FINNISH CENTRE FOR RADIATION AND NUCLEAR SAFETY
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Transient and accident analyses for justification of technical solutions at nuclear power plants

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Authorisation

By virtue of section 55, second paragraph, point 3 of the Nuclear Energy Act (990/87) and section 29 of the Council of State Decision (395/91) on General Regulations for the Safety of Nuclear Power Plants, the Finnish Centre for Radiation and Nuclear Safety (STUK) issues detailed regulations concerning the safety of nuclear power plants.

YVL Guides are rules an individual licensee or any other organisation concerned shall comply with, unless STUK has been presented with some other acceptable procedure or solution by which the safety level set forth in the YVL Guides is achieved. This Guide does not alter STUK's decisions which were made before the entry into force of this Guide, unless otherwise stated by STUK.

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1 General

General safety requirements for nuclear power plants are presented in the Council of State Decision (395/91). The most essential safety principle is that provision shall be made for anticipated transients and postulated accidents. According to this principle the reactor and its cooling system shall be designed in such a way that there are sufficiently good starting points for maintaining the plant in a safe condition. This also requires that the plant is equipped with reliable safety systems the operation principles of which are either passive or active. 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 Council of State Decision (395/91) the fulfilment of the safety requirements shall be demonstrated by the necessary experimental and analytical methods. In this Guide requirements for the transient and accident analyses of the nuclear power plant are presented. With the help of the analyses, plant behaviour, potential releases and the radiation doses caused by the releases during postulated design basis events are studied. By these analyses, the appropriateness of the technical solutions employed in the fulfilment of pre-determined safety requirements is justified. With the help of the analyses at least the following is confirmed:

— The reactor and its cooling system do not contain special features which could significantly aggravate the consequences of transients or 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.
— Radiation doses of population in the vicinity of the plant are limited by means of systems and structures which prevent the spreading of radioactive substances.

In the analyses, deterministic assumptions are made of the occurring faults and the functioning of components and systems. A so called conservative approach is characteristic of these analyses. This means i.a. the following choices and assumptions which have an unfavourable effect on the results:

— Faults which are obviously unlikely are also assumed in safety systems.
— Unknown parameters or parameters which normally vary within a certain range are selected from the worse end of a potential range.
— Deficiencies in the computation model are compensated by assumptions which aggravate the results.

The analyses include studies by which the sensitivity of the results with regard to analysis methods and initial parameters are examined. A significant part of the analysis is to identify processes and edge phenomena essential for the object and situation analysed and to clarify the effects arising from them.

The quantitative assessment of a nuclear power plant's safety, the compilation of emergency operating procedures, the clarification of the effects of external incidents, the assessment of a site's acceptability and emergency planning to protect the public in the vicinity of the plant require analyses the method of treatment and coverage of which differ from the analyses in this Guide. Such analyses are included i.a. in the PSA analysis which is dealt with in Guide YVL 2.8. In Guides YVL 2.6, YVL 4.3 and YVL 6.8 correspondingly deal with earthquakes, fire protection, handling and storage of fuel and the related safety analyses.

The Finnish Centre for Radiation and Nuclear Safety reviews the safety analyses of the plant and their appropriateness in connection with the review of the applications for construction and operation. 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 either in the Safety Analysis Report or topical reports.

In the analyses carried out for the construction licence, it is essential to look into such plant features in particular the modification of which is not possible in the later stages of design. As regards the safety systems, simplified assumptions can be made within such limits as are technically feasible alternatives. For the operating license, the analyses are completed and the structure of the plant is described in such a way that it as closely as possible corresponds to the final design of the plant.

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

2 Events to be analysed

The analyses shall be focused on events which by their nature and severity cover different types of incidents and accidents as well as possible. From the viewpoint of the representability of the events, it is essential that the plant's characteristic behaviour which is due to its structure and operational practices will be analysed thoroughly and events which are the most restricting with regard to the function and dimensioning of each safety system are analysed.

In sub-section 2.1, requirements are given on the analyses relating to plant behaviour. In these analyses, the course of events is studied as a function of time and the requirements for the approval of their results are given in sub-sections 5.1–5.5. In sub-section 2.2, analyses relating to releases and offsite radiation doses are dealt with. The acceptance criteria for these results are presented in sub-section 5.6. It is appropriate to use initial assumptions of a more general nature in them which cover several different cases simultaneously. Analyses of radiation doses do not necessarily relate directly to any case dealt with in the analyses concerning plant behaviour.

2.1 Analyses of plant behaviour

Anticipated operational transients and postulated accidents

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. In 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. Inaccuracy in power adjustment shall be taken into account. If the worst initiating event cannot be reliably concluded, the consequences of the same initiating event in several operational states (e.g. at various powers 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 taken into account.

The events to be selected as initiating events shall

— cause a significant change in some essential main process parameter while the reactor is in operation
— prevent normal plant shutdown
— 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 feedwater temperature.

A transient or an accident relating to each initiating event has to be analysed using the parameters and assumptions in sub-sections 4.1.

The cases to be analysed are classified into two groups as follows:
1) anticipated operational transients: probability not less than 10^{-2}/year.
2) postulated accidents: probability less than 10^{-2}/year.

Each event has to be classified according to the probability of the initiating event. Should the need arise for some initiating event's part to analyse several alternatives (for example alternatives starting from different plant conditions or containing additional faults), the instructions given in sub-section 4.1.6 must be complied with.

Also anticipated operational transients during which a scram fails (the so called ATWS cases) shall be treated as postulated accidents.

Severe accidents

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

Severe accident analyses shall be used to study factors which affect 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 i.e.:
— total, long lasting loss of AC power
— total loss of feedwater
— leak of primary coolant without emergency cooling during power operation or a maintenance, refuelling or other outage
— leak of primary coolant and blockage of coolant recirculation.

The analyses in this Guide do not deal with such in which a containment isolation valve or air lock would have remained in the open position already prior to the analysed incident.

2.2 Analyses of releases and radiation doses

Anticipated operational transients

If an anticipated operational transient may 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 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 by assuming that also the safety valve of the steam generator has stuck open in a case 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 (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 situation where the cask is not tightly closed, or dropping of the 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. 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. A case has to be included in the analyses, in which the containment pressure is reduced by a filtered venting relief system, but the containment stays otherwise intact.

3 Methods of calculation

Methods of analysis mean i. a. methods based on hand calculations, computer programs and the application of experimental information. The reliability of the methods of calculation employed in the analyses shall be justified. A description shall be presented for the used methods of calculation which includes the general principles of the methods of calculation, physical models and numerical methods.

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 may be sufficient.

The methods of calculation 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. Also, 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 to, for instance, the long term coolability of reactor core debris after a severe accident.

4 Assumptions used in analyses

4.1 Analyses of plant behaviour

In the analyses, it shall be taken into account that, due to the internal processes of the object of examination, or physical edge phenomena, 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.

4.1.1 Parameters of calculation

Parameters affecting the final results of the analysis which are essential for the acceptance requirements shall be selected from the edge of their likely range of variation so that the 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.

From the acceptance criteria point of view, the sensitivity of essential results of analysis due to internal processes of the object of examination and physical edge phenomena shall be clarified both by making several analyses by some chosen method and by making analyses in several ways which are independent of each other as regards the examined phenomenon, for instance by experiments or by methods of calculation which are mutually sufficiently different from each other.

4.1.2 Protection systems

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

4.1.3 Safety systems

Safety systems are assumed to operate at the 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, and
— performance parameters are determined for each operating component which, taking the appropriate safety margin into account, conform to the acceptance limit 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 a premature loss of water), also this possibility shall be examined as a separate alternative (for comparison, see the acceptance requirement in sub-section 5.1).

Faults which directly affect some safety function need not be included in the failures mentioned later in sub-section 4.1.6, since they are already taken into account when the minimum output of the systems is defined.

4.1.4 Normal operating systems

Normal operating systems can be assumed to operate in the way estimated as most probable. In sub-section 4.1.6 the need to analyse several alternatives of a certain case has been dealt with so that the assumptions concerning the functioning of normal operating systems are modified.

4.1.5 Operator actions

Operators can be assumed to act according to the written procedures concerning each analysed event. The time of consideration preceding actions shall be chosen conservatively and shall be justified. Actions for the mitigation of an incident or an accident can be considered likely if an event is clearly identifiable.

Also several alternative operator actions have to be analysed according to the principles stated in sub-section 4.1.6. When operator actions are evaluated it shall be particularly considered whether some incorrect action is likely.

4.1.6 Evaluation of various event alternatives

A sufficient number of alternatives shall be analysed of all events following the principles stated in sub-sections 4.1.1–4.1.5. If the failure of any individual component of the normal operating systems or operator action which deviates from the assumed would essentially affect the course of events and might aggravate the consequences, other alternatives affecting the same initiating event shall, in addition to the variations stated in sub-section 4.1.1, be scrutinised at discretion.

An analysis of alternatives containing operator errors or component faults may be considered for such events in particular as otherwise are analysed as anticipated operational transients but which in consequence of operator errors may turn to postulated accidents. The concept of a postulated accident can be applied to alternatives in which the frequency of an anticipated operational transient and the malfunction which aggravates it can be justified as being clearly smaller than 10^{-2}/year.

Typical examples of malfunctions which require alternative studies are:
- loss of external grid
- 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 which actuates in connection with an accident
- faulty operator action which is estimated possible on the basis of an operator's erroneous assessment of the situation
- delay of a necessary operator action.

4.1.7 ATWS analyses

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

- the reactor scram is assumed to fail because of a fault in the protection system which 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 which prevents the insertion of the control rods into the reactor core
- a single failure is assumed in the functioning of relief and safety valves
- normal operational systems, and operators are assumed to act in the most probable way
- safety systems are assumed to operate 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.

4.1.8 Severe accidents

Systems the functioning of which does not presuppose the operation of active components may be taken into account as factors which alleviate accident conditions or restrict 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 the operation of which is independent of the causes and consequences of a severe accident.

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

Accident mitigating actions for which sufficient instructions have been issued in advance and which are started after the accident has occurred, can be taken into account. They can be based on e.g. the utilisation of equipment independent of the plant's fixed systems. The time needed for the actions shall be justified.

When analysing the pressure behaviour of the containment, non-condensable gases have to be taken into account. When estimating the amount of released hydrogen, it shall be especially 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 according to Guide YVL 1.0.

4.2 Assumptions employed for radiation dose calculations

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

At the beginning of the accident the amount of radioactive materials in the primary coolant have to be assumed at least the same as is intended to be set as the limit in the Technical Specifications of the plant. The distribution of nuclides and their isotopes have to be chosen so that the distribution corresponds in practice to the distribution noted in plants of the same type.

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

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

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

Releases which are caused by the liquid part of the leaking coolant and releases which are caused by the vaporisable part shall be examined separately. It can be assumed that the concentration of radioactive materials in the vaporisable part is lower than in the coolant immediately before the leak. The coefficient indicating a decrease in concentration shall be justified by a reference to practical observations or test results. As an exception from the above, it shall 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, 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 which corresponds to the normal functioning of the ventilation systems.

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 shall be selected conservatively.

4.2.2 Loss of coolant by a large primary circuit break

The duration of 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, taking into account the accuracy of the calculating method. Assumptions concerning

— 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 steam shall be made according to sub-section 4.2.1.

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

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

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

A certain share of the radioactive substances released from the failed fuel rods to the coolant enters the containment airspace directly. 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. These shares of radioactive substances which initially end up in the water shall be justified by experimental research, or the assumptions concerning them shall be made conservatively.

Assumptions of the transport of radioactive substances within the containment can be based on experimental research if the results are applicable to the situation in question. 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 to take place 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 which have leaked from the containment shall be assumed to be in inorganic compounds and part in organic compounds. The distribution into the various kinds of compounds shall be justified.

The releases caused by the leaks and the potential malfunctions of the emergency core cooling systems and the leaks of the containment cooling systems outside the containment boundary have 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 the 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 possible 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 reached a full discharge burn-up
— has cooled down for 1 day after reactor shutdown
— is damaged so 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 which 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
— the fuel burn-up is the highest and the cool-down time the shortest possible in the accident situation under review
— 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 the 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 was 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 entered 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 is assumed to isolate the ventilation ducts within 30 minutes. If isolation is automatic and implemented by an appropriate protection system, also an earlier timing for the isolation can be assumed which 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 have to be carried out according to sub-section 2.2. 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, and the fuel loading shall be assumed to represent an equilibrium core at the end of a fuel cycle.

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 leak-tightness 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 which analyse the consequences of a containment leak or an event in which the 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 on site 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 7.2.

5 Acceptance criteria for the analyses

According to Guide YVL 1.0, the safety level of a nuclear power plant must be raised as high as practicably 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 this chapter thus is not sufficient justification for not implementing a solution which would essentially improve safety.

5.1 Operation of systems designed for accident mitigation

It shall be shown that the 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.

5.2 Bringing the plant to a safe state

For every transient and accident it shall be shown that the reactor is maintained in the 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 term be brought to a state where fuel removal from the reactor pressure vessel is possible.

5.3 Pressure control of the plant

Requirements for pressure control are presented in Guide YVL 2.4. Analyses which are in conformity with the mentioned Guide may, where applicable, also be used as transient and accident analyses.

5.4 Fuel failures

In section 15 of the Council of State Decision (395/91) the following is required:

"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 are presented in YVL Guide 6.2.

5.5 Containment integrity

In section 17 of the Council of State Decision (395/91) the following is required:

"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 presented in YVL Guide 1.0, sub-section 3.3.

5.6 Releases and radiation doses

According to section 7 of the Council of State decision (395/91), 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 above Decision and in Guide YVL 7.1 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

The following is prescribed in section 10 of the Council of State Decision (395/91):

“The limit for the dose of the 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

The following is prescribed in section 11 of the Council of State Decision (395/91):

“The limit for the dose of the 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

The following is prescribed in section 12 of the Council of State Decision (395/91):

“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 transients

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.

Postulated accident

A postulated accident means such a nuclear power plant safety system design-basis event as the nuclear power plant is required to manage without any serious damage to the fuel, and 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 mean 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 components. 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 shall mitigate the consequences of accidents.

Safety system

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

Safety functions

Safety functions are safety-significant functions to prevent the occurrence or propagation of transients and accidents 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. More detailed instructions concerning single failures are given in Guide YVL 2.7.