TRANSIENT AND ACCIDENT ANALYSES FOR JUSTIFICATION OF TECHNICAL SOLUTIONS AT NUCLEAR POWER PLANTS

continues

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Authorisation

By virtue of the below acts and regulations, the Radiation and Nuclear Safety Authority (STUK) issues detailed regulations that apply to the safe use of nuclear energy and to physical protection, emergency preparedness and safeguards:

- Section 55, paragraph 2, point 3 of the Nuclear Energy Act (990/1987)
- Section 29 of the Government Resolution (395/1991) on the Safety of Nuclear Power Plants
- Section 13 of the Government Resolution (396/1991) on the Physical Protection of Nuclear Power Plants
- Section 11 of the Government Resolution (397/1991) on the Emergency Preparedness of Nuclear Power Plants
- Section 8 of the Government Resolution (398/1991) on the Safety of a Disposal Facility for Reactor Waste
- Section 30 of the Government Resolution (478/1999) on the Safety of Disposal of Spent Nuclear Fuel.

Rules for application

The publication of a YVL guide does not, as such, alter any previous decisions made by STUK. After having heard those concerned, STUK makes a separate decision on how a new or revised YVL guide applies to operating nuclear power plants, or to those under construction, and to licensees' operational activities. The guides apply as such to new nuclear facilities.

When considering how new safety requirements presented in YVL guides apply to operating nuclear power plants, or to those under construction, STUK takes into account section 27 of the Government Resolution (395/1991), which prescribes that *for further safety enhancement, action shall be taken which can be regarded as justified considering operating experience and the results of safety research as well as the advancement of science and technology.*

If deviations are made from the requirements of the YVL guides, STUK shall be presented with some other acceptable procedure or solution by which the safety level set forth in the YVL guides is achieved.

Translation. Original text in Finnish.

1 General

The general safety requirements for nuclear power plants are presented in the Government Resolution (395/1991). The most essential safety principle is to consider anticipated operational transients and postulated accidents in the design of nuclear power plants. This principle requires the design of the reactor and its cooling system to be such that the plant can, with a sufficient safety margin, be maintained in a safe state under the aforementioned plant conditions. This also requires that the plant be equipped with reliable safety systems having passive or active operational principles. In addition to anticipated operational transients and postulated accidents, provision shall be made for the possibility of severe accidents.

According to section 6 of the Government Resolution (395/1991), the fulfilment of the safety requirements is to be demonstrated by the necessary experimental and analytical methods. In this guide, the requirements for the transient and accident analyses of the nuclear power plant are given. With the help of the analyses, plant behaviour, potential releases and consequent radiation doses during postulated design-basis events are studied. By these analyses, the appropriateness of technical solutions employed in the fulfilment of pre-determined safety requirements is justified. With the help of the analyses, at least the following is assured:

- The reactor and its cooling system do not contain features which could significantly aggravate the consequences of transients and accidents.
- The safety systems fulfil the set requirements.
- Actuation of safety systems occurs in the right situation and at the right moment.
- Events taken into account in design do not bring about loads or conditions which are likely to lead to further damage and via that to the deterioration of the situation.
- The radiation doses of the population in the vicinity of the plant are limited by means of systems and structures sufficiently preventing the spreading of radioactive susbstances.

The Finnish Radiation and Nuclear Safety Authority reviews the safety analyses of the plant and their appropriateness when reviewing the applications for a construction and an operation licence. The most essential results of the analyses are presented in the Preliminary and Final Safety Analysis Reports. More detailed information on the assumptions and calculational methods used in the analyses may be presented in either the Safety Analysis Report or topical reports.

In the analyses carried out for the construction licence, the focus is on plant features which are difficult to modify in the later stages of design. As regards safety systems, simplified assumptions may be made within technically feasible limits. For the operating licence, these analyses are completed and the structure of the plant is then described such that it, as closely as possible, corresponds to the final plant design.

Guide YVL 1.1 deals in detail with the procedure of applying for a construction and operating licence for a nuclear power plant and the control exercised by the Finnish Radiation and Nuclear Safety Authority.

2 Events to be analysed

2.1 General requirements

The analyses shall focus on events which by their nature and severity cover various types of incidents and accidents as well as possible. To ensure representativeness, it is essential that the plant's behaviour, which is characteristic of it due to its structure and operational practices, is thoroughly analysed as well as events most restricting as regards the function and dimensioning of each safety function and system.

Subsection 2.2 states requirements for analyses of plant behaviour. The analyses study the course of events as a function of time; and the requirements for the approval of their results are given in subsections 5.1–5.5. Subsection 2.3 deals with analyses relating to releases and off-site radiation doses. The acceptance criteria for their results are given in subsection 5.6. In these analyses it is appropriate to use initial assumptions of a more general nature, which cover several different cases simultaneously. Analyses of radiation doses do not necessarily relate directly to cases that are dealt with in analyses of plant behaviour.

2.2 Analyses of plant behaviour

Anticipated operational transients and postulated accidents

The cases to be analysed are classified into three groups as follows:

1. Anticipated operational transients; frequency at least 10^{-2}/year.
2. Postulated accidents
   a. Level 1 postulated accidents; frequency $10^{-3}\ldots10^{-2}$ /year.
   b. Level 2 postulated accidents; frequency less than 10^{-3} /year.

Each event is to be classified according to the frequency of the initiating event in the first place. Classification shall consider individual initiating events, events including additional failures or erroneous operator action, and events referred to in Guide YVL 2.7, which relate to the management of common cause failures. Level 2 postulated accidents shall include also anticipated operational transients without a scram (ATWS). The requirements for postulated accidents in this Guide apply to both classes of accident.

The course of anticipated operational transients and postulated accidents shall be analysed as a function of time, starting from the initiating event and ending in a safe and stable operational state. At the beginning of an initiating event, the plant shall be assumed to be operating at rated power (inaccuracy in power adjustment shall be taken into account) unless some other operational state is worse from the consequences point of view. If the worst initiating event cannot be reliably concluded, the consequences of the same initiating event in several operational states (e.g. at various power levels or fuel burn-ups) shall be analysed. When choosing the initial conditions, the possible sensitivity of the consequences to failure assumptions, calculational parameters and models have to be considered. For the analyses, events shall be selected which

- cause a significant change in some essential main process parameter while the reactor is in operation, or
- prevent normal plant shutdown, or
- jeopardise sub-criticality of the reactor or removal of decay heat while the reactor is in normal shutdown state.

Examples of initiating events are faults which have the following consequences

- leaks from the primary circuit during power operation, change in operational state, refuelling and/or outage
- leak from secondary circuit (PWR)
- leak from primary to secondary circuit (PWR)
- disturbance in the reactor power control or other disturbance, which causes a change in reactivity
- disturbance in primary circuit flow, pressure control or water volume control
- disturbance in steam pressure or steam flow
- disturbance in feedwater flow or temperature.

Typical examples of additional faults and erroneous operator action to be analysed in addition to initiating events include

- loss of offsite electrical power supply
- a stuck-open safety valve during the course of an accident
- a valve which is required for the isolation of a leak remains open
- malfunction of an automatic control that actuates during an accident
- erroneous operator action, which is considered possible on the basis of an operator's erroneous situation assessment.
- delay in necessary operator action.

These events shall be analysed using the assessments described in point 4.

Severe accidents

In addition to anticipated operational transients and postulated accidents, also severe accidents shall be analysed.

According to Guide YVL 1.0, the possiblity of a severe accident shall be considered in the design of the nuclear power plant. The nuclear power plant shall therefore have a severe accident strategy to assure the fulfilment of safety goals for severe accidents set forth in the Government Resolution (395/1991) and in Guide YVL 1.0.

The essential functions of the severe accident management strategy shall be justified by suitable experimental and analytic means. As part of the strategy, it shall be assured in particular that initiating events endangering containment integrity or the prevention of the dispersion of fission products as well as rapid and/or energetic physical phenomena have been prevented with a good certainty.

Severe accident analyses shall be conducted to study factors affecting containment integrity, leak tightness and the operability of containment systems. Analyses have to be carried out for cases which may be the worst from the viewpoint of the functioning of the containment. They could include e.g.

- total loss of AC power
- total loss of feedwater
- leak of primary coolant without emergency cooling during power operation; or during a maintenance, refuelling or other outage
- leak of primary coolant and blockage of coolant recirculation.

In addition, for the purpose of emergency planning, safety analyses are to examine events not considered in the severe accident management strategy proper. These include severe accident sequences whose prevention has been implemented with such certainty that they are excluded from the severe accident management strategy. Emergency planning is dealt with in Guide YVL 7.4.

The PSA (Probabilistic Safety Assessment) analysis uses the results of anticipated operational transients, postulated accidents and severe accidents analyses in assessing the success criteria for the systems and the consequences of events.

2.3 Analyses of releases and radiation doses

Anticipated operational transients

If an anticipated operational transient could cause an exceptional release of radioactive substances (e.g. a release of reactor coolant into the environment), the radiation doses caused by the release shall be estimated.

Postulated accidents

Separate radiation dose analyses shall be made for postulated accidents in case the dose upper limit caused by them cannot be concluded from the results of other analyses. For example the following cases can be such events:

- Large leak of coolant from the primary circuit during power operation. This shall be used as a typical example of accidents during which radioactive substances are first released within the containment and only gradually leak out.
- Leak of reactor coolant out from the containment as a consequence of an instrument line rupture.
- Leak from steam generator primary to secondary side. The total rupture of one or multiple steam generator tubes shall be analysed, assuming that also the safety valve of the steam generator has stuck open in a case where it is expected to open. Also a leak larger than the one mentioned above shall be analysed if estimated possible on the basis of the structure of the steam generator and/or applicable operating experiences (PWR).
- Leak out of the primary circuit during a maintenance, refuelling or other outage.
- Leak outside the containment in an unisolated steam line connecting to a steam generator in which, before the initiation of the accident, the largest primary to secondary circuit leak (PWR) allowable in the Technical Specifications has occurred.
- Leak in a steam line outside the containment or in a reactor coolant purification line (BWR).
- Damage outside the containment in a system containing radioactive gases.
- Damage outside the containment in a system containing radioactive liquids.
- Damage of a fuel assembly which has been removed from the reactor.
- Dropping of a transfer or transport cask containing spent fuel during hoisting, in a situ-

ation where the cask is not tightly closed, or dropping of a fuel cask during transfer.
- Dropping of a heavy object on top of stored fuel or an open reactor.

Severe accidents
Releases of radioactive substances and radiation doses caused by a severe accident shall be analysed. The analyses shall be carried out for cases which, on the basis of containment behaviour and conditions and the concentration of radioactive substances in the containment, are estimated to cause the most extensive releases. The analyses are to include events in accordance with the management strategy in the first place.

3 Methods of analysis

Methods of analysis mean i.a. methods based on hand calculations, computer programs and the application of experimental data. The reliability of the analysis methods used shall be justified. A description of the analysis methods used shall be given, including their general principles as well as the physical models and numerical methods used.

The experimental correlations used in the calculations shall be justified by presenting the measurement data from which the correlations have been derived. If the correlation is commonly known and the measurement data are publicly available, a bibliographic reference is sufficient.

The analysis methods shall be adequately verified for the treatment of the events in question. Both numerical methods and physical models shall be verified.

Numerical methods shall be verified by adequate reference calculations. Physical models shall be verified by demonstrating their ability to depict suitable separate effects tests or integral tests for complete systems or nuclear power plant transients. In addition, comparison with other, earlier verified models may be utilised.

If sufficiently reliable calculation methods are not available, the analysis shall be justified by experiments. This requirement applies especially to most phenomena essentially relating to severe accident management, for example, the long term coolability of reactor core debris after a severe accident.

4 Assumptions used in analyses

4.1 Analyses of plant behaviour

4.1.1 General requirements

Characteristic to the analyses is the so called conservative approach in which the uncertainties associated with the calculation methods and initial assumptions are taken into account in such a way that the real behaviour of the plant is, with sufficient certainty, more favourable than the most disadvantageous analysis results. The analyses shall take into account that it is not always possible to define unambiguously in advance how influence of the uncertainties connected to the calculation models, parameters or initial assumptions is conservative considering the final results. The thermal conductivity of the fuel gas gap is an example of that type of initial assumption.

Additionally it shall be considered that a single initiating event may have several consequences which are different as regards the fulfilment of the safety goal. Several acceptance criteria (based on different parameters) may thus apply to a transient or an accident starting from a certain initiating event. The small break loss of coolant accident is an example of that type of initiating event in a PWR. Safety goals connected to it are on the one hand ensuring fuel coolability and on the other consideration of brittle fracture risk of the reactor pressure vessel on the design of the emergency core cooling system.

The main objective of the analyses is to identify the essential processes and threshold phenomena for the analysed cases and to define their effects.

4.1.2 Parameters of calculation

The analyses shall include sensitivity studies which define how sensitive the results are for the analysis methods and initial assumptions. The requirements presented in subsection 4.1.1 shall be taken into account final result can be considered conservative. Such parameters are particularly
- process parameters (power, pressure, temperature, etc.) at the beginning of accident

- accuracy of the trip limits used in the protection systems
- capacity of the equipment and their performance characteristics
- inaccurately known factors (manufacturing tolerances, heat transfer coefficients, mixing phenomena, condensing phenomena, etc)
- decay heat of the fuel.

The conservativeness of the parameters chosen shall always be justified. If the conservativeness is ambiguous due to the nature of the phenomenon under analysis, or for some other reason, analysis results covering a parameter's entire range of variation shall be presented to facilitate identification of the choice least favourable for the analysis results.

4.1.3 Assumptions for systems

Safety systems are assumed to operate at their designed minimum output, unless an accident directly affects their operability. Minimum output is attained when

- a combination of faulty and inoperational components, which most hinders system operation, is assumed according to Guide YVL 2.7. The most reactive control rod in the reactor scram system is assumed jammed.
- performance parameters are determined for each operating component which, taking the appropriate safety margin into account, conform to the acceptance limits of components in periodic tests.

If the operation of a safety system at a higher output may have a detrimental effect (e.g. too quick a cooling or premature loss of water), also this shall be examined as a separate alternative (for comparison, see the acceptance requirement in subsection 5.1).

Protection systems are assumed to operate in the designed manner, unless an event directly affects their operability. A scram failure during ATWS analyses is an exception to this principle.

Transient and accident analyses shall cover situations in which normal operating systems are assumed to operate in their most probable manner as well as situations in which they operate erroneously or not at all. In all these cases, the acceptance criteria set for each situation shall be fulfilled.

4.1.4 Operator actions

Operators can be assumed to act according to the written procedures for each analysed event. The time of consideration preceding actions shall be chosen conservatively and be justified. Actions for the mitigation of an incident or an accident can be considered likely if an event is clearly identifiable. However, several alternative operator actions have to be analysed in which the impact of erroneous control manoeuvres and the necessary corrective measures on the course of the accident are analysed. When operator actions are evaluated it shall be specifically considered whether any incorrect action is sufficiently unlikely.

4.1.5 ATWS analyses

The following assumptions shall be made in those analyses of anticipated operational transients in which the reactor scram has failed (ATWS analyses):

- The reactor scram is assumed to fail because of a fault in the protection system that hinders the initiation of the reactor scram function, or because of a mechanical common cause failure in the reactor scram system, or in the control rods, that prevents the insertion of the control rods into the reactor core.
- A single failure of relief and safety valves is assumed.
- Normal operational systems and operators are assumed to act in the most probable way.
- Safety systems are assumed to operate in the same way as in other postulated accidents.
- Calculational parameters are chosen the same way as in other postulated accidents.
- Xenon concentration in the reactor core is assumed to be in equilibrium when accidents starting at full power are analysed.
- The reactor core is assumed to be xenon-free when accidents starting at low power are analysed.

4.1.6 Severe accidents

The analyses justify that the systems and actions designed to implement the severe accident management strategy are acceptable. The analyses may base on so called best estimate methods but apply conservatism in balance with the strategic significance of the function: the more essential the function, the better assurance for success shall be demonstrated. Also the conservative factors for the choices shall be justified.

In addition to the systems proper belonging to the actual management strategy, other systems whose functioning does not presuppose the operation of active components may be taken into account as factors mitigating accident conditions or restricting releases. An example of such a system is the heat transfer circuit in which the medium circulates by natural circulation. In addition, even such active components may be assumed operable whose operation is independent of the causes and consequences of a severe accident.

If relevant justification is provided, component faults that have resulted in a severe accident may be assumed to be fixed later, unless a high radiation level, or some other reason, prevents repairs. The time spent in repairs shall be chosen conservatively and be justified.

The time spent in actions in accordance with the management strategy and other factors connected to the implementation of the actions (e.g. accessibility of locally controlled equipment) shall be justified.

In analysing the pressure behaviour of the containment, non-condensable gases have to be taken into account. When estimating the amount of released hydrogen in particular, it shall be assumed that 100% of easily oxidising material in the area of the reactor core reacts with water. Also other hydrogen sources shall be taken into account in accordance with Guide YVL 1.0. In assessing the hydrogen release rate, it shall also be considered that the emergency cooling function may resume operation.

4.2 Assumptions employed for releases and radiation dose analyses

4.2.1 Events during which radiation doses arise from radioactive substances contained in primary coolant

At least the same amount of radioactive substances shall be assumed in the primary coolant at the beginning of an accident as is intended to be set as the limit in the Technical Specifications of the plant. The distribution of nuclides and their isotopes has to be chosen so that the distribution corresponds in practice to distributions observed in power plants of the same type.

As of the moment of time when reactor power starts to significantly change (to decrease or increase), an increase in the iodine and caesium concentrations shall be assumed that corresponds to the most extensive increase in connection with power changes observed at the type of plant in question.

The primary coolant leak rate shall be conservatively estimated. The time until the potential isolation of the leak shall be conservatively estimated on the basis of the alarms and measurement results available to the operators.

If some action affecting the isolation of a leak or the dispersion of radioactive substances is automatic and single failure tolerant, the system can be assumed to function in the designed manner in this respect.

Releases caused by the liquid part of leaking coolant and those caused by the vaporisable part shall be separately considered. It can be assumed that the concentration of radioactive substances in the vaporisable part is lower than in the coolant upstream of the leak. The coefficient indicating a decrease in concentration shall be justified by a reference to practical observations or test results. It shall then be assumed, however, that all the noble gases in the leaking coolant are always discharged to the environment in their entirety.

If a leak occurs directly into the environment and the coolant is in water form when entering the leak point, all the radioactive substances in the leak shall be taken into account when calculating offsite doses.

The steam which has leaked into the plant interior and the radioactive substances which have mixed with it shall be assumed to be transferred into the environment in a way corresponding to the normal functioning of the ventilation systems in the plant condition in question.

Part of the iodine which has mixed with the steam shall be assumed gaseous. The distribution of iodine into gas and aerosols shall be justified.

If the use of filters is assumed in the ventilation systems, the retention factors of the filters are to be conservatively selected.

4.2.2 Loss of coolant caused by a large primary circuit break

The duration of a primary coolant discharge into the containment shall be selected on the basis of thermohydraulic analyses. The time shall be shorter than the shortest calculated length of time considering the accuracy of the calculating method. The below assumptions shall be made according to subsection 4.2.1

- radioactive substances in primary coolant
- distribution of radioactive substances into vaporisable and condensing part of leak
- the state of the iodine, which has become mixed with the steam.

The time of failure of fuel rods and the number of failed rods shall be conservatively selected taking into account the results of analyses relating to plant behaviour.

The reactor shall be assumed to have operated at full power since the previous refuelling until the accident. The core loading shall be assumed to represent an equilibrium core and an analysis shall be made corresponding to the state at the end of the fuel cycle.

The percentages of radioactive substances released from failed rods shall be conservatively chosen on the basis of experimental research and operating experience of the fuel type in question.

A certain share of the radioactive substances released from failed fuel rods to the coolant directly enters the containment airspace. The distribution between airspace and cooling water shall be justified.

An additional release of radioactive substances from the failed rods shall be assumed later when cooling water enters the rods and dissolves the fuel. The share of radioactive substances ending up in the water in the first phase is to be justified by experimental research, or the assumptions concerning them are to be made conservatively.

Assumptions on the transport of radioactive substances within the containment may be based on experimental research if the results are applicable to the situation. Alternatively, a conservative model may be used, which gives a slower than normal disappearance of radioactive substances from the airspace.

If air is discharged from the containment during normal plant operation, the mixing of radioactive substances with the discharged air shall be estimated conservatively. The isolation of ventilation may be assumed according to the design of the plant protection system so that any changes in the parameters used as protection limits during accidents are assessed conservatively. Before isolation, ventilation shall be assumed to function in the normal way.

After potential isolation of the containment, radioactive substances shall be assumed to mix evenly in the airspace of the whole containment. The containment leak rate has to be selected taking into account the tightness requirement set for the containment and the containment overpressures calculated during the analysis of postulated accidents. Appropriate safety margins shall be employed during the selection.

Part of the halogens that have leaked from the containment shall be assumed to be in inorganic and part in organic compounds. The distribution into the various kinds of compounds shall be justified.

Releases caused by leaks and potential malfunctions of the emergency core cooling systems and leaks of the containment cooling systems outside the containment boundary are to be taken into account conservatively.

The ventilation of the space surrounding the containment shall be assumed to function in the way designed for accident conditions and releases arising from a containment leak shall be calculated accordingly. If the ventilation system is used in the normal way, with the filters bypassed, the time spent in the potential switch-over to the filters shall be justified.

If the use of filters in the ventilation systems is assumed, the retention factors of the filters shall be selected conservatively.

4.2.3 Accidents in spent fuel handling

In the analysis of the drop of a spent fuel assembly, it shall be assumed that the assembly

- has been in the reactor core during the whole cycle at full power
- has been located in the most heavily loaded position of the reactor core and has reached a full discharge burn-up

- has cooled down for one day after reactor shutdown
- is damaged in such a way that all fuel rods lose their tightness.

In the analysis of the drop of a spent fuel transfer or transport cask, it shall be assumed that

- an accident can happen in any room and at any time when a transport cask is being lifted with the lid open or insufficiently bolted
- the cask has been loaded with fuel that has reached a full discharge burn-up
- the cooling time required for fuel prior to transfer is the minimum time required in the administrative restrictions
- the number of failed fuel assemblies is with a sufficient safety margin higher than the number estimated on the basis of loads caused by an accident.

In the analyses of the drop of a heavy object, it shall be assumed that

- an accident can happen in any location where the handling of heavy objects above fuel is possible
- the falling object possible in the respective location causes the most extensive damage possible
- the fuel burn-up is the highest and the cool-down time the shortest possible in the accident situation under consideration
- the number of damaged fuel assemblies is, with a sufficient safety margin, higher than the number estimated on the basis of the loads caused by the accident.

Such percentages shall be assumed to be released from the radioactive substances in the failing fuel rods as represent the potential upper limit for the event in question. Assumptions concerning the percentages shall be justified on the basis of studies made for the type of fuel in question.

All released noble gases shall be assumed to enter the airspace of the building in question. If fuel damage occurs under water, in estimating the release of iodine, it is assumed that part of the iodine isotopes remains in the water and only part of them are released to the airspace above water.

Part of the iodine, which is released to the airspace, shall be assumed to be in inorganic and part in organic compounds. The distribution into the various types of compounds shall be justified.

The radioactive substances, which enter the airspace, shall first be assumed to be transported to the environment via the ventilation system in a way which corresponds to the normal functioning of the ventilation system. If the ventilation system can be used in several different ways in the above-mentioned situation, the way shall be chosen in the analysis which leads to the most extensive releases. The personnel are assumed to isolate the ventilation ducts within 30 minutes. If isolation occurs automatically and is implemented by an appropriate protection system, also an earlier timing for the isolation can be assumed that corresponds to the construction and operation of the system.

If the use of filters is assumed in the ventilation systems, the retention factors of the filters shall be selected conservatively.

4.2.4 Severe accidents

Analyses are to be carried out according to subsection 2.3. In analyses of power operation, the reactor shall be assumed to have been operating at full power before the accident and since the previous refuelling. In addition, the fuel loading shall be assumed to represent an equilibrium core and the situation at the end of a fuel cycle shall be analysed.

Assumptions of the amounts of radioactive substances released into the containment airspace as a result of reactor core degradation shall be based on adequate experimental studies. Appropriate safety margins shall be employed when selecting the amounts.

If the pressure and temperature inside the containment during an accident exceed the values for which the containment leaktightness requirements have been set and during which the leak rate is experimentally measured, the leak rate used for the release calculations shall be justified separately. In addition to the interdependency between pressure difference and leak rate, any additional leak caused by deformations in the sealings of containment penetrations and air locks shall be taken into account when the leak rate is determined.

Assumptions of the decontamination effect of components and potential filters along the release route shall be justified with experimental studies in release calculations that analyse the consequences of a containment leak or an event in which containment pressure is reduced by a filtered venting system.

When examining the hazard of acute health effects caused by a severe accident to the local population, the actual conditions onsite and in its vicinity shall be taken into account. Based on these conditions, the local distribution of the members of the critical group during the initiation of the accident as well as the duration of evacuation from various distances shall be selected for the assumptions required in the radiation dose calculations.

4.2.5 Dispersal of radioactive substances into the environment

Assumptions of the dispersal of radioactive substances into the air are presented in Guide YVL 7.3 and assumptions of radiation dose calculations in Guide YVL 7.2.

5 Acceptance criteria for the analyses

5.1 General requirements

According to Guide YVL 1.0, the safety level of a nuclear power plant must be raised as high as practically achievable. The more severe an accident's consequences could be, the smaller the likelihood of its occurrence shall be. The fulfilment of the acceptance criteria presented in the following chapters thus is not sufficient justification for not implementing a solution that would essentially improve safety.

5.2 Operation of systems designed for accident mitigation

It shall be shown that systems designed for accident mitigation fulfil their safety function without subjecting the power plant structures and components to such loads or conditions as would exceed the design limits applicable to the operating and accident conditions of the components.

5.3 Bringing the plant to a safe state

For every transient and accident it shall be shown that the reactor is maintained in shutdown state and that the plant can be brought to a safe and stable state. In addition, it shall be shown that the plant can, in the long run, be brought to a state where fuel removal from the reactor pressure vessel is possible.

5.4 Pressure control

Requirements for pressure control are presented in Guide YVL 2.4. In applicable cases, analyses conducted in accordance with it may be utilised in transient and accident analyses.

5.5 Fuel failures

Section 15 of the Government Resolution (395/1991) prescribes as follows:

The probability of significant degradation of fuel cooling or of fuel failure due to other reasons shall be low during normal operational conditions and anticipated operational transients.

During postulated accidents, the rate of fuel failures shall remain low and fuel coolability shall not be endangered.

The possibility of a criticality accident shall be extremely low.

Design requirements for fuel failures and coolability as regards events discussed in subsection 2.2 are given in Guide YVL 6.2.

5.6 Containment integrity

Section 17 of the Government Resolution (395/1991) prescribes as follows:

The containment shall be designed so that it will withstand reliably pressure and temperature loads, jet forces and impacts of missiles arising from anticipated operational transients and postulated accidents.

Furthermore, the containment shall be designed so that the pressure and temperature created inside the containment as a consequence of a severe accident will not result in its uncontrollable failure.

The possibility of the creation of such a mixture of gases as could burn or explode in a way which endangers containment integrity shall be small in all accidents.

The hazard of a containment building failure due to core melt shall also be taken into account *in other respects in designing of the containment building concept.*

Detailed design requirements for the containment are given in subsection 3.3 of Guide YVL 1.0. If the severe accident management strategy incorporates cooling of core melt at the bottom of the containment, it shall be demonstrated that the molten core can be cooled down without endangering the integrity of the containment. This means, among other things, that there is to be no interaction (e.g. erosion or development of gas) between the molten core and the pressure-bearing materials of the containment walls or ceilings.

5.7 Releases and radiation doses

According to section 7 of the Government Resolution (395/1991), radiation exposure arising from the operation of a nuclear power plant shall be kept as low as reasonably achievable. The fulfilment of the set limits contained in the aforementioned resolution and guide is not sufficient justification for not implementing a solution which would essentially reduce the radiation doses of workers or the population, or environmental pollution.

Anticipated operational transients

Section 10 of the Government Resolution (395/1991) prescribes as follows:

The limit for the dose of an individual of the population, arising, as the result of an anticipated operational transient, from external radiation in the period of one year and the simultaneous radioactive materials intake, is 0.1 mSv.

The limit value applies to the effective dose commitment of an individual of the critical group.

In addition, it shall be shown that, as a result of any anticipated operational transient, the global collective 500 years effective dose commitment of the population does not exceed the limit value of 5 manSv/GWe (per installed electrical power).

Postulated accidents

Section 11 of the Government Resolution (395/1991) prescribes as follows:

The limit for the dose of an individual of the population, arising, as the result of a postulated accident, from external radiation in the period of one year and the simultaneous radioactive materials intake, is 5 mSv.

The limit value applies to the effective dose commitment of an individual of the critical group. Collective dose commitments caused by a postulated accident shall also be analysed.

Severe accidents

Section 12 of the Government Resolution (395/1991) prescribes as follows:

The limit for the release of radioactive materials arising from a severe accident is a release which causes neither acute harmful health effects to the population in the vicinity of the nuclear power plant nor any long-term restriction on the use of extensive areas of land and water. For satisfying the requirement applied to long-term effects, the limit for an atmospheric release of caesium-137 is 100 TBq. The combined fall-out consisting of nuclides other than caesium-isotopes shall not cause, in the long term, starting three months from the accident, a hazard greater than would arise from a caesium release corresponding to the above-mentioned limit.

The possibility that, as the result of a severe accident, the above-mentioned requirement is not met, shall be extremely small.

6 Definitions

Loss of coolant conditions

Loss of coolant conditions mean those postulated accidents in which, due to a leak of the primary circuit, the coolant is lost faster than can be replaced by the make-up systems designed for normal operational conditions.

Operational conditions

Operational conditions mean a nuclear power plant's normal operational conditions and anticipated operational transients.

Final heat sink

The final heat sink means the atmosphere, the ground and also surface water and groundwater to which heat from various sources is transferred during operational conditions and accidents.

Normal operational conditions

Normal operational conditions mean that the nuclear power plant is operated according to the Technical Specifications and operational procedures. These also include tests, plant start-up and shutdown, maintenance and refuelling.

Anticipated operational transient

An anticipated operational transient means a deviation from normal operational conditions, which is milder than an accident and which can be expected to occur once or several times over a period of a hundred operating years.

Accident

An accident means such a deviation from normal operational conditions as is not an anticipated operational transient. There are two classes of accident: postulated accidents and severe accidents. Based on the initiating event, postulated accidents are further divided into two sub-classes whose acceptance criteria are described in Guide YVL 6.2.

Postulated accident

A postulated accident means such a nuclear power plant safety system design-basis event as the nuclear power plant is required to withstand without any serious damage to the fuel and without discharges of radioactive substances so large that, in the plant's vicinity, extensive measures should be taken to limit the radiation exposure of the population.

Fuel design limits

Fuel design limits denote the limits to prevent fuel failures during operational conditions and to ensure fuel coolability in postulated accidents.

Primary circuit

The primary circuit means pressure-retaining components of the reactor cooling water system, such as pressure vessels, piping, pumps and valves or other components connecting to the reactor cooling water system. The boundaries of the primary circuit are defined in Guide YVL 2.1.

Design parameters

Design parameters mean the design basis loads of a structure or component. Different design parameters are defined for normal operational conditions, anticipated operational transients or postulated accidents.

Structures, systems and components important to safety

Structures, systems and components important to safety are such that

- their malfunction or breakage can significantly increase the radiation exposure of the plant's workers or the environment
- they prevent the occurrence and propagation of transients and accidents
- they mitigate the consequences of accidents.

Safety system

A safety system is a system that carries out a certain safety function.

Safety functions

Safety functions are safety-significant functions to prevent the occurrence or propagation of transients or to mitigate the consequences of accidents.

Severe accident

A severe accident means an event during which a significant part of the fuel in the reactor sustains damage.

Single failure

A single failure means a random failure and its consequent effects, which are assumed to occur either during a normal operational condition or in addition to the initial event and its consequent effects. Further instructions concerning single failures are given in Guide YVL 2.7.