the system thermohydraulic calculations would provide the thermohydraulic boundary conditions for the fuel rod calculations. These results would then be reviewed, and the process may be repeated using the transient fuel rod temperatures in the transient reactor physics codes, etc.

The results of both the steady state and transient calculations may also be used to support separate structural calculations and to determine the margins between the actual response and the relevant steady state and transient safety criteria as shown in Fig. 10. The reactor core structure analysis shown in the figure may, for example, look at the potential twisting and bowing of the fuel assembly during operation (due to creep, irradiation growth, power and temperature gradients).

6.4. Necessary code features

As described in Ref. [1], three general criteria can be used as an initial guide to judge the adequacy of the reactor analysis codes, including those used specifically to support fuel behaviour analysis. Firstly, the use of internationally recognized and accepted codes provides some assurance that the codes are adequate for their intended application. Secondly, for DBA conditions, the important phenomena to be considered have been established internationally and may be documented on an individual code basis. Thirdly, individual codes need to be evaluated on a systematic basis, comparing the intended application of the code with the actual conditions for which the code is being applied.

In the case of steady state and transient fuel behaviour codes, the general selection process is relatively straightforward. Firstly, steady state and transient fuel behaviour codes are quite mature, with many of the internationally recognized codes listed in Ref. [1] being readily available to the general user community. (Vendor developed codes may be less widely available for the general user community but are typically available for vendor applications.) Secondly, many, if not all, of these codes have validated models that address the important phenomena described in Section 4. Thirdly, since the modelling approaches used in these codes are, in many cases, quite similar, the systematic evaluations can be simplified. For example, the application of the codes to different reactor designs can focus on the selection of appropriate material properties, library options rather than appropriate models or correlations. Nevertheless, special attention must be paid to the applicability of these codes to different fuels or reactor designs. In some cases, systematic evaluations have been performed for general reactor designs.

The selection of the other supporting codes, such as reactor physics codes or thermohydraulics codes, may be somewhat more complicated since there are more types of codes available with more modelling options like 2-D/3-D, coupled reactor physics/system thermohydraulics, etc. However, there are also many internationally recognized codes that are generally available to the general user. An example is the use of computational fluid dynamics codes to calculate natural circulation flows in-vessel, in the hot leg and in the lower plenum of the steam generators of PWRs, then using the output to help define noding schemes, cross-flow resistances, etc., for use in 1-D reactor pressure vessel and primary circuit models that are commonly found in system-level codes. Another example is the use of output from a system-level code to generate input for a detailed-level code where the system-level code’s modelling is not adequate for the purpose in mind.

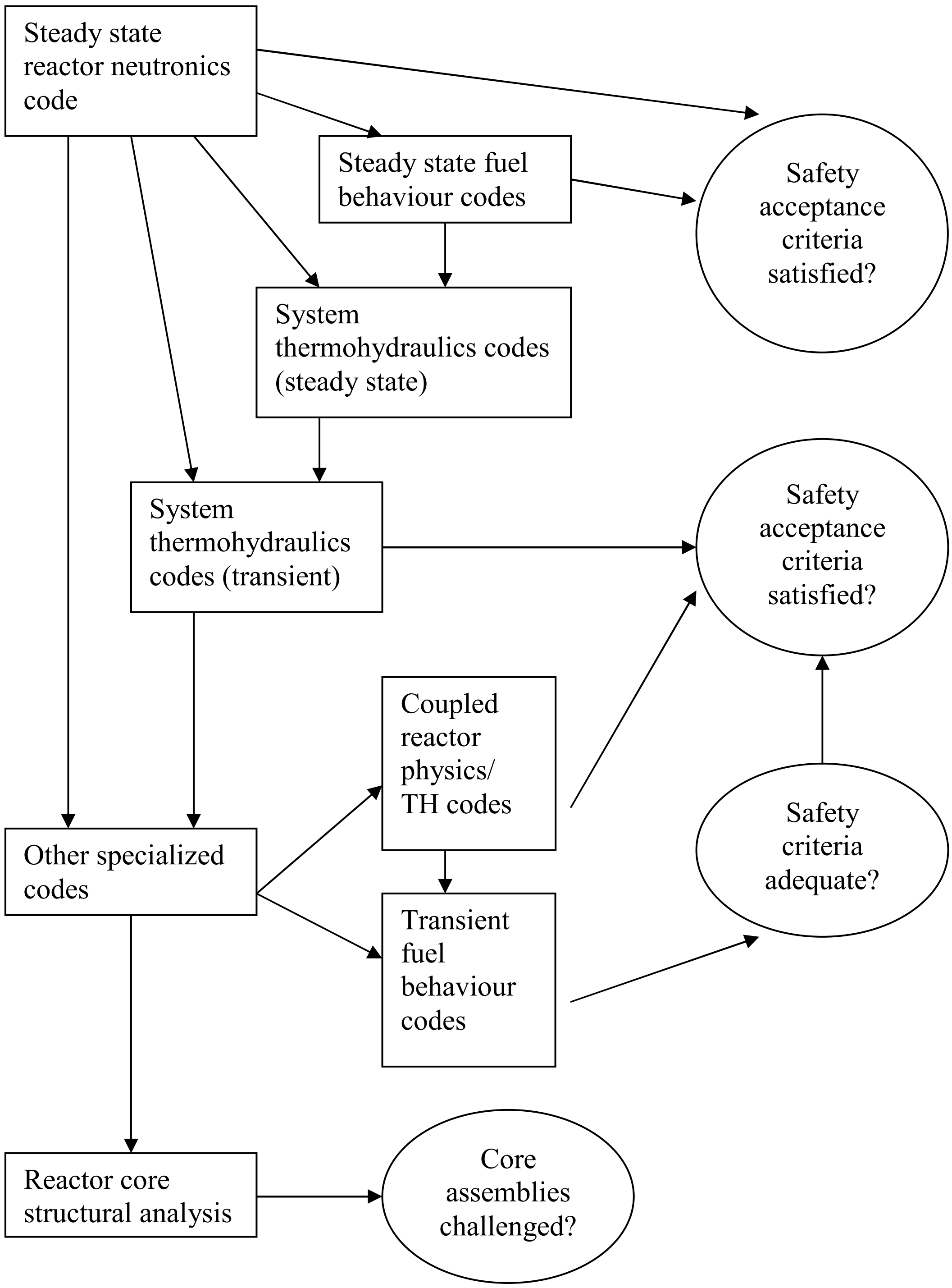


FIG. 10. Use of steady state and transient codes to determine safety margins.

The intended application of the codes is also a very important factor in the selection process, particularly for applications associated with licensing of the reactor fuel. For licensing applications, regulatory requirements may define specific features required for the code to be considered adequate for these applications. In some cases, these requirements may specifically identify what type of models and correlations must be used in the codes, particularly for codes used for conservative licensing approaches. In other cases, the requirements may provide some leeway for the selection of specific models but instead define in general terms how the uncertainties in these models must be determined in order for the code to be used. The latter approach typically applies to BE codes and models. Thus, the determination of what modelling features are necessary for codes used to support fuel behaviour licensing may be straightforward since they are defined by regulation. In this case, the analyst may have to select from a more limited set of codes with the regulatory mandated modelling features.

Therefore, the final selection process for licensing applications may be based on other criteria such as the general availability of candidate codes, the cost of qualifying these codes for licensing applications, availability of user support and training, and the longer term maintainability of the codes. (Since the cost of qualifying codes for licensing may greatly exceed the cost of acquiring the codes, the ability to maintain and use the software for a long period of time will determine the overall cost of using the codes.)

For other applications that may be not subject to direct regulatory requirements, the analyst can select the appropriate codes based on a more balanced emphasis on modelling features, initial cost of the software, ease of use of the software, the potential for future code improvements, and availability of user support and training. In this case, the ability to upgrade the codes to take advantage of improved computer technology and better and more powerful modelling and user options may be a dominant factor in the selection process.

Since there is potentially a large number of codes that could be required to support fuel behaviour analysis, the ease of use of the codes, and their ability to pass data between the different codes, may be a very important factor in the overall adequacy of the codes. As shown in the examples provided in Annex II, the general trend to merge the general capabilities of these codes will have a beneficial impact on fuel behaviour calculations.

The ability of codes to link reactor physics and systems thermohydraulics models together will greatly reduce the time and effort needed to establish realistic steady state conditions for the core and fuel as well as reduce the time to perform transient calculations where the reactor power and core thermohydraulics are tightly coupled.

The incorporation of detailed fuel rod models into system thermohydraulic codes also has a number of advantages for the analyst. Firstly, the accuracy of the fuel behaviour calculations is improved by accounting for the thermal and mechanical interactions within a fuel assembly. For example, the impact of radiation exchange within the fuel assemblies between hot fuel rods and colder structures such as water tubes or control rods can be considered. The impact of rod-to-rod contact on fuel rod deformation is another example. Secondly, the accuracy of the core thermohydraulic calculations is improved because the changes in the fuel assembly flow area and volume due to cladding deformation can be considered.

7. PRACTICAL APPLICATION OF FUEL BEHAVIOUR CODES UNDER ACCIDENT CONDITIONS

The analysis of fuel element behaviour under accident conditions is performed using detailed fuel behaviour codes. As discussed in previous sections and the examples shown in Annex II, a variety of other codes may also be used to support the fuel behaviour analysis. These codes provide thermohydraulic and neutronic initial and boundary conditions. As shown in Fig. 11, the calculations may be performed in stages as described in the following example. This figure also shows that there is a tremendous amount of input data that may be required to support each of the codes and their calculations.

One of the most comprehensive sets of fuel behaviour calculations may be performed to support the design and/or licensing activities associated with the installation of new reactor units or the use of new fuel for existing plants. For example, for the installation of a new unit, design calculations precede the performance of calculations by more specialized codes. At the first stage, neutronics and thermohydraulic calculations of steady state conditions are carried out. At this stage, limiting values of the basic characteristics that are used at the following stages as initial conditions are determined. At the second stage, calculations for all initiating events as described in Section 3 are carried out. On the basis of the analysis of results, the most conservative initial events and scenarios from the point of view of influence on a fuel element can be determined. At the third stage, detailed fuel element behaviour calculations under these selected initiating events are carried out.

In an example for the installation of new fuel in an operating nuclear power plant, the same calculations, as a rule, are carried out as in the design calculations. However, some simplifications are possible. For example, for the application of new fuel, the neutronic characteristics of the core will be given in design limits, and the hydraulics of a reactor and a core will not change, so the first stage may not be necessary. At the second stage, the limiting design data are used as initial conditions.

7.1. Collection of input data

As described in Ref. [1], the first important step in developing the input data is to collect the necessary documentation and other sources of reliable data. These data may come from a variety of sources as listed in Ref. [1]. In the case of detailed fuel behaviour calculations, an important source of information may be the fuel vendor. These data can then be collected in a formal database containing all of the pertinent information. In some cases, the basic input data must be converted into a form that can be used as input for a specific code. The conversion process should be documented as well, along with other specific information such as the nodalization scheme to be used. This information should be documented in an engineering handbook as well as in the input deck, to the extent possible.

The sources that serve as a basis for data collection may be as follows:

- Documentation on plant design;
- Technical specifications of equipment;
- Documentation gathered during the startup and commissioning of the installation;
- Operational documentation for the plant (limits and conditions, operating instructions and records of operational regimes);
- ‘As built’ plant documentation.

7.2. Important input parameters

Specific input parameters for each code are unique to the individual codes and should be described in the user guidelines and detailed input manuals provided for each code. However, Fig. 8-1 provides a good example of the general types of input that may needed at each stage of the calculations. This figure also shows that the input required at each stage may also come from the results of earlier calculations. In general terms, the following general information is needed:

(1) Input data relevant to core characteristics (reactor physics, thermohydraulic and steady state fuel behaviour calculations to determine initial and boundary conditions)

- Core geometry, fuel assembly geometry, fuel rod specifications;
- Core power, primary coolant flow rate, pressure, core inlet temperature;
- Fuel state at the start, including fuel burnup, power distribution in the fuel element and power history, gap gas content and pressure, gap conductance or gap width, thickness of cladding oxide, etc.).

(2) Input data relevant to transients (output from the reactor physics and thermohydraulic calculations)

- Power transient data;
- Transient data on coolant flow rate, pressure, inlet temperature, inlet quality, etc.

7.3. Construction of the fuel rod input

In those cases when the interactions between fuel rods and other elements of the fuel assembly are small, the use of a single representative fuel rod may be sufficient. Since heat conduction in the longitudinal and circumferential directions is generally negligible, except possibly in the vicinity of the quench front during the reflood state of a large break LOCA, a stacked one dimensional radial model of the rod may be used. In other cases, a three dimensional model may be employed. However, in both cases, nodalization studies should be performed to determine the appropriate nodalization. For example, the number of axial nodes should be decided, to be able to take into account the effect of axial power distribution and distribution of axial coolant conditions. The number of lateral (radial) nodes should be determined to be able to take into account the effect of the time constant of the fuel element as well as the influence of the temperature gradients in the fuel and gap. The nodalization may also depend on the type of transients to be analysed. RIA transients which deposit energy near the surface of the fuel may require different radial nodalization than that for LOCAs or PCMAs.

In the case where the interactions between the elements in the fuel assembly cannot be neglected, it may be necessary to describe additional representative fuel rods and other elements in the fuel assembly. As discussed in Annex II, this may be necessary where the coolability of the bundle due to fuel rod ballooning may be in question.

7.4. Selection of calculation model

A verified set of calculation models should be selected. For example, calculation of the fuel temperature is usually assessed by the experimental data from fuel centreline temperature measurements. This means a set of fuel temperature calculation models in which fuel thermal conductivity model, gap conductance model and heat transfer model on the cladding surface are included, should be used in the calculation.

7.5. Verification of input data

When the new input deck is being developed, errors could be introduced by a developer at any stage of the development process from preparation of the engineering handbook to preparation of the final input deck. Since these errors could result in unacceptable faults in the analysis, their early detection and correction are important.

Verification of the input deck is needed to check its formal correctness, that is, that no erroneous data were introduced into the input deck and that all formal and functional requirements are fulfilled accurately and therefore will permit successful use of the input deck.

Current practice in accident analysis is to apply, in a systematic manner, the complete verification process. The verification process provides the confidence that the modelling needs have been met. Verification of input data is the process of reviewing and cross-checking the input deck and confirming that no mistakes were made and that the input deck is ready for application. An effective way to avoid possible subjective errors in the development of the code input deck is the application of code specific pre-processing software, if available.

The verification of the input deck needs to be performed and documented by qualified individuals or groups who have not been involved in the development of the input data. The reviewers can be from the same organization or from a different organization. They need to have access to all relevant documentation. All errors that were detected and corrections that were made in the verification process need to be properly documented.

7.6. Validation of input data

Validation is performed after the verified input deck is completed and before the accident analysis is started. The purpose of validating input data is to demonstrate that the model adequately represents the functions of the modelled systems. Experience gained in the validation of the computer code and from the analysis of similar problems would be utilized in such a validation.

Validation of input data is an iterative process by means of which the correctness and adequacy of the plant models are confirmed so as to provide a good representation of the behaviour of the plant systems. The validation needs to assess whether the key performance parameters behave in correspondence with reality. The validation would include, but not be limited to, the following:

- Checking the spatial and temporal convergence of the calculation, such as by performing sensitivity analysis in relation to changes in nodalization for a typical case of analysis under consideration, and by changing/refining the timestep history in the transient analysis.

- Checking the energy and mass balances in the systems modelled, including long term system energy and mass balance; this can be done by: comparing power generation in heat structures with surface heat flux, comparing power generation in individual components with corresponding enthalpy rise, comparing evaporation rate with surface heat flux, comparing changes in mass inventories with the difference between injection and leak rates, and checking consistency of the flows in adjacent junctions.
- Checking the behaviour and response of individual components of equipment or of separate systems through determination of respective boundary conditions.
- Checking the steady state conditions for different operational states, preferably by comparison with real plant data.
- Comparing fluid volume and pressure distribution of the model with the height and pressure drop of the real installation.
- Performing a comparison between nuclear power plant behaviour predicted by calculations with relevant data from measurements in integral test facilities.
- Checking the computational results against real plant data from operational events.
- In relation to each of the aforementioned items, quantitative acceptance criteria for the code input deck could be available or established.

The plant data collected during commissioning and startup tests, conducted under well controlled conditions and with additional instrumentation, are very useful and need to be applied for validation of the input data. However, in some cases, such data may differ from data obtained during plant operation. Consideration needs to be given to such differences where applicable.

For the validation process it is advisable to use tools for graphical display of the nodalization and animation of the plant states.

7.7. Actual procedure of transient fuel behaviour analysis

Since the accident analyses, especially transient fuel behaviour analyses, need almost all data relevant to reactor and core, the analyses should be performed at the final phase of a sequence of the plant design process. As shown in Fig. 11, reactor and core data must be gathered from documentation of the plant design and technical specifications of equipment. These data are used to perform at first nuclear design and thermohydraulic design. Fuel rod data are created by performing the fuel rod design. In fuel rod design, fuel rod specifications such as cladding and fuel pellet diameters and rod internal pressures have to meet the fuel rod design criteria. Based on the results of fuel rod design analyses, the fuel rod and the time (burnup) of fuel behaviour analysis performed are determined in order to give the most conservative results.

The plant transient analyses are performed using the plant data, and the results of nuclear and thermohydraulic design.

The transient fuel behaviour analysis is performed following these design analyses and safety analyses. Fuel rod geometry, initial conditions, boundary conditions and transient plant performance are given as input parameters. In the code, a stacked one dimensional radial heat conduction equation is solved where the effects of variation of heat transfer mode, cladding oxidation, cladding deformation and burst, fission gas release, etc., are taken into account.

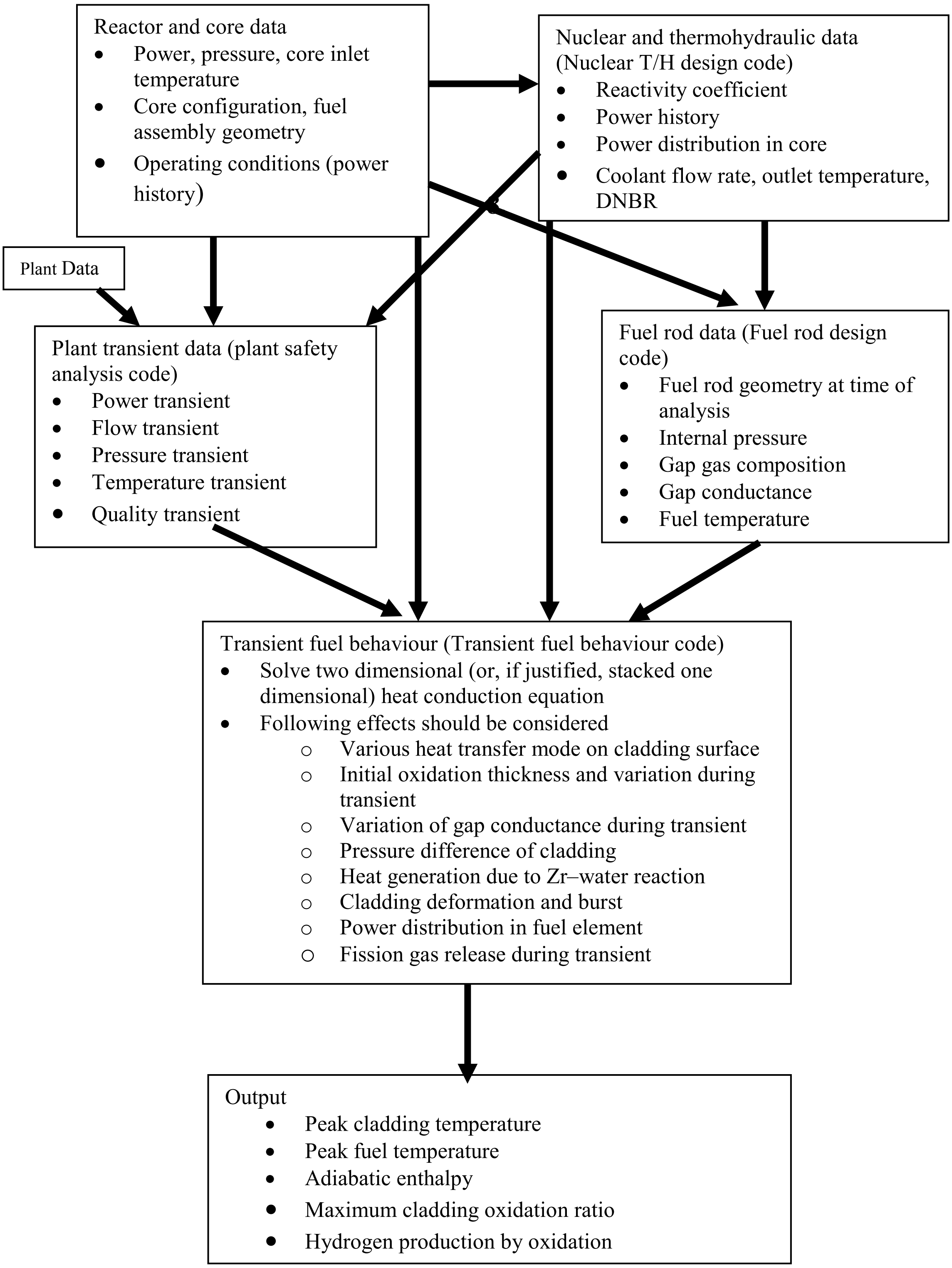


FIG. 11. Transient fuel behaviour analysis.

8. CURRENT ISSUES AND TEST PROGRAMMES

8.1. Current issues

The database, fuel behaviour models and computer codes used to support the analysis of fuel behaviour during accident conditions are relatively mature. The current issues for DBA related experimental test programmes and fuel behaviour model development activities are largely related to three general trends. The first important trend for many current reactor and fuel designs is to increase fuel burnup levels. As discussed in the following section, this is probably the most pressing issue from an experimental point of view. The second important trend is the increasing utilization of MOX fuels. The third, and more long range, trend is the development of more advanced reactor and fuel designs. Such trends require the development of an extended database as well as the development of additional fuel behaviour models (primarily the extension of material property models and correlations) capable of analysing high burnup in current fuels and analysing other types of fuels.

The development of more reliable and more ’user friendly‘ computer codes is also an activity that is currently receiving much attention. This activity does not require additional data or models but takes advantage of the tremendous improvements in computer technology and the use of more advanced programming and numerical techniques. Coupling the codes and models together in a more effective way is one area where more reliable and more ’user friendly‘ computing is helping the fuel behaviour analyst. For example, coupling 3-D reactor kinetics, system thermohydraulics and detailed fuel behaviour models is becoming much more common, as shown in the examples included in Annex II. As a result, the analyst has to become expert in fewer codes and can perform more accurate calculations more quickly.

As a further result, a transfer by the analyst of boundary conditions from one code to another should be developed and qualified in order to reduce the possibility of errors.

8.2. Highlights of fuel behaviour research and associated test programmes

8.2.1. Introduction of safety research programmes

The basic role of fuel behaviour experimental research programmes for accident conditions is to ensure the adequacy of fuel behaviour safety acceptance criteria, to determine the margins between the actual performance of the fuel and its associated safety criteria, and to help develop and validate fuel behaviour codes.

As shown in Fig. 12, there are two basic ways of ensuring that safety acceptance criteria are satisfied during an accident. Firstly, since many of the experiments use prototypical fuel elements, these experiments can be used to establish the actual performance of the fuel under conditions such as DNB, to determine the maximum fuel enthalpy during RIAs, and to investigate other design limiting conditions. Secondly, validated codes on analytical tests can be used to predict the performance of the fuel under conditions not explicitly addressed by integral experiments. For example, validated codes can predict local fuel melting during RIAs, peak cladding temperatures during LOCAs, internal gas pressure, fission product release, etc.

The fuel behaviour experiments are needed to assess material property models.

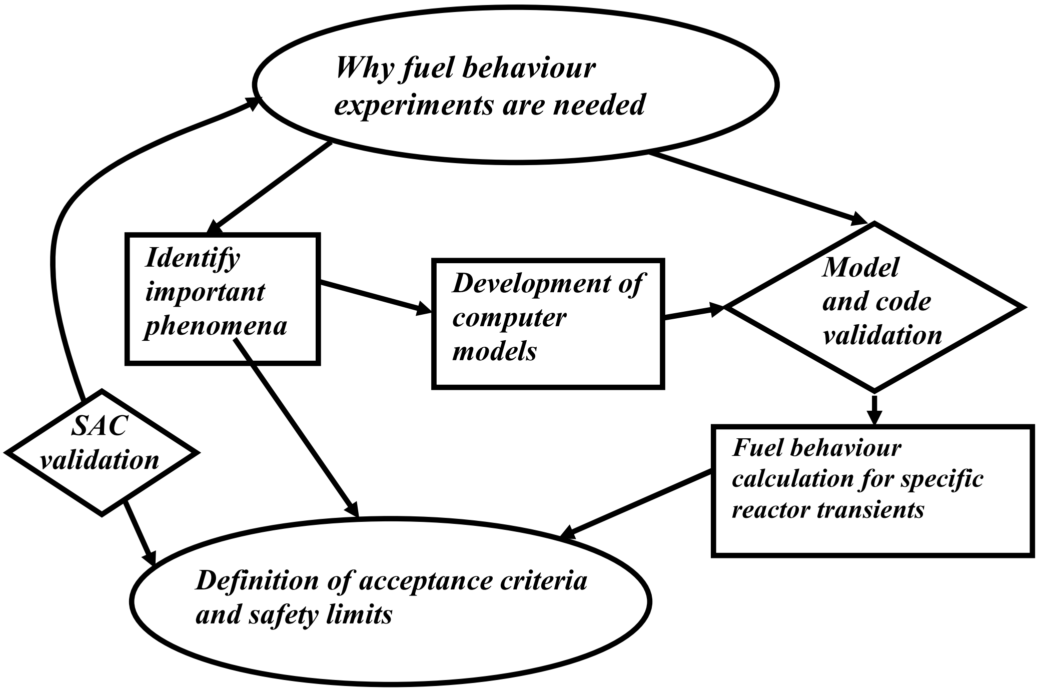


FIG. 12. Use of fuel behaviour experiments to define acceptance criteria and safety limits.

The specialized experimental installations can be divided into three basic types, covering the following phenomena, respectively:

- Thermophysics of a coolant stream near to a surface of a fuel element;
- Thermomechanical and physicochemical processes in a fuel cladding;
- Thermomechanical and physicochemical processes in a fuel pellet and their influences on its cladding.

8.2.2. Highlights of experimental programmes

Experimental programmes have been conducted worldwide since the early 1960s to evaluate fuel behaviour under transient and accident conditions. The fuel behaviour depends on the temperature, system pressure and other factors which vary widely according to the type of transient and accident considered.

The general trend to increase fuel burnup levels is likely to continue in the near future in many countries. In the 1970s, high burnup was considered to occur at around 40 GW·d/t. Data out to that burnup had been included in databases for criteria, analysis codes and regulatory decisions. However, by the mid 1980s, a unique pellet microstructure, the so-called rim structure, had been observed in high burnup fuel, along with an increase in the rate of cladding waterside corrosion. It thus became clear that other phenomena influencing fuel integrity could occur at high burnup and that extrapolation of data from a low burnup database was not appropriate. High burnup fuel behaviour under steady state operation has been widely studied by many fuel irradiation programmes, and the database is now adequate for safety assessment to burnup levels of about 70 GW·d/t (peak pellet). However, the database on transient fuel behaviour of high burnup fuel is limited and is based on rather old fuel rod designs. Thus the current issue of fuel behaviour research is to acquire and develop the requisite understanding of the performance of high burnup fuel under abnormal transient and accident conditions.

Several experimental programmes are currently under way to generate data on the behaviour of high burnup fuel under transient conditions representative of RIAs and LOCAs. Such programmes include the RIA simulation experiments performed at the CABRI facility in France and the NSRR in Japan. These programmes aim to provide data that can be used to develop safety criteria for extended burnup level applications and to validate analytical codes for high burnup UO_2 and MOX fuel behaviour.

Here, an overview of fuel behaviour research relevant to the following accidents is summarized.

8.2.3. Power cooling mismatch (PCM)

In-pile PCM experiments originated in the early 1960s with the General Electric Test Reactor (GETR) tests in the USA for BWR fuel development [18] and tests at the National Research Experimental (NRX) reactor in Canada to investigate BWR fuel behaviour in channel blockage accidents [19]. In both tests, fuel failure occurred due to lack of cooling. These tests were followed by many experiments concerning post-DNBR fuel behaviour, including low flow tests with a fuel bundle of 36 rods at the Steam Generating Heavy Water Reactor (SGHWR) in the UK, dryout tests by GE [20] and tests in the OECD Halden Reactor Project [21].

The most systematic PCM experiments had been conducted at the Power Burst Facility (PBF) in the USA since 1975 [22]. Dryout durations of 1– 15 min were applied to single or bundled PWR fuel rods of 90 cm length under PWR operational conditions. Unirradiated and irradiated fuel rods up to a burnup of around 16 GW·d/tU were tested, respectively, in PBF/PCM and PBF/IE test series. The experiments showed that cladding temperature reached approximately 350°C under forced convection or nucleate boiling conditions. Power increase or flow reduction caused DNB and unstable film boiling leading to cladding temperature oscillation. Further reduction of cooling produced stable film boiling, which resulted in a temperature increase at the fuel centre as well as at the cladding surface. The cladding surface temperature could reach 650– 1000°C at film boiling. The fuel temperature increase could enhance fission gas release and pellet thermal expansion leading to a rod internal pressure increase, but the high temperature caused cladding collapse rather than ballooning because of degradation of cladding strength and a high system pressure.

The fuel failure at power cooling mismatch was mainly caused by cladding embrittlement due to oxidation during film boiling and a mechanical load generated by thermal shrinkage at quenching. Cladding burst failure did not occur except in one case, with a fuel rod pre-pressurized to 85 kgf/cm^2 at room temperature. Cladding melting failure did not occur even in the most severe test, PBF/PCM-1, in which film boiling was sustained for 900 s at a linear heat rate of 790 W/cm. The fuel melted in the region of 85% in radius at the power peak elevation, but the cladding did not melt. The cladding failed due to oxidation at 520 s into film boiling, and fractured due to thermal shock at quenching at 920 s. In a burnup range up to 16 GW·d/t, no irradiation effect on fuel behaviour under PCM was observed, because any irradiation defect in cladding was quickly recovered at high temperature under film boiling [23].

Another series of PCM related experiments has recently been conducted at the OECD Halden Project. Repeated in-pile dryouts were performed with fresh/irradiated BWR and PWR fuel rods at the Halden Boiling Water Reactor (HBWR) [24]. Cladding examinations including tensile tests of ring specimens, Vickers microhardness measurements, scanning electron microscope and transmission electron microscope analyses, etc., were carried out to find dryout induced changes in cladding mechanical properties.

8.2.4. Power excursion accidents (RIAs)

8.2.4.1. SPERT and PBF tests

Parametric RIAsimulating experiments were conducted at the Special Power Excursion Reactor Test– Capsule Driver Core (SPERT–CDC) and the succeeding PBF in the USA in the early 1960s to 1980s [25]. The experiments provided data on fuel behaviour, fuel failure mode, failure threshold in terms of fuel enthalpy and mechanical generation at the failure, etc., for unirradiated and irradiated LWR fuel rods up to a burnup of 32.7 GW·d/tU. Unirradiated fuel rods failed due to cladding oxidation embrittlement or melting, while PCMI failure occurred on irradiated rods. These experiments provided the necessary input data for defining the RIA related fuel criteria for low irradiated fuel.

8.2.4.2. NSRR and ALPS tests

A great number of RIA experiments have been performed at the Nuclear Safety Research Reactor (NSRR) in Japan since 1975 [26]. In the first phase of the NSRR programme, unirradiated fuel rods were tested with variations of parameters such as peak fuel enthalpy, rod internal pressure, pellet–cladding gap, gap gas composition, coolant temperature, water logging level, etc. Most experiments were conducted on the NSRR standard test rod, which was a short 14 x 14 PWR type with a 135 mm fuel stack of 10%-enriched pellets, in stagnant coolant water of around 20°C and 0.1 MPa. These experiments provided enthalpy limits for DNB occurrence, fuel failure and fuel melting leading to energetic fuel coolant interaction. The results were utilized to establish Japanese safety guidelines giving tolerable enthalpy limits to avoid fuel failure and mechanical energy generation in a reactor vessel.

In the second phase of the NSRR programme starting in 1989, irradiated fuel rods were tested to assess burnup influence on fuel behaviour and integrity under RIA conditions. In test HBO-1 in 1994, PCMI failure occurred on a PWR rod which had been irradiated in a commercial reactor up to a burnup of 50 GW·d/t, at a fuel enthalpy increase of only 60 cal/g [27]. A large axial crack was generated on the cladding over the length. Cladding metallographs indicated brittle fracture due to hydrogen embrittlement, which could lower the enthalpy level for PCMI failure. Dispersal of fine fuel fragments occurred in the coolant water. Over 30 wt% of the fuel were fragmented to smaller than 50 μm. The fragmentation was probably caused by thermal expansion of accumulated fission products at fuel grain boundaries. Test TK-2 in 1997 showed that thermal interaction between fuel fragments and coolant water could cause explosive steam generation leading to water hammer even in the case that fuel did not melt. The NSRR programme is ongoing and is being extended to higher burnup UO_2 and MOX fuels, up to 79 GW·d/t, within the framework of the ALPS programme started in 2006. High temperature/pressure tests are planned to investigate temperature effects on cladding ductility, which determines the failure threshold.

8.2.4.3. CABRI tests

The CABRI REP Na test series has been conducted in France since 1993 [28]. High burnup PWR UO_2 and MOX rods up to 64 GW·d/t were pulse irradiated in sodium flow at 280°C and 0.5 MPa. Some PCMI failures (assisted by hydride concentrations in the cladding of UO_2 rods) and fuel dispersal into the coolant were observed. Twelve tests have been performed in the sodium loop that is going to be replaced by a pressurized water loop for the OECD Cabri Waterloop International Project (CIP). The corresponding loop modifications (Na → Water and CABRI Facility renewal) have started in 2002 and the first tests within the CIP programme in the water loop are scheduled for September 2009, although two tests were already performed in 2002 in the Na loop on 2 PWR high burnup fuel rods (74 GW·d/t). The CIP programme launched in 2000 under OECD auspices with broad international cooperation aims at providing, under typical pressurized water conditions, the necessary knowledge for assessment of new RIA related criteria for advanced high burnup UO_2 and MOX fuels. The CIP is a follow-up of the CABRI REP Na programme that was carried out in the sodium loop of the CABRI reactor and which mainly showed the deleterious influence of a high clad corrosion level with hydride concentration (rim or blisters) on clad failure and the contribution of grain boundary gases on fission gas release and potential gas loading, especially in MOX fuel, during the early phase of a fast power transient with limited clad heat-up. Moreover, the failure of some UO_2 and MOX fuel rods at enthalpy levels ranging from 30 to 113 cal/g demonstrated the need for evolution of the present safety criteria pertaining to fuel behaviour. The CIP should address the remaining questions concerning transient fission gas behaviour and its impact on clad loading during the entire transient, the rod behaviour with high clad temperature and internal pressure, and the post-failure phenomena (fuel ejection, fuel coolant interaction with finely fragmented solid fuel).

Separate effect tests were also launched for mechanical characterization of the cladding material (Zr4, Zirlo and M5) and for the study of clad–water heat transfer under fast transients; they underlined the influence of the cladding heating rate on boiling crisis conditions as compared to steady state conditions (increase of critical heat flux and critical temperature). Data and modelling are thus derived for implementation in the SCANAIR code that has been developed for quantitative interpretation of the results and translation to reactor conditions.

8.2.4.4. BIGR tests

RIA tests for WWER fuel rods have been carried out at the Impulse Graphite Reactor (IGR) and Baykal Impulse Graphite Reactor (BIGR) [29]. Tests showed that the PCMI failure threshold for cladding made of Zr/1%Nb, which provides higher ductility in relation with lower corrosion thickness after long irradiation, was higher than that of zircaloy-4 cladding (WWERs, PWRs) due to clad ballooning with high clad temperature.

8.2.4.5. Out-of-pile tests

Out-of-pile experiments including cladding mechanical property measurements have extensively been conducted to complement in-pile data and to perform separate-effect tests. With the assistance of computer codes, alternative indices such as strain energy density may be introduced to describe cladding failure conditions [30].

8.2.5. *Loss of coolant accident*

In safety analyses for a postulated LOCA, the fuel cladding would be exposed to high temperature steam for several minutes until the emergency core cooling water quenched the fuel bundle. The zircaloy–steam reaction is highly exothermic and results in hydrogen production. In addition, it promotes embrittlement of the cladding when the reacted amount becomes significant. Therefore, extensive experiments were performed in the USA in the 1960s to early 1970s to investigate the zircaloy–steam reaction rate [31, 32] and cladding embrittlement [33] due to high temperature oxidation. These and other relevant experiments are reviewed in Ref. [34]. The current criteria in many countries are based on the knowledge and data obtained in these experiments, with an objective to avoid excessive embrittlement of the fuel cladding and maintain coolability of the reactor core.

The fuel cladding may balloon and rupture due to increase of the rod inner pressure and decrease of the cladding integrity with increase of the fuel rod temperature. Information on rupture conditions to predict the timing of the rupture is important from the standpoint of the time required to isolate the containment building and the overall source term analysis. The increase of the cladding diameter due to ballooning may reduce the coolant channel, which results in a reduction of reactor core coolability. Bundle burst experiment programmes as well as separate-effect tests concerned with ballooning/rupture behaviour under LOCA conditions were performed in the USA, Germany, Japan, France, etc., in the period from the mid-1970s to early 1980s [35].

In the 1980s it was found that secondary hydriding occurs in the vicinity of the ballooned region, and that cladding embrittlement is influenced by secondary hydriding as well as oxidation under LOCA conditions [36]. JAERI and ANL conducted LOCA simulation tests involving rod burst, oxidation and reflooding, to examine embrittlement and the failure-bearing capability of the cladding on quenching [37, 38].

Some experimental programmes are ongoing.

8.2.5.1. The OECD Halden Reactor

Although the OECD Halden Reactor Project addresses, amongst other things, the fuel high burnup capabilities in normal operating conditions, as well as the fuel reliability issue and the fuel response to transients such as LOCAs. In particular, it includes:

- In-reactor tests with high burnup PWR and BWR rods;
- Investigation of axial fuel relocation.

8.2.5.2. The JAEA LOCA programme and ALPS programme

These programmes will investigate the following:

- Failure threshold of the cladding on quenching in terms of oxidation amount, equivalent cladding reacted;
- High burnup fuel (up to 75 GW·d/t);
- New cladding alloys for high burnup fuels (Zirlo, MDA and NDA);

- Oxidation rate measurements;
- Mechanical properties of cladding that experienced LOCA transients.

8.2.5.3. The ANL LOCA programme

This programme will investigate the following:

- High burnup BWR (57 GW·d/t) and PWR (67 GW·d/t) fuel rod behaviour;
- Ballooning and burst, oxidation, quench;
- Mechanical testing, ring compression and bending tests;
- High temperature oxidation behaviour of the new alloys Zirlo, M5, E110.

9. CONCLUSION

Detailed modelling of transient fuel behaviour is an important part of the analyses of the safety of the nuclear power plant. Investigations of fuel behaviour are carried out in close connection with experimental research and computational analyses. The overall objective of experimental research is to provide computing analyses with data used for the validation of models adopted in the codes and numerical values of safety acceptance criteria.

Depending on the type of influence on a fuel element, initiating events which may challenge fuel safety can be grouped into three basic categories, in general: power excursion accidents (including reactivity initiated accidents), power cooling mismatch accidents and decrease of reactor coolant inventory (LOCAs).

The initial and boundary conditions are provided for the related analysis. The choice of related analyses and computer codes depends on the purposes of the carried out calculations. Thus, the codes for the related analysis can be grouped according to the reactor phenomena and type of modelled state into the following categories:

- System thermohydraulics codes;
- Steady state fuel behaviour codes;
- Transient fuel behaviour codes;
- Steady state reactor neutronics codes;
- Reactor dynamics codes.

In this report the initiating events and related important phenomena in the fuel behaviour analysis are introduced. The highlights of relevant fuel behaviour experiments and the related safety criteria on fuel are briefly summarized. For the selection and use of computer codes for the analysis, the selection of methods, types of accident analysis, types of computer codes and the necessary code features are discussed. Finally the practical application of fuel behaviour codes such as input data, input model and actual procedure of transient fuel behaviour analysis are provided. An example of code calculations is introduced in Annex II.

The current issues for DBA related experimental test programmes and fuel behaviour model development activities are largely related to three general trends: (1) high burnup fuel, (2) utilization of MOX fuels, and (3) development of more advanced reactor and fuel designs (including cladding).

In this regard, for improving the understanding of nuclear fuel behaviour under steady state, transient and accident conditions, it is recommended to:

- Extend the experimental database, especially for LOCAs and RIAs for high burnup as well as for advanced cladding and fuels;
- Develop more accurate and precise computer codes:
  - Coupled codes between various disciplines,
  - Advanced codes such as computational fluid dynamics;
- Improve the reliability of computer codes and their utilization (‘user friendly‘ codes)
- Improve the design and accident studies’ methodologies, including the use of BE analysis associated with uncertainty quantification.

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ABBREVIATIONS

ANL	Argonne National Laboratories
ALPS	Advanced LWR fuel performance and safety
BWR	Boiling water reactor
CANDU	Canadian deuterium–uranium reactor
CIP	CABRI International Project
DBA	Design basis accident
DNB	Departure from nucleate boiling
DNBR	Departure from nucleate boiling ratio
FGR	Fachgemeinschaft Guss-Rohrsysteme
GETR	General Electric test reactor
IGR	Impulse graphite reactor
JAEA	Japan Atomic Energy Agency
LOCA	Loss of coolant accident
LWR	Light water reactor
MOX	Mixed oxide fuel
NSRR	Nuclear safety research reactor
PBF	Power burst facility
PCMA	Power cooling mismatch accident
PCMI	Pellet–cladding mechanical interaction
PHWR	Pressurized heavy water reactor
PWR	Pressurized water reactor
RBMK	High power boiling reactor with pressurized channels (Russian design)
RIA	Reactivity induced accident
WWER	Water moderated, water cooled power reactor (Russian design)

ANNEX I

IMPORTANT PHYSICAL PARAMETERS UNDER ACCIDENT CONDITIONS

I.1. Reactivity initiated accident

I.1.1. Ejected control rod worth

The course of a RIA is mainly governed by the amount of reactivity inserted at the beginning of the accident. For the purpose of safety analysis, the control rod having the maximum worth in the core should be assumed to eject. The control rod worth depends on the control rod design, loading pattern, cycle burnup, fuel burnup and control rod position in the core. These kinds of information are given by the nuclear design, fuel design and control rod design.

I.1.2. Cycle burnup and power level

Several cases relevant to cycle burnup and power level need to be considered in the analysis. These include beginning of cycle and end of cycle, hot full power and hot zero power, as well as intermediate power levels. The beginning of cycle case is the minimum feedback case since it typically has the least negative moderator temperature coefficient. The end of cycle case has a slightly smaller Doppler coefficient but a much larger moderator feedback effect. This causes the end of cycle transient to be less severe than the beginning of cycle transient for the same reactivity excursion measured in dollars. However, for end of cycle the reactivity in dollars may be larger due to the reduction of the fraction of delayed neutrons with core lifetime. These cases therefore have to be analysed to cover sufficiently the range of expected conditions. Sometimes intermediate stages between beginning of cycle and end of cycle have also to be considered to ensure conservatism of results.

From the fuel behaviour point of view, almost all of the input parameters such as thermal conductivity, gap conductance and cladding oxide thickness are functions of fuel burnup. In particular the initial gap width affects fuel performance during the transient. The wider initial gap suppresses the heat transfer to the cladding in the very early stages of the accident; later some of the energy stored in the fuel is transmitted through the gap region just after the gap closes. This will cause a sudden cladding temperature rise. Therefore, the effect of burnup on fuel behaviour should be taken into account as well as the effect on neutronic analysis.

I.1.3. Coolant conditions

Core inlet coolant temperature, core flow and coolant pressure might be kept almost constant for the analysis of the reactor power increase, due to the fact that changes of the above mentioned parameters are not substantial during the short time interval analysed. Natural circulation conditions should be taken into account when hot zero power is selected as an initial condition, because the timing of DNB may affect the transient behaviour of the fuel.

I.1.4. Reactor trip reactivity

Conservative values of reactor trip reactivity (conservative time delay and reactivity versus control rod position dependence) are used, typically assuming a stuck control rod in addition to the ejected rod. The stuck rod selected is the highest worth rod at hot zero power with all rods inserted less the ejected rod, and will usually be a rod adjacent to the ejected rod. This assumption is made to account for the possibility of the ejected rod causing damage to an adjacent rod drive housing and preventing that rod from tripping.

A weighting factor is applied to the calculated Doppler feedback to account for the expected increase in feedback as a result of the skewed power distribution after the control rod is ejected. The Doppler weighting has to be calculated conservatively.

I.1.5. Single failure criterion

In applying the single failure criterion, considering that the first signal to actuate the reactor trip system has to come from an excessive increase rate in neutron flux, a potential failure in a reactor power measurement channel also needs to be taken into account.

I.2. Power cooling mismatch accident

I.2.1. Power distribution

Since the DNB heat flux is highly dependent on rod power, the peak power rod should be selected for the analysis. The rod power should be determined to maintain consistency with the design peaking factors specified in the plant design. The radial power distribution in the core should be determined in order to envelop all power distributions expected to occur during the core lifetime.

The DNB heat flux is also influenced by axial power distribution. It should be determined based on the sensitivity study so that the design axial power distribution selected could give a conservative result compared to power distributions occurring in normal operation.

I.2.2. Coolant flow rate and flow distribution in reactor vessel

The thermal design flow rate should be conservatively determined based on both uncertainty analyses of the main coolant pump characteristics and the primary system flow resistance. In addition, flow coast-down after the event should be conservatively determined. The flow distribution between the core flow and the bypass flow needs to be conservatively considered.

I.2.3. Selection of DNB correlation

Selection of an adequate DNB correlation is essential. Usually, a specific DNB correlation is developed corresponding to the fuel assembly types or vendors. If there is no such specific DNB correlation, a generic and well known DNB correlation should be used, taking into account the applicable range of the correlation. Statistical analysis (validation plus definition of uncertainty against experimental data) is required.

I.2.4. Single failure criterion

Sensitivity studies of the results to single failure are required. Usually the single failure criterion is applied to the first signal of emergency protection.

I.3. Loss of coolant accident

A series of plant specific thermohydraulic analyses are required to assess plant safety during a LOCA. These analyses include break sizes, break locations and types of break. The fuel behaviour analysis is performed according to initial and boundary conditions and transient plant behaviour derived from the plant specific thermohydraulic analyses. Since the core becomes subcritical just after the accident and the core power drops to decay heat level, variations in input parameters to be taken into account are relatively rare compared to those for plant analyses. They are as follows:

I.3.1. Power peaking factors

Since the peak cladding temperature during an accident strongly depends on the linear heat rating, it is important how radial and axial power peaking factors used in the analysis are determined. The radial peaking factor should be determined conservatively to cover the power distribution expected to occur in the whole lifetime of the plant. Since the timing of quenching differs according to the axial position of the fuel rod, an axial power distribution skewed to the top generally gives a conservative result. Therefore, the axial power distribution used in the analysis should be determined based on a sensitivity study of axial power shapes taking into account the various operational modes.

I.3.2. Fuel rod burnup

In general, the power history of the fuel rod depends on loading and operation patterns. In addition, precise fuel rod geometry and gap gas content depend on the burnup. The fuel rod burnup used in the analysis should be determined based on a sensitivity study.

I.3.3. Single failure criterion

Sensitivity studies for results on single failure are required. Usually the single failure criterion is applied to the emergency core cooling system.

ANNEX II

EXAMPLES OF FUEL BEHAVIOUR CALCULATIONS

II.1. Introduction

Depending on the objectives of the fuel behaviour calculations, the analyst(s) may be required to perform a series of calculations using a variety of codes, or may simply be able to perform calculations using a single integrated code. For example, as described by Framatome-ANP [II.1] for the optimization of BWR fuel assembly design and performance, a large suite of codes may be used. The codes are used to look at all of the aspects of steady state and transient fuel assembly and core analysis including neutronics, thermohydraulic and mechanical fuel assembly analysis, in-core fuel management and plant transient analysis. Table II.1 shows a listing of the types of codes and calculations used for this type of analysis.

At the other extreme, the analysis to audit or verify the performance of the fuel under design basis accidents such as LOCAs or power cooling mismatch may simply require an integrated system thermohydraulic fuel behaviour calculation using a detailed system thermohydraulic code with representative fuel rod components or models. An example of such a calculation for a PWR plant using RELAP/SCDAPSIM is given in Section II.2.

For operational or other transients such as ATWS or those that may involve power excursions, such as RIAs, an iterative process using neutronic codes, system thermohydraulic codes and detailed fuel behaviour codes may be used. An example of the type of neutronic calculation that might be required in this case is shown in Section II.3.

It should be noted that, with the increases in computer performance, there is a trend to replace iterative calculations using separate codes for neutronics, system thermohydraulics and fuel behaviour with more integrated code systems that are capable of adding all of the aspects simultaneously. For example, in the Framatome-ANP example, a coupled code called RAMONA/S-RELAP5 was used to perform the best estimate plant analysis using 3-D special reactor kinetics. RAMONA [II.2] was used for the 3-D spatial reactor kinetics while S-RELAP5, Framatome's licensing version of RELAP5, was used for the system thermohydraulics.

II.2. Example of integrated system thermohydraulics – Fuel behaviour analysis using RELAP/SCDAPSIM

In the following example, the features of an integrated systems thermohydraulic and detailed fuel behaviour analysis are demonstrated using the RELAP/SCDAPSIM code for the Surry plant for a simple transient involving the simple uncovery and reflood of the core when temperatures in the core exceed 1300 K. The general features of the RELAP/SCDAPSIM code, the Surry plant, and nodalization of the reactor coolant system and representative fuel assemblies are described in the example along with sample results showing the calculated behaviour of one of the representative fuel rods.

The RELAP/SCDAPSIM code [II.3], as indicated in Fig. II.1, includes detailed system thermohydraulic, fuel assembly behaviour and fission product behaviour models. The RELAP5 portion of the code describes the overall thermohydraulics of the system including the convective and radiative heat transfer within each representative fuel assembly. The influence of changes in flow areas and volumes associated with ballooning and rupture are also included in the calculations. The fuel assembly behaviour is calculated using detailed

SCDAP core component models. These fuel assembly component models are selected by input from a set of representative fuel rod components, Ag–In–Cd control rod components and other detailed core components to describe the detailed response of representative fuel assemblies during normal and accident conditions. In the event that the accident exceeds DBA conditions, these models will automatically calculate the loss of fuel assembly geometry including the formation of molten pools and debris beds. For DBA conditions, the calculated fuel assembly response includes 2-D heat conduction within each representative assembly component, fuel rod ballooning and rupture, fission product release, and oxidation of the zircaloy cladding. The fission product transport and deposition models track the behaviour of noble gases, other fission products and, in the case of severe accident conditions, aerosols through the reactor coolant system.

TABLE II.1. EXAMPLE OF CODE CALCULATIONS USED TO SUPPORT BWR FUEL ASSEMBLY DESIGN AND PERFORMANCE [II.1]

Fuel assembly design	Monte Carlo neutronics	Lattice neutronics	2-phase thermo-hydraulics	Steady state subchannel thermo-hydraulics	Assembly thermo-mechanical analysis	Fuel channel analysis
In-core fuel managementand monitoring	Core simulator	Loading pattern optimization	On-line core monitoring			
Core transient and stability analysis	3-D space–time kinetics	Transient subchannel thermo-hydraulics	Stability in the frequency domain			
Plant transient and accident analysis	Best estimate plant analysis, 3-D core representation					

The Surry nuclear power plant is a typical Westinghouse three loop PWR with a rated thermal power of 2441 MW(t). The core consists of 157 15 x 15 fuel assemblies with an active fuel height of 3.66 m. Each of the three primary coolant loops contains a U-tube steam generator, a reactor coolant pump (RCP) and associated piping. A single pressurizer is attached to the hot leg piping of one of the three primary coolant loops. Additional details about the plant and the RELAP5 input models are included in Ref. [II.4].

The RELAP5 nodalization of the reactor coolant system and reactor core and vessel are shown in Figs II.2 and II.3. In this case, the reactor core is represented using five flow channels and ten axial nodes. Each flow channel is connected to its neighbouring flow channels by cross-flow models so that any flow diversion associated with fuel rod ballooning and rupture can be considered. Thus any flow in the lateral direction through the core is modelled.

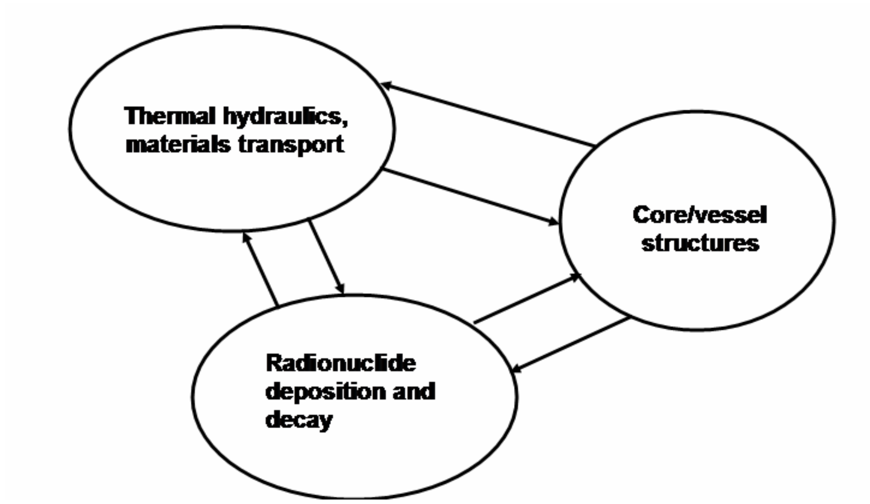


FIG. II.1. Modelling capabilities of RELAP/SCDAPSIM for integrated system thermohydraulic and detailed fuel assembly behaviour.

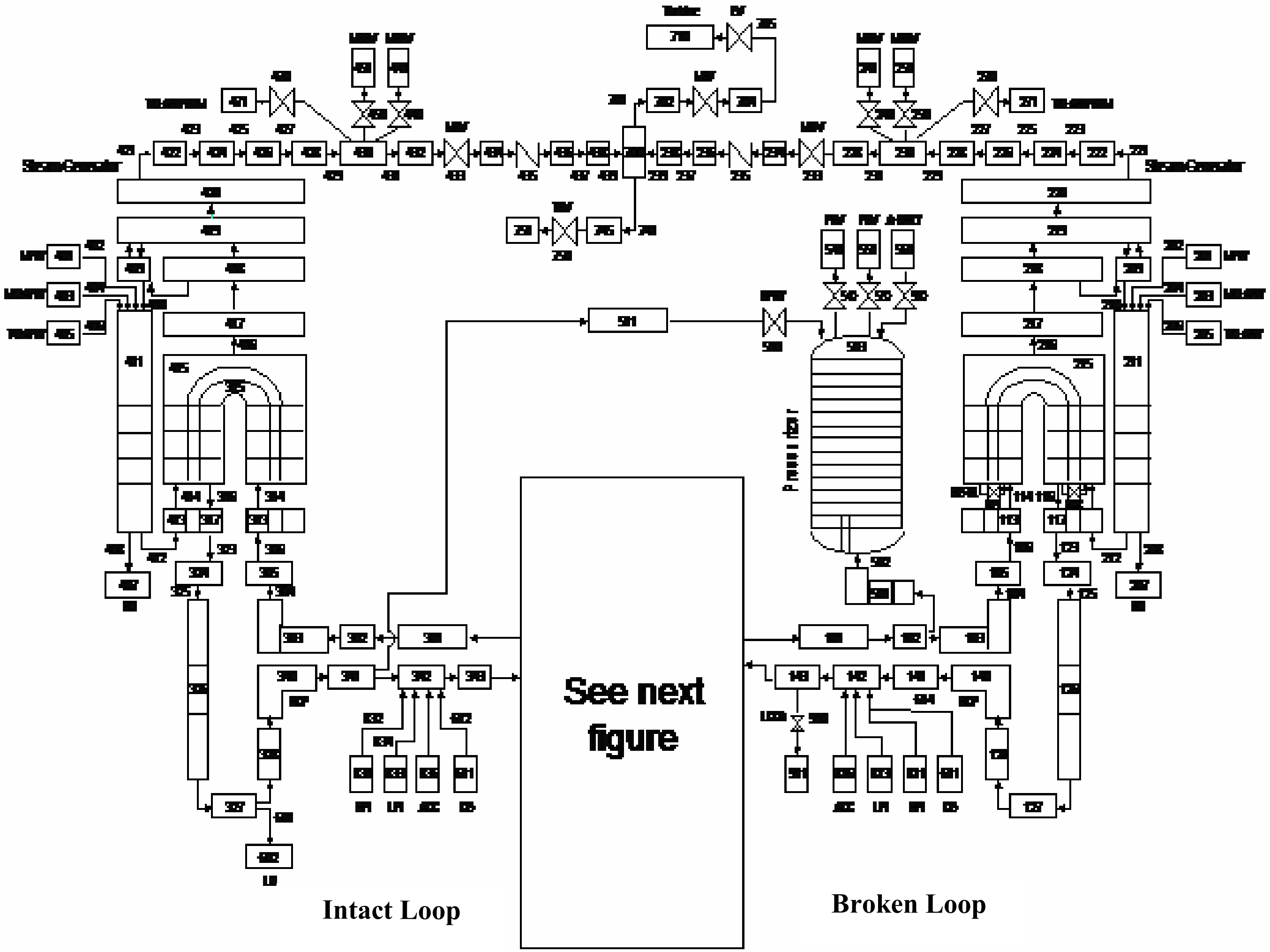


FIG. II.2. RELAP/SCDAPSIM system thermohydraulic nodalization for the Surry plant.

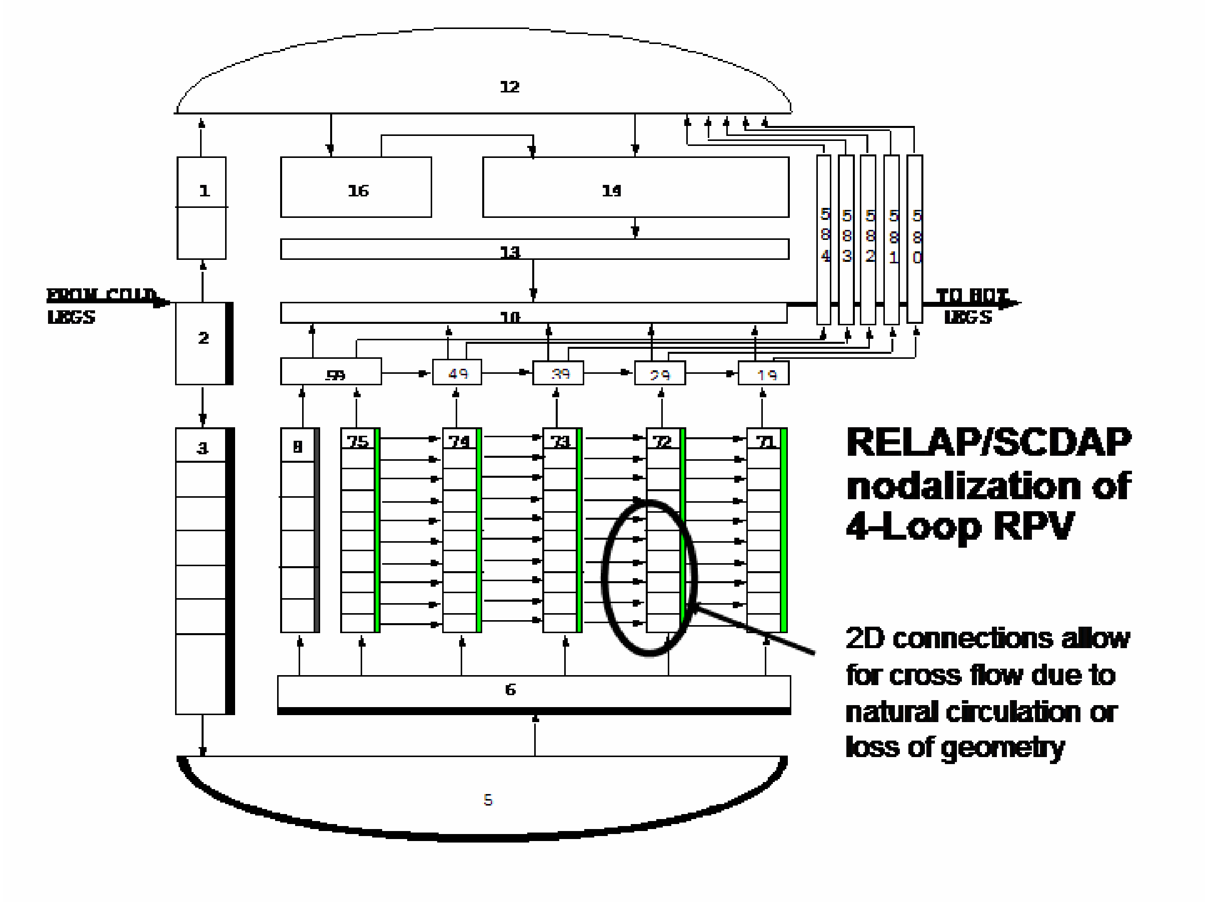


FIG. II.3. RELAP/SCDAPSIM nodalization of the Surry vessel and core.

A radial cross-section of the nodalization of the reactor core is shown in Fig. II.4 showing the number of assemblies considered within each flow channel. The fuel rods in each fuel assembly were represented by an SCDAP fuel rod component and the control rods were represented by an SCDAP PWR control rod component. Each individual fuel assembly was represented as having 204 fuel rods and 21 control rods. Each fuel rod and control rod was divided into ten axial nodes. The water rods in the assemblies were represented by an SCDAP control rod component model containing an infinitesimal amount of control material. Empty zircaloy guide tubes were modelled in a similar way using a fuel rod component model. The physical arrangement of the representative components within the assembly was also described through input, so the proper radiation heat transfer view factors and absorption terms within the assembly could be calculated.

For the purposes of this example, a very simple transient was run where the core was very quickly uncovered and then reflooded when peak fuel cladding temperatures exceeded 1300 K. Figure II.5 shows the representative cladding temperatures at several different axial elevations on one of the representative fuel rods. The start of the heating of the cladding as the water level drops below a given elevation can be very clearly seen, as well as the rapid cooling of the cladding at different elevations as water is added to the bottom of the core. The influence of a cosine shaped axial power profile can also be seen as cladding temperatures in the middle of the assembly rise at a more rapid rate even though the upper portion of the assembly started to heat up earlier due to falling water level. Figure II.6 shows representative fuel centreline and cladding temperatures at one axial position along with the calculated hydrogen production rate for the one representative rod. The results clearly show the beginning of noticeable cladding oxidation as cladding temperatures exceed 1300 K, and then a rapid reduction in the oxidation rates as the temperatures decrease. It should be noted that this transient was intentionally designed to avoid rapid cladding oxidation as cladding temperatures above 1500 K are reached.

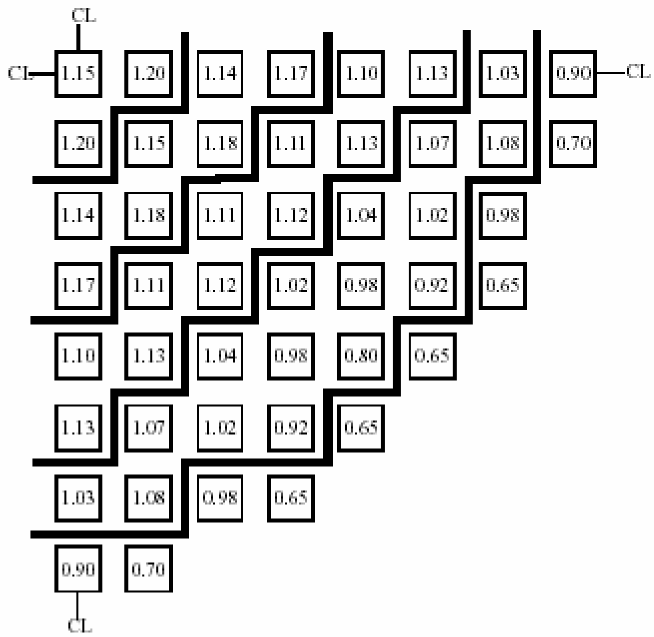


FIG. II.4. Quarter symmetry cross-section view of the Surry plant fuel assemblies represented by the five radial flow channels.

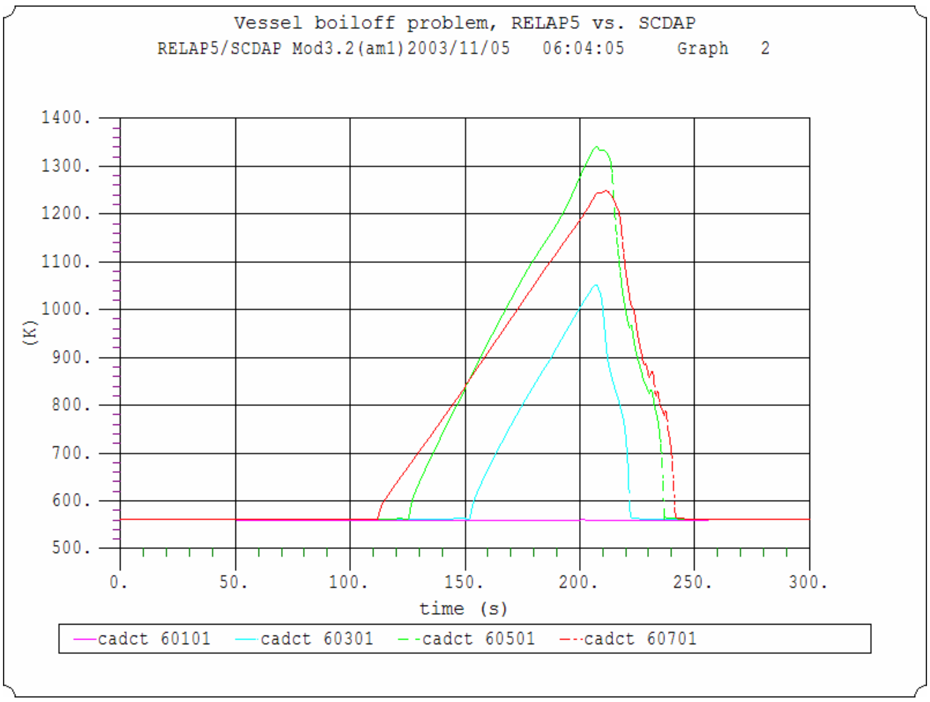


FIG. II.5. Representative cladding temperatures at different axial elevations during the rapid uncovery and reflooding of a representative assembly.

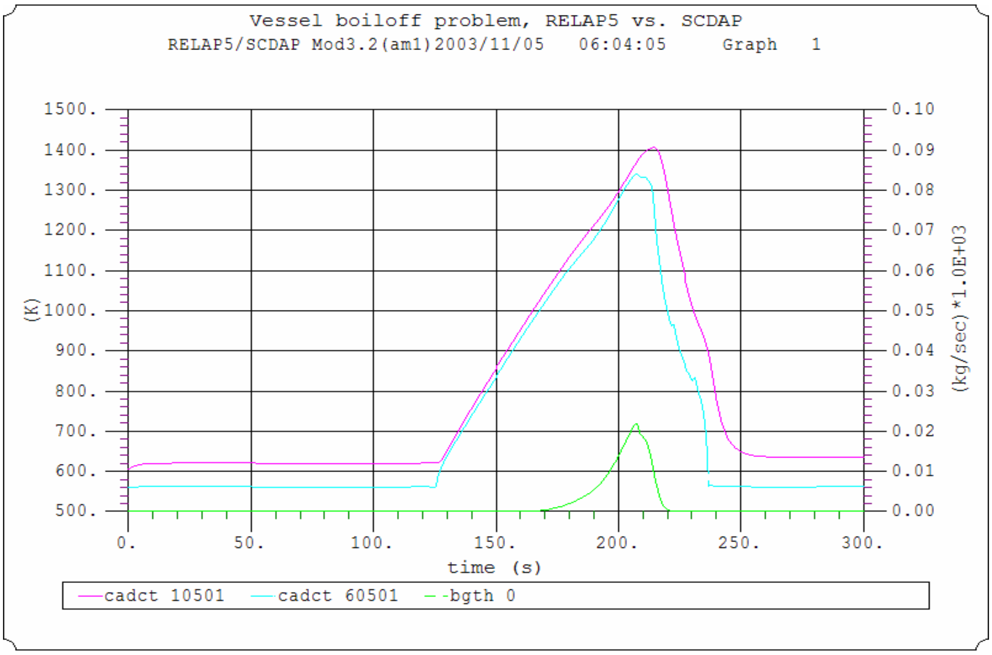


FIG. II.6. Representative cladding temperatures at one axial position during the rapid uncovery and reflooding of a representative assembly.

II.3. Example of reactivity analysis

II.3.1. Calculation results of some reactivity induced accidents

As an example of application of the updated coupled code system DINAMIKA-97+KAMERA+MAZ-3 (DKM), accidents with ejection of one of the most effective core protection system (CPS) control rod are considered.

II.3.2. Conditions with CPS control rod ejection

In calculations of the conditions with CPS control rod ejection, realistic choices were made when selecting the core neutronic and thermohydraulic characteristics.

For the calculations, two fuel loads of a WWER-1000 unified core are selected, namely the first and the equilibrium ones. The conditions with CPS control rod ejection are realized for beginning of the first fuel cycle. The absorber rods of regulating group 10 are located in fuel assemblies 31, 52, 58, 106, 112 and 133 when the complete core pattern is considered, or in fuel assembly 15 when a sector equal to 1/6 of the reactor core is modelled.

Calculations by the DKM code system are performed using the 1.5-group approximation. A spatial model according to the IAEA standard MS ISO 10645-92 is used for the decay heat calculation. The calculation is done using an assembly-by-assembly approximation for 10 sections along the height of the fuel. Fuel burnup is determined by reactor operation within one effective day at 100% of the nominal power, then the 10th regulating group is completely inserted into the lower extreme, with the reactor power increasing to 104% of the nominal power. Stationary xenon poisoning was obtained with this state.

Ejection of one CPS control rod is assumed from fuel assembly 112 within 0.1 s. It is conservatively assumed that the coolant flow rate through the reactor in the stationary state is minimum and amounts to 80000 m^3/h, and the coolant temperature at the reactor inlet is maximum and amounts to 291.7°C. The initial coolant pressure at the core outlet is the accepted minimum, namely, 15.4 MPa. The number of sections of axial and radial core discretization is ten.

The calculation results of the initial state, when the regulating group is 0.35 m above the core top at nominal power level for the beginning of the first fuel cycle and at the end of the equilibrium fuel cycle, and their benchmarking by the computer codes BIPR-7 and MAZ-3, are given in Table II.2 and in Figs II.7 and II.8. For the beginning of the first fuel cycle, the calculations of the states were also performed by the MAZ-3 code with the regulating group completely inserted and group 10 or one CPS control rod from group 10 withdrawn at nominal power level during the specified poisoning with the xenon and samarium nuclei. By the calculation results, on the basis of the change in effective multiplication coefficient (k_{eff}) by the MAZ-3 code, the worths of one rod and of the group of CPS control rods for the beginning of the first fuel cycle are determined and given in Table II.2 in comparison with the results obtained by the BIPR-7 code.

TABLE II.2. COMPARISON OF CALCULATION RESULTS OF THE STATIONARY STATE AT NOMINAL POWER LEVEL USING THE COMPUTER CODES BIPR-7 AND MAZ-3

Parameter designation	Beginning of the first fuel cycle		End of equilibrium fuel cycle	
	Code BIPR-7	Code MAZ-3	Code BIPR-7	Code MAZ-3
CB_{cr},gB/kg H_2O)	0.835	0.830	0	0
K_q (N_k)	1.243 (21)	1.230 (21)	1.305 (8)	1.29 (8)
K_v (N_k,N_z)	1.700 (21,4)	1.683(21,4)	1.450 (8,2)	1.41 (8,2)
$\partial\rho/\partial C_B$, 1/(g$H_3BO_3$/kg)	-0.0205	-0.0203	-0.0150	-0.0149
$\partial\rho/\partial t_c$ 10^{-4}, 1/°C	-1.99	-2.30	-5.80	-5.50
$\partial\rho/\partial t_u$ 10^{-5},1/°C	-2.25	-2.23	-2.14	-1.93
$\beta_{eff,}$ 10^{-2}	0.67	0.673	0.510	0.507
L (μs)	30.07	30.01	22.63	22.56
Efficiency of one CPS CR (%)	0.14	0.146	—	—
Efficiency of CPS CR group (%)	0.780	0.778	—	—

To analyse the hottest fuel rods as regards spatial kinetics at the transients, the power peaking factors inside the chosen hottest fuel assemblies were conservatively assumed so that the maximum linear rating of a fuel rod was about 44.8 kW/m.

During ejection of the CPS CR regulating group, the inserted positive reactivity taking account of the total fraction of delayed neutrons, 0.673%, amounts to 1.113 β, and as a result there can be a prompt neutron runaway that leads to a neutron burst whose restriction in the initial period of the accident is basically due to the Doppler effect.

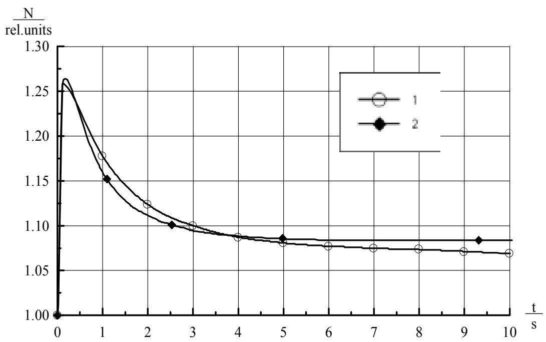
In calculations by the DINAMIKA-97 code using the point kinetics model, the axial power distribution used is that calculated by the MAZ-3 code for the hottest fuel assembly. In this calculation, three channels were used to characterize respectively the average channel, the hottest channel and the flow channel. For the hottest channel, the value of the radial power peaking factor was such that the maximum linear heat rate for the hottest core section amounted to 44.8 kW/m. After ejection of one CPS control rod, an increase in radial power peaking factor of the physical calculations of 1.2 times was accepted. The change was made conservatively for a time of 0.05 s, and later the radial power peaking factor was assumed, again conservatively, to be unchanged. The coefficients of reactivity, efficiency of one CPS control rod and other point kinetics parameters for all the conditions considered (fractions of delayed neutrons, disintegration constants of the nuclei-predecessors of delayed neutrons, prompt neutron lifetime) in the DINAMIKA-97 calculations are assumed equal to the values obtained using DKM for the stationary state. The change in external reactivity for conditions with CPS control rod ejection is found to be linear within 0.1 s. Calculation results from DKM and from DINAMIKA-97 for conditions with ejection of one CPS control rod are given in Figs II.7–II.10.

Benchmarking of the core main parameters in the stationary state calculated by DKM and by the DINAMIKA-97 code shows that there is good agreement in the results obtained. In practice, values of the maximum and average fuel temperatures and maximum radially averaged enthalpies are similar.

Maximum fuel temperatures in the DINAMIKA-97 calculation within the first several tenths of seconds for the accidents with ejection of one CPS control rod do not differ significantly from the change in maximum fuel temperatures determined by DKM.

Calculation results of the considered condition with ejection of one CPS control rod show that analysis of this condition by the DINAMIKA-97 code with use of traditional assumptions on increase of the radial power peaking factor leads to conservative results.

Thus, calculation by the point kinetics method allows a conservative analysis of conditions with ejection of one CPS control rod with minimum quantity calculations. The main condition of conservatism is correct consideration of the change in power peaking ’before‘ and ’after‘ ejection, and the assigning of conservative values of the fuel and coolant temperature feedback.



1 – calculation by DINAMIKA-97
2 – calculation by complex DKM

FIG. II.7. Change in average reactor power.

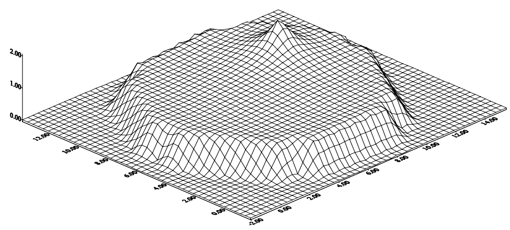
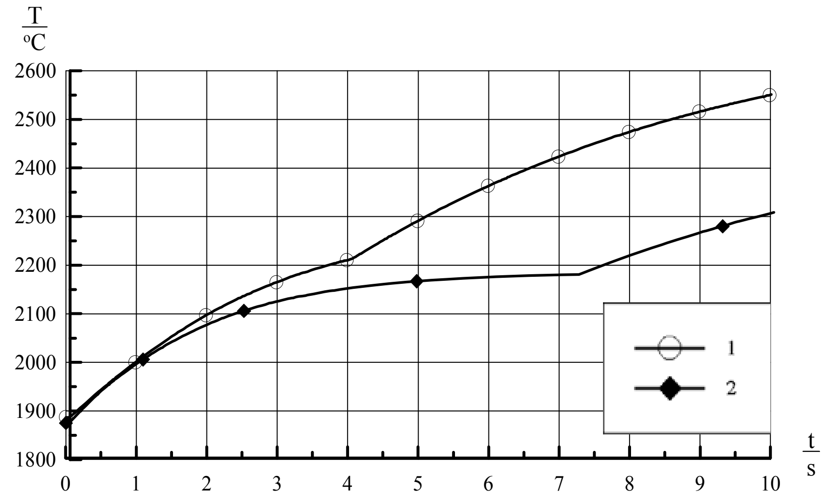


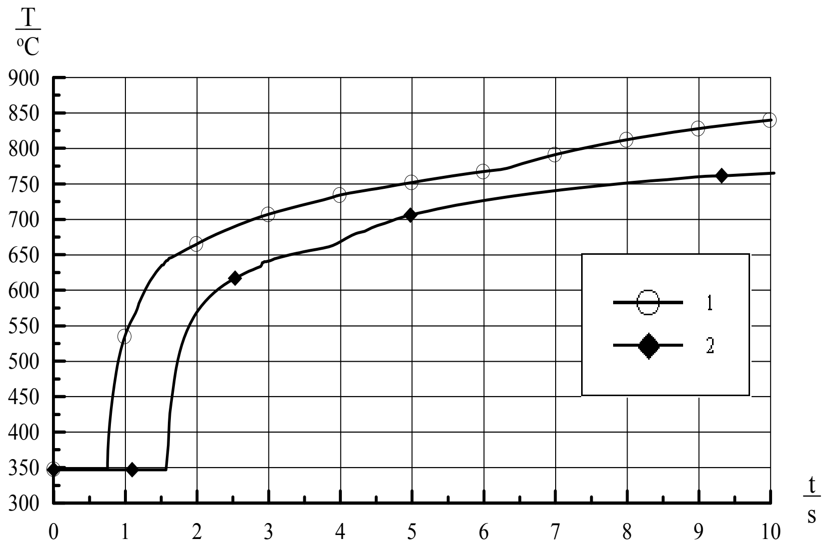
FIG. II-8. Relative calculated change of Kq for the assemblies, at a time 0.1 s off



1: Calculation by DINAMIKA-97

2: Calculation by complex DKM

FIG. II.9. Change in maximum fuel temperature.



1: Calculation by DINAMIKA-97
2: Calculation by complex DKM

Fig. II.10. Changes in coolant temperature.

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CONTRIBUTORS TO DRAFTING AND REVIEW

Allison, C.M.	Innovative Systems Software, L.L.C., United States of America
Bykov, M.	FSUE EDO “Gidropress”, Russian Federation
Haste, T.	Paul Scherrer Institut, Switzerland
Lee, S.	International Atomic Energy Agency
Makihara, Y.	International Atomic Energy Agency
Tricot, N.	International Atomic Energy Agency
Uetsuka, H.	Japan Atomic Energy Agency, Japan