Safety Reports Series
No. 43

Accident Analysis for Nuclear Power Plants with Graphite Moderated Boiling Water RBMK Reactors

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ACCIDENT ANALYSIS FOR NUCLEAR POWER PLANTS WITH GRAPHITE MODERATED BOILING WATER RBMK REACTORS

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SAFETY REPORTS SERIES No. 43

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FOREWORD

Deterministic safety analysis (frequently referred to as accident analysis) is an important tool for confirming the adequacy and efficiency of provisions for the safety of nuclear power plants in accordance with the defence in depth concept. Owing to the close interrelation between accident analysis and safety, an analysis that lacks consistency, is incomplete or is of poor quality is considered a safety issue for a given nuclear power plant. The development of IAEA guidance publications for accident analysis is thus an important step towards resolving this issue.

Various IAEA safety publications have provided details of the requirements as well as guidance for accident analysis, in particular for nuclear power plants of Russian design with water moderated, water cooled power reactors (WWERs) and graphite moderated, boiling water reactors (RBMKs). In particular, the IAEA has developed several guidance publications relevant to accident analysis within the Extrabudgetary Programme on the Safety of WWER and RBMK nuclear power plants. Likewise, several of the revised IAEA Safety Standards Series publications, for example the Safety Requirements on Safety of Nuclear Power Plants: Design (NS-R-1) and the Safety Guide on Safety Assessment and Verification for Nuclear Power Plants, address this topic.

Consistent with these publications, the IAEA in 2002 issued a detailed report on Accident Analysis for Nuclear Power Plants (Safety Reports Series No. 23) that provides practical guidance for performing accident analysis. That report covers the steps required for accident analyses, i.e. selection of initiating events and acceptance criteria, selection of computer codes and modelling assumptions, preparation of input data and presentation of the calculation results. It also discusses aspects that need to be considered to ensure that the final accident analysis is of acceptable quality. Separate IAEA Safety Reports deal with specific features of individual reactor types, such as pressurized water reactors, boiling water reactors, pressurized heavy water reactors and RBMKs.

The present Safety Report provides further guidance by considering specific design features of nuclear power plants with RBMK reactors. In particular, the guidance given focuses on classification of initiating events, on selection of acceptance criteria and on initial and boundary conditions, and specific suggestions are offered for the analysis of different groups of events. This report is aimed primarily at analysts, whether from regulatory bodies or from utilities, who coordinate, perform or review computational analyses of transients and accidents for RBMK reactors. The report is also intended as guidance for IAEA activities in this domain, such as training courses and workshops.

The IAEA staff member responsible for this publication was J. Mišák of the Division of Nuclear Installation Safety.

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CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The IAEA Safety Report on Accident Analysis for Nuclear Power Plants [1] comprehensively describes the methodology for accident analysis. The report is in concert with the revised Nuclear Safety Standards Series and, in particular, with the safety requirements set out in Safety of Nuclear Power Plants: Design [2] and in Safety Assessment and Verification for Nuclear Power Plants [3].

Reference [1] is generic in that it considers all reactor types — it reviews the following issues:

— Classification of initiating events and acceptance criteria;
— Analysis methodology;
— Types of accident analysis;
— Computer codes;
— User effects on the analysis;
— Input data preparation;
— Presentation and assessment of results;
— Quality assurance.

Reference [1] also discusses the analysis of uncertainties and provides a practical example for preparing input data and documentation for the analysis. The annexes to Ref. [1] provide further examples of practical applications and describe the main steps in accident analysis.

Specific guidelines for accident analysis need to take into account the specific characteristics of the plant, and publications can only be developed for specific reactors or, more generally, for a group of reactors belonging to the same type. Reactor specific guidelines have been issued as separate Safety Reports for various types of reactors, including: pressurized water reactors, pressurized heavy water reactors, Canadian deuterium–uranium (CANDU) reactors as a special case of pressurized heavy water reactors, boiling water reactors and graphite moderated, boiling water reactors, also known as RBMK reactors, pursuant to their acronym in Russian.

This Safety Report on safety analysis for nuclear power plants with RBMK reactors has been developed taking into account Russian national regulations [4–6], experience gained with safety analysis reports for RBMKs and international reviews of these reports.

1.2. OBJECTIVE

The objective of this report is to provide specific guidance on accident analysis for nuclear power plants with RBMK reactors. Licensing type safety analyses, aimed at demonstration of sufficient safety margins, are mainly addressed. This guidance includes a detailed list of initiating events and their direct causes, as well as an overview of the safety aspects of an event that may result in failure of the barriers designed to prevent the release of radioactive materials. Suggestions on the selection of acceptance criteria as well as initial and boundary conditions are provided. Specific methodological instructions on how to perform the analysis of individual events are given. A list of output parameters to be presented for various events is suggested.

1.3. SCOPE

Methods for accident analysis have been considerably improved over the past two decades owing to better insights into physical phenomena through research, and enhancement of computer codes and computational capabilities. In parallel, the development of an experimental database and computer code validation studies have made it possible to switch from simplified codes to more sophisticated and mechanistic integral (system) codes. Finally, the ongoing improvements in computer capabilities have removed the main constraints to the use of computational tools.

In the past, safety analyses for facilities using RBMK reactors relied on a conservative approach, using conservative models and computer codes along with conservative input data. Such an approach permitted assessment of the 'worst' consequences of an accident, but was of little use in developing emergency operating procedures and accident management guidelines and, more generally, in planning mitigation activities.

This Safety Report is intended for use in the performance of safety analyses of nuclear power plants both under construction and in operation. While focusing on the performance of the reactor and its systems, including the accident localization system (ALS), during transients and accidents, this Safety Report takes account of best estimate analysis and conservative analysis. The application of best estimate codes that use well grounded acceptance criteria and conservative input data provides a more reasonable assessment of the safety margins in various situations. Adequate conservatism in input data is normally achieved by setting the parameter values at the 'worst' boundary of the range of deviations allowed by the technical specifications of the nuclear power plant [7].

This Safety Report covers situations associated with both design basis accidents (DBAs) and beyond design basis accidents (BDBAs), but consideration of the latter only goes as far as accidents with loss of the core geometry (i.e. the start of core damage). This means that severe accidents with substantial core damage are beyond the scope of this report. The primary focus is on the thermohydraulic and neutronic aspects of the analysis, with some consideration of the relevant radiological and structural issues. Accident progression is covered from the initiating event to the assessment of the radioactive material released. The analysis of the dispersion of radioactive material outside the reactor building is not discussed.

This Safety Report addresses only the 'internal' events that originate in the reactor or in its associated process systems. Some initiating events that affect a broad spectrum of activities at a nuclear power plant (often referred to as internal or external risks), such as fires (internal and external), flooding (internal and external), earthquakes and local external impacts, such as aircraft crashes, are not discussed in detail. Nevertheless, the guidance provided may be used for analysing the consequences of such events from the viewpoint of neutronics and thermohydraulics.

This Safety Report is intended primarily for computer code users who analyse accidents in nuclear power plants with RBMK reactors. Russian and Lithuanian regulatory authorities might wish to use it for revising requirements or setting up new ones, as needed. This Safety Report, together with the other reports dealing with different types of installation, is meant for independent use. However, it is suggested that the user first become acquainted with the general Safety Report [1] before turning to this guidance on RBMK transient and accident analysis.

1.4. STRUCTURE

This report is consistent with the contents and, to a large extent, with the format and structure of Ref. [1]. Section 2 presents the main characteristics of RBMK reactors, with particular emphasis on the peculiarities of systems and special design features that distinguish RBMKs from other types of reactor from the point of view of safety analysis. Section 3 describes the initiating events and breaks them down into classes. The selection and classification of events are based on physical phenomena that result from the initiating events. Section 4 discusses the acceptance criteria that are applied to accident analyses for RBMKs, as well as the logic that underpins these criteria. The methodology of accident analysis is the subject of Section 5. This section deals with the approach to the analysis and the definition of accident scenarios. Section 6 provides suggestions for selection of the initial and boundary conditions for accidents. The safety aspects of various initiating events are discussed in Section 7. In analysing DBAs and some beyond design basis events (those without significant degradation of the geometry of the systems), a correlation is made with the relevant acceptance criteria, as dictated by the logic of the analysis. In the case of BDBAs, including anticipated transient without scram (ATWS) events, it is always important to check whether the particular thermohydraulic and neutronic parameters of a system remain inside the scope of the computer code models. Finally, Section 8 describes the format and structure for presenting the results of the analysis.

2. STRUCTURAL CHARACTERISTICS OF RBMKs

2.1. REACTOR CORE AND CIRCULATION CIRCUIT

RBMKs are boiling water cooled, graphite moderated, channel type reactors (Fig. 1). The graphite stack together with the fuel and other channels make up the reactor core. The key structural component of the stack is a column made of graphite blocks (parallelepipeds of square cross-section). All graphite blocks have a hole in the centre. The central holes of the columns accommodate the tubes of the fuel channels, channels with absorbers placed in them for reactor control and protection, as well as the tubes of other special purpose channels. The thermal contact between the moderating graphite and the channel tubes is provided by means of solid contact split rings and sleeves also made of graphite. In order to improve the thermal contact between the graphite and the tubes, and thus to reduce the graphite temperature, the entire free space of the stack, including the clearances in the block/ring/channel/tube or block/sleeve/channel/tube systems, is filled with a gas mixture consisting mostly of helium. The helium–nitrogen mixture enters the stack from the bottom at a low flow rate and exits from the top through the standpipe of each channel via an individual pipeline. Increases in the moisture and temperature of the mixture provide evidence for detecting coolant leaks from the pressure tubes. The core has top, bottom and side reflectors. The first two are made up from the same graphite blocks and the latter is formed by the columns.

The core is enclosed in the reactor cavity, formed by the top and bottom plates and a cylindrical barrel (KZh in Fig. 1) hermetically welded to the plates.

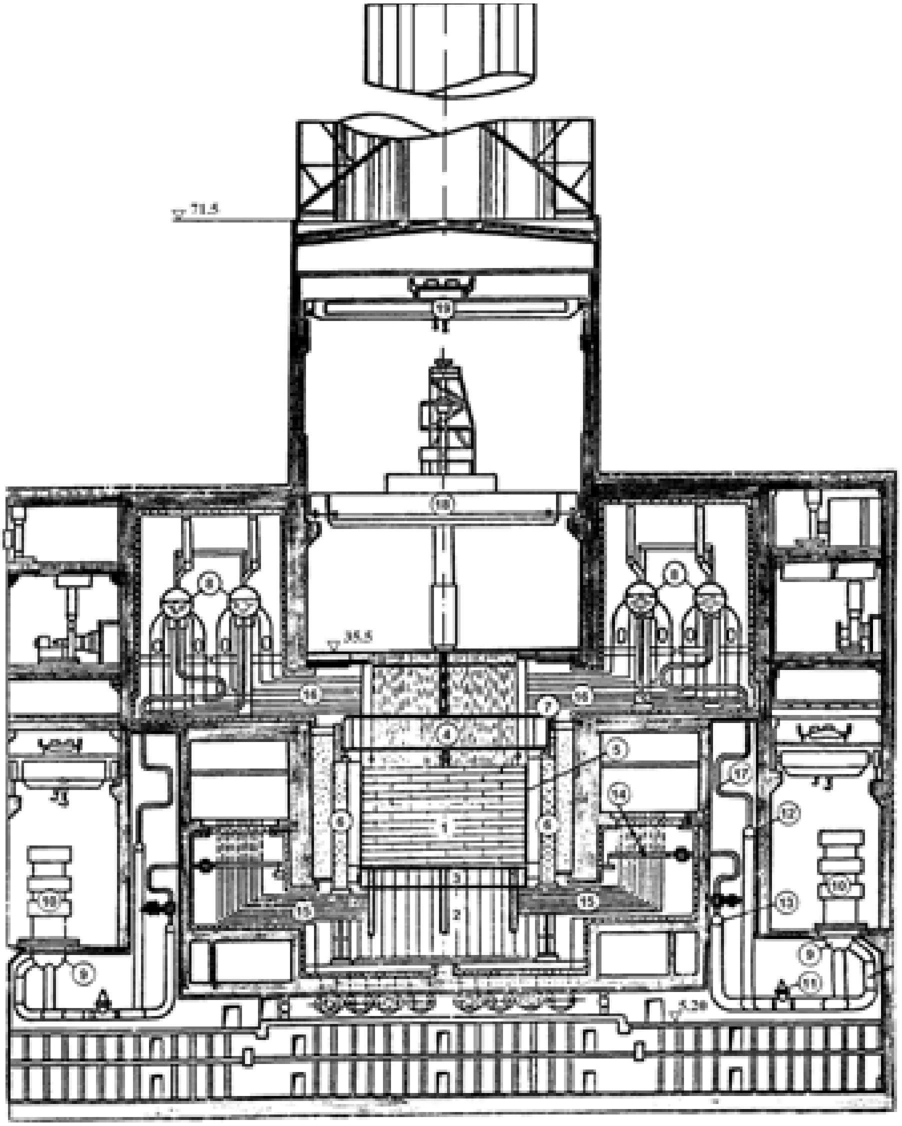

FIG. 1. Schematic diagram of an RBMK reactor: 1, graphite stack; 2, structure S; 3, structure OR; 4, structure E; 5, structure KZh; 6, structure L; 7, structure D; 8, drum separator; 9, MCP casing; 10, MCP motor; 11, pressure gate valve; 12, suction header; 13, pressure header; 14, distribution group header; 15, water lines; 16, steam and water lines; 17, downcomers; 18, refuelling machine; 19, central hall crane.

The top and bottom plates (the metal structures E and OR) are pierced by tube lines to house the fuel and other channels. The top plate, mounted on roller supports, rests on the structural components of the reactor building. It takes the weight load from all the reactor channels together with their internal components, as well as part of the weight load from the steam–water pipelines and other service lines (pipes and cables). The bottom plate supports the graphite stack of the core.

Fuel assemblies installed in their channels consist of two subassemblies connected in series. The container type fuel rods are filled with pellets of low enrichment uranium dioxide with the addition of a burnable absorber (erbium). Fuel claddings are made of zirconium alloy (Zr 1% Nb), and channel tubes inside the core are fabricated from another zirconium alloy (Zr 2.5% Nb). Corrosion resistant steel is employed for the inlet and outlet pipelines of the channels outside the core.

The circulation circuit of the reactor is divided into two loops, each including a group of the main circulation pumps (MCPs), the suction, pressure and distribution group headers (DGHs), drum separators, as well as the downcomers between the drum separators and the MCP suction header. Each of the two circuit loops has half of the fuel channels connected to it.

Figure 1 shows schematically the layout of the reactor and the circulation circuit components in the reactor building. In the central hall, a refuelling machine is placed above the top plate of the reactor. Its function is to unload spent fuel assemblies and to load fresh ones under reactor operating conditions.

2.2. REACTIVITY AND POWER CONTROL

Solid absorber rods are employed to control the reactivity and, thus, the reactor power. The control rods travel on hangers in special channels cooled by water from a separate circuit. The control rods are suspended by the steel strips of drive mechanisms mounted on the channel cappings. The absorber rods fall into several functional groups.

The related functions at an operating reactor include:

— Monitoring of the neutron power and its rate of increase;
— Automatic maintenance of the specified power in accordance with the signals of the ex-core ionization chambers and in-core sensors;
— Control of the specified radial power distribution;
— Preventive controlled power reduction in response to variations in neutron flux signals;

— Fast controlled power reduction to safe levels;
— Complete shutdown of the reactor by all control rods except the functional group of emergency protection (EP) rods (fast power reduction (FPR) mode);
— Complete shutdown of the reactor by all the rods of the system (emergency protection mode).

All these functions are performed by an integrated monitoring, control and protection system (CPS) designed in compliance with the requirements specified by the national regulatory authorities (see, for example, Ref. [5]). The system has a two suite arrangement, with either of the two suites being capable of carrying out all of the system's functions. The system was introduced at Kursk-1 in 2002 and will be fitted to all RBMK plants.

2.3. SAFETY SYSTEMS

An RBMK reactor is equipped with the following safety systems:

(a) An emergency core cooling system (ECCS), consisting of two (one fast acting and the other providing long term cooling) subsystems. The fast acting subsystem, using hydroaccumulators, is designed for immediate supply of water to the reactor channels in response to a corresponding emergency signal, while the long term cooling subsystem employs pumps; the longest delay at the beginning of its operation depends on the time it takes diesel generators to start in response to the emergency protection signal. Both subsystems are capable of feeding water into the circuit at its nominal pressure.
(b) The overpressure protection system of the circulation circuit. Its key components are three groups of main safety valves (MSVs) installed on a loop pipeline that integrates all the ducts collecting steam from the drum separators. From the MSVs, steam flows into a pressure suppression pool (PSP) of the ALS[1].
(c) A reactor cavity overpressure protection system (reactor cavity venting system), composed of two parts. One part, consisting of outlet pipelines and condensing devices, is designed for localization of a DBA, for example rupture of one pressure tube. The other part, equipped with a group of relief devices opening to the atmosphere, is designed to prevent

[1] Applies to all but first generation RBMKs.

overpressure of the reactor cavity in the case of BDBAs associated with multiple pressure tube ruptures (MPTRs).

(d) An ALS intended for confining accidental coolant releases in leaktight compartments.[1] The ALS does not cover the whole circulation circuit. Some pipelines at the top of the circuit, the drum separators and the steam ducts are located outside the system's hermetic boundary (Fig. 1).

(e) An emergency CPS that consists of two independent shutdown systems (see also Section 2.2).

3. CLASSIFICATION OF INITIATING EVENTS

Accident analysis is intended to assess the capability of the plant systems and personnel to cope with abnormal and accident conditions. For analysis, it is helpful to classify the initiating events. Various approaches to this classification are possible. The Safety Report [1] suggests using the following attributes to classify initiating events:

(a) Adverse impact of initiating events on the fundamental safety functions;
(b) Root causes of the initiating events;
(c) Consideration of the events in the original nuclear power plant design;
(d) Phenomenology, reflecting the thermohydraulic and neutronic aspects of the transient;
(e) Frequency and scenarios of events.

It is also essential to take into account the national regulatory requirements such as those in Ref. [4], where classification into DBAs and BDBAs is related to item (c) in the list above. The aim of safety analysis is to demonstrate that the consequences of a DBA are coped with by the plant safety systems, considering the single failure criterion so that the doses at the boundary of the controlled area are kept within the permissible limits, or more generally that the acceptance criteria are not exceeded. However, it is possible that design basis limits will not be exceeded even with a more serious initiating event or with a greater number of failures, beyond the single failure criterion. Should the initiating event initially have been classified as a BDBA, this event

[1] Applies to all but first generation RBMKs.

TABLE 1. CLASSIFICATION OF INITIATING EVENTS ACCORDING TO FREQUENCY OF OCCURRENCE

Initiating event	Frequency (1/reactor-year)	Characteristic	Identification
Design basis	10^{-2}–10^{-1}	Anticipated during nuclear power plant service life	Anticipated operational occurrence
Design basis	10^{-4}–10^{-2}	Possible occurrence during nuclear power plant service life with probability >1%	DBA
Beyond design basis	10^{-6}–10^{-4}	Possible occurrence during nuclear power plant service life with probability <1%	BDBA without core damage
Beyond design basis	$<10^{-6}$	Extremely unlikely event	BDBA with substantial core damage (severe accident)

can be moved to the DBA class as long as it does not violate the classification scheme based on frequency. Therefore, the boundaries between classes of initiating events are not necessarily rigid and the classes themselves are established only for convenience of analysis.

An important indication for assigning an initiating event to one class or another is the frequency of its occurrence. Table 1 gives an example of events classified according to their frequency. The approach employed for analysing events of various classes differs mainly with respect to the use of certain acceptance criteria and in the level of conservatism of the assumptions made in the computational analysis.

Another approach, which relies on items (b) and (d) above, resulted in the different list of initiating events recommended in Ref. [6]. The class of DBAs in accordance with Ref. [6] includes:

— Operational transients;
— Deterioration of core cooling;
— Loss of coolant accidents;
— Reactivity initiated accidents.

In addition, in Ref. [6], it is suggested that 'other accidents' should also be considered, such as accidents in fuel handling, internal events (flooding, fire and explosion) and external events (flooding). BDBAs include ATWS and 'other accidents' such as dropping down of the refuelling machine and total station blackout.

On the basis of existing experience of safety analyses and various regulatory and guidance reports applicable for RBMK reactors, a tentative classified list of initiating events has been developed. Such a list of initiating events is provided in Table 2. In this table, two classes of events are distinguished: anticipated operational occurrences (denoted by T for transients) and accidents. Accidents are further subdivided into DBAs and BDBAs. The classification of events for RBMKs of different generations is dictated by two factors:

(1) Frequency of occurrence;
(2) Ability of the plant systems to cope with the accident.

For instance, the probability of a break with 800 mm equivalent diameter (D_{eq} = 800 mm) or in the MCP pressure header is estimated to be less than 10^{-6} per reactor-year, i.e. this initiating event could be classified as a BDBA. However, ALSs at RBMKs of the second and third generation are capable of confining this accident so that the acceptance criteria for exposure at the control area boundary and beyond are not violated. Therefore, a break in the MCP pressure header may be regarded as a DBA for RBMK units of the second and third generations in spite of the fact that its frequency is considered to be very low.

For RBMKs of the first generation, guillotine breaks of the main feedwater pipe and of the main steam line should be treated as BDBAs [6]. Intensive in-service inspection and other feasible actions are undertaken to prevent breaks in these pipes (with the break probability reduced to below 10^{-6} per reactor-year).

A partial break of the DGH leads to flow stagnation in the fuel channels of this DGH with possible damage to the fuel rods. The size of the flow area of this break is ~35% of the flow area corresponding to the header (D_{eq} = 300 mm). Detailed computational analysis of the stressed–strained state of the DGH with crack development by two different mechanisms shows that a critical crack (with its growth leading to major pipe rupture) may have a flow area of no more than 1.37% of the corresponding header flow area. With the DGH operating, a larger crack is unstable and develops to the point of complete rupture of the header, i.e. the existence of a DGH break with 35% area is physically impossible. Break-off of a pipe from the DGH at the welded joint is an event of very low frequency, while a dependent failure of adjacent

TABLE 2. EVENTS TO BE CONSIDERED IN THE SAFETY ANALYSIS OF RBMK NUCLEAR POWER PLANTS

Event group	Initiating event	Class
Anomalies of core coolant temperature	Reduction of feedwater temperature	T
	Excessive feedwater flow	T
	Excessive steam discharge from the drum separator by: — inadvertent opening of bypass valve — inadvertent opening of MSVs — reduction of grid frequency	T
	Loss of feedwater flow	T
	Loss of service water supply	T
	Loss of intermediate cooling circuit	T
	Feedwater control failure	T
	Pressure control failure	T
	Inadvertent safety relief valve opening	T
Anomalies of system pressure	Total loss of in-house power supply	T
	Loss of main heat sink	T
	Loss of feedwater	T
	Failure of one or two turbogenerators	T
	Generator load surge	T
	Loss of residual heat removal	T
	Pressure control failure	T
	Inadvertent closure of main steam isolation valves	T
Anomalies of core coolant flow rate	Trip of one MCP	T
	MCP seizure	T
	Trip of several MCPs	T
	Spurious partial closure of the MCP throttling valve in an operating reactor	T
	Failure of the isolation disc of the DGH check valve	T
	Actuation of an idle MCP	T
	Shaft break of one of the MCPs	T
	Break of an MCP check valve plate or of an MCP gate valve disc	T
Increase of core coolant inventory	Spurious ECCS actuation	T

TABLE 2. EVENTS TO BE CONSIDERED IN THE SAFETY ANALYSIS OF RBMK NUCLEAR POWER PLANTS (cont.)

Event group	Initiating event	Class
Spectrum of loss of coolant accidents (LOCA) events	Guillotine break of DGH	DBA
	Guillotine break of downcomer	DBA
	Break in the inlet pipeline of a fuel channel	DBA
	Break in the outlet pipeline of a fuel channel	DBA
	Break of a channel tube inside the reactor cavity	DBA
	Break of the main feedwater pipeline	DBA
	Break of the main steam duct	DBA
	Failure to close the MSV	DBA
	Break of a small diameter pipeline outside the ALS	DBA
	Inadvertent safety/relief valve opening	DBA
	MSV jammed open	DBA
	Guillotine break of the MCP pressure header	BDBA
	Partial (critical) break of the DGH	BDBA
	Rupture of water communication line	DBA
	Rupture of steam–water communication line	DBA
	Rupture of pressure tube inside the reactor cavity	DBA
	Rupture of a pipeline in the blowdown and cooling system	DBA
Reactivity initiated accidents	Voiding of the CPS cooling circuit	DBA
	Erroneous refuelling	DBA
	Prolonged withdrawal of a control rod from the core at both nominal and low power	DBA
	Prolonged withdrawal of a bank of control rods at both full and low power	DBA
	Control rod drop, including the absorber part of short rods falling out of the core	DBA
	Spurious actuation of the ECCS	T
	Nitrogen ingress into reactor coolant system after actuation of ECCS	DBA

TABLE 2. EVENTS TO BE CONSIDERED IN THE SAFETY ANALYSIS OF RBMK NUCLEAR POWER PLANTS (cont.)

Event group	Initiating event	Class
ATWS	Loss of ultimate heat sink	BDBA
	Partial flow loss in the main circulation circuit (MCC)	BDBA
	Loss of feedwater	BDBA
	Loss of in-house power supply	BDBA
	Prolonged control rod withdrawal at full and low power	BDBA
Fuel handling accidents	Fuel assembly jamming or breaking off during its installation in the spent fuel pool by the refuelling machine	BDBA
	Canister with spent fuel falling or becoming jammed in a hanging position during refuelling	BDBA
	Fuel assembly jamming or breaking off during its removal from the channel by the refuelling machine under reactor operational conditions	BDBA
	Fuel assembly falling or becoming jammed in a hanging position during its handling by the central hall crane	BDBA

pipes if struck by the broken pipe is not credible as the impact will not be strong enough to damage the neighbouring pipe. Therefore, the practical impossibility of a critical rupture and the very low probability of a consequential water pipe break suggest that this initiating event belongs in the class of BDBAs.

Analysis of BDBAs is not discussed fully in this report, although some of the report provisions are applicable for analysing the initial phase of the BDBAs, when the geometry of the core is still intact. The results of such an analysis of the early phase may be helpful for working out actions to manage accidents with the aim of mitigating their consequences.

Beyond design basis accidents are analysed for the following purposes:

(a) To assess the degree of reactor protection and the time available for taking countermeasures;
(b) To determine the emergency and non-emergency signals available to the operator for identifying the plant status and to devise appropriate accident management steps;
(c) To develop a package of organizational and technical measures (accident management strategy) for the prevention and mitigation of accident consequences;
(d) To assess the possible consequences of the BDBA as input information for developing emergency plans for the population and personnel.

The approach to the analysis of ATWS events for different nuclear power plants should be determined in conformity to the as-built CPS configuration. According to the upgrading plan, all RBMK plants should be equipped with a modern integrated monitoring CPS (Section 2.2) that should meet all the current safety requirements, such as those for redundancy and diversity. In such a case, loss of the shutdown function during transients with equipment failures is disregarded in the international practice as physically impossible. Therefore, ATWS analysis focuses on proving the adequacy (from the viewpoint of preventing unfavourable process development) of the adopted set points for operation of the second shutdown system.

Multiple pressure tube rupture events are BDBAs. In this case, the potential hazards are loss of reactor cavity integrity and damage to the metal structures of the reactor. To define the scope of an MPTR beyond which there is the threat of reactor cavity destruction, it is necessary to perform an analysis of the venting capacity of the system for reactor cavity protection against overpressure. The results of this analysis are needed in assessing the consequences of the BDBA leading to an MPTR.

The list of initiating events in Table 2 is not intended as exhaustive or mandatory. It may be used for reference in compiling a similar list for a specific RBMK unit. Other recommendations that may be relevant to this effort are listed in Ref. [6].

4. ACCEPTANCE CRITERIA

The safety of a nuclear power plant is understood as its capability to keep the radiation exposure of personnel and the population within specified limits (see, for example, Ref. [4]). This capability is ensured by maintaining the integrity of safety barriers, which are part of the plant's defence in depth concept. A series of barriers prevents the release of radioactive fission products beyond the reactor containment and into the environment. In analysing the safety of a nuclear power plant, it is essential to assess the integrity of the barriers and to decide to what degree the response of the whole nuclear power plant and its systems to a certain initiating event is acceptable from the viewpoint of plant safety. For the sake of simplicity, the integrity of the barriers is related to certain threshold values, which are referred to as acceptance criteria. Essentially, these are the design limits for DBAs (see, for example, Ref. [6]), adopted with a conservative margin, so that the integrity of the safety barrier is guaranteed as long as the parameters do not exceed the relevant criteria.

In normal operating conditions, the integrity of the safety barriers is assured by adherence to the limits and conditions for operation of the nuclear power plant (or so-called operating regulations, see, for example, Ref. [7]). These limits and conditions define the normal operation parameters of nuclear power plants and their tolerable deviations from the nominal values. The set points for initiation of the emergency protection are established so that, in the case of a significant increase or decrease of a parameter, the corresponding safety feature should prevent this parameter from going beyond its safe operating limit (see, for example, Ref. [8]). Analysis of a transient should demonstrate whether the emergency protection set points for a certain parameter were chosen correctly.

Exceeding the safe operating limit by some parameter (if such a limit is set for this parameter) may lead to an accident, i.e. to failure of the barrier(s) preventing the release of radioactivity.

A safety analysis for an RBMK nuclear power plant should assess the integrity of the following barriers against release of radioactivity:

(a) Fuel matrix;
(b) Fuel cladding;
(c) Circulation circuit pressure boundary and, in particular, the components most susceptible to damage, namely the fuel channel (pressure) tubes;
(d) Metal structures forming the reactor cavity;
(e) Structural components of the leaktight ALS compartments and other compartments of the nuclear power plant housing circulation circuit pipelines.

Should any barrier fail, thus opening the pathway for the release of radioactivity beyond the plant boundaries, the amount of radioactivity and the population exposure need to be assessed.

For RBMKs designed and licensed before the main safety criteria and requirements came into force in 1993, the radiation criterion is absolute in the sense that the consequences of a DBA (in terms of radioactive releases and discharges to the environment) should never result in such a population exposure as to require emergency protective actions in the early phase of an accident (i.e. for about ten days after the accident).

As far as the conditions of barrier integrity are concerned, exceeding a given threshold value will only mean that the integrity of the barrier in question is not guaranteed and that an additional, more comprehensive, analysis should be performed to assess the actual state of this barrier.

The logical procedure of accident analyses for RBMKs is shown in a general form in Fig. 2. The whole set of acceptance criteria or some part of it may be used, depending on the problem being considered. For instance, the criteria related to the ALS may be disregarded if the pipelines and components of the circulation circuit (pressure boundaries) have not lost their integrity.

However, if the initiating event is a break in the MCC pipeline, the analysis takes two routes: the first route is to assess the integrity of the fuel claddings, and the second is to check the integrity of the ALS and other nuclear power plant compartments (involved in the accident). In the first route, if the acceptance criterion for cladding integrity is met, the analysis proceeds to assessment of the pressure tube integrity. If the latter criterion is also complied with, the outcome is an 'acceptable plant response'. However, a final answer will not be found until the second route is covered, which lies in the assessment of the ALS integrity, with possible additional analysis needed even in the case of intact compartments, as ALS compartments are not sufficiently leaktight in some RBMK units. In these cases, both 'assessment of off-site fission product release' and 'exposure assessment' are required. If the acceptance criterion for doses is met, the second route of analysis also leads to an 'acceptable response'. This means that the safety of the nuclear power plant under the conditions of the accident in question is assured.

Safety analysis involves anticipated operational occurrences (transients) as well as DBAs and BDBAs. It should be borne in mind, as indicated in Section 3, that the latter two classes have a quite arbitrary demarcation line between them.

4.1. NORMAL OPERATION

Operational limits should be observed during normal operation of RBMKs. The most significant limits related to the RBMK-1000 and RBMK-1500 reactor designs are listed in Table 3.

All relevant parameters are available to the operator either as a result of measurement or as a result of on-line computer calculations. An example is the calculation of the safety factor for the critical power of the fuel channel (critical power ratio).

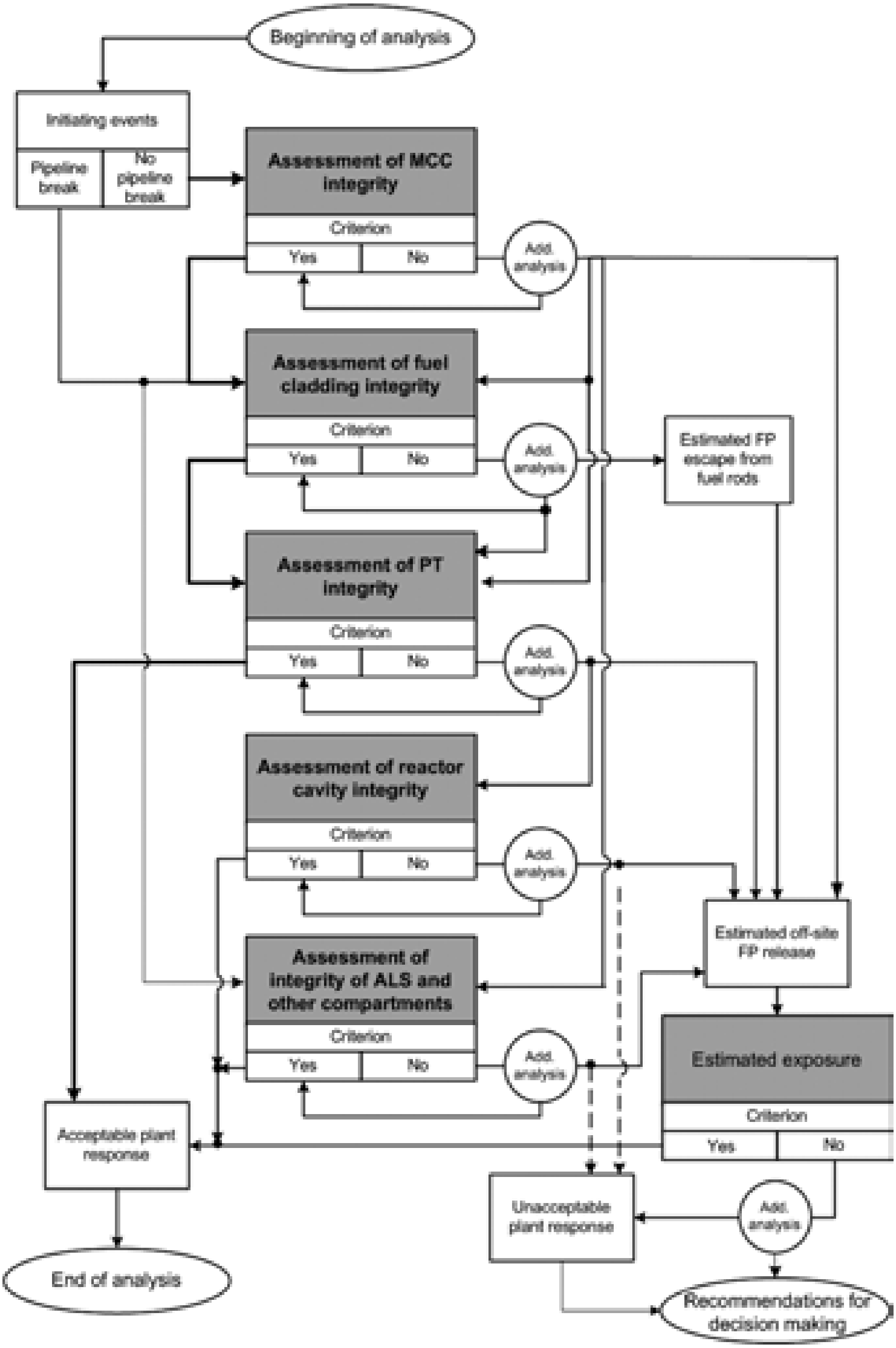

FIG. 2. Flow chart for accident analysis (FP, fission power; PT, pressure tube).

TABLE 3. OPERATIONAL LIMITS OF RBMKs

Parameter	Dimensions	RBMK-1000	RBMK-1500
Reactor thermal power	MW	3200 + 10%	4200 + 10%
Reactivity margin	Number of effective control rods	43–48	53–58
Maximum power of fuel channels	MW	3.0	3.75
Maximum linear power of fuel channels	kW/m	35.0	42.5
Safety factor for critical power of channels	Non-dimensional	>1.0	>1.0
Steam flow rate at full power of the reactor	t/h	5440–5600	7400–7650
Water flow rate in reactor	m^3/h	$(46–48) \times 10^3$	$(39–48) \times 10^3$
Overpressure in drum separator	MPa	$6.9^{+0.1}_{-0.4}$	$6.9^{+0.1}_{-0.4}$
Temperature of feedwater	°C	155–165	177–190
Temperature of water at the core inlet	°C	265–270	260–266
Maximum temperature of graphite	°C	≤730	≤750
Reactor and MCC heat-up rate	°C/h	10	10[a]
Reactor and MCC cooldown rate	°C/h	30	10[a]
Water flow rate in CPS circuit	m^3/h	1030–1220	1250–1350

[a] Reactor and MCC heat-up or cooldown rate in emergency cases should not be more than 30ºC/h. The rate of 30ºC/h is a safe operating limit for RBMK-1500 reactors.

4.2. ANTICIPATED OPERATIONAL OCCURRENCES

Anticipated operational occurrences (transients) requiring emergency protection of the reactor should be analysed by considering the possibility of reaching and/or exceeding the safe operating limits. A properly chosen set point of the emergency protection in relation to a certain parameter allows power to be reduced or the reactor to be shut down before this or another parameter can reach or exceed the safe operating limit.

The safe operating limits proposed for the RBMK-1000 and RBMK-1500 reactors are shown in Table 4.

TABLE 4. SAFE OPERATING LIMITS OF RBMKs

Parameter	Dimensions	RBMK-1000	RBMK-1500
Reactor thermal power	MW(th)	≤3840	≤4800
Maximum linear power of fuel rods	kW/m	≤56.0	≤48.5
Safety factor for critical power of fuel channels	Non-dimensional	≥1.0	≥1.0
Overpressure in drum separator	MPa	≤7.84	≤7.84
Water flow rate in a CPS channel with an inserted rod	m^3/h	≥2.0	≥3.0
Reactivity margin in the effective manual control rods	Number of effective control rods	[a]	≥30

[a] The reactivity margin in effective control rods is regarded as the operational limit for RBMK-1000 reactors [8].

Both reactor thermal powers in excess of the value given in Table 4 (20% above the nominal value) and linear heat rates in excess of the specified limit indicate that fuel may begin to melt in the channels with maximum power. Fuel melting leads to failure of the first two barriers against the release of radioactivity. Reduction of the critical power ratio to less than 1.0 does not necessarily mean immediate failure of the fuel claddings, but prolonged operation of fuel rods under degraded heat transfer is likely to cause failure of the claddings in spite of their relatively low temperatures. A pressure rise in the drum separators to the permitted limit will cause all three groups of MSVs to open, and its further increase will jeopardize the integrity of steam ducts and drum separators. Water flow reduction in the CPS channels with inserted rods to values below 2 m^3/h will affect the rod cooling conditions, with the possibility ensuing of thermomechanical deformation of absorber claddings and jamming of control rods.

Exceeding a single safe operating limit in some transient requiring emergency protection can lead to an accident, i.e. to the failure of a barrier accompanied by the release of radioactivity. This means that safe operating limits may serve as acceptance criteria in the analyses of transients involving emergency protection features.

4.3. DESIGN AND BEYOND DESIGN BASIS ACCIDENTS

The acceptance criteria applicable for analyses of DBAs are listed in Table 5 [9].

4.4. FUEL CLADDING INTEGRITY

The design functions of the fuel cladding are as follows:

(a) To keep all fission products and actinides confined inside the fuel rods throughout the fuel lifetime (function of a physical barrier against release of radioactivity);
(b) To maintain the fuel rod geometry and configuration throughout its lifetime (heat removal or cooling function).

The safety barrier function is expected to be lost if thermomechanical analysis has demonstrated a loss of cladding integrity.

The other function of the cladding (fuel rod cooling) may be lost even without its loss of integrity, merely due to thermomechanical deformation caused by inward and outward directed fuel rod pressure differences (Fig. 3). In the latter case, cladding ballooning (without its failure) and its subsequent operation will pose a threat of overheating both of the fuel (due to increase of the clearance and distortion of the fuel column geometry) and of the cladding itself (due to abnormal flow patterns and flow redistribution in the fuel assembly). Considerable deformation of the cladding impairs the thermal conditions of fuel and cladding due to the temperature increase and the variation in temperature distribution.

Extensive experimental results for RBMK fuel claddings permit estimation of the conditions for cladding failure without a detailed thermomechanical analysis of cladding behaviour (Figs 4 and 5). The results show that the RBMK cladding may be expected to fail with the temperature exceeding 700°C both in the event of ballooning and during squeezing of the fuel column. The probability of cladding rupture as a result of ballooning increases from zero at 700°C to unity as the temperature rises. For instance, if the internal pressure in the fuel rod is 4.0 MPa (spent fuel rods) and the pressure in the circuit is close to atmospheric, the heating by residual heat to 800°C will result in cladding rupture. For fuel rods with lower burnup (i.e. with lower internal pressure), rupture is probable at higher temperatures.

TABLE 5. RBMK ACCEPTANCE CRITERIA FOR DESIGN BASIS AND BEYOND DESIGN BASIS ACCIDENTS

Parameter	Acceptance criterion
Fuel pellet	Temperature ≤2800°C Volume averaged fuel enthalpy ≤710 kJ/kg
Fuel cladding	Maximum temperature ≤1200°C for DBAs[a] Maximum local cladding oxidation ≤18% Maximum core-wide hydrogen generation ≤1%
Fuel channel tube	Nominal wall temperature ≤350°C at 13.4 MPa Maximum wall temperature ≤650°C at 4.0–8.0 MPa
Circulation circuit	Overpressure ≤10.4 MPa
Reactor cavity	Excessive pressure: ≤210 kPa for a DBA ≤300 kPa for a BDBA with MPTR
ALS	Maximum permissible pressures in compartments, operating pressure difference of a safety device or maximum water temperature in the PSP Hydrogen concentration in any compartment no higher than 4 vol.%
Maximum permissible radiation doses in the early phase of the accident	No greater than: 0.5 cSv for the whole body 5.0 cSv for the thyroid

[a] Temperature ≤ 700°C to avoid fuel cladding collapse.

In the event of squeezing of the fuel cladding by inward directed pressure variation, no perceptible deformation is observed if temperatures do not exceed 700°C and pressure differences are no greater than 8.0 MPa. At higher temperatures, the cladding may cling to the fuel column. Under these conditions, a real threat appears to be the loss of cladding integrity through the following mechanisms:

(a) A crack resulting from deformation at a break in the fuel column, if the break width is larger than two times the cladding thickness;
(b) Brittle failure during overcooling as a result of degraded mechanical properties due to metal saturation with oxygen and formation of hydrides;
(c) Fatigue failure (during continued operation) aggravated by degradation of the mechanical properties of the cladding during accidental heating and/or cooling.

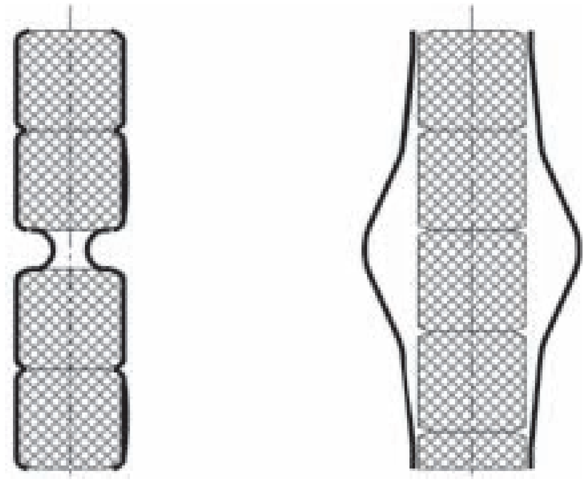

FIG. 3. Cladding deformation modes.

The first two mechanisms can take place during an accident.

It is known that fuel claddings experience radiation hardening and embrittlement in an operating reactor. Over the fuel lifetime the yield strength can increase twofold while the ductility can be reduced by one half.

The cladding temperature value of ≤700°C adopted as a fuel failure criterion corresponds to the absence of any perceptible cladding deformation under possible fuel rod pressure variations, directed either inwards (Fig. 4) or outwards (Fig. 5). Assuming that no changes in the alloy properties occur when irradiated claddings are heated up to 700°C, it can be concluded that the acceptance criterion for irradiated claddings was adopted with a certain margin, i.e. with a higher yield strength and reduced ductility. A tangible cladding deformation is very unlikely under abnormal conditions if the temperature is below 700°C.

Recovery of the mechanical properties of irradiated claddings heated to abnormal temperatures is discussed in an analytical review [10] which shows that the mechanical properties of irradiated Zircaloy-4 will be no different from those of an unirradiated alloy even with a heating rate of 28°C/s and temperatures as high as 700°C. Another conclusion of this review is that at temperatures of 600°C and more, even with high heating rates (possible in accidents), the mechanical properties of irradiated zirconium alloys may be recovered by annealing.

Fuel rod failures, associated with the thermomechanical interaction of fuel with claddings, are mainly typical of reactivity initiated accidents. The

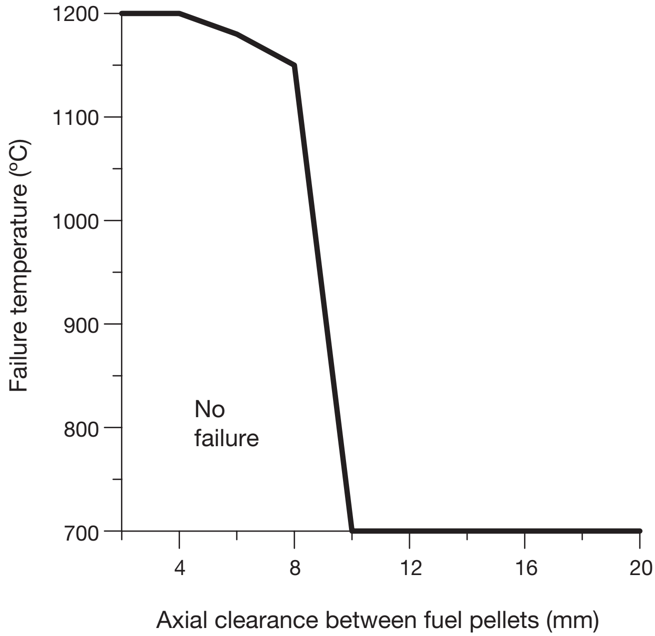

FIG. 4. Cladding failure under squeezing.

nature of fuel rod damage depends primarily on the power produced in the fuel rod and on the temperature of its cladding. The fuel enthalpy (the average for the pellet volume), leading to cladding failure, decreases as the rate of power increase becomes higher. If the power increase takes long enough for relaxation of stresses in the fuel and cladding, the fuel enthalpy corresponding to cladding failure will increase. The fuel of RBMKs is fundamentally no different to the fuel of light water reactors, for which a sound database has been collected using experimental results obtained both in the Russian Federation (pulsed mode tests of WWER fuel rods in the Gidra reactor and in the Impulse Graphite Reactor) and in other countries (tests at the SPERT and PBF facilities in the United States of America and at NSRR in Japan). On the basis of these data, the fuel enthalpy corresponding to the onset of cladding failure is taken as ~1000 kJ/kg UO_2 (240 cal/g UO_2). The conservative margin adopted in the US standard implies that the fuel enthalpy should be no greater than ~710 kJ/kg (170 cal/g) if fuel damage is to be avoided.

As noted above, if the time of power increase is in seconds rather than microseconds, the thermomechanical fuel cladding interaction becomes weaker and the limiting enthalpy approaches its value at the beginning of fuel melting. As the typical time of reactivity insertion in RBMKs is measured in seconds

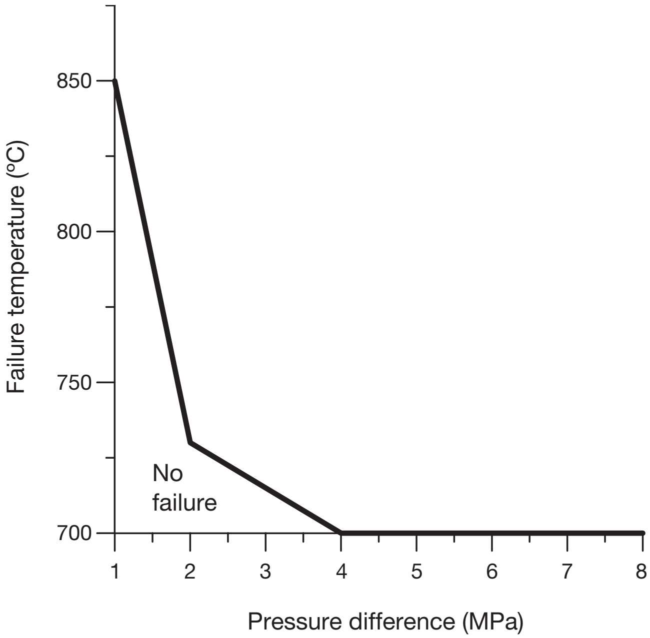

FIG. 5. Cladding failure under ballooning.

rather than in microseconds, the limits for fuel enthalpy and temperature become practically equal in value.

For the integrity of fuel claddings to be confirmed, it is therefore essential to ensure that the following maximum values of fuel rod parameters are not exceeded:

— Pellet volume averaged fuel enthalpy of 710 kJ/kg;
— Fuel temperature of ~2800°C;
— Cladding temperature of 700°C.

Additional analysis may have to take into account another mechanism of cladding damage, i.e. the acceleration of the steam–zirconium reaction at temperatures above 800°C. At 1000°C and above, the heat release from zirconium oxidation in steam becomes commensurate with the residual heat release in the fuel.

The effect of the interaction of the cladding with steam is normally evaluated by the local depth of cladding oxidation, which implies an equivalent thickness of the oxidized layer. This is calculated as the conditional thickness of the metal layer converted to ZrO_2 provided that all the oxygen absorbed by the cladding metal has formed the oxide (in terms of the stoichiometric ratio). The local depth of fuel cladding oxidation should not exceed 18% of the original thickness. It is assumed that the integrity of oxidized claddings is assured for smaller values of local oxidization.

The additional analysis should check compliance with the criterion that the mass of zirconium cladding that reacted with steam should not exceed 1% of the total mass of fuel claddings in the core. This sets a limit to the release of hydrogen into the compartments of the nuclear power plant.

4.5. CHANNEL TUBE INTEGRITY

The fuel channel tube is installed so that its zirconium part (Zr 2.5%Nb alloy) is located inside the core, with the part made of corrosion resistant steel outside. The acceptance criterion addressed in this section is only applicable to the zirconium part. Steel portions of the fuel channel are treated as circulation circuit pipelines according to the respective acceptance criteria (Section 4.6).

During an accident, a channel tube rupture may be expected either on account of a steep pressure rise at near operational temperatures or as a result of thermomechanical deformation at rather high pressures in the circuit. The admissible pressure of hydraulic tests, equal to 13.4 MPa, may be taken as a conservative acceptance criterion for cases when the tube temperatures are close to their operating values (i.e. ≤350°C). The conservatism of this value can be easily proven by evaluating the pressure corresponding to the onset of plastic deformation of a tube. For the temperature of 300°C, the result is 20 MPa, and a real threat of pressure tube rupture will not appear until this pressure is exceeded.

The definition of the acceptance criterion for conditions of thermomechanical deformation relies on the results of investigations into the high temperature behaviour of RBMK and CANDU pressure tubes. The temperature related failure criterion, i.e. tube failure temperature as a function of pressure, was established for pressure tubes of both CANDUs and RBMKs on the basis of experimental data. This dependence varies with the increase in heating rate (Fig. 6). The rupture temperature (in °C) of the tube varies with pressure (in MPa) according to:

$T_{\text{rupture}} = 790.5\,P^{-0.099}$ for heating rates <1°C/s
$T_{\text{rupture}} = 987.7\,P^{-0.139}$ for heating rates >1°C/s

The experimental results depicted in Fig. 6 pertain to unirradiated tubes. As shown in Ref. [9], heating of irradiated zirconium alloys (such as Zr 2.5%Nb) to 600°C and higher makes their mechanical properties similar to

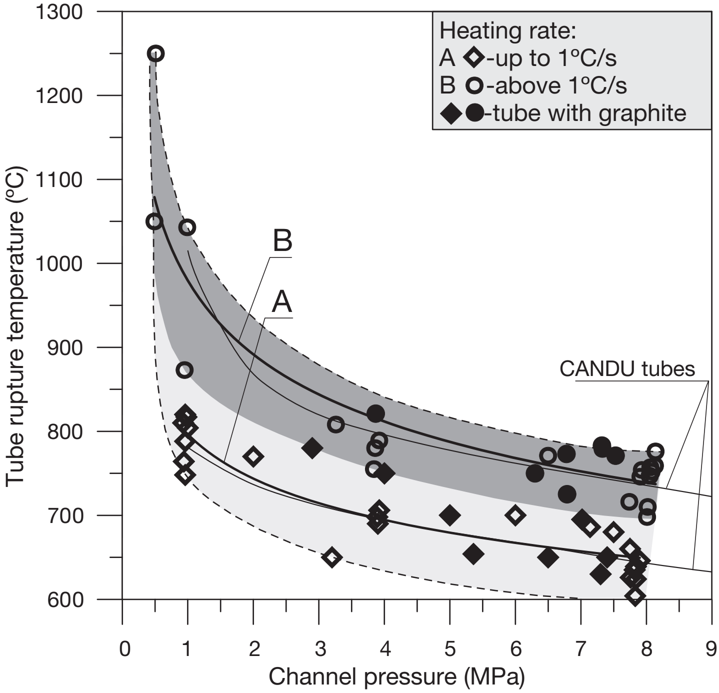

FIG. 6. Pressure tube rupture temperature as a function of internal pressure.

those of an unirradiated alloy, which basically amounts to annealing of radiation induced defects.

The constant temperature of 650°C may be taken as a conservative value of the acceptance criterion for the pressure range from 4 to 8 MPa. This value corresponds to the low heating rates (<1°C/s) common to relatively late time intervals when heating is affected by residual heat release. Low pressures in the circulation circuit are most often found during such time periods. At pressures below 4.0 MPa, graphite blocks may prove strong enough to prevent tubes deforming to the point of rupture.

If computational analysis shows that the channel tube temperature approaches or exceeds 650°C at any point in time and at any elevation, additional analysis should be carried out using a thermomechanical code. Such analysis takes into account both time dependent variations of thermohydraulic parameters and the characteristics of the tube material together with the changes in the tube rupture parameters.

Heating of the zirconium part at a higher rate results in higher failure temperature values at the same pressure. For instance, if the tube heating rate is 10–12°C/s, the failure temperature at about 7 MPa will be 750°C, i.e. 100°C higher. Such sensitivity of the criterion to heating rate poses certain difficulties in its application. For the temperature criterion to be effectively used, it could be represented by a family of curves $T(P)$ with various heating rates, $dT/d\tau$, where τ is time. This, however, would require a large number of expensive experiments. This difficulty may be obviated by introducing a more versatile criterion dependence. Thus, an energy criterion that is only slightly sensitive to the heating rate may be adopted for the thermomechanical code employed for calculating the deformation and for assessing pressure tube integrity, as shown in Fig. 7. This criterion is a specific rupture strain power j_i (W/kg) which is determined by the stress intensity σ_i, material density ρ_w and strain rate intensity ζ_i:

$$j_i = \frac{\sigma_i \zeta_i}{\rho_w}$$

The criterion curve in Fig. 7 is obtained by approximation of empirical data by two conjugated sixth degree polynomials:

$$j_{ip} = \sum_{k=0}^{k=6} \alpha_k T^k$$

with polynomial coefficients α_k given as in Table 6.

The thermomechanical code calculates the stress intensity and the strain rate intensity. Then, given the material density, ρ_w, ratio (1) is taken to calculate the current value of j_i, which is compared with the criterion value j_{ip} at every time step using ratio (2). The condition $j_i \geq j_{ip}$ corresponds to channel tube rupture.

4.6. INTEGRITY OF MCC PIPELINES AND COMPONENTS

The MCC is shown schematically in Fig. 8. The integrity of the MCC pipes and components is the subject of safety analysis for accidents that are not initiated by pipe breaks. Pressure tube integrity is investigated separately (Section 4.5), as only the pressure tubes can experience the simultaneous impacts of forces and heat loads. All the other MCC components are only exposed to loads at temperatures close to those of coolant saturation. Criteria in the form of certain pressure values are adopted to prove the MCC integrity during DBAs or to assess the integrity during beyond design basis events.

Various MCC sections are capable of withstanding different maximum pressures. The MCC section between the gate valves of the MCP inlet pipes

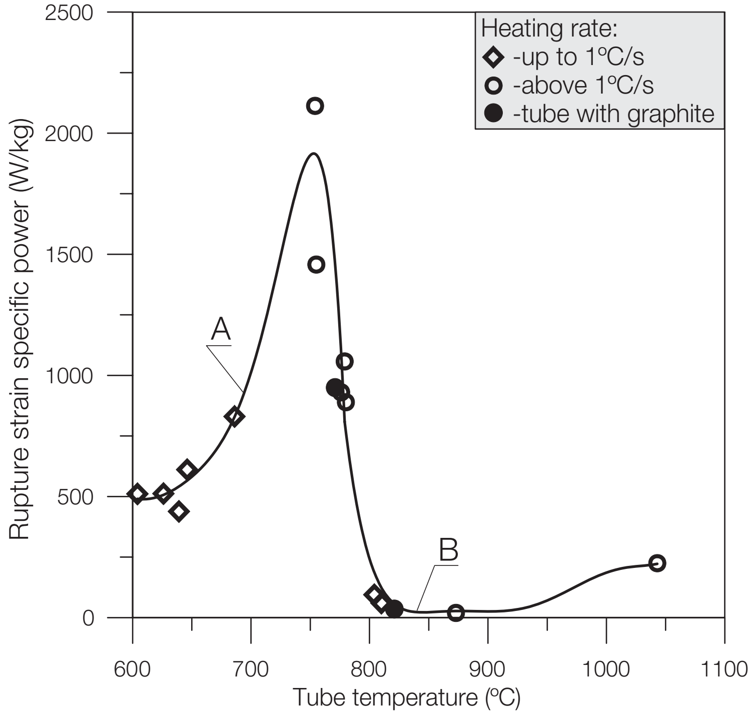

FIG. 7. Specific rupture strain power as a function of tube temperature: A, first polynomial in Table 6, $T \leq 780°C$; B, second polynomial, $T > 780°C$.

TABLE 6. POLYNOMIAL COEFFICIENTS USED IN FIG. 7

Temperature (°C)	α_k (k = 0, ... 6)						
	0	1	2	3	4	5	6
$600 \leq T \leq 780$	-8.58388×10^{7}	797 061.0	3079.41	6.336043	-7.322469×10^{-3}	4.506571×10^{-6}	-1.15385×10^{-9}
$780 < T \leq 1050$	9.00264×10^{7}	–581 258.59	1561.12	–2.23237	1.7925482×10^{-3}	$-7.6633326 \times 10^{-7}$	1.326675×10^{-10}

and the gate valves at the DGH inlets, which can be shut off by isolating valves, can tolerate the highest pressure. The permissible hydraulic test pressure at this section is 13.4 MPa. Fuel channels are also tested under 13.4 MPa. The hydraulic test pressure adopted for the remaining MCC components, which is determined primarily by the strength of the drum separators and steam lines, ranges from 10.1 to 10.4 MPa. Since the MCC operates as a single system, the lowest of these values should be taken as an acceptance criterion.

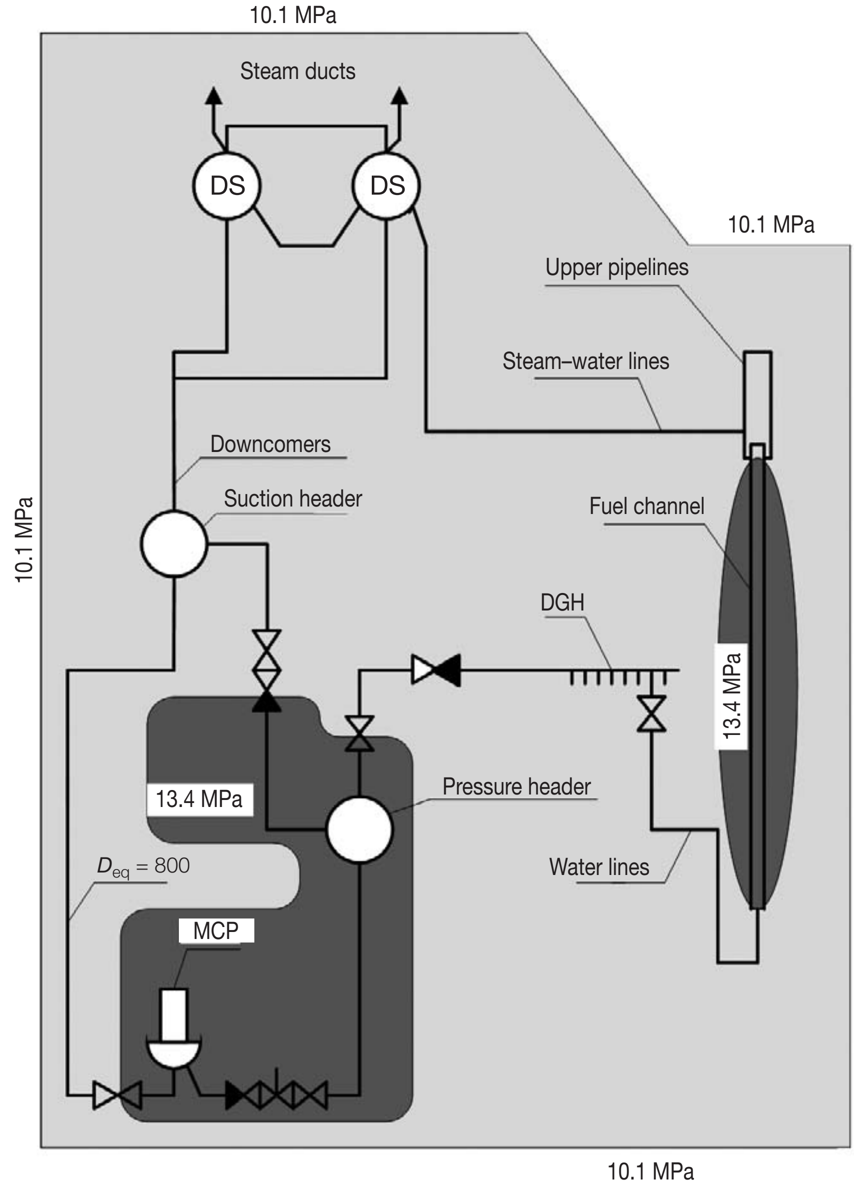

FIG. 8. Basic diagram of one loop of the MCC (DS, drum separator).

4.7. INTEGRITY OF THE REACTOR CAVITY

The reactor cavity is formed by three metal structures: the top plate (E), the bottom plate (OR) and the barrel with a thermal expansion compensator (KZh). The barrel is hermetically welded to the top and bottom plates. The metal structures of the reactor cavity are shown schematically in Fig. 9. The permissible internal pressure loads in the reactor cavity are also presented in Fig. 9. The lowest value of 214 kPa (excess) is the top plate uplift pressure. This pressure was calculated without regard for the mechanical links of the top plate with other structural components, such as the graphite columns. In using the top plate uplift pressure (214 kPa) as an acceptance criterion, it would be implicitly assumed that the graphite stack is not deformed in the event of pressure tube ruptures and that there is no mechanical interaction whatsoever between pressure tubes and graphite blocks. These assumptions are conservative and unrealistic as, in the case of MPTRs, the bulk of the stack will inevitably be deformed due to the high flexibility of the pressure tubes and columns (low lateral stiffness) [10]. If, however, in the event of MPTR all the graphite is found 'suspended' on pressure tubes, this effect alone will increase the top plate uplift pressure to about 300 kPa (excess). In such a case, it would seem that the permissible barrel (KZh) pressure of 255 kPa is the lowest of the three pressures. This pressure was determined in keeping with the rules for strength analysis of reactor components.

For normal operating conditions, the nominal permissible stress $[\sigma]$ is assumed to be the lesser of the following two values:

$$[\sigma] = \min\{R^T_{p0,2}/1.5;\ R^T_m/2.6\}$$

where

$R^T_{p0,2}$ is the minimum yield strength at operating temperature and
R^T_m is the minimum ultimate strength at operating temperature.

For accident conditions, the calculated set of stresses is determined by the following conditions:

$$(\sigma)_1 \leq 1.4[\sigma]$$

if the overall membrane stresses alone are taken into account, or

$$(\sigma)_2 \leq 1.8[\sigma]$$

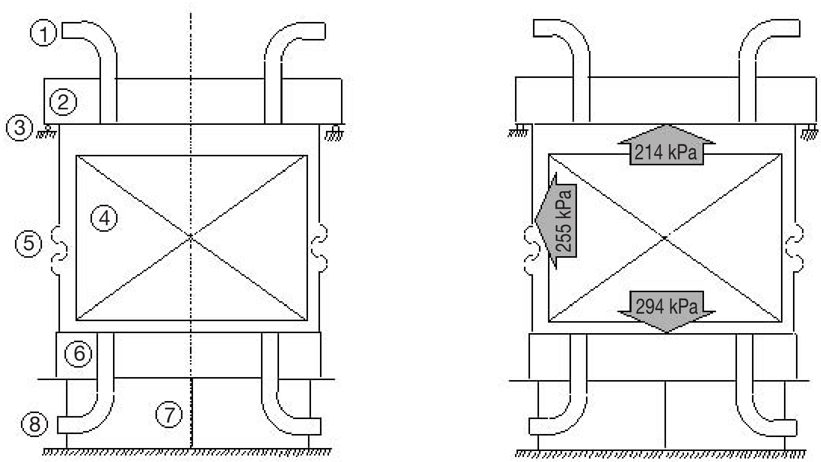

FIG. 9. Reactor cavity components and ultimate pressures: 1, upper steam dump pipes; 2, top plate; 3, roller support; 4, core; 5, barrel; 6, bottom plate; 7, support; 8, lower steam discharge pipes.

if allowance is made for the overall or local membrane stresses and for the overall bending stresses.

For normal operating conditions, the nominal permissible stress was determined by the ultimate strength value as

$$[\sigma] = 173.54 \text{ MPa}$$

Thus, the following values were derived for abnormal conditions:

$$(\sigma)_1 = 242.96 \text{ MPa}$$
$$(\sigma)_2 = 312.37 \text{ MPa}$$

Taking the lesser of the two stresses, $(\sigma)_1$, the maximum permissible pressures were determined for the barrel (255 kPa) and the bottom plate (294 kPa). From the above it follows that:

$$(\sigma)_1 = 1.4[\sigma] = \frac{1.4}{2.6} R_m^T = 0.54 R_m^T$$

i.e. the adopted permissible stress is almost half as high as the ultimate strength of the material. This value of $(\sigma)_1$ is roughly 30% below the yield point. This means that a real threat of barrel failure under internal pressure will arise when the pressure in the reactor cavity exceeds the level of $1.3 \times 255 = 331.5$ kPa.

Therefore, an excessive pressure of 300 kPa in the reactor cavity may be regarded as a less conservative, realistic, criterion for reactor cavity failure.

4.8. INTEGRITY OF THE ACCIDENT LOCALIZATION SYSTEM AND COMPARTMENTS

An ALS is provided at RBMK plants of the second and third generations. An accidental release due to a break in a circulation circuit pipe will be confined in a system of leaktight compartments equipped with devices for emergency steam condensation (pressure suppression system). However, the ALS does not cover all the circulation circuit pipes. Upper parts such as steam–water lines, the top part of the downcomers, the equalizing pipes of the drum separators and all the steam lines are located outside the ALS, in compartments designed according to general building standards and rules. This means that these compartments are not leaktight and are not equally strong as most of the ALS rooms.

There is no ALS at RBMK plants of the first generation.

Analysis of the integrity of the ALS and other compartments of a power plant is an essential requirement of the safety analysis of nuclear power plants for all DBAs.

4.8.1. Power plants of the first generation

The maximum permissible pressures in compartments for MCC pipes and components are given in Table 7 for the Leningrad and Kursk power units 1 and 2.

4.8.2. Power plants of the second generation

The units of the second generation are Leningrad 3 and 4, Smolensk 1 and 2, Kursk 3 and 4, and Ignalina 1 and 2. There are two types of ALS:

(1) At Leningrad 3 and 4 and Ignalina 1 and 2, the system for condensation of accidental steam is housed in accident localization towers.
(2) At the other power plants, the system is found at lower elevations of the main building. In the case of a coolant leak, pressure relief in the ALS relies on passive condensing devices, i.e. pipes submerged under water on two decks of the PSP.

TABLE 7. MAXIMUM PERMISSIBLE PRESSURES IN COMPARTMENTS (FIRST GENERATION RBMKs)

Compartment with MCC components	Permissible (excess) pressure (kPa) at Leningrad 1, 2 and Kursk 1, 2
MCP pipes, suction and pressure headers	40.0
DGH, lower water lines (LWLs)	40.0
Steam–water lines, drum separator, downcomers and steam lines	25.0
Downcomers	40.0

The maximum permissible pressures in the ALS and other compartments at Smolensk 1 and 2 and Kursk 3 and 4 are presented in Table 8. Leaktight compartments have safety valves with an opening pressure difference of 270 kPa. Opening them allows the steam and gas mixture to be vented from the ALS to the atmosphere.

The ALSs at Leningrad 3 and 4 and Ignalina 1 and 2 operate in the following manner: in the case of MCC pipe breaks inside leaktight compartments or in DGH–LWL compartments, the steam and gas mixture will be vented into the accident localization tower (ALT) via a steam discharge passage. With pipe breaks in the upper part of the circuit (drum separator compartments or the space above the reactor), excess steam will be vented

TABLE 8. MAXIMUM PERMISSIBLE PRESSURES IN COMPARTMENTS (SECOND GENERATION RBMKs: SMOLENSK 1 AND 2, KURSK 3 AND 4)

Compartment	Permissible (excess) pressure (kPa)
DGH, LWLs	80
Leaktight compartments	440 (270)
Steam distribution passage	440
Air space of PSP	440
Air space of the enclosure	440
Drum separator compartment and space above the reactor	25.0
Central hall	5.0

TABLE 9. MAXIMUM PERMISSIBLE PRESSURES IN COMPARTMENTS (SECOND GENERATION RBMKs: LENINGRAD 3 AND 4, IGNALINA 1 AND 2)

Compartment	Permissible (excess) pressure (kPa)
DGH, LWLs	80
Leaktight compartments	300
Steam release corridor	80 (300)[a]
Drum separator compartments	25.0
Central hall	2.0 (5.0)[a]
Accident localization tower (ALT)	80

[a] Numbers in brackets refer to Ignalina 1 and 2.

directly to the atmosphere. The maximum permissible pressures in ALS compartments at Leningrad 3 and 4 and Ignalina 1 and 2 are given in Table 9.

4.8.3. Power plants of the third generation

Of the two third generation RBMK-1000 plants, one is in operation (Smolensk 3) and the other is under construction (Kursk 5). The ALSs of these units have two major distinctions from their counterparts at plants of the second generation: they have no enclosure to receive steam and gas from the reactor cavity and their PSP has only one elevation.

The maximum permissible pressures in compartments of RBMK-1000s of the third generation are provided in Table 10.

Tables 7–10 present excess pressure values that are limits for the assured integrity of building structures. However, the ALS may lose its function of a safety barrier at a lower excess pressure due to opening of safety valves in the leaktight compartments, whereupon a radioactive release may occur at upper elevations of the reactor building. The leaktight compartments and the air space of the PSP may suffer overpressure during a LOCA with failure of the PSP cooling system (loss of ultimate heat sink), when steam condensation in the pool water is less effective at water temperatures exceeding about 85°C. Therefore, either the maximum permissible water temperature (for example ≤85°C) or the operating pressure difference of the safety valves may be adopted as an acceptance criterion for the PSP.

TABLE 10. MAXIMUM PERMISSIBLE PRESSURES IN COMPARTMENTS (THIRD GENERATION RBMKs)

Compartment	Permissible (excess) pressure (kPa)
DGH, LWLs	80
Leaktight compartments	440 (270)
Steam release corridor	440
Air space of the PSP	440
Drum separator compartments and space above the reactor	25.0
Central hall	5.0

4.8.4. Permissible hydrogen concentration

Hydrogen ignition as a threat to ALS integrity may be prevented if its concentration is kept below the flammability limit.

There are two flammability limits for binary hydrogen and air mixtures: a limit for a depleted mixture (with the H_2 concentration below the stoichiometric level) and a limit for an enriched mixture (with the H_2 concentration above the stoichiometric level). These limits are roughly estimated as 4.5 vol.% H_2 and 74 vol.% H_2, respectively, at standard temperature and pressure (i.e. 298 K and 100 kPa). The maximum permissible hydrogen concentration in any single ALS compartment is taken equal to 4 vol.% in the analysis. Should this criterion value be reached, H_2 flammability must be comprehensively assessed, taking into consideration the time dependence and the characteristic boundary conditions of the accident scenario for the ALS.

4.9. MAXIMUM PERMISSIBLE RADIATION DOSES

According to the rules laid down by the national nuclear regulatory authority (see, for example, Ref. [11]), the consequences of a DBA should never result in any population exposure that would require countermeasures to protect people in the early period of the radiation accident.

The early phase (initial period) of an accident covers the time from its beginning to the time when the atmospheric release of radioactive substances is arrested. This period is assumed to be up to ten days.

According to national regulatory requirements (see, for example, Ref. [12]), the dose limits below which no urgent decisions have to be made during the early period of a radiation accident are:

— 0.5 cSv (rem) for the whole body;
— 5.0 cSv (rem) for the thyroid.

The design radius of the control area around an RBMK nuclear power plant is 3 km.

The main pathways of radiation effects on the population during this period are:

— External gamma and beta irradiation during the passage of the radioactive cloud;
— Internal irradiation through inhalation of radioactive substances.

The permissible radiation doses in DBA analysis should be confirmed with the following conservative assumptions:

(a) The radioactive release to the environment is a single event of short duration, and the release height is equal to the source altitude above ground level.
(b) The plume rise due to its buoyancy is disregarded.
(c) The radiation doses are calculated for the worst weather conditions and specific elevation of the release source, with the wind speed and atmospheric conditions producing the greatest possible near ground concentrations of radionuclides.

5. REQUIREMENTS FOR ANALYSIS

Analysis of transients and accidents is used mainly for the following purposes:

(a) Establishment and validation of design characteristics;
(b) Demonstration of compliance with the acceptance criteria to fulfil licensing requirements;

(c) Analytical support for development of emergency operating procedures and accident management programmes;
(d) Development of success criteria for probabilistic safety analysis;
(e) Analysis of operational occurrences.

Transient and accident analysis may use conservative or best estimate methods [1]. Selection of a method varies according to the purpose of the analysis.

The IAEA Safety Guide [3] suggests for licensing type analysis one of the following two approaches:

(1) Use of best estimate codes with conservative input data;
(2) Use of best estimate codes with realistic input data associated with evaluation of uncertainties.

A discussion of methods for best estimate analysis is provided in Refs [13, 14].

The following are prerequisites for the analysis:

(a) Computer codes must be qualified. The adequacy of the nodalization should be demonstrated as recommended, for instance, in Ref. [15].
(b) Integrated computer codes are preferable. For instance, analyses of reactivity initiated accidents with spontaneous control rod withdrawal should employ a three dimensional (3-D) neutronic code with a built-in multichannel thermohydraulics code in order to determine the distortion of the neutron field and the redistribution of thermohydraulic parameters in the group of fuel channels affected by the distorted power density distribution.
(c) Code users will inevitably affect the results in their selection of the computational model. Users should have the qualifications and experience needed for the code selected.
(d) There may be variations in the consideration of the availability of nuclear power plant systems, such as inclusion of a whole series of failures instead of a single failure or deliberate omission of the performance of normal operating systems from the analysis, if they mitigate the accident processes.
(e) In preparing input data for analysis, allowance should be made not only for the nominal values of nuclear power plant parameters but also for their deviations within the process tolerance range. It is also necessary to take into account possible deviations in the boundary conditions, such as system set points and characteristics.

5.1. REFERENCE STATE OF THE POWER PLANT

RBMK reactors are undergoing phased renovation and upgrading of systems with the aim of bringing their features into compliance with modern safety requirements. The neutronic characteristics of the core are being improved primarily by reducing the void reactivity feedback coefficient, which was achieved first by installing supplementary absorbers in the core, then by increasing fuel enrichment (to 2.4% ^{235}U) and, finally, by the ongoing replacement of fuel with a new mixture containing a burnable poison (erbium). A long period of time is needed to carry out the stages associated with loading of new fuel and with replacing the old CPS (CPS + emergency process protection) by new systems. It is therefore necessary to identify the specific core conditions, configurations and characteristics of plant systems that are to be considered in the analysis. Identification of the reference state of a nuclear power plant is very important for the analysis of transients and accidents.

5.2. INPUT DATA PREPARATION

The use of modern computer codes requires preparation of detailed input data. This effort involves not only selecting the appropriate information from the documents on design and operation and the other documents pertaining to a particular power plant, but also performing preliminary analyses. This is especially the case when defining the reference neutronic state of the core.

In general, the input data for a computational model of an RBMK plant are comprised of the following:

(a) The neutronic characteristics of the core, including the fuel inventory, burnup and position of the control rods, as well as the definition of the basic initial state of the core at rated (nominal) power;
(b) The thermohydraulic characteristics of the fuel channel paths between the DGH and the drum separator, with a description of the configurations, dimensions, elevations and properties of the structural materials used;
(c) The geometry and thermal characteristics of the fuel rods, with a description of the design, dimensions (with tolerances) and material properties, including the effect of irradiation;
(d) The geometry and thermal characteristics of the graphite stack, with allowances made for changes caused by operational factors, for example irradiation;

(e) The geometry and thermal characteristics of the CPS channels and control rods, including the energy released into structural components;
(f) The structural and thermohydraulic characteristics of the circulation circuit components, steam lines and feedwater pipes, including the characteristics of pumps, throttling control valves, check valves, gate valves, drum separator and de-aerator;
(g) The structural and thermohydraulic characteristics of the cooling circuit for the CPS channels and those of the reflector cooling channels;
(h) The characteristics of the ionization and fission chambers;
(i) The characteristics of the system for monitoring the integrity of the fuel claddings;
(j) The characteristics of the information and monitoring system;
(k) The characteristics of the system for monitoring the power density distribution;
(l) The characteristics of the integrated monitoring, control and protection system, including a description of the logic and sensors;
(m) The characteristics of the emergency core cooling system with a description of the operation algorithms;
(n) The key characteristics of the ALS, i.e. those of the leaktight compartments, valves and condensing devices, as well as the performance of the ALS during accidents;
(o) The key characteristics of the MCC overpressure protection system;
(p) The key characteristics of the reactor cavity venting system;
(q) The characteristics of the compartments housing the piping outside the leaktight ALS.

5.3. DESIGN BASIS ACCIDENT SCENARIOS

The methodology for accident analysis should include identification of the initiating event and the subsequent scenarios and selection of conservative initial and boundary conditions (Section 6).

When considering scenarios for DBAs, including transients with additional equipment failures, the following aspects need to be taken into account:

(a) Initial nuclear power plant conditions and their uncertainties;
(b) The basis for selection of initial conditions and parameter values (for example, reactivity coefficients);
(c) Single failure criteria and consequential failures;
(d) Assumed operator actions;

(e) Identification of the desired method and the assumptions made in the analysis;
(f) Availability of systems and equipment;
(g) Time delays for instrumentation and/or actions;
(h) Impact of plant control system and its influence on plant protection actuation;
(i) List of all set points and other appropriate boundary conditions, such as system characteristics.

6. SELECTION OF INITIAL AND BOUNDARY CONDITIONS

6.1. INITIAL CONDITIONS

The initial and boundary conditions for analysis can be adopted on the basis of either a realistic approach or a conservative approach. The choice is made within the range of possible values depending on the operating mode and on the ranges of the parameters in question. Such ranges may include technological tolerances, calculation and/or measurement errors.

Depending on the purpose of a specific analysis, the input parameters can be set at different ends of the range. For instance, for a conservative assessment of the coolability of the core channels during a LOCA, the input parameters will be set to have the smallest coolant inventory in the circulation circuit on the one hand, and the greatest rate of discharge through the break on the other, the longest possible delays in ECCS water delivery and its lowest possible flow rate. The greatest variations in neutron flux distribution allowed by the operating regulations should be taken as an initial condition in the neutronic calculation. Since the void effect of reactivity is greatly dependent on the reactivity margin (in effective control rods), the least permissible value of the margin should be used in the analysis.

The choice of initial conditions will depend on the safety aspects and acceptance criteria considered for analyses. Table 11 may serve as a guide to selecting initial conditions for analyses of accidents associated with degradation of heat removal from the core and of accidents leading to a pressure rise in the coolant circuit.

TABLE 11. TYPICAL INITIAL CONDITIONS FOR ANALYSES OF RBMK ACCIDENTS

Parameter	Conservative values	
	Analysis of fuel channel coolability	Analysis of pressure behaviour in the circuit
Reactor power	Max.	Max.
Residual heat release in the core	Max.	Max.
Coolant flow rate in the circuit	Min.	Min.
Feedwater temperature	Max.	Max.
Pressure in drum separators	Max.	Max.
Coolant level in drum separators	Min.	Max.
Feedwater flow rate	Corresponding to the power	Corresponding to the power
Power peaking factors	Max.	Max.
Reactivity margin in effective control rods	Min.	Min.
Worth of CPS rods in terms of arresting fission reaction in the core	Min.	Min.

6.2. NEUTRONIC PARAMETERS

The reference (initial) core state needs to be established for the analysis of accidents, especially reactivity initiated accidents.

The actual core characteristics of the reactor under consideration should be taken into account together with the maximum permissible deviations of the parameters, first of all characterizing neutronic field non-uniformity.

The reference core state is established with regard to:

— Power level and power distribution;
— Operating reactivity margins in terms of effective control rods;
— Number of fuel channels with the variation in fuel enrichment in different regions;
— Number of fuel channels with uranium–erbium fuel;
— Number of additional absorbers.

All the reactivity effects are calculated for the reference state of the core.

6.3. INSTRUMENTATION AND CONTROL

The time lag that exists between the generation of appropriate signals and the start of actions must be specified. The analysis involves conservative time lag values, i.e. maximum time intervals either given in the system specifications or based on specific tests.

6.4. ACCIDENT LOCALIZATION SYSTEM

The initial conditions of the ALS, i.e. the pressures in the compartments and the air and water temperatures, should be taken (within their permissible ranges) so as to have the least heat capacity of the system and thereby the fastest temperature rise. On the other hand, for assessment of the radioactive release outside the ALS, compartment leakages should be set at their maximum within the given range, even if the pressures in the compartments become somewhat lower in this case.

6.5. RADIOACTIVITY SOURCE TERM

In assessing the source term, it is necessary to take into account the operational experience and specific design features of RBMKs such as on-line refuelling. Long operational experience shows that the reference (initial) activity of the radionuclide ^{131}I in the coolant is normally well below the operational limit. It is obvious, however, that there is no direct relation between the number of leaking fuel rods and the operational limit for the radioactivity. On the basis of operational experience with RBMK reactors, it can be stated that:

(a) Rare surges of ^{131}I activity in the coolant beyond the operational limit were caused by the appearance of individual rods with a direct fuel–coolant contact, rather than by a large increase in the number of leaking fuel rods.
(b) Not a single case has ever been observed throughout the operation time of any RBMK in which more than one rod at a time developed a leak with direct exposure of its fuel to the coolant (an event with ^{131}I activity in excess of the operational limit).
(c) Any fuel assembly with a leaking rod is promptly detected by the system for fuel cladding integrity monitoring and is immediately removed from

the reactor, whereupon the level of ^{131}I activity in the coolant falls to values below the operational limit.

6.6. PARAMETRIC ANALYSIS OF THE REACTOR CAVITY VENTING SYSTEM

It has been repeatedly shown in previous RBMK safety analyses that none of the DBAs would lead to MPTRs. Should analysis of any BDBA encounter a sequence of events with the possible rupture of more than one fuel channel, the extent of the MPTR should be determined and the maximum capacity of the reactor cavity venting system analysed in order to assess the likely structural and mechanical consequences of this BDBA, including the reactor cavity integrity.

The methodology for assessing the venting capacity of the reactor cavity venting system involves modelling of the system thermohydraulics for an MPTR with various boundary conditions. Parameters conservative from the point of view of the maximum venting capacity should be selected on the basis of a phenomenological analysis. The ranges of possible variations of such parameters need to be justified. Consideration should be given to the parameters affecting the flow rate of steam entering the stack and steam generation in the case of pressure tube ruptures, as well as to the parameters that influence the rate of steam venting from the reactor cavity under various boundary conditions.

Phenomenological analysis of the effect of various parameters on the maximum venting capacity allows a classification of them into key parameters and those of secondary importance. Variation ranges are determined for the former, while the latter have the ‘worst’ values set for them, i.e. the values detracting from the maximum capacity of the reactor cavity venting system.

The maximum venting capacity should be defined in terms of the number of broken pressure tubes and the resultant pressure in the reactor cavity. The parametric results should demonstrate compliance with the acceptance criteria. Analysis of the maximum venting capacity should determine the dependence of the number of broken tubes on such an important parameter as the pressure in the coolant circuit.

7. DISCUSSION OF EVENTS

7.1. ANTICIPATED OPERATIONAL OCCURRENCES

7.1.1. Initiating events leading to reduction in coolant flow

7.1.1.1. Initiating events

The reactor may be adversely affected by events that lead to an abrupt reduction of the coolant flow when there is a risk of a significant mismatch between flow rate and power. These events include MCP seizure, tripping of one or several MCPs, failure of the MCP throttling control valve in the closed position and break-off of the DGH check valve disc. Startup of an idle MCP is a routine action with no hazard to the reactor facility.

7.1.1.2. Safety aspects

Main circulation pump seizure or trip leads to coolant flow reduction. The reactor CPS should restrict the degree and duration of the flow–power imbalance. The system will respond by reducing the reactor power or by shutting it down, depending on the power level at which the initiating event occurs. Analysis should explore the implications of the imbalance for the thermal conditions of fuel channels, i.e. the possibility of violating the safe operating limit in terms of the critical power ratio of the fuel channels.

Main circulation pump trip in the group of pumps serving the common pressure header results in overloading of the pumps remaining in operation. Assessment of the possibility of failure of these pumps due to cavitation is an essential component of the analysis to be made for these initiating events.

If the isolating disc has broken away from the check valve at the DGH inlet, the coolant flow will decrease in the channels of this DGH since water flow can cause this separated disc to be pressed against a cruciform restraint in some unidentified position. The increase in hydraulic resistance leads to a reduction in channel flow.

7.1.1.3. Specific suggestions for analysis

Reduction of the critical power margin should be assessed in the analysis.

7.1.2. Events related to performance of turbogenerators

7.1.2.1. Initiating events

This group includes failure of one or two turbogenerators, loss of the main heat sink and turbogenerator load surge.

7.1.2.2. Safety aspects

Turbine trip causes a pressure increase in the coolant circuit due to abrupt interruption of steam flow. Therefore, consideration must be given to the performance of the systems for emergency pressure relief in the coolant circuit. Although the worst conditions arise in the case with simultaneous trip of two turbogenerators at full power, a detailed analysis of this process may prove superfluous as this case is fully covered by the scenario for loss of the main heat sink, i.e. trip of two turbogenerators with simultaneous failure of the turbine condensers. As the supply of in-house loads (including MCPs and feedwater pumps) is not interrupted, due to automatic switching to the external grid, the analysis should focus primarily on the pressure dynamics in the circulation circuit and on the possibility of exceeding the safe operating limit and/or the acceptance criterion for pressure in the coolant circulation circuit.

Generator load surge is attributed mainly to the reduction in frequency of the grid. The resulting increase in steam consumption by the turbine will cause the pressure to drop in the coolant circuit, with the appearance of steam possible at the MCP inlets, cavitation failure of these pumps and impaired thermal conditions of the fuel channels.

7.1.2.3. Specific suggestions for analysis

Special attention should be paid to the pressure dynamics and the possible cavitation failure of the MCPs.

7.1.3. Loss of alternating current power supply

7.1.3.1. Initiating events

Loss of in-house electric power supply results in tripping of all pumps (for example MCPs, feedwater pumps and service water pumps) and of both turbogenerators.

7.1.3.2. Safety aspects

With the transition of coolant flow from forced to natural circulation, a flow–power mismatch is likely to occur. Turbogenerator trip leads to a pressure increase that should be restricted by operation of MSVs. It will take some time for the emergency pumps to deliver water to the coolant circuit as the diesel generators will first have to be started and loaded.

7.1.3.3. Specific suggestions for analysis

Consideration should be given to the pressure dynamics in the coolant circuit and to the behaviour of the fuel rod and pressure tube temperatures.

7.1.4. Events related to feedwater supply

7.1.4.1. Initiating events

Such initiating events include loss of feedwater, excessive feedwater flow and reduction of feedwater temperature.

7.1.4.2. Safety aspects

Total loss of feedwater flow may be caused by de-energization of the feedwater pump motors and by failure of the oil cooling of these pumps, as well as by inadvertent closing of isolating and/or control valves on the feedwater lines. The transient is characterized by a pressure drop in the coolant circuit and by a reduction of the MCP cavitation margin.

Excessive feedwater flow may result from improper operation of the feedwater controlling device. The fast controlled power reduction and FPR systems protect the reactor against overfilling of the coolant circuit, acting on the corresponding set points for water levels in the drum separators.

A reduction of feedwater temperature due to failure of the heaters leads to an increase in steam flow through the fast acting steam dump valves to the de-aerators. However, cavitation can occur in the feedwater pumps, tripping them as a result of the pressure drop in the de-aerators.

7.1.4.3. Specific suggestions for analysis

The rate of feedwater flow reduction is different in all these cases. For analysis it is practical (even if unrealistic) to consider the flow to decrease to zero so as to produce an enveloping scenario for the transient.

Consideration needs to be given to pump cavitation conditions and to the reactor and pressure tube temperatures.

7.1.5. Excessive steam discharge from drum separators

7.1.5.1. Initiating events

Such a situation is possible in the case of pressure controller failure or improper operation of the fast acting valves (safety relief valves) dumping steam to the condensers.

7.1.5.2. Safety aspects

In a similar way to the generator load surge discussed in Section 7.1.2, this event is also associated with a sharp increase in the steam flow from the drum separators. It is necessary to compare the possible rate and degree of pressure reduction, given an excessive steam discharge, with those found during generator load surges.

7.1.5.3. Specific suggestions for analysis

The case with the greatest pressure drop may be regarded as an enveloping scenario.

7.1.6. Spurious operation of the ECCS

7.1.6.1. Initiating events

In this case, cold (about 40ºC) water enters the coolant circuit. Owing to this event, water subcooling at the fuel channel inlets will become larger, leading to an increase in the coolant density resulting from both a lower water temperature and a smaller void fraction in the fuel channels.

7.1.6.2. Safety aspects

Depending on the core conditions (power level), the reactivity feedback coefficient due to the coolant density may either be negative or slightly positive. Ingress of gas into the core is likely during injection by the hydroaccumulators. This also adds to the coolant density variations, which are bound to affect the reactivity.

7.1.6.3. Specific suggestions for analysis

Considering the ECCS characteristics and operating procedures (including water delivery to one or both sides of the core), the initiating event should be analysed using a 3-D neutronics code coupled with multichannel thermohydraulic models.

Primary attention should be paid in this analysis to the reactor power behaviour and to the performance of the CPS.

7.2. DESIGN BASIS ACCIDENTS

7.2.1. Loss of coolant accidents

7.2.1.1. Initiating events

These events include postulated breaks within the reactor coolant circuit ranging in size from small breaks up to guillotine breaks of about 800 mm in equivalent diameter, as well as failure of MSVs to close.

7.2.1.2. Safety aspects

Loss of cooling accidents have several safety aspects that can jeopardize compliance with the acceptance criteria:

(a) Cooling of fuel channels may become worse despite reactor shutdown; fuel rods and pressure tubes may heat up and experience thermomechanical strain and failure.
(b) Ballooning of fuel claddings can impair emergency cooling of fuel channels.
(c) At higher temperatures (>800ºC) and in the presence of steam the steam–zirconium reaction is accompanied by generation of hydrogen that can escape through a pipe break into the ALS compartments and present fire and explosion hazards.
(d) Discharge of the high energy coolant into the compartments causes vigorous pressurization that may be a threat to the integrity of the plant building structures.
(e) Excessive pressure due to the radioactive steam and air mixture in the ALS compartments results in the escape of radioactive substances to the environment.

(f) High velocity coolant flows from pipe breaks can produce forces that will be dangerous to the integrity of the ALS structures and adjacent pipelines. The analysis of these effects should follow appropriate guidelines that are beyond the scope of this report.

7.2.1.2.1. Accident localization system considerations

When the behaviour of the thermohydraulic parameters of the ALS is analysed, all possible heat sources and sinks should be taken into consideration, especially if the time intervals are long enough (several hours).

The heat sources are:

(a) Residual heat release manifesting itself in coolant evaporation;
(b) Heat stored in the coolant circuit components;
(c) Metal–water reactions;
(d) Possible hydrogen combustion.

The heat sinks are:

(a) Condensation in the pressure suppression system;
(b) Metal structures (including wall linings) in the ALS compartments;
(c) Concrete building structures;
(d) Operation of sprays;
(e) Operation of the recirculation ventilation systems, including the hydrogen venting system;
(f) Operation of the ECCS, when subcooled water enters an ALS compartment through a pipe break.

The following safety aspects are associated with the ALS during LOCAs:

(a) Given excessive pressure in the ALS, radioactive substances escape into the environment via existing gaps.
(b) Hydrogen buildup and the combustion mixture are a potential challenge to ALS integrity.
(c) Rarefaction inside ALS compartments due to intensive condensation of steam may result in implosive conditions.
(d) Dynamic processes in the PSP represent a potential impact on structures.
(e) Pipe break outside the ALS leads to a pressure rise in compartments, operation of relief devices and release of the steam–water–air mixture to the atmosphere.

The acceptance criteria and other assumptions for the analysis should be selected in a way that is consistent with the safety aspects under consideration.

7.2.1.2.2. Main circulation circuit considerations

For analysis of the coolant circuit in the case of LOCAs, consideration should be given to the following:

(1) The location and size of the pipe break should be chosen with the worst possible consequences in mind (the largest temperature increase of the fuel channels).
(2) The emergency protection system is assumed to miss the first signal, and the lowest tolerable characteristics are taken for the ECCS.
(3) Usually LOCA events are examined assuming coincident loss of the in-house power supply. Failure to start one diesel will then be a typical assumption of a single failure. This assumption leads, in turn, to failure of one train of the ECCS pumps.
(4) All sources of heat in the coolant circuit need adequate consideration, with realistic time characteristics.
(5) Detailed nodalization is required for the area near the break so that ECCS water carry-over to the break can be adequately modelled.
(6) The internal parts of drum separators, including the system for measuring the water levels in drum separators, should be subject to detailed modelling. Proper modelling of this section, together with the level measurement system of the drum separator, is of great importance as the level signals are used in initiating the operation of the plant protection systems.
(7) Modelling of MCPs should include the characteristics of the four quadrants.
(8) Assessment of the radioactivity source term should be made with regard to the operational limit for leaking fuel rods, plus one rod with direct exposure of fuel to the coolant.

7.2.1.3. Specific suggestions for analysis

In analysing thermohydraulic processes in the ALS, especially over long time intervals, it is important to use best estimate codes. This allows adequate modelling of the strong interrelations between the key thermohydraulic parameters and various local effects (such as thermodynamic non-equilibrium conditions).

The degree of nodalization should be adapted with regard to the possible phenomena, such as stratification of the steam and gas mixture and formation of natural convection paths.

In general, it is recommended that parametric calculations be performed for the key factors. This sensitivity analysis should cover at least the following parameters:

— Nodalization;
— Coefficients of hydraulic resistance between computational volumes;
— Size of the break;
— Geometry and properties of fuel rod materials;
— Neutronics dependent fuel parameters, including power peaking factors and residual heat rate;
— ECCS flow rate and actuation time;
— Response time of the emergency protection (shutdown) system;
— Heat exchange with structures and components inside the ALS;
— Water discharge through gaps in the ALS.

7.2.2. Reactivity initiated accidents

7.2.2.1. Initiating events

Reactivity initiated accidents include events such as uncontrolled withdrawal of CPS rods, erroneous refuelling and voiding of the CPS cooling circuit.

7.2.2.2. Safety aspects

Reactivity initiated accidents in the large cores of RBMKs are accompanied by both global and local power variations, with the local effects dominating in a number of cases. Therefore, a meaningful analysis is only possible using 3-D neutronic codes. The methodology for assessing the response of the reactor and its systems to reactivity events is based on integrated dynamic calculations, including mutually consistent 3-D neutronic and thermohydraulic calculations of fuel and graphite temperatures with regard to CPS operation.

Withdrawal of a group of rods from the core causes a great variation in reactivity with the ensuing early generation of emergency protection signals.

Erroneous reloading of a fuel assembly is typically considered as an event in which a fresh fuel assembly is placed in a cell adjacent to a fuel channel that has low burnup and, therefore, high power. Such insertion of reactivity due to loading of fresh fuel results in an increase in fuel channel power in this region in excess of the operational limit. Analysis should be aimed at assessing the probability of violating the safe operating limits, as well as the probability of balancing out the increase in power by CPS rods or reactor scram.

The addition of positive reactivity due to voiding of the CPS cooling circuit will not be fast due to the fact that, even in the case of the largest possible pipe break in the lower part of the circuit, it will take several tens of seconds for the water to leave the core. The emergency signals of the protection systems will be generated within this time interval. Compliance with the safe operating limits or acceptance criteria for fuel rods should be verified.

7.2.2.3. Specific suggestions for analysis

Analysis should verify the effectiveness of the reactor protection systems.

7.2.3. Fuel handling accidents

7.2.3.1. Initiating events

These events include all the initiators resulting from fuel handling accidents, as listed in Table 2.

7.2.3.2. Safety aspects

Analysis of accidents during fuel handling and transfer operations inside a power unit should assess:

(a) The possibility of mechanical damage caused to fuel assemblies and rods;
(b) The adequacy of fuel cooling in abnormal conditions during fuel handling operations;
(c) The capability of equipment and personnel to cope with such abnormal conditions and, in doing so, to meet the acceptance criteria for radiological consequences.

7.2.3.3. Specific suggestions for analysis

The preserved integrity of fuel claddings and compliance with the dose criteria are indicative of an acceptable response of the plant to fuel handling accidents.

7.3. BEYOND DESIGN BASIS ACCIDENTS

7.3.1. Anticipated transients without scram

7.3.1.1. Initiating events

Anticipated transients without scram are the most significant group of BDBAs. These include failures of the reactor protection system in addition to the following anticipated operational occurrences:

— Loss of feedwater;
— Reduction of flow in one loop of the cooling circuit;
— Loss of alternating current power supply;
— Loss of the main heat sink;
— Maximum reactivity increase, owing to continuous withdrawal of a single control rod.

7.3.1.2. Safety aspects

Owing to complete loss of the emergency shutdown function, there is an imbalance between heat production and heat removal from the reactor core. This imbalance leads to a deterioration of the core cooling and a pressure increase in the reactor coolant system. The analysis should determine the time window available either for restoration of the shutdown function or for activation of the independent shutdown system. Owing to the lack of effective inherent safety features of RBMKs, a second shutdown system is very important for preventing loss of reactor coolant system integrity and for ensuring safe long term shutdown and sufficient heat removal capacity.

7.3.1.3. Specific suggestions for analysis

Owing to the very low probability of the initiating event, its further worsening by additional equipment failures is unlikely. Therefore ATWS modelling may adopt the following principles:

(a) All systems, with the exception of the failed reactor protection system, are assumed to be available.
(b) The systems actively involved in normal operation remain active during the accident.
(c) The systems that have no effect on reactor power, which are activated by emergency signals, are available.

The initial and boundary conditions should be selected such that the system response will be the most challenging with respect to the relevant acceptance criteria.

In modelling ATWS accidents, it is essential to determine the critical points in time for intervention, i.e. when the recovered shutdown function is able to reduce the consequences to an acceptable level. It is necessary to determine the time when FPR and emergency protection signals are generated and when the set points are reached for warning signals to appear at the operator's console.

The degree of plant safety from emergency protection failures can be inferred from the number of FPR and emergency protection signals generated during an accident and from the time taken by parameters to reach the acceptance criteria.

7.4. GENERAL RECOMMENDATIONS

Best estimate codes in combination with conservative input data are preferably to be used for analyses, in particular for use in the safety analysis report. Special attention should be paid to the assessment of uncertainties. The current status of the uncertainty assessment methodology is described in Ref. [13].

RBMK reactors have large cores, where local effects play a significant role in safety analyses of the several initiating events. For this reason, it is appropriate to use a 3-D neutronics code with a built-in multichannel thermohydraulic model. In addition, a 3-D neutronics code should be employed for compiling input data for point neutron kinetics (commonly involved in thermohydraulic codes) when events such as LOCAs are analysed.

Considering the fact that the MCC is divided into two symmetric loops, a pipe break in one of these directly affects the thermohydraulic and neutronic behaviour on one side of the core. The resulting asymmetry in reactivity and power has to be assessed by a 3-D code. Optionally, the information obtained can be compiled as input for use by a point kinetic model, which, however, usually requires some iterations.

User effects constitute a significant aspect of the analysis [14]. Ways to reduce these effects are discussed in Ref. [16].

Accident analysis of any of the events should justify the proper selection of conservative initial and boundary conditions, single failure, actions of the operator and plant control systems and other relevant assumptions.

Transient and accident analyses for RBMK reactors should reflect the peculiarities of the design, such as the use of adequate correlations for critical heat flux, and consideration should be given to hydrodynamic instabilities.

All transient and accident analyses should be carried out for a sufficient time period in order to demonstrate that a safe and stable configuration has been achieved after the event.

As noted in Ref. [1], quality assurance should be addressed at various stages of the process, from verification of codes to interpretation of analysis results and their uncertainties.

8. PRESENTATION OF RESULTS

The results of analyses are the calculated plant parameters — both global (MCC) and local (individual fuel channels) — determined as a function of time. There are some key parameters that are of interest as they reflect the nature and complexity of the accident process. This set of parameters to be identified in each particular case should allow for:

— A fundamental understanding of the development and interrelations of the processes and phenomena involved in the accident;
— The possibility to compare any relevant parameter with acceptance criteria;
— Knowledge of the overall process duration until the plant reaches a new safe and stable state.

Graphic and tabular presentations of the accident progression are more convenient for interpretation of the results. A typical table with the characteristics of the accident process will list the events (for example, actuation of devices and parameters reaching their maximum and/or minimum values) in chronological order, supplemented with additional comments if necessary.

It will be easier to understand variations in parameter behaviour, especially those involving a stepwise change, if the time when equipment is switched on or off, the time at which protective features come into action, etc., are indicated in a plot showing time dependent parameter variations.

The following are a typical set of parameters to be presented as a function of time:

(a) Global parameters:
- Pressure in the drum separators (on the affected and intact MCC sides);
- Coolant flow rate in characteristic areas of the MCC;
- Steam flow rate to turbines;
- Water levels in the drum separators (on the affected and intact sides of the MCC);
- Thermal power of the reactor (including fission and decay power);
- Feedwater temperature (on each side of the MCC);
- Feedwater flow rate (on each side of the MCC);
- Break flow rate and enthalpy (for LOCAs);
- ECCS flow rate;
- Total core reactivity, as well as its feedback components from the Doppler effect, the moderator, temperature changes, coolant voiding and the effect of xenon.

(b) Local parameters:
- Core inlet temperature;
- Flow rates in fuel channels;
- Void fraction in fuel channels;
- Maximum temperatures of claddings and fuel;
- Maximum temperatures of fuel channel (pressure) tubes;
- Maximum temperature of graphite;
- CPS flow rate;
- CPS exit temperature.

If additional analysis is performed, the set of plots should be supplemented with the following:

- Thickness of the oxidized layer on fuel claddings;
- Axial distributions of cladding and fuel temperatures;
- Total fraction of zirconium reacted and hydrogen produced;
- Maximum heat flux and the margin to critical heat flux.

If transport of radionuclides is involved in the analysis, the list of reported parameters should include:

- Concentration of radionuclides in the fuel, coolant and reactor coolant structures;
- Concentration of radionuclides in the ALS compartments (airborne, deposited, etc.);
- Amount and composition of radioactive releases to the environment.

If a 3-D neutronic code coupled with a multichannel thermohydraulic model is used, it is appropriate to use post-processing capabilities for visualization of the results.

The plots demonstrating variations in the parameters important for safety evaluation should have a timescale with sufficient resolution along the time axis.

The parameters characterizing the thermohydraulic response of the ALS should include:

— Pressure in compartments;
— Steam and air mixture temperatures in compartments;
— Hydrogen concentration;
— Rates of fluid flows between compartments and through gaps;
— Air concentration in compartments;
— Water temperature in the PSP;
— Liquid quantities in compartments;
— Thermal power removed by various heat sinks including suppression pool and passive structures.

Similarly as for the coolant circuit and core channels, the plots for the ALS parameters should also indicate the time at which the switching on and off of the equipment took place.

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This report provides guidance on accident analysis for the design features specific to RBMKs. In particular, advice is given regarding categorization of initiating events, selection of acceptance criteria, and initial and boundary conditions. The report is intended primarily for analysts coordinating, performing or reviewing computational analyses of transients and accidents for RBMKs, on both the utility and regulatory sides.

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