NUCLEAR FUEL AND REACTOR

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 Guides YVL 6.2, YVL 6.3 and YVL 6.8.

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Authorisation

According to Section 7 r of the Nuclear Energy Act (990/1987), *the Radiation and Nuclear Safety Authority 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.*

According to 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.*

Translation. Original text in Finnish.

1 Introduction

101. The Government Decree on the Safety of Nuclear Power Plants (717/2013) sets forth the general safety principles that shall be applied in the design, construction, operation and decommissioning of nuclear power plants.

102. Section 3 of Government Decree (717/2013) prescribes that the safety of a nuclear power plant shall be assessed when applying for a construction licence and an operating licence, in connection with plant modifications, and at regular intervals during the operation of the plant. Subsections 2 and 3 of the Section set forth provisions on the methods that shall be used in substantiating nuclear power plant safety and the technical solutions of the plant's safety systems.

103. Section 14 of Government Decree (717/2013) prescribes that in ensuring the safety functions of a nuclear power plant, inherent safety features attainable by design shall be primarily utilised. There shall be systems in place for shutting down the reactor and maintaining it in subcritical state, for removing decay heat, and preventing the dispersion of radioactive substances. Redundancy, separation and diversity principles must be applied in designing the systems in question.

104. To assure nuclear fuel integrity, Section 13 of Government Decree (717/2013) prescribes as follows:

- the probability of fuel failure shall be low during normal operational conditions and anticipated operational occurrences
- 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.

105. This Guide presents criteria and detailed requirements to ensure and demonstrate the implementation of the regulations contained in the aforementioned Government Decree (717/2013) during the design of the nuclear power plant, reactor core and nuclear fuel. Criticality safety requirements apply to all nuclear facilities where fissile material is used, stored or handled. The requirements for the reactor core and reactivity control systems are given in chapter 3 of this Guide, those for nuclear fuel and fuel design in chapter 4 and the requirements for the prevention of a criticality accident in chapter 5.

106. Chapter 4 of this Guide presents the design criteria whose fulfilment shall be demonstrated by a fuel suitability study. The fuel suitability study is part of the acceptance procedure for nuclear fuel procurement described in Guide YVL E.2, which comprises four parts as follows:

- quality management in design and manufacture
- suitability study
- construction plan
- control of manufacturing

107. Approval of the nuclear fuel suitability study is, in accordance with Guide YVL E.2, a prerequisite for the final review of the construction plan for the fuel.

2 Scope of application

201. This Guide shall be applied to the design of the reactors, reactivity control systems, nuclear fuel as well as fuel handling and storage systems of nuclear facilities.

202. In addition to fuel design, control rod design shall comply with the requirements of chapters 4 and 5 of this Guide for applicable parts.

203. Requirements concerning the manufacture and design of nuclear fuel are presented, in addition to this Guide, in Guide YVL E.2. Requirements for spent nuclear fuel storage and handling are given in Guide YVL D.3. The design criteria of systems relating to the reactivity control of a nuclear power plant are given in Guide YVL B.1 and the requirements for safety analyses to demonstrate safety in Guide YVL B.3.

3 Requirements for the reactor and reactivity control systems

3.1 Structural compatibility of reactor and nuclear fuel

301. The structure of the nuclear fuel and reactor internals shall be designed to be compatible in such a way that when the reactor is assembled each component fits reliably in the correct location and position. It shall be possible to check after reactor loading that the nuclear fuel and reactor internals are correctly positioned.

302. The reactor pressure vessel internals shall be designed and installed in such a way that they maintain their position during different operational conditions and that they are not permanently shifted during postulated accidents. The reactor pressure vessel internals shall withstand all design-basis scenario loads without reactor shutdown or cooling being endangered.

3.2 Reactivity control and reactor shutdown

303. The combined effect of the nuclear reactor's physical feedbacks shall be such that it mitigates the increase in reactor power in normal operation as well as in anticipated operational occurrences and accidents in which an initiating event causes a reactivity increase or compromises nuclear fuel cooling. In situations during which physical feedbacks cause a positive reactivity increase, the reactor protection system shall be capable of limiting the increase of reactor power in such a way that the nuclear fuel design limits are not exceeded.

304. A single failure of the control system or a single control error by the operator shall not cause a power increase reaching a limit that requires the reactor to be shut down.

305. The nuclear power plant shall be provided with two independent reactivity control systems that implement the diversity principle and fulfil the special requirements of Guide YVL B.1 for systems needed for reaching and maintaining a controlled state.

306. The reactor and the associated systems shall be designed and implemented in such a way that an anticipated operational occurrence or a postulated accident cannot cause a significant reactivity increase due to reduction, weakened efficiency or uneven distribution of neutron absorbers needed to maintain reactivity control.

307. It shall be possible to maintain a reactor that has sustained damage in a severe reactor accident, or its remains, in a subcritical condition.

4 Requirements for nuclear fuel

4.1 General

401. The integrity of nuclear fuel shall be ensured during its operation, handling, transport, long-term storage and final disposal. To ensure this, design criteria for nuclear fuel shall be determined, including adequate safety margins. The criteria shall be based on experimental results for an equivalent fuel type.

402. The nuclear fuel design criteria shall be stated in the Safety Analysis Report of the plant unit or in the nuclear fuel design documents.

403. In the nuclear fuel suitability study, it shall be stated which materials and components of the fuel are specific to a delivery batch and which are not. Alternatively, a reference can be made to a list submitted in some other context, where the matter has been presented.

404. The only modifications allowable to nuclear fuel are those reviewed and approved by organisational units responsible for design and quality management. The modifications shall be justified by applicable analyses, experimental research and possible operational experience. The potential effects of the modifications on the safety analyses of the facility in question shall be taken into account.

405. In order for a fuel type used earlier by the licensee to be modified substantially, the licensee shall review in detail the design documentation pertinent to the modification and conduct the necessary comparative analyses as well as evaluate the modification's effects on the behaviour of nuclear fuel. The modification's compatibility with the reactor and the other systems of the facility shall be ensured. The potential effects of the modification on the safety analyses of the facility shall also be analysed.

4.2 General design criteria for nuclear fuel

406. In determining the design criteria for nuclear fuel, the physical, chemical and mechanical phenomena that affect the durability of the nuclear fuel during operational and accident conditions shall be comprehensively analysed. The analyses shall cover all design basis scenarios.

407. In determining the design criteria for nuclear fuel, furthermore, the structural and material properties that are relevant to final disposal and the long-term safety of final disposal shall be addressed.

408. A nuclear fuel assembly shall be designed in such a way that its components maintain their position in all operational conditions and that they are not permanently shifted during postulated accidents. The nuclear fuel assembly must withstand all design-basis scenario loads in such a way that reactor shutdown and coolability are not endangered.

409. Irradiation-induced changes that affect nuclear fuel properties shall be taken into account in determining the limits for safe use of the fuel, including the effects on the final disposal of spent nuclear fuel. Burn-up limits to be applied to nuclear fuel shall be presented, and they shall be based on experimental data. If irradiation affects heat transfer between cladding and coolant it shall be taken into account also in correlations used for evaluating the heat transfer crisis.

410. Control rods shall endure wear and other stresses during operation so that their normal function is not endangered. The control rods shall retain their ability to absorb neutrons during operation in compliance with the assumptions of the Safety Analysis Report of the plant unit.

411. The normal function of control rods must not be prevented by deformations of a fuel assembly or fuel rods.

4.3 Design criteria for normal operational conditions

412. In normal operational conditions, the nuclear fuel shall fulfil the following conditions:

- No melting shall occur in fuel pellets.
- Cladding temperature shall not substantially exceed coolant temperature.
- Fuel rod cladding shall not collapse.
- The internal pressure of a fuel rod shall not increase to the extent that cladding deformations caused by it would negatively affect the heat transfer between fuel pellets and coolant (lift-off).

413. Deformations in the fuel assembly and control rod components shall remain so minor that

- no significant increase in power in the fuel rods occurs
- coolability of nuclear fuel is not endangered
- reactor scram or other movement of control rods is not obstructed
- handling of fuel rods is not hampered.

414. The probability of a fuel failure caused by mechanical interaction between fuel pellet and cladding shall be extremely low. To ensure this, operating limits for changes in power and the rates of change shall be determined for nuclear fuel, taking into account, i.a., the stress corrosion of the cladding.

4.4 Design criteria for anticipated operational occurrences

415. In anticipated operational occurrences, the nuclear fuel shall fulfil the following conditions:

- No melting shall occur in fuel pellets.
- Adequate cooling of the cladding shall be ensured. Cooling of the cladding is considered adequate if there is a 95% probability at 95% confidence level that the hottest fuel rod does not reach heat transfer crisis. Alternatively, it may be demonstrated that the number of rods reaching heat transfer crisis does not exceed 0.1% of the total number of fuel rods in the reactor.

- The probability of fuel failure caused by mechanical interaction between fuel and cladding shall be extremely low.

4.5 Design criteria for postulated accidents and design extension conditions

Class 1 postulated accidents

416. A Class 1 postulated accident shall not cause significant changes to the original fuel geometry. To ensure this, the nuclear fuel shall fulfil the following criteria:

- The number of fuel rods reaching heat transfer crisis shall not exceed 1% of the total number of fuel rods in the reactor.
- The maximum temperature of the nuclear fuel cladding shall not increase to the extent that oxidation of the cladding or changes in the cladding material properties could endanger the integrity of the cladding during an accident. This requirement can be considered fulfilled without a separate justification if the temperature does not exceed the value of 650°C.
- The number of fuel failures caused by mechanical interaction between nuclear fuel and cladding shall not exceed 0.1 % of the total number of fuel rods in the reactor.

Class 2 postulated accidents

417. The number of fuel rod failures in a Class 2 postulated accident shall not exceed 10% of the total number of fuel rods in the reactor.

418. In determining the total number of fuel rod failures owing to a temperature rise of the cladding, the changes in cladding temperature, chemical reactions, deformations, such as ballooning and collapse of the cladding, as well as damage to the cladding caused by an increase in the fuel enthalpy shall be taken into account.

419. Limits employed in assessing the loss of cladding integrity shall be based on experimental study. In determining the limits, chemical, physical and mechanical factors affecting the phenomena in question as well as the dimensional tolerances of the fuel rod shall be comprehensively taken into account. The properties of the cladding material and fuel pellets that change as a result of irradiation of the nuclear fuel shall be taken into account in assessing the comprehensiveness of the experiments and in determining fuel burn-up dependent limits for fuel failure based on them.

420. Nuclear fuel failure is assumed if the radial average enthalpy of a fuel rod at any axial location exceeds the value 586 J/gUO_2. The failure criterion can be changed if it is demonstrated by sufficiently comprehensive tests on the fuel type in question that the fuel is highly probable to withstand the corresponding enthalpy without a failure.

Class 2 postulated accidents and design extension conditions

421. The coolability of the nuclear fuel shall not be endangered due to, for example, ballooning or bursting of the cladding, deformation of parts of the fuel assembly or reactor internals, or debris possibly introduced into the reactor as a result of an accident.

422. Excessive embrittlement of the cladding shall be prevented. To ensure this, it shall be demonstrated that

- the cladding is not oxidised during an accident to a degree where it cannot withstand the loads caused by the accident. The estimate shall take into account both the oxidation of the cladding (external and possibly internal) during the accident and the preceding oxidation during normal operation as well as chemical interactions between fuel pellets and cladding material
- the cladding withstands loads caused by the handling, transport and storage of the fuel assembly after an accident
- the hydrogen absorbed during normal operation and during an accident does not excessively deteriorate the properties of the cladding. The effect of the absorbed hydrogen on cladding integrity shall be experimentally determined
- the highest temperature of the cladding during an accident does not exceed the value of 1,200ºC.

423. The temperature rise of the cladding shall be limited to a level where the oxidation of the cladding in consequence of a metal-water reaction does not accelerate uncontrollably.

424. Fragmentation and melting of the fuel rod shall be prevented. The radial average enthalpy at any axial location of any fuel rod shall not exceed the value of 963 J/g UO2. In preventing the melting of the cladding, the interactions between the different components of the fuel assembly that (e.g. due to eutectic properties of materials) may decrease the melting temperature of the cladding shall be taken into account.

425. The amount of hydrogen generated by the chemical reaction between coolant and cladding shall not exceed 1% of the amount that would be generated if the part of the cladding surrounding the fuel pellets in the whole reactor core would react with the coolant.

426. No melting shall occur in the control rods. Structural deformations in fuel rods, fuel assemblies, control rods or reactor internals shall not obstruct the movement of control rods in the reactor.

5 Requirements for preventing a criticality accident

501. This Chapter presents the general requirements for preventing a criticality accident from the point of view of designing the reactor core, fuel as well as fuel storage and handling systems. Guide YVL D.3 presents other requirements related to handling, encapsulation and final disposal of spent nuclear fuel.

5.1 Requirements for nuclear fuel outside the reactor

502. In designing nuclear fuel and its storage and handling systems, it shall be ensured that the requirements set forth for criticality safety are fulfilled. Analyses demonstrating fulfilment of the requirements shall be presented as part of the suitability study for nuclear fuel or for a system relating to fuel handling or storage. In licensing a new fuel type, it shall be demonstrated that the nuclear fuel fulfils the criticality safety requirements in all phases of its planned handling, storage and final disposal.

503. Structural means shall be used to prevent criticality of nuclear fuel located outside the reactor. The arrangements for ensuring the subcriticality of storage and handling systems shall not be based on substances dissolved in water. Only fixed absorber structures may be credited in criticality analyses of fuel storages.

504. The storage locations and the handling and transfer systems shall be so designed that, when the storage is full of nuclear fuel, the effective multiplication factor k_{eff} will not exceed the value 0.95 under normal conditions or in anticipated operational occurrences and the value 0.98 in other design basis scenarios. In criticality safety analyses pertaining to dry storage, cases where water or other possible moderator enters the storage shall also be examined as an accident.

505. In criticality safety analyses, the effect of uncertainties arising from e.g. structures, dimensions and storage conditions that may increase the multiplication factor shall be taken into account in such a way that the analysis results are conservative with high confidence. The possible deviations from normal storage conditions during accidents shall be taken into account in the analyses.

506. The nuclear fuel isotope concentration used in criticality safety analyses shall be so determined that the analyses cover all nuclear fuel irradiation histories considered possible with a high confidence. As regards dry storage and transport packaging of fresh nuclear fuel, it is sufficient to consider only fresh nuclear fuel. In analyses of storages and handling systems solely intended for decommissioned nuclear fuel, the nuclear fuel burn-up may be taken into account in the criticality safety analyses (burn-up credit). For all other storage facilities, a burn-up that maximises the reactivity of the nuclear fuel shall be assumed.

507. In criticality safety analyses, the entire fuel rack or other structure under consideration shall be assumed to be filled up with nuclear fuel to the extent technically feasible.

508. In criticality safety analyses, all fissile nuclides significantly affecting reactivity shall be taken into account. Only such non-fissile, neutron-absorbing nuclides whose reactivity effect over the entire planned storage time is, with high confidence, at least equal to what is assumed in the analyses may be taken into account. As regards unstable nuclides, the reactivity effect of daughter nuclides may be taken into account in such a way that the combined reactivity effect of nuclides in a decay chain starting from an unstable nucleus is conservative. The uncertainties of the computational system used for burn-up calculations shall be taken into account in determining the isotope composition of nuclear fuel.

5.2 Criticality safety requirements for nuclear fuel in the reactor

509. Inadvertent criticality of nuclear fuel in a nuclear reactor shall be prevented primarily by technical means. If no technical barriers exist to prevent nuclear fuel becoming critical, the reactor shall be equipped with neutron flux measurement that must be capable of detecting and signalling imminent criticality so that a criticality accident can be prevented. Ensuring that fuel assemblies are placed into the reactor according to plan is not by itself a sufficient technical barrier against inadvertent criticality.

510. While making modifications to the core (fuel assembly or control rod transfers), the reactor neutron flux and the possible boron concentration of the coolant shall be monitored.

6 Regulatory oversight by the Radiation and Nuclear Safety Authority

601. STUK reviews the nuclear fuel suitability study.

602. STUK oversees the design of the systems of nuclear power plants by reviewing their pre-inspection documentation and witnessing their fabrication and use.

603. STUK oversees nuclear fuel integrity and criticality safety related matters by inspections in accordance with the periodic inspection programme.

Definitions

Subcritical state

Subcritical state shall refer to a state where no chain reaction sustained by neutrons released by nuclear fission occurs.

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)

Criticality

Criticality shall refer to a state where the output and loss of neutrons, created in nuclear fission and maintaining a chain reaction, are in equilibrium so that a steady chain reaction continues. (Government Decree 717/2013)

Criticality accident

Criticality accident shall refer to an accident caused by an uncontrolled chain reaction of nuclear fissions.

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).

Fuel failure

Fuel failure shall refer to a scenario where a fuel rod loses its integrity.

Shut down reactor

Shut down reactor shall refer to a reactor in a subcritical state with an effective multiplication factor, taking uncertainties into consideration, of less than 0.995.

Design basis scenario

Design basis scenario shall refer to the reactor's normal operation, anticipated operational occurrences, postulated accidents, and design extension conditions.

Products specific to a delivery batch

Products specific to a delivery batch shall refer to products (materials, parts, components) that have been allocated to a nuclear fuel delivery batch at the time of their manufacture. Other nuclear fuel products are not considered allocated to a delivery batch.

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)

Loss of coolability of the nuclear fuel

Loss of coolability of the nuclear fuel shall refer to a scenario where nuclear fuel loses its coolable shape due to fuel failure or a deformation that exceeds the design basis, or where a flow cooling fuel rods is blocked due to impurities in the fuel assembly.

Common cause failure

Common cause failure shall refer to a failure of two or more structures, systems and components due to the same single event or cause.

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).
4. Government Decree on the Security in the Use of Nuclear Energy (734/2008).
5. Government Decree on Emergency Response Arrangements at Nuclear Power Plants (716/2013).
6. Government Decree on the safety of disposal of nuclear waste (736/2008).
7. IAEA Safety Guide NS-G-1.12: Design of the Reactor Core for Nuclear Power Plants.
8. IAEA Safety Guide NS-G-1.4: Design of Fuel Handling and Storage Systems in Nuclear Power Plants.
9. IAEA Safety Guide SSG-2: Deterministic Safety Analysis for Nuclear Power Plants.