## APPENDIX C

# 1. Accident Analysis for WWER

### 1.1 Aim of the analysis

According to the Federal German codes and standards, safety precautions must ensure that if an accident occurs the residual heat of the reactor can be safely removed, the reactor can be shutdown, that long-term subcriticality can be maintained and radiation exposure of staff and environment can be kept as low as reasonably achievable and also below the dose limits determined by the regulations of the Atomic Energy Act and the subordinated ordinances, taking into account the state of the art. Additionally, for many accidents it is required that further protective targets are met. Thus it must be demonstrated for accidents or incidents with a higher probability of occurence that the heat flux densities at the fuel-rod-cladding tubes are sufficiently remote from the critical heat flux density, that the release of energy in the fuel rods is so low that melting is avoided and the pressure in the primary system is so low that safety valves do not open.

To prove precautions against inadmissible effects of accidents, an accident analysis is to be performed for the plant under consideration, in which sequence and effects of the accidents are investigated. The qualification of the methods of analysis and of the computing programs must be verified with tests in experimental plants or experiments in the reactor plant. The requirements and the boundary conditions for accident analysis are conservatively defined by the Federal German codes and standards.

During the last years very intensive work has been performed in the western countries concerning different safety studies for the East European NPP. Considerable know-how has been transfered to the eastern countries, especially in the field of reactor safety analysis. In all of the performed analysis in GRS the bases for comparison has been the German rules and standards which in some aspects are rather different from the Russian ones, that have been adopted also as national rules in all eastern countries where Russian NPPs with WWER are in operation.

On the basis of the German experience, international practice and also IAEA recomendations and Russian radiological requirements a list of initiating events for

NPP with WWERs has been generated. Till now exists no final PSA for WWER therefore the list should be considered preliminary. Some of these events were not the original design accidents for existing WWERs, especially for older plant types (i.e. WWER-440/230). This fact should be taken into account when using recommendation of this document, concerning conservative assumptions, boundary conditions and acceptance criteria.

## 3.1 List of Design Basis Accidents

According to the western international practice the accidents are grouped according to the main physical processes involved in the accident sequence. The following event groups (EG) are used:

**A** Increase in Heat Removal by Secondary System

**B** Decrease in Heat Removal by Secondary System

**C** Decrease in Reactor Coolant System Flow Rate

**D** Reactivity and Power Distribution Anomalies

**E** In crease in Reactor Coolant Inventory

**F** Decrease in Reactor Coolant Inventory

**Additional accidents and issues (special group of cases):**

**G** Anticipated Transients without Scram (ATWS)

**H** Confinement

**I** Accidents in Shutdown conditions

**K** Pressurized Thermal Shock (PTS)

**L** Radioactive Release from a System or Component

The events are classified according to the frequency of occurence of the initiating event. Since complete and reliable probabilistic studies for VVER-1000 and VVER-440 are not yet available, the categorisation is preliminary. It has been performed taking into account international experience and recent results. Still there is no categorisation for accidents in shutdown. The following event categories (EC) are suggested:

1st category: includes normal operating conditions. In this class, there is no actuation of scram; however, spurious actuation of the emergency protection (reactor scram) can happen and it is classified in this category.

2nd category: includes abnormal transients which present deviations with regard to the normal operating conditions. The transients of this second group can lead (but not necessary) to reactor scram. Transients of this category belong to the design basis accidents.

3rd category: includes most of the remaining design basis accidents like control rod ejection, LOCA, main steam line break, etc.

4th category: includes beyond design basis conditions.

The initiating events within the different groups and categories are as follows:

**Increase of Heat Removal by Secondary System**

| **Initiating Events** | **Event Category** |
|---|---|
| Spectrum of steam line breaks including the worst case with respect to recriticality. The following parameter must be varied: initial power (from hot standby to full power), with and w/o loss of power, break spectrum (from 0.1 A to double ended guillotine break), with and w/o additional steam generator tube leakage, break location (between SG and MSIV, main steam header and inside containment), the following cases are examples<br>- 2 A break size (with and without SG tube damage)<br>- 0,5 A break size<br>- 0,1 A break size | 4 or 3<br>3<br>3<br>3 |
| Inadvertent opening and stuck-open of valves, including the variation of additional parameters as above, as far as these are not covered by the spectrum of steam line breaks, for example<br>- SG safety valve<br>- Turbine bypass valve (BRU-K)<br>- SG relief valve (BRU-A) | <br>3<br>3<br>3 |
| Feedwater system malfunction at nominal power that results in a<br>- reduction of FW temperature<br>- increase of FW flow<br>The worst case with respect DNBR, at low power (hot stand-by) | <br>2<br>2<br>2 |
| Malfunction of main steam pressure control which results in increasing steam flow | 2 |

| | |
|---|---|
| Long term failure of main heat sink with operational leakage at SG tubes | 3 |

**Decrease in Heat Removal by Secondary System**

| **Initiating Event** | **Event Category** |
|---|---|
| Failure to reach house load | 2 |
| Loss of feedwater supply<br>- with auxiliary feedwater supply<br>- with emergency feetwater supply only | <br>2<br>2 |
| Closing of SG shut-off valves (MSIV) | 2 |
| Turbogenerator disconnection from power | 2 |
| Loss of offsite power<br>- short term<br>- long term | <br>2<br>2 |
| Break of feedwater pipeline<br>- non isolable<br>- isolable<br>- main feedwater header break | <br>3<br>3<br>3 |

| | |
|---|---|
| Malfunction of main steam pressure control which results in decreasing steam flow | 2 |

**Decrease in Reactor Coolant System Flow Rate**

| | |
|---|---|
| Seizure of one reactor coolant pump (RCP) | 3 |
| De-energization of one RCP | 2 |
| De-energization of all RCPs | 2 |
| RCP shaft break | 3 |
| Coolant flow rate decrease as a consequence of low grid frequency | 2 |

**Reactivity and Power Distribution Anomalies**

| Initiating Event | Event Category |
|---|---|
| Uncontrolled withdrawal of a group of control members at<br>- nominal power<br>- low power (subcritical, start-up) | <br>2<br>2 |
| RCP start in a loop being inoperative before (erroneous start up) | 2 |

| | |
|---|---|
| Ejection of most effective control member at<br>- full power<br>- intermediate power<br>- low power | <br>3<br>3<br>3 |
| Decrease of boric acid concentration (e.g. by means of CVCS malfunction) | 2 |
| Drop of control rod | 2 |
| Cold water injection (e.g. by bypassing the heat exchanger in the CVCS) | 2 |

**Increase of Reactor Coolant Inventory**

| | |
|---|---|
| CVCS malfunction (or operator error) which results in an increase of reactor coolant inventory | 2 |
| High pressure borated water system malfunction (TQ14, 24, 34) (inadvertent operation) | 2 |

**Decrease in Reactor Coolant Inventory**

| **Initiating Event** | **Event Category** |
|---|---|
| Inadvertent opening of pressurizer pulse safety device | 3 |

| | |
|---|---|
| Break of pulse tube of instrumentation outside the containment or other pipelines carrying primary coolant outside containment | 3 |
| Break of primary system make-up-blowdown pipeline | 3 |
| LBLOCA spectrum, for instance:<br><br>- 2A RCP-RPV, $C_D$ = 1.0<br><br>- 2A RCP-RPV, $C_D$ = 0.8<br><br>- 2A RCP-RPV, $C_D$ = 0.6<br><br>- 2A RPV-SG, $C_D$ = 1.0<br><br>Other LBLOCA (break of surgeline, ECC-injection lines) | 3 or 4<br><br><br><br><br><br><br><br><br><br>3 or 4 |
| LBLOCA Analysis with N2 injection | 3 or 4 |
| SBLOCA spectrum | 3 |
| Break of SG heat exchange tube including radiological consequences | 2 |
| SG collector cover break (100 mm) with radiological consequences | 3 |

**Anticipated Transient without Scram (ATWS)**

| **Initiating Event** | **Event category** |
|---|---|
| ATWS Total Loss of Main Feedwater | 3 or 4 |

| | |
|---|---|
| ATWS Loss of Normal Onsite and Offsite Power (LONOP) | 3 or 4 |
| ATWS Inadvertent control assembly withdrawal | 3 or 4 |
| ATWS Loss of condenser vacuum with and without LONOP | 3 or 4 |
| ATWS Inadvertent start-up of pressurizer spray | 3 or 4 |
| ATWS Inadvertent opening / stuck-open of a pressurizer safety valve | 3 or 4 |
| ATWS Maximum increase of steam flow (opening of BRU-K or BRU-A, or safety valves) | 3 or 4 |
| ATWS Maximum reduction of core inlet temperature | 3 or 4 |

**Confinement**

| | |
|---|---|
| Containment Calculation for 2A-Break of main coolant line including structure analysis | 3 |
| Containment Calculation for SLB inside containment | 3 |

**Accidents in Shutdown Conditions**

| | |
|---|---|
| Spectrum of leaks/breaks on ECC suction line | 3 or 4 |
| Loss of residual heat removal system while reactor is shutdown for maintenance with minimal water inventory | 3 or 4 |
| Dilution of primary coolant circuit while reactor is in shutdown states, including the analysis of possible slug formation | 3 or 4 |
| Primary breaks occuring while reactor is in intermediate or cold shut down | 3 or 4 |

**Pressurized Thermal Shock (PTS)**

| **Initiating Event** | **Event category** |
|---|---|
| PTS analysis for stuck open pressurizer valves which close later | ? |
| PTS analysis of SBLOCA which can be compensated | ? |
| PTS analysis of secondary side leakage up to SLB | ? |
| PTS analysis of primary-to-secondary side leakage | ? |
| PTS analysis of feedwater system malfunction | ? |

**Radioactive Release from System or Component**

| | |
|---|---|
| Gaseous waste treatment system failure | 2 |
| Radioactive liquid waste system leak or failure | 2 or 3 |
| Postulated radioactive release due to liquid containing failure | 3 |
| Fuel handling accident | 3 |
| Spent fuel cask drop accident | 4 |

The case of ejection of the most effective control rod and intermediate power (see group D) is introduced additionally because of the fact that on the one hand the initial fuel enthalpy is larger at full power, on the other hand the ejected control member worth and the hot spot factor are higher at zero power. Some Western calculations, for instance for French 900 MWel PWR, have confirmed the interest for this case which could be the worst one.

The steam generator collector cover break has up to now been included within category 3 as DBA (see group F). This is also in agreement with recent information about probabilistic studies for VVER-1000/V-320. The probability of the initiating event is calculated as $3.10^{-3}$ (Balakovo NPP) .

For ATWS 9 cases have been suggested as examples. The complete treatment of ATWS should include the combination of the transients of category 2 involving the failure of reactor scram. Pressurized thermal shock analysis has been included in the list for reasons of more completeness.

### 1.2 List of Beyond Design Basis Accidents

It is recommended that the following beyond design basis accidents should be analysed in order to determine the additional measures to cope with such situations. A probabilistic approach should be used to justify the categorisation of these events as BDBA.

Combination of steam line break with one or more steam generator tube rupture

- Combination of steam line break with steam generator collector cover break
- Total loss of steam generator feedwater
- Total loss of heat sink
- Total loss of power (total blackout)
- Total loss of low pressure safety injection or containment condensation capabilities in the case where it is required
- Total loss of high pressure safety injection in the case where it is required.

Besides, it is recommended that care should also be taken to control situations leading to core melt.

The issue book IAEA 95 recommends that the work on severe accident analysis should be accelerated. It requests that the quality of the analysis should be ensured to reach an internationally acceptable level.

It is recommended that two categories of BDBA should be considered:

- prevention of core melt: studies, implementation of procedures and if necessary of measures to be formed before start-up (at least for events having the biggest consequences and estimated probability of occurence)

- mitigation if core melt: some studies have to be launched, but not finalised before start-up as well as improvement measures. Specifically, issues regarding severe accidents with significant damage of core (including also $H_2$, analysis) should be considered also.

This is in principle the same strategy as recommended for the french-german safety approach .

**1.3 Initial Conditions and Assumptions**

The initial conditions first include those variables which can be measured in the plant, either directly or by calculation. Examples of direct measurement variables are the process variables; power, pressure, temperature, and flow. Variables which are indirectly measured include the neutronics variables; peaking factors, moderator temperature coefficient, control rod worth, etc. In addition, initial conditions may include design conditions which are calculated but are not measured, such as decay heat or delayed neutron fraction. Peaking factors are both measured and calculated as part of the core design.

The values selected for the initial conditions should be those which result in the most conservative (least favourable) results with respect to the acceptance criteria being evaluated. These should normally be selected from the edge of their operating range for process variables, which may be based on for example, instrument uncertainties, plant procedures, or technical specification limits. They may also be based on a defined confidence range (e. g. 95 %) for the calculated variables. Note however, that this may not always be conservative, or may not be appropriate for the basis against which the acceptance criteria are evaluated. (Some methods incorporate the initial condition uncertainties into the limit value of the acceptance criterion.) It is the responsibility of the designer to make the appropriate determination of this.

Independent selection of all various parameters in conservative way can lead to inconsistent data, which may not be appropriate for computational analysis. If this is the case, it is recommended to select conservatively the parameter(s) with strongest influence on results regarding acceptance criterion under consideration. Remaining parameters can afterwards be specified consistently.

An example of a list of initial  conditions:

Reactor Power

Coolant Temperature

Primary Side Pressure

Reactor Coolant Flow

Steam Generator Level

Steam Pressure

Steam Flow

Feedwater Flow

Feedwater Temperature

Power Peaking Factors and Power Distribution

Linear Power in Fuel

Gap conductance

Moderator Feedback

Fuel Temperature Feedback

Delayed Neutron Fraction

Prompt Neutron Lifetime

Decay Heat

Control Rod Worth

Pressurizer level

### 1.4 System availability and functionality assumptions

The availability of the protection and safety functions is, to a large extent, determined by the application of the single failure criterion (SFC). The sense of SFC is explained later in the same chapter. The analysis should also consider those failures which could occur as a consequence of the event itself. If such failures can occur, they must be considered in addition to the single failure. For example, if a valve is exposed to an adverse environment as a result of a steamline rupture, and the valve is not qualified to perform in that adverse environment, failure of the valve should be considered if the failure results in a more conservative (less favourable) result. Generally, equipment not qualified for specific accident considerations should be assumed to fail unless its normal operation leads to more conservative results.

In addition to the single failure and any consequential failures, postulated accidents should be analysed for a loss of off-site power. The loss of offsite power may be assumed to occur either at the initiation of the event or as a consequence of reactor and turbine trip. The analyst should consider both cases (with an without offsite power) to determine which is the most limiting.

In most cases, the availability of operational control systems serves to mitigate the consequences of the event. Thus, it is often conservative to assume that the automatic features of the control system are turned off, if it is possible to turn them off. However, in certain situations, the control systems may aggravate the transient or delay actuation of the protection features. For example, the actuation of pressurise spray will reduce the pressure during an overpressurization event. This will delay the time at which reactor trip on high pressurise pressure is reached. If power continues to rise during this delay, this may produce a more conservative result. Therefore, it might be more conservative to assume that automatic pressure control is available. The analyst should investigate these situations (not necessarily performing detailed calculations) starting with the assumption of full operability of control systems.

An example of a list of system availability and functionality assumptions follows:

Reactor Control (manual or automatic)

Offsite power

Main Isolation Valve closure

Condenser Bypass (BRU K)

Atmospheric Relief Valve (BRU A)

ECCS pumps (flow rates and number of pumps)

Main Circulation Pump operation

Minimum or maximum flows (ECCS, accumulators, etc)

Containment leak rates

Diesel generators

Containment spray

Stuck Rod

Emergency Feedwater

The WWER protection system design includes four features associated with control rod actuation, AZ-1 through AZ-4. Of these, only AZ-1 (free fall of all control rods by gravity) should normally be assumed for providing protection, i. e., reactor trip. It is acceptable to assume that the AZ-1 reactor trip occurs on the first signal generated. All rods should be assumed to fall except for the most effective rod (stuck rod). The stuck rod assumption may be considered as either a deterministic requirement or as one possible single failure. AZ-2, AZ-3 and AZ-4 should be treated as control systems in that they do not function unless their actuation produces more conservative results.

The analyses should take into account conservative values for delay times of safety system actuations, for protection set-points and for critical parameters of safety system (e. g. flow rate of ECCS or safety valves).

In the German practice an additional assumption is made that one train of a safety system is not available due to maintenance (or repair). As in Russian plants maintenance is not done during the time when the safety system is needed, Russian plants are not designed according to this principle. This has been considered in the analyses here.

### 1.4.1 Single failure criterion application:

Objectives of the single failure criterion

The single failure criterion provides protection against isolated random events. Multiple failures which can result in a single event are considered to be part of the single failure. This criterion makes it possible to guarantee that the design of a safety-related system will enable it to perform its function even in the event of failure of any one of its components.

Active failures

In a mechanical or fluid system, active failure is:

- either refusal of an item of equipment, whose function necessitates mechanical movement of one of its components, to accomplish its function when ordered to do so,
- or the inadvertent operation of such equipment.

The single active failure of the type consisting of refusing to operate on demand is a fault affecting a system on standby in normal operation of the reactor, otherwise the fault would be an initiating event. This type of failure is a fault occurring after the initiating event.

Active single failure of the inadvertent operation type (for example the starting of the pump or the changing of the position of a valve) is a fault which can occur in a pre-accident period (incorrect positioning of a valve), during an accident or during the post-accident period.

Passive failure

In a fluid system, a passive failure is considered to be:

- either a leak in the shell retaining the fluid,
- or a mechanical failure affecting the fluid flow inside the system.

The passive failure considered in design of the system is a leak at a valve spindle seal or pump packing; this leakage must be detectable and isolable, its rate being conventionally taken to be 200 l/min for 30 minutes.

It is then examined whether, in the event of passive failure, considered to be a leak at any point in the shell retaining the fluid, the required safety function or functions would be provided.

If a leak is not detectable and isolable, it is necessary to consider that it is liable to develop to the point where it would be equivalent to total rupture of the pipe.

Short term and long term

a) Short term: this is the period which immediately follows the accident, during which automatic nuclear steam supply system action takes place, the response of the systems is verified, the nature of the accident is identified and the procedure to be followed is decided upon. By convention, the short term consists of the first 24 hours after the onset of the accident.

b) Long term: this is the post-accident period in which systems are operated following the short term while safeguard and safety functions are necessary. Whereas in the short term, the essential goal is to limit radioactive releases, the long term comprises, when necessary, all action taken to reach the safe shutdown state, to gain access to the reactor buildung and to repair any damaged equipment.

The single failure criterion:

A system complies with the single-failure criterion for a given function if it is capable of performing the function even in the case of:

- a single active failure in the short term,
- a single active failure or single passive failure in the long term.

For electrical systems, no distinction is made between active and passive failures. Failures are always considered to be active as concerns the application of the single failure criterion.

For the design of the safety injection and containment spray systems, there is assumed to be a leak of 200 l/min for 30 min on switching to recirculation and it is assumed that there is an undetectable or non-isolatable leak only after 24 hours from the onset of the accident.

General conditions of application

When the single failure criterion is applied, the conditions are as follows:

- the single failure is only taken into consideration for the period when accomplishment of the safety functions is required,
- allowance for a single failure affecting an item of equipment must result in the worst case studied for the operating conditions considered,
- a single active failure is postulated at the time of demand of the equipment under consideration,
- when accomplishment of the safety function necessitates the simultaneous activation of a number of systems, the single failure criterion applies to the systems as a whole and not to each separately.

Specific application conditions

- Failure of a check valve to close results in partial leakage at the seat (leakthightness not assured),
- refusal to open is not a failure to be taken into consideration as a single failure in unpowered check valves, for example flap type check valves, but constitutes an active failure to be taken into consideration in power-assisted check valves,
- safety valves are considered to be vulnerable to active failure. The nature of the failure to be taken into consideration is the following:
  - loss of leakthightness after closure subsequent to operation with water or steam if the safety valve is qualified for the operating conditions,
- for certain active equipment which is the subject of special preventive measures which require justification, it may be considered that they are not vulnerable to active failure; this essentially concerns the motor-operated valves and pumps. Similarly, the following are not considered to be active failures:

- the inadvertent closure fo normally-open valves whose states are signalled in the control room and which receive confirmation of the opening command when operation of the system is required,
- complete closure of control valves for which a by-pass or a stop which prevents total closure is provided.

Field of applicability

The single failure criterion is applied:

- in design of systems,
- in accident analyses,
- in safety analyses of the installation as a whole.

The single failure criterion is applied to safety systems only for transients of categories 2 and 3.

The stucking of the most effective control member is considered in all cases where reactor scram occurs.

- Consideration of operational systems

Operational systems are principally considered only in category 1 and category 2. In emergency conditions (category 3), they are only considered if they worsen the situation

### 1.4.2 Operator Action

In German rules for accident analysis, no credit is taken of operator actions during the first thirty minutes after the beginning of the accident. In the eastern countries this rule is not so strictly considered. For certain events, operator action is required after a certain time in order to bring the plant to a safe shutdown condition. Typically, it is assumed that no operator action occurs for the first 30 minutes of the transient or accident. In cases of clear and reliable indication shorter reaction time periods can be assumed, but not less than 10 minutes.

When the operator takes an action, it is acceptable to assume that the operator takes the correct action. It is not required to assume that the operator makes an error in his action, either as a single failure, or as an additional failure. It should be demonstrated that the operators have appropriate time, training, procedures, etc., in order to substantiate this assumption. Specific operator actions which are required should be included in the operating procedures. The effects of operator errors can be considered in probabilistic safety studies, which are outside the scope of this document.

Note that operator error can be the initiating event for a transient or accident.

### 1.5 List of additional modelling assumptions

Quality of Fluid flowing out of a break (saturation, entrainment)

Core bypass flow

Primary to Secondary heat transfer coefficients

Flow diversions (e.g. ECCS flow) to break

Boron Injection and Transport

Stored Energy (metal structures)

Delay times for actuation of reactor trip and safeguards functions

Rod Drop Time for Reactor Trip

Coolant Mixing

### 1.6 Acceptance Criteria for WWER NPPs

The main ideas in this chapter are based on the recommendations made by the IAEA in ‘Guidelines for Accident Analysis of WWER Nuclear Power Plants’ , Dec’95.

### 1.6.1 Basic Protection Aims

Transient and accident analyses are performed to confirm that the nuclear power plant is capable to cope with the whole set of anticipated transients and accidents that has been selected as a design basis or as a basis for upgrading without exceeding acceptable limits.

The main protection aims that must be considered are:

- permanent control of reactivity,
- permanent cooling of fuel elements ,
- enclosure of radioactivity
- limitation of radiation

These protection aims should be concretised through acceptance criteria, such as:

- prevention or reduction of the possibility for fuel cladding damage,
- limiting the number of damaged fuel rods or the extent of the damage,
- preventing loss of leak tightness or damage to the integrity of the primary circuit,
- preventing damage to the integrity of the containment or confinement,
- direct limitation of the radiological impact of the event, implying a compounded limitation on all barriers.

Specific acceptance criteria should be defined for each initiating event that is being analysed. The results of the analysis should be compared with the acceptance criteria to confirm that an acceptable level of safety is ensured for the event.

The assessment of the accident analyses should be based on the German codes and standards. The respective paragraphs of the BMI Safety Criteria, the Accident Guidelines of the BMI, the respective paragraphs of the RSK Guidelines for Pressurised Water Reactors and the respective KTA-Rules must be referred to here. To examine the completeness of the accident range the List of Notes with Sub-division for a Standard Safety Report for Nuclear Power Plants with Pressurised Water Reactor or Boiling Water Reactor /BMI 76/ must further be referred to (see Appendix 1). Apart from the accidents applying to the WWER listed in the previous chapter, such accidents, which result from the structural peculiarities of each NPP, compared to a West German plant are also to be analysed.

RSK Guideline 22.1 is to be referred to for assessing the design calculations relating to loss-of-coolant accidents. The following specific acceptance criteria are limited to the major criteria related to the integrity of the fuel and the primary and secondary pressure boundaries. The numerical values are examples taken from actual practice in various countries. They may vary somewhat in different countries operating WWER NPPs and therefore they should be checked against actual regulatory requirements.

Acceptance criteria for the categories 1-3 (see 1.2):

- First category: the rated maximum damage of fuel elements for normal operation, which determines the established level of activity of the coolant of the primary system, comprises the following in the number and extent of defects of the fuel elements (the first rated limits of damage to fuel elements) : 1% of the fuel elements with defects of the gas leak type and 0,1% of fuel for which direct contact of the coolant and nuclear fuel occurs.

- Second category: for the transients of this class, departure from nucleate boiling is not allowed. Consequently, DNB ratio, calculated by taking into account uncertainties, must be higher than 1.0.

- Third category: the acceptance criteria are similar to those of appendix K of FR 50:
  - fuel cladding temperature must not exceed 1200 °C,
  - local percentage oxidation of fuel cladding not more than 18% of the initial cladding thickness,
  - fraction of reacted zirconium not more than 1% of its mass in the core.

In the case of control member ejection, the following acceptance criteria is considered: enthalpy rise in the fuel is limited to 230 cal/g (New studies show that this value must be lower in case of higher burnup. Work is undergoing to proove that and is expected in the near future the value to be changed).

Besides, for the operating conditions of the three classes, primary pressure must not exceed 115% of the design pressure.

### 1.6.2 Controlled Shutdown Condition

For all events it should be shown, justified with calculations where necessary, how the maintaining of the reactor in the shutdown state is ensured and how the plant is brought to a safe and stable state. Temporary return to power in the course of the event may be permitted, if other acceptance criteria are met.

### 1.6.3 Fuel Integrity

For anticipated transients (equivalent to category1 and 2) the probability of the fuel cladding failure resulting from a heat transfer crisis or some other reason should be insignificant. This will minimise radiological consequences and will permit the plant to resume operation after corrective action.

For accidents of category 3 fuel damage should in general be kept as low as reasonably achievable for each type of accident. In no accident should either the cladding temperature or the local energy input to a fuel rod be sufficient to cause a structural disruption of the cladding and dispersal of the fuel.

In LOCAs with fuel uncovery and heat up, a coolable geometry and structural integrity of the fuel rods upon cooling shall be maintained. Disassembly of the cooled core should be possible. Accordingly, limitations are placed on embrittlement of the cladding by oxidation, structural deformations of the fuel rods and other core components, loss of shutdown capability of control rods and generation of hydrogen.

Fuel cladding failure (leakage) criteria shall be established in order to evaluate the number of failed fuel rods for radiological consequences for postulated accidents.

### 1.6.4 Overpressure protection

The pressure in the primary and secondary circuits should not exceed the relevant design limits for the existing plant condition. Additional overpressure protection analyses may be required to study the influence of failures in safety and relief valves.

### 1.6.5 Containment function and integrity

The purpose of the containment is to enclose the entire primary system and to retain radioactive material in case of an accident. The full pressure containment is designed for receiving any design loads arising during LOCA up to double ended break of main circulating pipeline $D_{nom}$=850 mm (WWER-1000) and secures localisation of radioactive accident products inside the tight volume that secures radiation situation outside the boundaries of tight rooms during the accidents not worse than the permissible one by the respective norms. Design values are 500 kPa absolute for the maximum pressure and 150°C for the maximum temperature. In addition to the internal pressure load from the design basis LOCA, the load from the design earthquake is taken into account. The sprinkler system of the containment in the plant is of higher importance for pressure suppression than in Western plant with vertical steam generators. This is because of the horizontal steam generators in plants of VVER-type which are located nearer to the same elevation of the inlet nozzles and exit nozzles at the reactor pressure vessel than in plants with vertical steam generators. Consequently, the water injected by means of the emergency core cooling system, particularly by means of its low-pressure pumps, reaches the steam generators and the stored heat at secondary side of all four steam generators is faster transferred to the primary system and from there to the containment.

No event should cause the temperature, pressure, or pressure differences within the containment (or confinement) to exceed values which have been used as the containment design basis.

No jet forces or missiles caused by an event shall endanger the containment integrity.

### 1.6.6 Releases and Radiation Doses

All acceptance criteria defined so far are technical requirements which should guarantee a high quality of the design with sufficient design margins to fulfil the ultimate safety requirements, which are the ones related to the radiological effects on workers, the public and the environment.

The acceptance criteria include radiation criteria (RC) which are defined in the SAR. According to the russian national standarts RC are related to personal category and critical organ groups. Three groups of irradiated persons are defined:

A - personnel

B - limited part of population

C - population of the district, region, republic, country

Three groups of critical organs are established:

I - whole body, gonads, red bone marrow

II - muscles, kidneys, spleen, gastroenteric tract, lungs, eye lenses and other organs exept for those belonging to groups I and III

III - skin, bone tissues, hands, forearms, malleoli, feet

The maximum permissible doses of personnel exposure (A) according NRB-76/87 and limited doses for category B caused by radioactive wastes according to SPAES-79 during normal operation is given as follows:

Table 1.7.6-1 Maximum permissible doses for different personal categories

| | | | Group of critical organs | | |
|---|---|---|---|---|---|
| Personal category | | Dimension | I | II | III |
| A | Maximum permissible dose | rem/year | 5 | 15 | 30 |

| | | | | | |
|---|---|---|---|---|---|
| B | Gas aerosol release | mrem/year | 20 | 60 | 120 |
| C | Liquid effluents for separate types of water usage | mrem/year | 5 | 15 | 30 |

During DBA, the personnel is allowed 5 times the maximum permissible dose, according to NRB-76/87.

During maximum DBA, the expected individual exposure to the children's thyroid gland caused by the isotopes shall not exceed 30 rem whereas the expected external exposure shall not exceed 10 rem, according to SPAES-79.

The Technical Projects of the WWER NPP are based on the radiation protection limits contained in **Table 1.7.6-1**

**German regulations**

Basic requirements to be met by the radiation protection monitoring of NPP according **German regulations** are contained in **BMI Criterion 10.1**. They concern the personnel, organisational, spatial and equipment related preconditions for radiation protection monitoring in the plant and they refer to the scope of the necessary measurement equipment. The KTA Rules comprise a specification of these requirements. The regulations of Sec. § 49 StrlSchV represent the basis for detailed assessment. In contrast to NRB-76, category B according to StrlSchV for professionally exposed personnel applies. In addition, the sum of the effective doses of persons professionally exposed to radiation, determined in all calendar years may not exceed 400 mSv (age-related dose) according to Sec. § 49, Subsec. 1 StrlSchV.

A comparison with the equivalent dose limits stated in Table 3.2.7 for different groups of organs with the body dose limits shows that the radiation protection limits on which the design of the WWER NPP is based correspond to the criteria of the StrlSchV (Radiation Protection Ordinance), although it must be mentioned that an age-related dose is not determined in the Soviet radiation protection standard NRB-76. It must therefore be examined whether measures are required for special maintenance personnel to keep them within their age-related dose.