Translation

Transient and accident analyses for justification of technical solutions at nuclear power plants

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1 Introduction

This guide deals with the transient and accident analyses required to support the application for a construction and an operating licence of a new nuclear power plant.

The principles which aim at ensuring nuclear power plant safety are presented in the Guide YVL 1.0, *Safety criteria for design of nuclear power plants* /1/. Provisions made for Anticipated Transients and Postulated Accidents are of vital importance, too. This presupposes the design of the reactor and its cooling system in such a way that sufficiently good starting-points exist for maintaining the plant in a safe condition under the above-mentioned circumstances. This also presupposes the fitting of the plant with safety systems the principles of operation of which are passive or which are activated when the need arises. Safety functions such as reactor shutdown, reactor core cooling, removal of decay heat as well as the prevention of the dispersal or radioactive materials into the environment shall take place reliably. The Guide YVL 1.0 also presupposes that provisions are made for the possibility of Severe Accidents.

With the help of the analyses according to this Guide, plant behaviour, potential releases and the radiation doses during postulated design basis events are examined. By analyses, the appropriateness of the designed technical solutions in the carrying out of pre-determined safety functions is justified. This means that i.a. the following items are studied:

- reactor and reactor core cooling system do not contain special features which, by aggravating the effects of analysed transients or accidents, would significantly hinder the maintaining of a safe state,
- safety systems designed for each event to be studied carry out their tasks,
- automatic 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 and
- radiation doses received in the plant surroundings are restricted by means of adequate systems.

Analyses are based on deterministic assumptions on the occurring faults and the functioning of components and systems. A so called conservative method of treatment is characteristic of these analyses, also in other respects. This means i.a. the following choices and assumptions which have an unfavourable effect on the results:

- faults which are obviously unlikely are assumed in safety systems,
- unknown parameters or parameters which vary normally within a certain range are selected from the worse end of a potential range and
- deficiencies in the computation model are compensated for by assumptions which aggravate the results and simplify the analysis.

Owing to the conservative method of treatment, the analyses according to this Guide do not picture the most likely course of a transient or an accident. It is therefore to be carefully considered to what extent these analyses can be used for purposes other than the assessment of the acceptability of certain technical solutions at a nuclear power plant.

The quantitative assessment of a nuclear power plant's safety as a whole, the compilation of the emergency operating procedures, the assessment of a site's acceptability and the emergency planning for the protection of the public in the plant's vicinity require analyses the method of treatment and coverage of which differ from the analyses in this Guide. Such analyses are included as part of the so called PSA analysis which is dealt with in the Guide YVL 2.8, *Probabilistic safety analyses (PSA) in the licensing and regulation of nuclear power plants* /2/. The effects of fires are analysed according to the Guide YVL 4.3, *Fire protection at nuclear facilities* /3/.

The Finnish Centre for Radiation and Nuclear Safety checks in connection with the review of the applications for construction and operating licences for nuclear power plants that a plant's technical solutions have been sufficiently justified with the help of incident and accident analyses. The results are presented in the Preliminary and Final Safety Analysis Reports. More detailed information on the initial assumptions and methods of calculation used in the analyses may be presented either in the Safety Analysis Report or topical reports.

For the construction licence, it is essential to demonstrate a plant type's general acceptability and to look into such plant features in particular the modification of which is not possible in the later stages of design. As regards e.g. the safety systems, however, simplified assumptions may be made within such limits as are technically feasible alternatives. For the operating licence, the analyses are completed and the plant's structure is described so that it corresponds with the final design to the extent possible.

The Guide YVL 1.1 /4/ deals in more detail with the procedures for approval of the Preliminary Safety Analysis Report and the Final Safety Analysis Report and the role of the Finnish Centre for Radiation and Nuclear Safety in the review of licence applications.

2 Events to be analysed

Such events shall be analysed as, by 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 such events will be analysed as are the most restricting ones with regard to the dimensioning of each safety system.

In the following, the events to be analysed have been classified into two groups according to what each analysis is intended to show. In sub-section 2.1, instructions 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 - 5.6. Another group consists of analyses relating to the releases and offsite radiation doses. They are discussed in sub-section 2.2. and the requirements for their approval are set forth in sub-section 5.7. 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 is 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 powers or fuel burn-ups) shall be analysed. The events to be selected as initiating events

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

Examples of initiating events are faults which have the following consequences:

- leak from primary circuit,
- leak from secondary circuit (PWR),
- leak from primary to secondary circuit (PWR),
- reactor power control malfunction,
- disturbances in primary circuit flow, pressure control or water volume control,
- steam pressure or steam flow transient and
- feedwater flow or feedwater temperature transient.

A transient or an accident relating to each initiating event is analysed using the parameters and assumptions given in sub-sections 4.1 and 4.2.

The cases to be analysed are 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.

The basic alternative for each case is classified according to the probability of the initiating event. Should the need arise for some initiating event to analyse several alternatives (the basic alternative and the alternatives starting from the different plant conditions or containing further faults), the instructions given in sub-section 4.2.5 are complied with. Depending on the case, the alternatives containing further faults may be Postulated Accidents even though they could be classified as Anticipated Operational Transients on the basis of the initiating event.

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 dealt with which lead to core degradation e.g. as a result of the loss of a safety function.

The objective of the analyses of Severe Accidents is, in this connection, to study factors which affect containment integrity, leak tightness and the operability of containment systems. They are conducted for cases which may be the worst from the viewpoint of the functioning of the containment. They could include i.e.:

- total loss of AC power,
- total loss of feedwater,
- leak of primary coolant without emergency cooling, and
- leak of primary coolant and blockage of coolant recirculation.

The analyses in this Guide do not deal with such cases 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 materials (e.g. release of reactor coolant into the environment), the radiation doses caused by the release shall be calculated.

Postulated Accidents

Radiation dose calculations shall be conducted for Postulated Accidents for which the dose upper limit cannot be concluded from the results of other analyses. E.g. the following cases may be dimensioning from the radiation doses' point of view:

- Loss of coolant caused by an extensive primary circuit rupture. This is a typical example of accidents during which radioactive materials are first released within the containment and only gradually leak out. Dose calculation covers most incidents and accidents analysed according to sub-section 2.1.

- Leak of reactor coolant out from the containment as a consequence of an instrument line rupture.

- Leak from steam generator primary to secondary side. At least the total rupture of one steam generator tube shall be dealt with here, as well as a more extensive leak if such is estimated possible on the basis of the structure of the steam generator (PWR).

- Unisolated leak in a steam line outside the containment, also assuming that a maximum primary to secondary circuit leak as stipulated by the Technical Specifications has occurred in the respective steam generator already long before the accident's initiation (PWR).

- Outside-the-containment leak in a steam line or a reactor coolant purification line (BWR).

- Damage in an outside-the-containment system containing radioactive gases.

- Damage in an outside-the-containment system containing radioactive liquids.

- Damage of a fuel assembly which has been removed from the reactor.

- Drop during hoisting of a transfer or transport cask containing spent fuel, in a situation where the cask is not tightly bolted.

- Drop of a heavy object on top of stored fuel or an open reactor.

Severe Accidents

Releases caused by a Severe Accident shall be calculated for a case which on the basis of containment pressure and temperature conditions and the concentration of radioactive materials in the containment air space, is estimated to cause the most extensive releases.

If a specific Severe Accident imposes such loads on the containment that a controlled venting is inevitable for preventing containment rupture, releases are calculated for that case in particular.

If, in conjunction with a Severe Accident, such loads are created as may cause a local leak in a containment penetration, a personnel or a material air lock or a through-the-containment pipeline, also releases via the leak pathway in question shall be taken into account in the release calculatios. Such a leak pathway could be established e.g. when the steam generator pipe of a PWR plant ruptures under high pressure and at high temperature.

If significant releases are calculated to occur at such short notice that the possibilities of evacuating the local population prior to the initiation of the release are questionable, also short-term radiation doses to the member of the critical group are calculated in addition to the relases.

3 Methods of calculation

The reliability of the methods of calculation employed in the analyses shall be justified. A description of all the used methods of calculation shall be made available in which the general principles of the methods of calculation, physical models and numerical methods of solution are presented.

The experimental correlations potentially 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 are verified by means of adequate reference calculations.

Physical models are verified by demonstrating their ability to depict suitable tests of independent phenomena, tests of complete systems or incidents at a nuclear power plant. Also, comparison with other, earlier verified models may be utilized.

4 Assumptions used in analyses

4.1 Parameters

Parameters affecting the final outcome of the analysis which is essential from the acceptance requirements' point of view such as:

- process parameters (power, pressure, temperature, etc) at the accident's initiating moment,
- accuracy of the limits of trip used in the protection systems,
- component performance parameters, and
- inaccurately known factors (manufacturing tolerances, heat transfer coefficients, mixing phenomena, condensing phenomena, etc),

shall be selected from the edge of their likely range of variation (e.g. 95 % point in the cumulative distribution) so that the final result can be considered conservative.

Decay heat power shall be defined using the standard ANSI/ANS-5.1-1979, Decay Heat Power in Light Water Reactors. In applying the standard, the actual time of use of fuel in the reactor may be taken into account. Uncertain factors affecting the decay heat power shall be chosen conservatively, however, with the objective of decay heat not exceeding the value defined for it with 95 percent's probability and using a 95 % reliability level.

4.2 Functioning of components and operator activities

4.2.1 Protection systems

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

4.2.2 Safety systems

Safety systems operate at the designed minimum output unless an accident directly affects their operability. Minimum output is attained when

- a combination of faulty components and components under maintenance which most hinders system operation is assumed according to the Guide YVL 2.7 /5/ 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 requirements in sub-section 5.1)

The faults meant later in sub-section 4.2.5 do not contain such faults as would have any direct effect on any safety function since they are already taken into account when the minimum output of the systems is defined.

4.2.3 Normal operating systems

Normal operating systems operate in the way estimated as most probable in the base case of each event to be analysed. In sub-section 4.2.5 the need to analyze several alternatives of a certain case has been dealt with so that the assumptions concerning the functioning of normal operating systems are modified.

4.2.4 Operator activities

Operators act in the way assumed as most likely in the basic alternative of each case to be analysed. When estimating operator activity, the probability of any faulty action shall be assessed in particular.

Actions for the mitigation of an incident or an accident may be considered likely only on the following conditions:

- event is clearly identifiable,
- there are clear instructions in the control room on the actions to be taken and the circumstances under which an action is taken, and
- the time of consideration preceding the actions is estimated to be adequate.

Operator actions assumed in the analyses shall always be justified taking the aforementioned matters into consideration.

Where operator action is concerned, an analysis of several alternatives according to the principles stated in sub-section 4.2.5 may also be considered.

4.2.5 Evaluation of various event alternatives

First of all, the base case from various events is analysed using the assumptions in sub-sections 4.2.1 - 4.2.4 . If a fault in any individual component of the normal operating systems or operator activity which deviates from the assumed would essentially affect the course of events and might aggravate the consequences, several analyses of the same initiating event shall be conducted at discretion. In some cases, a simplified study may then be sufficient as the analysis of the base case by which the base case is shown to be milder than the alternative case which is analysed in detail. ATWS analyses are conducted for the base cases.

An analysis of several alternatives may be considered for such events in particular the base case of which is an Anticipated Operational Transient but milder Postulated Accident acceptance limits are used in case of an alternative which contains a faulty function.

Postulated Accident acceptance limits can be applied to such alternatives in which the frequency of an Anticipated Operational Transient and erroneous functioning which aggravates it can be justified to be below 10^{-2}/year. In those cases,it shall be specifically shown that the milder base case meets the acceptance requirements laid down for Anticipated Operational Transients.

Typical examples of malfunctions which require alternative studies are:

- loss of external electricity,
- jamming open of a safety valve which opens during the course of an accident,
- remaining open of a valve which is required for the isolation of a leak,
- malfunction of 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 and
- delay of a necessary operator action.

4.2.6 Mitigation of consequences of 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, such active components may be assumed operable the operation of which is independent of the causes and consequences of a Severe Accident.

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 estimated in such cases.

Accident mitigating actions for which sufficient instructions have been issued in advance and which will be started after the accident's initiation, can be taken into account. They may be based on e.g. the utilization of systems which are independent of the plant's fixed equipment. Actions shall be justified according to sub-section 4.2.4.

A controlled venting for restricting containment pressure may be assumed if appropriate facilities have been designed for this purpose and written instructions for their use are available.

4.3 Assumptions employed for radiation dose calculations

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

During the moment of initiation of the accident, the amount of radioactive materials in the primary coolant is assumed to be at least the same as is intended to be set as the limit in the plant Technical Specifications. The distribution of isotopes is chosen so that it corresponds in practice to the distribution noted in plants of the same type in cases of fuel leaks. As of the moment of time when reactor power starts changing significantly (to decrease or increase), an increase in the iodine and cesium concentrations shall be assumed which corresponds to the most extensive increase in the mentioned concentrations in connection with power changes which have been observed at the type of plant in question.

The primary coolant leak rate shall be estimated using a model which is known to be conservative. 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 materials is automatic and carried out and ensured by means of an appropriate protection system, the system may be assumed to function in the designed manner.

If a leak occurs inside the plant, it may be assumed that an environmental release will only be caused by those radioactive materials which are in the vaporizable part of the leak. In addition, it may be assumed that the concentration of radioactive materials (per steam weight unit) in the vaporizable part is lower than in the coolant before arrival to the leak. The coefficient indicating a decrease in concentration shall be justified by means of a reference to practical observations or test results. A corresponding assumption of a decreased concentration of radioactive materials in steam (in comparison to the water from which vaporization happens) can be made if the leak is a pure vapour leak direct into the environment (e.g. a steam line leak when the steam generators have not yet filled up with water). As an exception from the above, it is assumed, however, that all the noble gases present in the leaking coolant will always enter the environment in their entirety.

If a direct leak into the environment occurs and coolant is in water form when reaching the leak, all the radioactive materials the leak contains shall be taken into account in the calculation of offsite doses.

The steam which has leaked into the plant internals and the radioactive materials which have mixed with it are assumed to disperse 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.

4.3.2 Loss of coolant by large primary circuit break

The period of time the primary coolant takes to discharge into the containment is selected on the basis of thermohydraulic analyses. The time shall be appropriately shorter than the shortest calculated length of time, taking into account the accuracy of the calculating method. Assumptions concerning

- radioactive materials in primary coolant,
- division of radioactive materials into vaporizable and condensing share of leak, and
- the state of the iodine which has become mixed with steam

will be made as presented in sub-section 4.3.1.

The point of time of failure of fuel rods and the number of failed rods shall be selected conservatively taking into account the results of analyses which relate to plant behaviour. The chosen figure shall be at least as high as the most extensive result gained in the analyses of Postulated Accidents, regardless of whether the result relates to a loss of coolant or some other accident.

Before the accident, the reactor is assumed to have been operating at full power since the previous refuelling and the core composition is assumed to represent an equilibrium core at the end of the fuel cycle.

The percentages of radioactive materials assumed to escape from the failed rods are chosen so that they can be justified on the basis of experimental research and operating experience of the fuel type in question.

A certain share of the radioactive materials released from the failed fuel rods is assumed to enter the containment airspace direct. The rest of the released radioactive materials are first assumed to have become dissolved in or mixed with the cooling water. The distribution between airspace and cooling water shall be justified.

The failed rods are assumed to emit more radioactivity later when cooling water enters the rods and dissolves fuel. These shares of radioactive materials which initially remain in the water, shall be justified by experimental research, or the assumptions concerning them shall be made conservatively.

Assumptions concerning the transport of radioactive materials within the containment may be based on experimental research if the results are applicable to the situation in question and are reliably verified. Alternatively, a conservative code may be used which gives a slower than normal disappearance of radioactive materials from the airspace.

If air is discharged from the containment during normal plant operation, the mixing of radioactive materials with the discharged air is estimated conservatively. The isolation of ventilation is assumed to take place in a way equivalent 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 is assumed to function in the normal way.

After containment isolation, radioactive materials are assumed to mix evenly with the airspace of the whole containment. The containment leak rate is selected by taking into account the tightness requirement set for the containment and the containment overpressures calculated during the analysis of Postulated Accidents. Appropriate safety margins are employed during the selection.

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

The ventilation of the space surrounding the containment is assumed to function in the way designed for accident conditions and the releases arising from a containment leak are calculated accordingly. If the ventilation system is operated normally with the filters bypassed, the time spent in the possible switchover 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.

4.3.3 Accidents in spent fuel handling

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

- spent a full fuel cycle in a reactor which was operating at full power,
- was located in the reactor's most heavily loaded position and reached a full discharge burn-up,
- has cooled down for 3 days after reactor shutdown, and
- 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 is assumed that

- accident may happen in any quarters and at any time when a transport cask is being lifted with the lid open or insufficiently bolted,
- cask has been filled up with fuel which has reached a full discharge burn-up,
- cooling time allowed for fuel prior to transfer is the minimum time as prescribed in the administrative restrictions and
- with a suitable safety margin, the number of failed fuel assemblies exceeds the number estimated on the basis of loads caused by an accident.

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

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

Such percentages are assumed to be released from the radioactive materials 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 are assumed to get to the airspace of the building in question. A separately justified water decontamination factor may be used for iodine in case of an underwater fuel damage. This means that part of the iodine isotopes will remain in the water and only part will get to the airspace above water.

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

The radioactive materials which came into the airspace are first 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 in the above mentioned situation be used in several different ways, the way shall be chosen which leads to the most extensive releases. The potential isolation of ventilation ducts may be assumed to take place in 30 minutes. If isolation is automatic and uses an appropriate protection system, also an earlier point of time for isolation may be assumed which corresponds to the system's design. Releases may be assumed to cease after the accomplishment of isolation.

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

4.3.4 Severe accidents

Before the accident, the reactor is assumed to have been operating at full power since the previous refuelling and the fuel composition is assumed to represent an equilibrium core at the end of a fuel cycle.

Assumptions concerning the amounts of radioactive materials released into the containment airspace as a result of core degradation shall be based on experimental research adequately representative of each accident case. 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. As the first estimate, the general interdependency between pressure difference and leak rate which is based on gas dynamics may be used, assumed that no containment deformations occur which would increase leakage. In addition, the extent of any deformation-induced extra leakage in the sealings of various penetrations and air locks shall be assessed. If the consequences of controlled venting or containment local leakages need to be analysed in the release calculations, assumptions relating to the decontamination effect of components and potential filters along the release route shall be justified with appropriate experimental research.

When examining the hazard of acute health effects which a Severe Accident poses on the local people, the actual conditions on site and in its surroundings shall be taken into account. Based on these conditions, the local distribution during the initiation of the accident of the members of the critical group as well as the duration of evacuations from various distances shall be selected for the assumptions needed in the radiation dose calculations.

4.3.5 Dispersal of radioactive materials into the environment and radiation dose calculations

Releases are assumed to occur at the effective height of the release point.

Assumptions on the dispersal of radioactive materials into the air are presented in the Guide YVL 7.3, *Evaluating the dispersion of radioactive releases from nuclear power plants under operating and accident conditions* /6/.

Assumptions concerning radiation dose calculations are presented in the Guide 7.2, *Evaluation of population doses in the environment of nuclear power plants* /7/.

5 Requirements set for approval of results

Of the requirements presented in this paragraph, the sub-sections 5.1, 5.2, 5.3, 5.4 and 5.5 deal with Anticipated Operational Occurrences and Postulated Accidents. Sub-section 5.6 deals with Severe Accidents and sub-section 5.7 is applicable to all three classes of events.

5.1 Operation of systems designed for accident mitigation

It shall be shown that the systems designed for accident mitigation will not subject the power plant 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 of plant to safe state

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

This requirement applies to leakages in particular during which regular cooling methods cannot be used.

5.3 Cooling system overpressure protection

Requirements for the overpressure protection of PWR plants are given in the Guide YVL 2.4 /8/. Overpressure analyses which are in conformity with the mentioned Guide may, where applicable, also be used as transient and accident analyses.

Sub-sections 2.2 and 2.3 of the Guide YVL 2.4 are also to be applied to BWR plants, with the exception of the sections dealing with the overpressure protection of the secondary circuit, durability of steam generators and the consequences of a stuck-open safety valve.

5.4 Fuel failures

A fuel rod is assumed to fail if it undergoes a heat transfer crisis or if a local energy pulse results in the mean enthalpy on rod cross-section exceeding 586 J/gUO_2 (140 cal/g). Also other potential failure modes shall be taken into account in the assessment of the number of fuel rods which could fail.

Anticipated Operational Transients

The probability of a fuel damage resulting from a heat transfer crisis or some other reason shall be shown as insignificant.

If a correlation is used in the analyses which describes the likelihood of the occurrence (DNB relation or CHF relation) of a local heat transfer crisis, it shall be shown that even in the hottest fuel rod with 95 percent's probability and using the 95 % confidence level, no heat transfer crisis will occur during any single Operational Transient.

If the analysis is based on the critical power relation correlation, the minimum critical power relation (MCPR) during plant operation shall be selected so that 99.9 % of the fuel rods in the reactor core will avoid undergoing a heat transfer crisis during a transient.

Postulated Accidents

The general design criterium is to keep the number of fuel damages as low as reasonably achievable for each type of accident. If fuel damages are calculated to occur during some accident, it shall be separately studied how the damage rate could be restricted with the help of modifications in plant structure or fuel design and plant operation.

The requirement for all Postulated Accidents is that no single fuel rod shall undergo a local energy pulse as the result of which the mean enthalpy on rod cross-section would be in excess of 963 J/gUO_2 (230 cal/g).

For the part of LOCAs, it shall also be shown that the requirements presented in the Guide YVL 6.2 /9/ Section 3 relating to fuel cladding embrittlement and structural deformations, shall be fulfilled.

5.5 Containment integrity

No transient or Postulated Accident may cause such pressure or temperature within the containment as would exceed the value of the corresponding parameter which has been used as the containment design basis.

No jet forces or missiles caused by a transient or a Postulated Accident may endanger containment integrity.

5.6 Mitigation of consequences of severe accidents

The probability of the occurrence of such a mixture of gases as could burn or explode in a way which would endanger containment integrity shall be highly insignificant.

Accident-induced jet forces or missiles must not jeopardize containment integrity.

The pressure and temperature which may arise inside the containment as the result of a Severe Accident must not exceed the limit values which the containment may justifiably be estimated to withstand without a significant loss of tightness.

The long-term cooling of the core debris of the damaged reactor at the bottom of the containment shall be effective enough to restrict the release of radioactive isotopes into the containment airspace and to prevent the penetration of the debris through containment bottom as well as a containment failure caused by radiation heat emanating from core debris.

5.7 Releases and radiation doses

A general design criterium for nuclear power plants is to keep radiation doses as low as reasonably achievable (the so called ALARA principle). Staying below the limits presented in the following which are contained in the Guide YVL 7.1, *Limitation of public exposure from nuclear facilities* /10/, is therefore not alone an adequate reason for the non-implementation of any solution which would essentially decrease occupational doses, population doses or radioactive contamination in the environment. In addition to staying below the limits, the benefits of the solution and the incurring costs shall be assessed in the first place.

Anticipated Operational Transients

The individual dose limit is 0.1 mSv. It shall be shown by analyses that this limit would not be exceeded as a consequence of any single Anticipated Operational Transient. The limit is to be applied to the effective dose-equivalent commitment of the individual in the critical group. The assumptions concerning the individual in the critical group are presented in the Guide YVL 7.2.

The collective dose limit is 5 manSv/GW$_e$ (per installed electrical power). It shall be shown by analyses that this limit would not be exceeded as a result of any single Anticipated Operational Transient. The limit is to be applied to the global collective effective dose-equivalent commitment of the population truncated at 500 years.

Postulated Accidents

The individual dose limit for a Postulated Accident is 5 mSv. This limit is to be applied to the effective dose equivalent of the individual in the critical group calculated from the external radiation dose during one year and from the radioactive materials uptake by the body during the same time.

Collective doses arising from Postulated Accidents shall also be analysed.

Severe Accidents

The release of radioactive materials caused by a Severe Accident which is analysed according to this Guide shall not be so extensive as to cause acute radiation effects among the local population or to restrict the use of extensive land and water areas in the long term.

In order to meet the requirements relating to long-term effects, it shall be shown that

- a cesium release will not be in excess of 0.1 % of the cesium inventory in the reactor and that
- a combined release of other nuclides is not so extensive that the fallout consisting of them would in the long-term (period of time which starts 3 months after the accident) cause a heavier combined external and internal radiation dose than the aforementioned cesium release.

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4 Guide YVL 1.1, The Institute of Radiation Protection as the supervising authority of nuclear power plants.
5 Guide YVL 2.7, Failure criteria for the design of a light-water reactor.
6 Guide YVL 7.3, Evaluating the dispersion of radioactive releases from nuclear power plants under operating and accident conditions.
7 Guide YVL 7.2, Evaluation of population doses in the environment of nuclear power plants.
8 Guide YVL 2.4, Overpressure protection and pressure control during disturbances in the primary circuit and steam generators of a PWR plant.
9 Guide YVL 6.2, Fuel design limits and general design criteria.
10 Guide YVL 7.1, Limitation of public exposure from nuclear installations.

This guide is a translation of the Guide YVL 2.2 issued on 7 Oct. 1987.

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