

DETERMINISTIC SAFETY ANALYSES FOR A NUCLEAR POWER PLANT

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continues

With regard to new nuclear facilities, this Guide shall apply as of 1 December 2013 until further notice. With regard to operating nuclear facilities and those under construction, this Guide shall be enforced through a separate decision to be taken by STUK. This Guide replaces Guide YVL 2.2.

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Authorisation

According to Section 7 r of the Nuclear Energy Act (990/1987), *the Radiation and Nuclear Safety Authority (STUK) shall specify detailed safety requirements for the implementation of the safety level in accordance with the Nuclear Energy Act.*

Rules for application

The publication of a YVL Guide shall not, as such, alter any previous decisions made by STUK. After having heard the parties concerned STUK will issue a separate decision as to how a new or revised YVL Guide is to be applied to operating nuclear facilities or those under construction, and to licensees' operational activities. The Guide shall apply as it stands to new nuclear facilities.

When considering how the new safety requirements presented in the YVL Guides shall be applied to the operating nuclear facilities, or to those under construction, STUK will take due account of the principles laid down in Section 7 a of the Nuclear Energy Act (990/1987): *The safety of nuclear energy use shall be maintained at as high a level as practically possible. For the further development of safety, measures shall be implemented that can be considered justified considering operating experience, safety research and advances in science and technology.*

In accordance with Section 7 r(3) of the Nuclear Energy Act, *the safety requirements of the Radiation and Nuclear Safety Authority (STUK) are binding on the licensee, while preserving the licensee's right to propose an alternative procedure or solution to that provided for in the regulations. If the licensee can convincingly demonstrate that the proposed procedure or solution will implement safety standards in accordance with this Act, the Radiation and Nuclear Safety Authority (STUK) may approve a procedure or solution by which the safety level set forth is achieved.*

1 Introduction

101. The IAEA's general safety requirements [3] prescribe that the safety of nuclear power plants shall be assessed. Guide YVL B.3 presents the requirements for the nuclear power plant's deterministic safety analyses.

102. Under Section 3 of Government Decree 717/2013, *the safety of a nuclear power plant shall be assessed when applying for a construction license and operating license, in connection with plant modifications, and at regular intervals during the operation of the plant. It shall be demonstrated in connection with the safety assessment that the nuclear power plant has been designed and implemented in a manner that meets the safety requirements. The safety assessment shall cover all the nuclear power plant states. The safety of a nuclear power plant shall be assessed also after accidents and, whenever necessary, on the basis of the safety research results.*

103. Under Section 3 of Government Decree 717/2013, *nuclear power plant safety and the technical solutions of its safety systems shall be assessed and substantiated analytically and, if necessary, experimentally. Analytical methods include transient and accident analyses, analyses of internal and external hazards, strength analyses, failure resistance analyses, failure mode and effects analyses, and probabilistic risk assessments. The analyses shall be maintained and revised as necessary, taking into account operating experience from the plant itself and from other nuclear power plants, the results of safety research, plant modifications, and the advancement of calculation methods.*

104. Under Section 3 of Government Decree 717/2013, *the analytical methods employed to demonstrate compliance with safety requirements shall be reliable and well qualified for the purpose. The analyses shall demonstrate the conformity with the safety requirements with high certainty. Any uncertainty in the results shall be assessed and considered in determining safety margins.*

105. Sections 9 and 10 of Government Decree 717/2013 set release and dose limits for radioactive substances during anticipated operational occurrences, postulated accidents, design extension conditions and severe accidents.

106. Section 13 of Government Decree 717/2013 prescribes the principles for ensuring the integrity of the fuel, the primary and secondary circuit as well as the containment during plant normal operational conditions, operational occurrences and accidents.

2 Scope of application

201. Guide YVL B.3 applies to deterministic safety analyses for licensing of new nuclear power plants, plant modifications of operating nuclear power plants and periodic plant safety assessments.

202. A separate decision will be made on how the Guide applies to other nuclear facilities.

2.1 Other Guides concerning safety analyses

203. Requirements pertaining to the nuclear power plant's risk management and probabilistic risk assessments are given in Guide YVL A.7.

204. The requirement for the reactor and fuel behaviour analyses is set forth in para 608 of Guide YVL A.6.

205. The requirement for analyses made in order to prevent criticality accidents is set forth in chapter 5 of Guide YVL B.4.

206. Protection of the nuclear power plant from internal and external hazards and the analysis methods pertaining to the hazards are presented in Guide YVL B.7.

207. The requirements for analyses of releases and doses are set forth in Guide YVL C.4.

208. The requirements for analyses of emergency situations and the emergency plan are set forth in Guide YVL C.5.

209. The requirements for the stress analyses, brittle fracture analyses and leak-before-break analyses of the nuclear power plant's primary circuit and other important nuclear pressure equipment are given in Guide YVL E.4.

210. The requirements for analyses of failures of I&C systems are set forth in chapter 5.2 of Guide YVL B.1.

2.2 Associated guides

211. The requirements for failure criteria used in deterministic safety analyses and the requirements for achieving a controlled and safe state are given in Guide YVL B.1.

212. The acceptance criteria for nuclear fuel are given in Guide YVL B.4.

213. The acceptance criteria for the pressure control of the nuclear power plant's primary circuit are given in Guide YVL B.5.

214. The acceptance criteria for the integrity of the nuclear power plant's containment are given in Guide YVL B.6.

3 Events to be analysed

301. Analyses pertaining to the plant's behaviour as well as releases of radioactive substances and radiation doses shall cover the nuclear power plant's normal operational states, anticipated operational occurrences, postulated accidents, design extension conditions and severe reactor accidents. Examples of the events to be analysed are given in [4 and 5].

302. The scope of the analysed events shall provide a comprehensive assessment of the nuclear power plant's behaviour during incidents and accidents as well as releases and doses due to incidents and accidents.

303. Operator actions shall be assessed to identify essential operator actions needed in accident management and to assess the effects of potential operator errors.

304. The inadvertent actuation of every system accomplishing a safety function shall be addressed as an initiating event.

305. Pressure control analyses for the reactor coolant circuit shall consider cases during which the reactor pressure tends to increase or decrease in consequence of an initiating event, and situations where the coolant circuit pressure must be increased or decreased.

306. Pressure control analyses shall cover also low operating temperatures. Brittle fracture must be analysed for the most loaded parts of the ferritic steel components of safety class 1 pressure equipment of a nuclear power plant, as required in chapter 6 of Guide YVL E.4.

307. The service loadings of the nuclear power plant's main components during operation and their recurrence during the entire life cycle shall be analysed. The analyses shall also include allowable loadings under low and high operating temperatures. Based on the allowable loadings, safe pressure and operating ranges shall be determined for the normal operation of the equipment. The requirements for analyses of loadings and strength analyses of a nuclear power plant's primary system and other nuclear pressure equipment important to safety are set forth in Guide YVL E.4.

308. Severe reactor accident analyses shall cover all actions required for the plant's severe reactor accident strategy and the phenomena associated with the strategy.

4 Analyses of plant behaviour

4.1 General

401. Analyses shall cover anticipated operational occurrences and accidents that determine or limit the dimensioning of systems accomplishing safety functions.

402. Anticipated operational occurrences and accidents shall be analysed starting from the initiating event and ending in a safe state.

4.2 Methods of analysis

403. The suitability of analysis methods for their purpose shall be justified.

404. A description of the models and calculation methods used in the analyses shall be given. The models shall be described to a level of detail that facilitates conducting of verifying analyses. The information to be presented shall include the analysis model representing the plant or its component (e.g. the division into nodes applied in the model), justification of the selected model parameters as well as the plant data used for the analyses or a reference to the source of the available plant data.

405. The validation of the physical models and computer code used for the analyses shall be substantiated by comparing their calculation results to separate effects tests or tests carried out on entire systems, or to disturbances that have occurred at nuclear power plants. Comparison with models that have already been validated may also be utilised.

406. The plant and fuel type specific experimental correlations used in the calculation methods 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.

407. If reliable calculation methods are not available, the acceptability of the technical solution in question shall be justified by means of experiments.

408. The accepted methods to be used in the plant behaviour analyses are either the conservative analysis method supplemented with sensitivity studies or the best estimate method supplemented with uncertainty analysis.

409. Sensitivity studies supplementing conservative analyses shall define how sensitive the results are for the models used, the initial conditions and the main parameters.

410. Utilisation of the best estimate method shall be supplemented with an uncertainty analysis that is justifiable by statistical methods. Examples of such methods are given in [6].

4.3 Assumptions for the analyses

411. The initial conditions of the conservative analyses and the conservativeness of the parameters chosen shall be justified. If the choice that is the least beneficial in terms of the acceptability of the end result is not unambiguous, analysis results covering the parameter's entire range of variation shall be presented.

412. When using the best estimate method, the failure combination that is the least beneficial to the functionality of the systems shall be chosen in accordance with the failure criteria presented in chapter 4.3 of Guide YVL B.1.

413. The selected consideration time preceding operator actions and the time to accomplish the actions shall be sufficiently long. The durations chosen shall be justified. Operators can be assumed to act on each analysed event in accordance with the procedures available in written or electronic form.

4.3.1 Assumptions for anticipated operational occurrence analysis

414. Anticipated operational occurrences shall be analysed in two ways as follows:

1. All plant systems operate according to design, with the exception of the failure or operator error analysed as the initiating event and the consequences of the initiating event.
- 2a Actuation of non-safety classified systems shall not be postulated as systems mitigating the consequences of the initiating event. Operation of non-safety classified systems shall be postulated if a system's designed operation could aggravate the consequences of the initiating event.
- 2b The most penalising failure in accordance with the failure criteria given in chapter 4.3 of Guide YVL B.1 shall be postulated in safety class 2 or safety class 3 systems designed for anticipated operational occurrences or postulated accidents.

3c Performance values for functioning components shall be chosen conforming to the acceptance limits in periodic tests.

4.3.2 Assumptions for postulated accident analysis

415. Safety-classified systems shall be assumed to operate at their minimum system performance during postulated accidents.

416. Actuation of non-safety classified systems shall not be postulated as systems mitigating the consequences of the accident. Actuation of non-safety classified systems shall be postulated if a system's designed operation could aggravate the consequences of the initiating event.

417. In analyses of category 2 postulated accidents, only safety class 2 systems may be assumed to be systems mitigating the accident from the initiating event to the controlled state. Operation of systems in lower safety classes shall be postulated if a system's designed operation could aggravate the consequences of the initiating event.

418. Loss of the external grid shall be combined with postulated accidents if it could aggravate the consequences of the initiating event.

4.3.3 Assumptions for design extension condition analyses

419. For DEC A accidents, the most penalising single failure shall be assumed in one of the systems whose operation is required to accomplish a safety function in the event in question. For DEC B and C accidents, a single failure need not be assumed. The consequences of an initiating event shall be assumed in the analyses.

420. Loss of the external grid need not be combined with other initiating events in design extension condition analyses unless it is the likely consequence of an initiating event.

421. In design extension condition analyses, best estimate methods can be applied concerning assumptions of the plant's initial state and the performance of operating subsystems.

422. In design extension condition analyses, application of the best estimate method need not be complemented with an uncertainty analysis as required in para410.

4.3.4 Assumptions for severe reactor accident analysis

423. In analysing severe reactor accidents, best estimate methods can be applied concerning assumptions of the plant's initial state and the performance of operating subsystems. However, the more essential the function, the better assurance for its successful accomplishment shall be provided.

424. In severe accident analyses, application of the best estimate method need not be complemented with an uncertainty analysis as required in para410.

425. In severe reactor accident analyses, the most penalising failure according to the failure criterion presented in chapter 4.3 of Guide YVL B.1 shall be assumed for systems designed for severe reactor accident management. Consequences of the initiating event shall also be taken into account.

426. The time needed for actions required for the severe reactor accident management strategy and other factors relating to the implementation of the actions (e.g. accessibility of locally operated equipment) shall be justified.

427. Analyses justifying the hydrogen management strategy shall separately evaluate cases in which the hydrogen generation rate increases.

4.3.5 Assumptions for cooling circuit pressure control analysis

428. Reactor cooling circuit pressure control analyses for anticipated operational occurrences shall only be performed as determined by part 1 of para414.

429. Cooling circuit pressure control analyses for accidents shall be performed as required in chapters 4.3.2–4.3.4.

430. In analyses of postulated accidents leading to pressure increase, assumptions for the analyses shall be chosen with the following amendments and additions:

1. Reactor scram occurs from the second signal of the reactor protection system .
2. Pressure reduction systems other than safety valves and the equivalent blow-off valves fail.
3. Safety valves and equivalent blow-off valves are assumed to fail in the closed position as follows:

total number of valves	failing
2–3	1
4–8	2
≥ 9	one fourth of the total number, rounded up to the next whole number

4. The discharge flow capacity of safety valves and equivalent blow-off relief valves equals the nominal capacity determined on the basis of an applicable standard and the opening pressure equals to the nominal setting.
5. Safety valves and equivalent blow-off valve relief valves are arranged in accordance with decreasing capacity. Equal capacity valves are further arranged in relation to one another in accordance with increasing opening pressure. The valves thus arranged are assumed to fail as follows: first, fourth, ninth, etc.
6. If more than one control device is needed to control the operation of a safety valve or an equivalent blow-off relief valve and the control devices have been set at different pressures, the higher setting pressure shall be assumed as the opening pressure.

431. The pressure control analyses of low operating temperatures shall analyse disturbance transient and accident conditions as service loadings B, C or D as defined in Guide YVL E.4. The load grouping is presented in para 407–409 of Guide YVL E.4.

5 Release and radiation dose analyses

5.1 Events to be analysed

501. Release and radiation dose analyses shall be performed on those transient and accident cases required in para 301 which are limiting as regards the release of radioactive substances and radiation doses. Selection of the limiting cases shall be justified.

502. The analyses referred to in para 501 shall be supplemented with an analysis of the containment's retention capability, in which the source term into the containment is calculated according to the maximum number of failed fuel rods (10%) allowed in a class 2 postulated accident as required in para 417 of Guide YVL B.4 .

503. Chapter 3.2 of Guide YVL D.3 lists operational occurrences and accidents that shall be postulated for nuclear fuel handling and storage.

5.2 Methods of analysis

504. The requirements for plant behaviour analysis methods, which are given in chapter 4.2, apply to the release analysis methods.

505. Requirements for the analysis methods to be used in calculating population doses in the nuclear power plant's environment are given in Guide YVL C.4.

5.3 Assumptions for release and dose related analyses

5.3.1 General assumptions

506. In analysing releases, the same assumptions shall be used to describe the plant as are used in the analyses in chapter 4.3.

507. The inventory of radioactive substances in the primary coolant at the initiation of an event shall be assumed to be at least equal to the limit set in the Operational Limits and Conditions of the plant.

508. The number of leaking fuel rods before an accident shall be chosen in conformity with para 507.

509. The increasing release of fission products due to the pressure difference change between the failed fuel and coolant shall be taken into account in evaluating the concentration of radioactive substances in the primary coolant. The increase in concentration and its dependence on time shall be justified.

510. The effect of cooling water that enters the failed fuel rods on the release of radioactive substances shall be taken into account in the analyses.

511. The distribution of radioactive substances into gaseous and liquid phases of the leaking substance shall be justified.

512. A fraction of the iodine mixed with the steam shall be assumed to be gaseous. The distribution of iodine into gaseous and aerosol phases shall be justified.

513. The halogens released into airspace shall be assumed to be partly bound to inorganic and partly to organic compounds. The distribution into the various kinds of compounds shall be justified.

514. The radioactive substances entering the airspace shall first be assumed to be transported into the environment via the ventilation and filtering system in a way corresponding to the normal functioning of the system. If the ventilation system can be used in several different ways in the above-mentioned situation, the way leading to the most extensive releases shall be chosen for the analysis.

515. Isolation of ventilation may be assumed in accordance with the design of the plant's protection systems, so that any changes in the parameters used as protection limits during accidents are assessed conservatively.

516. 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 release calculations shall be separately justified.

5.3.2 Fuel handling related postulated accident assumptions

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

1. has been in the reactor core during the whole cycle at full power
2. has been located in the most heavily loaded position of the reactor core and has reached a full discharge burn-up
3. has cooled down for the shortest time of cooling possible in the accident analysed
4. is damaged in such a way that all fuel rods lose their leak-tightness.

518. If a transfer cask filled with spent fuel is lifted with the lid not tightly closed, it shall be assumed in the analyses that

1. an accident can occur in any room and at any time when a transfer cask is being lifted
2. the lid loses its leak-tightness in an accident
3. the cask has been loaded with fuel that has reached a full discharge burn-up
4. the cooling time required for fuel prior to transfer is the minimum time required in the administrative restrictions
5. the number of failed fuel rods shall be conservatively estimated.

519. In the analysis of the drop of a heavy object, it shall be assumed that

1. an accident can happen at any location where the handling of heavy objects above fuel is allowed
2. the falling object that is applied to the room in question is the most penalising one as regards the damage it causes
3. the fuel burn-up is the highest and the cool-down time the shortest possible in the accident under consideration
4. the number of damaged fuel rods shall be conservatively estimated.

520. During spent fuel handling accidents, all released noble gases shall be assumed to enter the airspace of the building in question. If fuel failure occurs under water, in estimating the release of other fission products, it can be assumed that a part of them is retained by the water and only part is released to the airspace above the water.

521. The halogens released into the airspace shall be assumed to be partly bound to inorganic and partly to organic compounds. The distribution into the various kinds of compounds shall be justified.

5.3.3 Dispersal of radioactive substances into the environment

522. Assumptions of the dispersal of radioactive substances into the environment and population dose calculations are presented in Guide YVL C.4.

6 Acceptance criteria for results

6.1 General requirements

601. In the analyses of anticipated operational occurrences, postulated accidents and design extension conditions, it shall be shown that the reactor can be shut down and maintained in shut-down state, and that the plant can be brought to a controlled state and, thereafter, to a safe 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 is possible.

602. The acceptance criteria set forth in chapters 6.2 and 6.3 are written for the conservative analysis method. In applying a best estimate method with uncertainty analysis, the result is acceptable if there is a 95% probability with 95% confidence that the examined parameter will not exceed the acceptance limit set for the conservative analysis method.

603. Chapter 4.3 of Guide YVL B.1 sets forth requirements for the reaching of a controlled and safe state.

604. Sections 9 and 10 of Government Decree 717/2013 set limits for releases of radioactive substances and doses during anticipated operational occurrences and accidents.

605. Acceptance criteria for the failure analyses of I&C systems are set forth in chapter 5.2 of Guide YVL B.1.

606. The acceptance criteria for the strength analyses of the nuclear power plant's pressure equipment are given in Guide YVL E.4.

607. The pressure control analyses performed at low operating temperatures shall demonstrate that the systems designed to prevent pressure increases are able to maintain pressure and temperature at such values that the integrity and operability of the equipment is not endangered, and that sufficient margins against rapid fracturing of the equipment are guaranteed at each service load.

6.2 Anticipated operational occurrences

608. An event to be analysed as an anticipated operational occurrence using the assumptions in item 1 of para 414 must not require the initiation of safety systems designed for postulated accidents.

609. The overpressure acceptance criterion for an event to be analysed as an anticipated operational occurrence using the assumptions in item 1 of para 414 is that design pressure is not exceeded, and that not a single safety valve opens.

610. The fuel integrity acceptance criteria for anticipated operational occurrences are given in Guide YVL B.4, chapter 4.

611. Acceptance criteria for the pressure control of the nuclear power plant during anticipated operational occurrences are given in Guide YVL B.5, chapter 4.2.

612. Acceptance criteria for the containment leak-tightness during anticipated operational occurrences is given in Section 13 of Government Decree 717/2013.

6.3 Postulated accidents

613. The fuel integrity acceptance criteria for postulated accidents are given in Guide YVL B.4, chapter 4.

614. Acceptance criteria for the pressure control and depressurisation during postulated accidents of a nuclear power plant are given in Guide YVL B.5, chapters 4.3 and 4.4.

615. The acceptance criterion for the overpressure protection in postulated accidents is that the pressure of the object to be protected stays below 1.1 times the design pressure of the protected object.

616. Acceptance criteria for the containment in postulated accidents are given in Section 13 of Government Decree 717/2013 and in Guide YVL B.6, chapter 3.

6.4 Design extension conditions

617. The fuel integrity acceptance criteria for design extension conditions are given in Guide YVL B.4, chapter 4.

618. The acceptance criterion for the DEC overpressure protection analysis is that the pressure of the object to be protected stays below 1.2 times the design pressure of the protected object.

6.5 Severe accidents

619. Requirements for the depressurisation of the primary circuit during severe accidents are given in Section 13 of Government Decree 717/2013, and in Guide YVL B.5, chapter 4.4.

620. Acceptance criteria for the containment's behaviour during severe accidents are given in Guide YVL B.6, chapter 3.

7 Documents to be submitted to STUK

701. The documents to be submitted to STUK for the nuclear power plant's licensing process are given in Guide YVL A.1.

702. As part of the suitability analysis to be submitted during the decision-in-principle phase, the licence applicant shall demonstrate that the organisation performing the analyses has adequate competence to conduct transient and accident analyses for the preliminary safety analysis report as required in Guide YVL B.3.

703. The preliminary safety analysis report shall present the calculation methods for transient and accident analyses and their validation, as well as the preliminary transient and accident analyses demonstrating the acceptability of the systems' technical solutions.

704. The final safety analysis report shall present the calculation methods for transient and accident analyses and their validation, as well as the final transient and accident analyses demonstrating the acceptability of the systems' technical solutions.

705. The essential results of the analyses shall be presented in the preliminary and final safety analysis reports. Detailed information on the assumptions and calculation methods used in the analyses may be presented in either the safety analysis report or the topical reports.

706. The description of the models and analysis methods as required in para 404 shall be delivered to STUK for information as part of the preliminary and final safety analysis reports.

707. The analyses of the preliminary safety analysis report shall describe the plant to the level of detail that is possible at this design stage, in order to facilitate analyses of the plant's operation in all operational conditions during anticipated operational occurrences and accidents.

708. The analyses conducted for the operating licence shall describe the plant in a way consistent with the plant for which the operating licence is applied.

709. The analyses conducted for an operating plant shall describe the plant in a way consistent with the plant's current status or consistent with

the plant after the implementation of the plant modifications.

710. An assessment on the effects of the planned modification to plant behaviour during transient and accidents, and a summary of design analysis results shall be provided a part of the conceptual plan required for modifications to an operating nuclear power plant's systems in safety classes 1, 2 and 3. Analyses verifying the acceptability of the technical solutions shall be provided as part of the pre-inspection documentation.

711. In connection with periodic safety assessments, the licensee shall evaluate the scope of and need for updates in transient and accident analyses, and update the analyses for the final safety analysis report, where necessary.

8 Regulatory oversight by the Radiation and Nuclear Safety Authority

801. During the decision-in-principle phase, STUK reviews the suitability analysis provided with the application for a decision-in-principle and the description of calculation methods used in the transient and accident analyses presented in it. STUK draws up a preliminary safety assessment based on the review.

802. STUK reviews the preliminary safety analysis report provided with the construction licence application and the transient and accident analyses contained in it, as well as the validation of the calculation methods used. STUK draws up a safety assessment based on the review.

803. STUK reviews the final safety analysis report provided with the operating licence application and the transient and accident analyses contained in it, as well as the validation of the calculation methods used. STUK draws up a safety assessment based on the review.

804. STUK reviews the conceptual plans, pre-inspection documents and changes to the final safety analysis report of systems modifications in operating nuclear power plants, and approves the above on the basis of the review.

805. During the construction and operating licence review phase, and also where necessary, STUK conducts or commissions to an external expert organisation independent verification analyses for the key initiating events affecting the acceptability of the plant's systems.

Definitions

Initiating event

Initiating event shall refer to an identified event that leads to anticipated operational occurrences or accidents.

Controlled state

Controlled state shall refer to a state where a reactor has been shut down and the removal of its decay heat has been secured. (Government Decree 717/2013)

Controlled state following a severe reactor accident

Controlled state following a severe reactor accident shall refer to a state where the removal of decay heat from the reactor core debris and the containment has been secured, the temperature of the reactor core debris is stable or decreasing, the reactor core debris is in a form that poses no risk of re-criticality, and no significant volumes of fission products are any longer being released from the reactor core debris. (Government Decree 717/2013)

System

System shall refer to a combination of components and structures that performs a specific function.

Minimum system performance

Minimum system performance can be determined by making the following assumptions:

1. Consider the consequential effects of the initiating event (component failure, for example).
2. Furthermore, select the failure combination that is most detrimental to the functionality of the system in accordance with the failure criterion presented in chapter 4.3 of Guide YVL B.1. The single failure with the highest reactivity effect is also assumed to occur in the reactor scram system.
3. Determine the performance parameters for each functioning component, which conform to the acceptance limits of components in periodic tests.

Qualification

Qualification shall refer to a process to demonstrate the ability to fulfil specified requirements (corresponds to the qualification process of the ISO 9000 standard).

Validation

Validation shall refer to confirmation, through the provision of objective evidence, that the requirements for a specific intended use or application have been fulfilled. (ISO 9000)

Conservative analysis method

Conservative analysis method shall refer to a manner of preparing a safety analysis that considers the uncertainties related to the calculation models and initial assumptions so that, with a high level of certainty, the consequences of the event analysed would be milder than the analysis results.

Criticality accident

Criticality accident shall refer to an accident caused by an uncontrolled chain reaction of nuclear fissions. (Government Decree 717/2013)

Loading analysis

Loading analysis shall refer to the computational analysis, covering the entire life cycle, of the mechanical and thermal loads (service

loads) to which a component is subjected in the operational conditions and accidents used as the facility's design bases over the course of its entire life cycle, when the procedures, specifications and analyses concerning operation, required functions and sequences of events are taken into account.

Normal operation (DBC 1)

Normal operation (DBC 1) shall refer to the planned operation of a nuclear power plant according to the Operational Limits and Conditions and operational procedures in place. These also include testing, plant start-up and shutdown, maintenance and refuelling. As far as other nuclear facilities are concerned, normal operation shall refer to similar plant operation.

Anticipated operational occurrence (DBC 2)

Anticipated operational occurrence (DBC 2) shall refer to such a deviation from normal operation that can be expected to occur once or several times during any period of a hundred operating years. (Government Decree 717/2013)

Postulated accident

Postulated accident shall refer to a deviation from normal operation which is assumed to occur less frequently than once over a span of one hundred operating years, excluding design extension conditions; and which the nuclear power plant is required to withstand without sustaining severe fuel failure, even if individual components of systems important to safety are rendered out of operation due to servicing or faults. Postulated accidents are grouped into two classes on the basis of the frequency of their initiating events: a) Class 1 postulated accidents (DBC 3), which can be assumed to occur less frequently than once over a span of one hundred operating years, but at least once over a span of one thousand operating years; b) Class 2 postulated accidents (DBC 4), which can be assumed to occur less frequently than once during any one thousand operating years.

Design extension condition (DEC)

Design extension condition (DEC) shall refer to:

- a. an accident where an anticipated operational occurrence or class 1 postulated accident involves a common cause failure in a system required to execute a safety function (DEC A);
- b. an accident caused by a combination of failures identified as significant on the basis of a probabilistic risk assessment (DEC B); or
- c. an accident caused by a rare external event and which the facility is required to withstand without severe fuel failure (DEC C).

Pressure control analysis

Pressure control analysis shall refer to an analysis used to demonstrate that the pressure control systems meet the design requirements set for them.

Best estimate method

Best estimate method shall refer to a method of preparing a safety analysis where the physical modelling of any phenomenon studied is as realistic as possible, and the initial assumptions for the analysis are realistically selected.

Safe state

Safe state shall refer to a state where the reactor has been shut down and is non-pressurised, and removal of its decay heat has been secured. (Government Decree 717/2013)

Safe state following a severe reactor accident

Safe state following a severe reactor accident shall refer to a state where the conditions for the controlled state of a severe reactor accident are met and, in addition, the pressure inside the containment is low enough that leak from the containment is minor, even if the containment is not leak-tight. (Government Decree 717/2013)

Safety classified system/structure/component

Safety-classified system/structure/component shall refer to a system, structure or component assigned to safety classes on the basis of its safety significance.

Safety functions

Safety functions shall refer to functions important from the point of view of safety, the purpose of which is to control disturbances or prevent the generation or propagation of accidents or to mitigate the consequences of accidents. (Government Decree 717/2013)

Severe reactor accident

Severe reactor accident shall refer to an accident in which a considerable part of the fuel in a reactor loses its original structure. (Government Decree 717/2013)

(N+1) failure criterion

(N+1) failure criterion shall mean that it must be possible to perform a safety function even if any single component designed for the function fails.

(N+2) failure criterion

(N+2) failure criterion shall mean that it must be possible to perform a safety function even if any single component designed for the function fails and any other component or part of a redundant system – or a component of an auxiliary system necessary for its operation – is simultaneously out of operation due to repair or maintenance.

Single failure

Single failure shall refer to a failure due to which a system, component or structure fails to deliver the required performance.

References

1. Nuclear Energy Act (990/1987).
2. Nuclear Energy Decree (161/1988).
3. Government Decree on the Safety of Nuclear Power Plants (717/2013).
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7. Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation. IAEA Safety Reports Series No. 52. IAEA, Vienna 2008.
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