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# III. OTHER PROVISIONS

## NUCLEAR SAFETY COUNCIL

**7715** *Nuclear Safety Council Instruction IS-27, revision 1, of June 14th 2017, on general nuclear power plant design criteria.*

Article 2.a) of Law 15/1980, of April 22nd, creating the Nuclear Safety Council, attributes to this Public Entity powers to «draw up and approve technical instructions, circulars and guidelines relating to nuclear and radioactive facilities and to activities associated with nuclear safety and radiological protection» ensuring the safe operation of such facilities, i.e. operation without undue risk for persons or the environment.

Royal Decree 1836/1999, of December 3rd, which approves the Regulation on Nuclear and Radioactive Facilities, regulates fundamentally the administrative and procedural aspects of the granting of authorisations. As regards the technical aspects, and in the absence of internal standards, the different authorisations have been based on the regulations in force in the country of origin of the design and on the technical standards enacting these regulations. Article 8.3 of the said Royal Decree provides that «The Licensee shall continuously strive to improve the conditions of nuclear safety and radiological protection of his facility. In this respect, he shall analyse the existing technical and practical improvements, in accordance with the requirements established by the Nuclear Safety Council, and implement those that the said organisation considers to be ideal», this introducing as a regulatory basis for the facility the continuous improvement of its safety and the powers of the Nuclear Safety Council to require practical and technical measures aimed at achieving this goal.

The general design criteria constitute the minimum set of requirements in accordance with which a nuclear power plant must be designed in order for it to be considered safe. The objective of the present instruction is to establish this set of criteria. In drawing up the instruction, consideration has been given to the standards of the country of origin of the technology of the Spanish plants, in particular the contents of Appendix A of part 50 of title 10 of the United States Code of Federal Regulations and the equivalent standards of the German BMI, as well as those of the IAEA. Consideration has also been given to the experience acquired in relation to the design of structures, systems and components. To date the Nuclear Safety Council has assessed and inspected compliance with this standard by the licensees of the nuclear power plants throughout all the phases of the lifetime of the facilities.

Additionally, consideration has been given in this Instruction to the work performed by «WENRA» («Western European Nuclear Regulators Association»), with a view to bringing into harmony the regulations of the different countries. As a result of these efforts, a set of common requirements or reference levels has been established, this to be reflected in the national standards. The drawing up of a Nuclear Safety Council Instruction contemplating these criteria is considered a necessity in order to give consistency to the standards development process addressed by the Nuclear Safety Council as a result of the aforementioned harmonisation efforts.

It has been considered necessary to review this Nuclear Safety Council Instruction due to the experience acquired as a result of its application in certain specific aspects. The changes incorporated in this revision 1 spring from the advisability of modifying the affected requirements in order to establish an adequate scope.

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By virtue of all the above and in accordance with the legal authorisation contemplated in article 2.a) of Law 15/1980, of April 22nd, creating the Nuclear Safety Council, and following consultations with the affected sectors and issuing of the appropriate technical reports,

This Council, in its deliberation of June 14th 2017, has provided as follows:

One. *Objective and scope of application*

1. The objective of the present Nuclear Safety Council Instruction is to set out the general criteria to be met in the design, manufacturing, construction, testing and general, operation of safety-related structures, systems and components at nuclear power plants.

2. The present Instruction is applicable to the licensees of the Spanish nuclear power plants, in relation to their operating permits.

Two. *Definitions*

The definitions of the terms and concepts contained in the present Instruction correspond to those contained in the following provisions:

Nuclear Energy Act, Law 25/1964, of April 29th.

Law 15/1980, of April 22nd, creating the Nuclear Safety Council.

Royal Decree 1836/1999, of December 3rd, approving the Regulation on Nuclear and Radioactive Facilities.

European Union Council Directive 2009/71/EURATOM, of June 25th 2009, establishing a community framework for the nuclear safety of nuclear facilities.

European Union Council Directive 2014/87/EURATOM, of July 8th 2014, establishing a community framework for the nuclear safety of nuclear facilities.

Furthermore, the following definitions are applicable within the context of the present instruction:

Loss of coolant accidents: In the case of pressurised water reactors (PWR), these are events in which the rupturing of the reactor coolant pressure boundary leads to a rate of loss of coolant greater than the capacity of the normal make-up systems, and in the case of boiling water reactors (BWR) leads to loss of containment pressure control by the normal containment cooling systems, regardless of break size, including the double-ended rupturing of the reactor coolant system pipe of largest diameter.

Design basis accidents: This is the set of accident conditions for which a nuclear power plant is designed. Under these conditions, the criteria used in designing the plant mean that the deterioration of nuclear materials and the release of radioactive materials remain within the authorised dose limits.

Reactor coolant pressure boundary: This is the set of all components subjected to the pressure of the reactor and that are part of its coolant system or are connected to it. The pressure boundary includes the following:

1. Plants of American design:

For piping systems penetrating containment, up to the outermost containment isolation valve.

For systems not penetrating containment, up to the second of two valves that are closed during normal reactor operation.

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For BWR reactors, the reactor coolant system includes up to the outermost containment isolation valve of the feedwater and main steam systems.

The reactor coolant system relief and safety valves.

2. Pressurised water reactor plants of German design:

Piping connecting to the reactor coolant system, up to the first isolation valve.

The reactor coolant system relief and safety valves.

Effective multiplication constant: Quotient between the numbers of neutrons in two successive generations in a chain reaction.

Site: Demarcated plot of land belonging to the licensee and subject to a series of controls, limits and regulations on which is located an authorised facility.

Safety (or safety-related) structures, systems and components: These are elements to whose operation credit is given in design basis accident analyses in order to:

Take the facility to a safe condition and keep it in this condition in the long term.

Limit the radiological consequences of foreseen operating events and design basis accidents to within the specified limits.

Single failure: a single failure is an independent event that leads to the loss of a component as regards the performance of its safety function. Whatever multiple failures might occur as a result of a single failure shall be considered single failures. Electrical and fluid systems are considered to be designed to withstand single failures if the system maintains its capacity to perform its safety functions in the event of a single failure of any active component (assuming that all the passive components operate correctly) or of any passive component (assuming that all the active components operate correctly).

Design limits: Set of values that establish limits for functional capacity and performance parameters and levels and that are considered acceptable because they guarantee compliance with the safety limits.

Safety limits: These are limits that are established for important process variables that have been seen to be necessary to reasonably maintain the integrity of the physical barriers that provide protection against the uncontrolled release of radioactivity off site.

Radioactive materials or substances: These are all substances or materials that contain one or more radionuclides and whose activity or concentration cannot be considered insignificant from the point of view of radiological protection.

Normal operation: This concept includes all operating modes in which the plant might find itself routinely, from outages for refuelling to full power operation.

Nuclear reactor: Any structure containing nuclear fuel arranged in such a manner that it may provide a self-sustaining process of nuclear fission without the need for an additional source of neutrons.

Reactor containment (or containment): This is one of the structures of a nuclear power plant that acts as a barrier, along with the fuel rods and the reactor coolant pressure boundary, to control the emission of radioactive material. The containment includes the following:

1. The containment structure and its access locks, penetrations and auxiliary systems.

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2. The valves, piping, closed systems and other components that make it possible to isolate the containment atmosphere from the outside, and

3. Those systems or parts of systems that, in view of their functions, extend the boundary of the containment structure and provide effective isolation (for example steam and feedwater piping).

Protection systems: This concept includes the reactor protection system and the systems or sub-systems that activate the engineered safeguards and allow for the performance of its safety functions.

Anticipated operational or operating events (also known as anticipated operational transients): These are operating conditions that deviate from normal operation and that are expected to occur once or twice during the lifetime of a nuclear power plant, such as loss of off-site power, turbine trip or reactor isolation. The criteria used in designing the plant mean that these events do not cause significant damage or give rise to postulated accident conditions.

Nuclear group: Each of the sets of a nuclear reactor and associated systems existing on a single site.

Three. *Criteria.*

In order to facilitate the implementation and documentary control of the design criteria for the licensees, the present Instruction adheres to the nomenclature, numbering and divisions of the structure of Appendix A of 10CFR50, approved by the United States Nuclear Regulatory Commission.

*Part 1. General requirements*

Criterion 1. Design of safety functions

1.1 The nuclear power plant must be capable of maintaining the following safety functions under normal operating conditions, anticipated operating events and design basis accidents:

1) Reactivity control.
2) Removal of residual heat from nuclear fuel.
3) Confinement of radioactive material.

1.2 The design of safety-related structures, systems and components (from hereon SSC) shall take into account the fail-safe criterion; i.e., in the event of a failure, the SSC should remain in the position most favourable to safety that its design allows.

1.3 The design shall prevent the failure of non safety-related systems from potentially affecting safety functions.

1.4 Actuations and manoeuvres required for the performance of safety functions shall be accomplished automatically or using passive means, such that operator actions are not required for 30 minutes following an initiating event. If the design requires the operator to take any action during this period, the said actions shall be justified and included in operating procedures, which shall be periodically exercised, whenever possible on a full-scope replica simulator.

1.5 The reliability of safety-related SSC's shall be ensured by choosing the most adequate options in each specific case, such as for example the use of intrinsic safety resources, passive safety resources, adequately tested components, redundancy, diversity or physical and functional separation.

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1.6 Safety-related SSC's shall be designed, manufactured, assembled and tested in accordance with quality standards in keeping with the importance of the safety functions performed. In this respect, all safety-related SSC's shall be identified and classified depending on their safety significance.

1.7 The classification of safety-related SSC's shall be based mainly on deterministic methods, complemented wherever necessary by probabilistic methods and the judgement of experts.

1.8 The design of safety-related SSC's shall take into account the principles and techniques of human factors engineering.

Criterion 2. Design basis for protection against natural phenomena

Safety-related SSC's shall be designed to withstand the effects of natural phenomena without losing their capacity to perform their safety functions. The design basis for these SSC's shall contemplate the following aspects:

1. Suitable consideration shall be given to the most severe natural phenomena that have been recorded throughout history at the site and in the surrounding area, and a sufficient margin shall be included in the design to account for the limitations of the historical data as regards accuracy, quantity and the period of time to which the information corresponds.

2. Consideration shall be given to credible combinations of normal operating and accident conditions with the effects of natural phenomena.

3. Consideration shall be given to the importance of the safety functions to be performed by these SSC's.

Criterion 3. Fire protection

3.1 Safety-related SSC's shall be designed and located such that the probability of fires or explosions, and their effects, are minimised, in all cases in a manner coherent with other safety requirements.

3.2 Wherever feasible, and especially in vital areas of the plant such as the containment and the control room, non-flammable heat resistant materials shall be used.

3.3 Fire detection and extinguishing systems of adequate effectiveness and capacity shall be installed in order to minimise the adverse effects of fire on safety-related SSC's. Fire extinguishing systems shall be designed such that, in the event of failure or undue operation of the system, the capacity of these SSC's to perform their safety functions is not significantly affected.

3.4 The protective measures required to limit the propagation of fires shall be provided, guaranteeing that fires are kept confined in fire-resistant areas.

Criterion 4. Ambient and dynamic effects design basis

4.1 The SSC's indicated below must be designed to withstand the effects deriving from, and be compatible with, the ambient conditions associated with normal operation, maintenance work, the performance of tests and design basis accidents, including loss of coolant accidents, throughout the entire lifetime of the plant.

Safety-related,

those that without being safety-related might prevent the performance of safety-related SSC functions in the event of their failing under postulated ambient conditions, and

post-accident instrumentation so requiring, in accordance with the specific applicable standards.

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4.2 In order to comply with this objective, a qualification programme shall be adopted confirming that the SSC's indicated in section 4.1 are capable of performing their function throughout their entire design lifetime, taking into account both the ambient conditions expected during plant operation and, where applicable, those corresponding to anticipated operating events and design basis accidents.

4.3 The SSC's indicated in section 4.1 shall be suitably protected against dynamic effects, including those due to missiles, pipe whipping effects and fluid discharges that might occur as a result of equipment failures, as well as against events and conditions occurring off site. However, the dynamic effects associated with postulated plant pipe ruptures may be excluded from the design basis if analyses approved by the Nuclear Safety Council demonstrate that the probability of such ruptures is extremely low under conditions consistent with the design basis of the affected piping.

Criterion 5. Shared structures, systems and components

At nuclear power plants having more than one group on one same site, safety-related SSC's may not be shared by different groups unless it can be demonstrated that this does not significantly affect the capacity of the SSC's to perform their safety functions, including, in the event of an accident in one group, the orderly shutdown of the remaining groups.

*Part 2. Protection against fission products by mans of multiple barriers*

Criterion 10. Reactor design

The reactor core and the cooling, control and protection systems associated with it shall be designed with sufficient margins to ensure that the fuel design limits are not exceeded under any normal operating condition, including the effects of anticipated operating events.

Criterion 11. Intrinsic reactor protection

The reactor core and the cooling systems associated with it shall be designed such that over the entire range of operation at power the net effect of intrinsic nuclear feedback tends to compensate for rapid increases in reactivity.

Criterion 12. Suppression of reactor power oscillations

The reactor core and the cooling, control and protection systems associated with it shall be designed in such a way as to guarantee that there are no power oscillations leading to conditions in which the fuel design limits are exceeded, or that there are guarantees that such oscillations may be detected and eliminated quickly and reliably.

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Criterion 13. Instrumentation and control

13.1 There shall be instrumentation adequate to watch over the behaviour of the main variables and plant systems within the ranges of values expected for normal operating conditions, anticipated operating events and postulated accident conditions, such that the plant may operate safely and reliably.

13.2 The instrumentation shall cover the variables and systems that might affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary and the containment and its associated systems. There shall be the resources required for the automatic registration of the measured values of safety significant variables.

13.3 The instrumentation shall be suitable for measurement of the plant variables and shall be qualified to perform its function under the ambient conditions expected for normal operating conditions, anticipated operating events and postulated accident conditions in accordance with the specific applicable standards.

13.4 Adequate control systems and methods shall be available to maintain the variables and systems within the prescribed operating ranges.

Criterion 14. Reactor coolant pressure boundary

The reactor coolant pressure boundary shall be designed, manufactured, assembled and tested in such a way as to ensure that the presence of abnormal leakage, rapidly propagating failures or the catastrophic rupturing of the barrier are extremely low.

Criterion 15. Design of the reactor coolant system

The reactor coolant system core and the auxiliary control and protection systems associated with it shall be designed with sufficient margin to guarantee that the reactor coolant pressure boundary design conditions are not exceeded under any normal operating condition, including anticipated operating events.

Criterion 16. Containment design

There shall be a reactor containment enclosure equipped with the associated systems required to provide an essentially leak-tight barrier preventing the uncontrolled release of radioactivity to the atmosphere. There shall be guarantees that the safety significant containment design conditions are not exceeded during the period associated with the development of design basis accidents.

Criterion 17. Electricity supply systems

17.1 There shall be an on-site electricity supply system and an off-site electricity supply system allowing for the operation of the safety-related SSC's. The safety function of each of these systems shall be to provide a supply of electricity sufficient to guarantee the following in the event of failure of the other:

1. In the event of an anticipated operating event, that the design limits of the fuel and the design conditions of the reactor coolant pressure boundary are not exceeded, and

2. That under design basis accident conditions the core may be cooled and the integrity of the containment and the other safety functions necessary under these conditions are maintained.

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17.2 The on-site sources of electricity supply, including the batteries and on-site electricity distribution system, shall have the independence, redundancy and testing capacity required to perform their safety functions in the event of a single failure.

17.3 The supply of electrical power from the off-site network to the on-site electricity distribution system shall be provided via at least two physically independent circuits (not necessarily running along separate paths) having the following characteristics:

1. The circuits shall be designed and located such that, wherever feasible, the possibility of their simultaneous failure under normal operating conditions and under postulated accident or ambient conditions is reduced.

The availability of a switchyard common to the two circuits is considered acceptable.

2. Each of these circuits shall be designed such that, in the event of simultaneous loss of the other circuit and of all the on-site sources of electricity supply, the circuit is immediately available as a guarantee at all times of the system being able to fulfil its safety function.

17.4 The design of the electricity supply systems shall include the measures necessary to ensure that, in the event of loss of the power generated by the nuclear power plant or of the on or off-site electricity supply, the probability of losing the electricity supply from any of the other sources as a result of or coinciding with the initiating event is minimised.

Criterion 18. Inspection and testing of electricity supply systems

Safety-related electricity supply systems shall be designed to allow for the periodic inspection and testing of their relevant components and characteristics, such as wiring, insulations, connections and cabinets, in order to verify the continuity of the systems and the condition of their components. The systems shall be designed with the capacity for the following to be periodically tested:

1. The operability and functional capacity of the system components, such as on-site sources of electricity supply, relays, switches and busses.

2. The operability of the systems overall.

3. The complete sequence of actuation initiating the operation of the system under conditions as close as is feasible to the design conditions. This includes the operation of the corresponding parts of the protection system and the transfer of electricity from the main plant generator, the off-site electricity supply and the on-site electricity supply.

Criterion 19. Control room

19.1 There shall be a control room from which the actions necessary to operate the plant safely under normal operating conditions may be taken and from which the plant may be taken to a safe condition and kept there in the event of an anticipated operational transient or design basis accident.

19.2 The design of the control room shall take human factors into account. The control room shall be equipped with visual and, where appropriate, acoustic aids identifying any processes and conditions that have strayed from their normal condition and that might impact safety. The operator shall be provided with the information required to check on the actuation and the effect of automatic actions.

19.3 Events internal and external to the control room that might impact its continued operation shall be identified and the design shall include whatever measures might reasonably be taken to minimise the effects of such events. In particular, suitable protection shall be provided against radiations, such that it be possible to access the control room and remain in it throughout the entire duration of an accident.

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19.4 In addition, instrumentation and control equipment providing the following design characteristics shall be available:

1. Installation in a single location physically and electrically separated from the control room. If the location were not unique, it would be necessary to demonstrate the possibility of operating all the equipment in an integrated manner by means of adequate procedures.
2. Availability of the capacity to take the plant to hot shutdown conditions with sufficient speed, including the capacity to keep the plant in safe hot shutdown conditions.

3. Availability of the potential capacity to subsequently take the plant to cold reactor shutdown conditions by means of adequate procedures.
4. In order to, fulfil the requirement for electrical separation established in previous section 1, the following requirements shall be met:

a) The instrumentation and controls necessary to shut down the plant after evacuating the control room from locations separated electrically from it shall be applicable to the minimum number of trains that by themselves would allow the plant to be taken to a safe condition and kept there.
b) Electrical separation shall be guaranteed by way of appropriate cut-off devices allowing for the transfer of the command and signals required to achieve and maintain the necessary shutdown situation after evacuating the control room, with sufficient speed to achieve the objective mapped out.
c) The electrical separation shall contemplate the existence of specific protective devices making it possible to quickly recover whatever circuits might be affected by the propagation of electrical faults from the control room prior to the transfer.
d) If the licensee considers that installing the devices mentioned in previous sections b) or c) is not feasible or is extraordinarily complex, he shall justify and provide alternatives compatible with the shutdown situation required, in the terms established for equivalent measures in the fifth article of this IS.

e)

*Part 3. Reactivity protection and control systems*

Criterion 20. Protection system functions

The protection system shall be designed to fulfil the following functions:

1. Automatic initiation of the operation of the systems, including reactivity control systems, required to guarantee that in the case of an anticipated operating event occurring, the fuel design limits are not exceeded.
2. Detection of conditions indicating that an accident has occurred and automatically initiating the operation of the safety-related systems and components required to mitigate its consequences.

Criterion 21. Reliability and possibility of testing the protection system

21.1 The protection system shall be designed such that it provides a high level of functional reliability and the possibility of being tested under operating conditions in a manner coherent with the safety functions to be performed.

cve: BOE-A-2017-7715
Verificable en http://www.boe.es

21.2 The design of the protection system shall include sufficient redundancy and independence to guarantee the following:

1. No single failure can cause the loss of the protection function.
2. The removal from service of any protection system component or channel cannot lead to the loss of the minimum redundancy required, unless it can be demonstrated that the operational reliability of the system continues to be acceptable under these conditions.

21.3 The protection system shall be designed to allow for the periodic testing of it complete operation (from the sensing instrument that provides the input signal to the final actuator) during normal plant operation, including the capacity to test the channels independently in order to identify whatever failures and losses of redundancy might have occurred. Exceptions to this criterion shall be adequately justified based on the specific characteristics of the design of the system.

21.4 The protection system shall be designed such that the possibility of an operator action potentially reducing the effectiveness of the protection system during normal operation and in the case of anticipated operating events is reduced. The protection system shall not, however, prevent the operators from taking the correct actions in the event of a design basis accident.

21.5 Digital systems used to perform protective functions, or that might affect such performance, shall provide the following characteristics:

1. The systems shall be designed, constructed, verified, validated, tested and controlled in accordance with the highest quality standards recognised internationally in the nuclear field.
2. The complete systems development process, including the control, testing and implementation of design modifications, shall be systematically documented and reviewed.
3. In those cases in which the reliability of the systems in the case of common cause failures cannot be demonstrated with a high degree of confidence, there shall be an alternative method guaranteeing compliance with their safety functions.

Criterion 22. Independence of the protection system

22.1 The protection system shall be designed such that the protective function is not lost as a result of conditions associated with natural phenomena, normal operation, normal operation, maintenance tasks, the performance of tests or design basis accidents, unless it can be demonstrated that the reliability of the protective function is acceptable using a different technical basis.

22.2 In order to avoid the loss of the protective function, design techniques such as functional diversity or diversity in the design of components and in their operating principles shall be used wherever feasible.

Criterion 23. Protection system failure modes

The protection system shall be designed such that it remains in the condition providing the highest level of safety or in which it can be demonstrated that an acceptable level of safety is provided using a different technical basis in those cases in which the following occurs:

a) Disconnection of the system
b) Loss of electrical feed
c) Loss of instrument air supply
d) Postulated adverse ambient conditions

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Criterion 24. Separation between the protection system and any control system

The protection system shall be separated from the control systems such that in the event of failure of a control system component or channel or failure or removal from operation of a protection system component or channel common to the control and protection systems, there is always an intact system satisfying all the reliability, redundancy and independence requirements of the protection system. The interconnection of the protection system with control systems shall be limited in order to guarantee that safety is not significantly affected.

Criterion 25. Protection system requirements in the event of reactivity control failures

The protection system shall be designed to guarantee that the fuel design limits are not exceeded in the event of any single failure of the reactivity control systems, such as for example the uncontrolled withdrawal of control rods. Control rod ejection or drop events are not considered single failures for the purposes of this criterion.

Criterion 26. Capacity and diversity of reactivity control systems

26.1 There shall be two independent reactivity control systems based on different design principles.

26.2 One of the systems shall use control rods, and it shall be preferable for there to be an active device for rod insertion. The system shall be capable of reliably controlling reactivity changes in order to guarantee that the fuel design limits are not exceeded under normal operating conditions, including anticipated operating events, with an appropriate margin for single failure of the system or functional failures such as rod seizure.

26.3 The second reactivity control system shall be capable of reliably controlling the rate of reactivity variation deriving from normal and scheduled power level changes (including changes in Xenon concentration), such that the fuel design limits are not exceeded.

26.4 At least one of the systems shall be capable of keeping the reactor core subcritical and in cold conditions.

Criterion 27. Combined capacity of reactivity control systems

The reactivity control systems, where appropriate in combination with the injection of neutron poison by the safety systems, shall be designed such that their combined capacity makes it possible to reliably control reactivity changes in order to guarantee that core cooling capacity is maintained under design basis accident conditions, with sufficient margin to cover the possibility of control rods getting stuck.

Criterion 28. Reactivity limits

28.1 The design of reactivity control systems shall include limits for potential increases in reactivity, and for the rate at which such increases may occur, in order to guarantee that the effects of postulated reactivity accidents are limited as follows:

1. There can be no damage to the reactor coolant pressure boundary, other than limited local damage.

cve: BOE-A-2017-7715
Verificable en http://www.boe.es

2. The core, its support structures or other vessel internals are not affected to such an extent that core cooling capacity is significantly impacted.

28.2 The postulated reactivity accidents shall include at least control rod ejection events (unless these may be prevented by positive means), control rod drop events, steam line break, reactor coolant pressure and temperature changes and the injection of cold water in the steam generators (PWR) or reactor vessel (BWR).

Criterion 29. Protection against anticipated operating events

The reactivity protection system and reactivity control systems shall be designed to guarantee that there is a high probability of their fulfilling their safety functions in response to anticipated operating transients.

*Part 4. Fluid systems*

Criterion 30. Quality of the reactor coolant pressure boundary

The components forming part of the reactor coolant pressure boundary shall be designed, manufactured, assembled and tested in accordance with the strictest quality standards. The means required to detect reactor coolant leaks and, wherever feasible, identify the origin and location of such leaks shall be available.

Criterion 31. Prevention of reactor coolant pressure boundary cracking

31.1 The reactor coolant pressure boundary shall be designed with sufficient margin for the following to be guaranteed when it is subjected to the stresses corresponding to the postulated operating, maintenance, testing and accident conditions:

1. The materials forming the barrier do not show brittle behaviour.
2. The probability of rapidly propagating cracking is minimised.

31.2 The design shall consider the in-service temperature and other conditions to which the barrier material is subjected during operation, maintenance, testing and design basis accidents and anticipated operating events, as well as the uncertainties existing in the determination of the following parameters:

1. Material properties.
2. Effects of irradiation on material properties.
3. Residual stresses under steady-state and transient conditions.
4. Defect size.

Criterion 32. Inspection of reactor coolant pressure boundary

Components forming part of the reactor coolant pressure boundary shall be designed such that the following is possible:

1. Periodic inspection and testing of the most important areas and characteristics, in order to check their structural integrity and leaktightness.
2. Implementation of a suitable reactor vessel material surveillance programme.

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Criterion 33. Reactor coolant make-up

33.1 As a protection against minor reactor coolant boundary breaks, there shall be a system providing the capacity to replace whatever coolant might leak from the reactor coolant circuit. The function of the safety system shall be to guarantee that the fuel design limits are not exceeded in the event of coolant losses due to leakage from the circuit or to the rupturing of small pipes or other minor components forming part of the pressure boundary.

33.2 +-The system shall be designed to guarantee that when in operation with the on-site electricity supply (assuming the off-site supply to be unavailable) or with the off-site supply (assuming the on-site supply to be unavailable), the safety function of the system may be carried out using the same piping, pumps and valves as are used to maintain coolant inventory during normal reactor operation.

Criterion 34. Removal of residual heat from the reactor core

34.1 There shall be systems for residual heat removal. The safety function of these systems shall be to transfer decay heat from fission products and other sources of core residual heat, at a heat removal rate sufficient for the design limits of the fuel and pressure boundary established as being acceptable not to be exceeded.

34.2 The capacity of these systems shall be sufficient to remove the residual heat from the core following reactor shutdown and during and after an anticipated operating event or design basis accident.

33.3 These systems shall be equipped with adequate redundancy in their components and characteristics and with the interconnections and leak detection and isolation capacities required to guarantee that when in operation with the on-site electricity supply (assuming the off-site supply to be unavailable) or with the off-site supply (assuming the on-site supply to be unavailable), the safety function of the system may be carried out assuming a single failure.

34.3 The residual heat removal systems shall be designed to allow for the adequate periodic inspection of components performing safety functions, in order to guarantee the integrity and capacity of the systems themselves.

34.4 The residual heat removal systems shall be designed to allow for the performance of suitable periodic pressure and functional tests guaranteeing the following:

1. The structural integrity and leaktightness of components performing safety functions.

2. The operability and actuation of active components performing safety functions.

3. The operability of the complete systems, under conditions as close as feasible to the design conditions, the functional capacity of the system and the operation of the associated cooling water system or systems.

Criterion 35. Emergency core cooling

35.1 There shall be a system providing abundant cooling for the core in the event of an emergency. The main safety function of the system shall be to transfer heat from the core in the event of a loss of reactor coolant at a removal rate sufficient to guarantee the following:

cve: BOE-A-2017-7715
Verificable en http://www.boe.es

1. The prevention of damage to the fuel and cladding that might hinder the effective and continuous cooling of the core.
2. Limitation of reactions between the cladding metal and water to insignificant levels.

35.2 The system shall be equipped with adequate redundancy in its components and characteristics and with the interconnections and leak detection and isolation capacities required to guarantee that when in operation with the on-site electricity supply (assuming the off-site supply to be unavailable) or with the off-site supply (assuming the on-site supply to be unavailable), the safety function of the system may be carried out assuming a single failure.

Criterion 36. Inspection of the emergency core cooling system

The emergency core cooling system shall be designed to allow for suitable periodic inspection of its most important components, in order to guarantee the integrity and capacity of the system.

Criterion 37. Testing of the emergency core cooling system

The emergency core cooling system shall be designed to allow for suitable periodic pressure and functional testing allowing the following to be guaranteed:

1. The structural integrity and leaktightness of its components.
2. The operability and actuation of the active components of the system.
3. The operability of the complete systems, under conditions as close as feasible to the design conditions, the actuation of the complete operating sequence placing the system in operation, including the operation of the applicable parts of the protection system, switching from the normal electricity supply to the emergency supply and operation of the associated cooling water system or systems.

Criterion 38. Containment heat removal

38.1 If necessary there shall be a system or systems for the removal of heat from the reactor containment. The safety function of these systems shall be to reduce containment pressure and temperature at a sufficient rate and in a manner consistent with the operation of other associated systems following any design basis accident, maintaining these variables at acceptably low values.

38.2 Each system shall be equipped with adequate redundancy in its components and characteristics and with the interconnections and leak detection and isolation capacities required to guarantee that when in operation with the on-site electricity supply (assuming the off-site supply to be unavailable) or with the off-site supply (assuming the on-site supply to be unavailable), the safety function of the system may be carried out assuming a single failure.

Criterion 39. Inspection of containment heat removal systems

The containment heat removal systems shall be designed to allow for suitable periodic inspection of their most important components, in order to guarantee the integrity and capacity of the systems.

Criterion 40. Testing of containment heat removal systems

The containment heat removal systems shall be designed to allow for suitable periodic pressure and functional testing allowing the following to be guaranteed:

1. The structural integrity and leaktightness of its components.

2. The operability and actuation of the active components of the system.

3. The operability of the complete systems, under conditions as close as feasible to the design conditions, the actuation of the complete operating sequence placing the systems in operation, including the operation of the applicable parts of the protection system, switching from the normal electricity supply to the emergency supply and operation of the associated cooling water system or systems.

Criterion 41. Scrubbing of the containment atmosphere

41.1 There shall be systems for the control of fission products, hydrogen, oxygen and other substances that might be released inside containment, allowing the following functions to be performed, in a manner consistent with the operation of other associated systems:

1. Reduction of the concentration of the fission products released to the environment during design basis accidents to values guaranteeing compliance with the established radiological limits.
2. Control of the concentration of hydrogen, oxygen and other substances in the containment atmosphere following a postulated accident, in order to guarantee that containment integrity is maintained.
1. Each of the systems shall be equipped with adequate redundancy in its components and characteristics and with the interconnections and leak detection and isolation capacities required to guarantee that when in operation with the on-site electricity supply (assuming the off-site supply to be unavailable) or with the off-site supply (assuming the on-site supply to be unavailable), the safety function of the systems may be carried out assuming a single failure.

Criterion 42. Inspection of containment atmosphere scrubbing systems

The containment atmosphere scrubbing systems shall be designed to allow for suitable periodic inspection of their most important components, in order to guarantee the integrity and capacity of the systems.

Criterion 43. Testing of containment atmosphere scrubbing systems

The containment atmosphere scrubbing systems shall be designed to allow for suitable periodic pressure and functional testing allowing the following to be guaranteed:
2. The structural integrity and leaktightness of its components.
3. The operability and actuation of the active components of the system.
4. The operability of the complete systems, under conditions as close as feasible to the design conditions, the actuation of the complete operating sequence placing the systems in operation, including the operation of the applicable parts of the protection system, switching from the normal electricity supply to the emergency supply and operation of the associated cooling water system.

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Criterion 44. Cooling water systems

44.1 There shall be a system or systems to transfer heat from safety-related SSC's to an ultimate heat sink. The safety function of these systems shall be to transfer the combined heat load of these SSC's under normal operating and postulated accident conditions.

44.2 These cooling water systems shall be equipped with adequate redundancy in their components and characteristics and with the interconnections and leak detection and isolation capacities required to guarantee that when in operation with the on-site electricity supply (assuming the off-site supply to be unavailable) or with the off-site supply (assuming the on-site supply to be unavailable), the safety function of the systems may be carried out assuming a single failure.

Criterion 45. Inspection of cooling water systems

The cooling water systems shall be designed to allow for suitable periodic inspection of their most important components, in order to guarantee the integrity and capacity of the systems.

Criterion 46. Testing of cooling water systems

The cooling water systems shall be designed to allow for suitable periodic pressure and functional testing allowing the following to be guaranteed:

1. The structural integrity and leaktightness of its components.

2. The operability and actuation of the active components of the system.

3. The operability of the complete systems, under conditions as close as feasible to the design conditions, the actuation of the complete operating sequence placing the systems in operation, both for reactor shutdown and in the event of a loss of coolant accident, including the operation of the applicable parts of the protection system, and switching from the normal electricity supply to the emergency supply.

*Part 5. Reactor containment*

Criterion 50. Containment design basis

The structure of the reactor containment and its internal compartments, including the access locks and penetrations, shall be designed such that they are capable of withstanding, with sufficient margin, the calculated pressure and temperature conditions that would occur in the event of a loss of coolant accident, without the containment design leak rate being exceeded. The determination of this margin shall be based on the following considerations:

1. The effects of all possible energy sources not considered for determination of the maximum pressure and temperature conditions, including the energy contained in the steam generators and that generated by the metal-water reaction and other chemical reactions that might occur as a result of the postulated degradation of the operation of the emergency core cooling system, without this function being lost completely.

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2. The limitations of the experience and experimental data available in relation to understanding of the phenomena occurring during the accident and the response of the containment to this accident.

3. The conservatism of the calculation models and input data used.

Criterion 51. Prevention of containment cracking

51.1 The reactor containment barrier shall be designed with sufficient margin to guarantee that under operating, maintenance, testing and postulated accident conditions the ferritic materials forming the barrier do not undergo embrittlement and the probability of rapidly propagating cracking is minimised.

51.2 The design shall consider the in-service temperature and other conditions to which the material forming the barrier is subjected during operation, maintenance, testing and design basis accidents, as well as the uncertainties existing in determination of the following parameters:

1. Material properties.
2. Residual stresses under steady-state and transient conditions.
3. Defect size.

Criterion 52. Capacity to perform containment leaktightness tests

The containment and all other items of equipment that might be subjected to the test conditions shall be designed to allow for the performance of leak testing at the containment design pressure.

Criterion 53. Containment inspection and testing

The reactor containment shall be designed to allow for the following:

1. The performance of suitable periodic inspections of all its important areas, including the penetrations.

2. The implementation of an appropriate surveillance programme.

3. The performance, at the containment design pressure, of periodic leak tests on penetrations with elastic seals or expansion joints, unless it can be demonstrated using a different technical basis that this is not necessary to guarantee the containment leak rate considered as a hypothesis in the applicable safety assessments.

Criterion 54. Systems with pipes penetrating the containment walls

54.1 Systems that include pipes penetrating the walls of the containment shall provide the capacity to detect leakage and isolate the containment, with a redundancy, reliability and actuation capacity in keeping with the safety significance of the isolation of these pipes.

54.2 These systems shall be designed to allow for the following:

1. Periodic testing of the operability of the containment isolation valves and associated equipment.

4. Periodic verification that containment isolation valve leakage is within acceptable limits, unless it can be demonstrated using an adequate technical basis that this is not necessary to guarantee the containment leak rate considered as a hypothesis in the applicable safety assessments.

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Criterion 55. Isolation of reactor coolant pressure boundary piping penetrating the containment walls

55.1 Each pipe belonging to the reactor coolant pressure boundary and penetrating the walls of the containment shall be fitted with containment isolation valves fulfilling one of the configurations indicated below:

1. One isolation valve locked closed inside the containment and one isolation valve locked closed outside.

2. One automatic isolation valve inside the containment and one isolation valve locked closed outside.

3. One isolation valve locked closed inside the containment and one automatic isolation valve outside.

4. One automatic isolation valve inside the containment and one automatic isolation valve outside.

Standard check valves, that is to say valves not fitted with an active mechanism for remote actuation, shall not be used as automatic isolation valves outside the containment.

Configurations other than those indicated above may be considered valid if it is demonstrated using a different technical basis that the containment isolation devices on one pipe or specific pipe type, for example the instrumentation lines, are acceptable.

55.2 Isolation valves located outside the containment must be positioned as close as possible to it.

55.3 In the case of penetrations with two automatic containment isolation valves, at least one of these valves shall be designed such that in the event of a loss of the power required for its actuation, it is left in the position most favourable for safety, unless it can be demonstrated using a different technical basis that the reliability of the containment isolation function is acceptable.

55.4 In order to guarantee an adequate level of safety, whatever additional requirements might be necessary to minimise the probability or consequences of the accidental rupturing of these pipes, or of others connected to them, shall be implemented. Examples of these additional requirements are: the use of higher levels of quality in design, manufacturing and testing; additional capacities for the performance of in-service inspections; protection against severe natural phenomena and the use of additional isolation valves. In order to determine whether these additional requirements are adequate, consideration shall be given to the population density and the physical characteristics and use of the land around the site.

Criterion 56. Isolation of piping open to the containment atmosphere

55.5 Each pipe penetrating the containment walls and connecting directly to the containment atmosphere shall be fitted with containment isolation valves fulfilling one of the configurations indicated below:

1. One isolation valve locked closed inside the containment and one isolation valve locked closed outside.

2. One automatic isolation valve inside the containment and one isolation valve locked closed outside.

3. One isolation valve locked closed inside the containment and one automatic isolation valve outside.

4. One automatic isolation valve inside the containment and one automatic isolation valve outside.

Standard check valves, that is to say valves not fitted with an active mechanism for remote actuation, shall not be used as automatic isolation valves outside the containment.

Configurations other than those indicated above may be considered valid if it is demonstrated using a different technical basis that the containment isolation devices on one pipe or specific pipe type, for example the instrumentation lines or the containment filtered venting system lines, are acceptable.

56.1 Isolation valves located outside the containment must be positioned as close as possible to it.

56.2 In the case of penetrations with two automatic containment isolation valves, at least one of these valves shall be designed such that in the event of a loss of the power required for its actuation, it is left in the position most favourable for safety, unless it can be demonstrated using a different technical basis that the reliability of the containment isolation function is acceptable.

Criterion 57. Isolation of closed system piping

57.1 Each pipe penetrating the containment walls that does not belong to the reactor coolant pressure boundary nor connects directly to the containment atmosphere (closed systems) shall be equipped with at least one containment isolation valve, which shall be automatic or locked closed or allow for remote manual operation. Standard check valves, that is to say valves not fitted with an active mechanism for remote actuation, shall not be used as automatic isolation valves.

57.2 The containment isolation valves of closed systems shall be located outside the containment and as close as possible to it.

*Part 6. Radioactivity control*

Criterion 60. Control of releases of radioactive materials or substances to the environment

60.1 The design of the nuclear power plant shall include adequate means to control the release of radioactive materials or substances in gaseous or liquid effluents and to manage the solid radioactive wastes produced during normal operation of the reactor and anticipated operating events.

60.2 There shall be sufficient capacity to retain gaseous and liquid effluents containing radioactive materials or substances, in particular for those cases in which adverse environmental conditions on site might impose limitations on the off-site release of such effluents.

Criterion 61. Surveillance of radioactive releases

In order to detect the radioactivity that might be released during normal operation, anticipated operating events and design basis accidents, there shall be means available for the monitoring of the atmosphere of the containment and of spaces external to it containing components through which the fluids resulting from an accident might circulate, as well as effluent discharge paths and the area around the plant site.

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*Part 7. Fuel and radioactive waste storage*

Criterion 70. Storage and handling of fuel and radioactive waste

Systems for the storage and handling of fuel, those for the handling of radioactive waste and other systems possibly containing radioactive substances shall be designed to guarantee an adequate level of safety under normal operating conditions and in the event of anticipated operating events and postulated accidents. These systems shall be designed with the following characteristics:

1. Capacity for the performance of suitable periodic inspections and tests on their safety-related components.
2. Shielding suitable for protection against radiation.
3. Adequate containment, confinement and filtering capacity.
4. Residual heat removal capacity, which in the case of storage under water shall be sufficient to maintain the temperature of the coolant within a range of values compatible with the design of the safety-related SSC's under all fuel storage operating conditions, with levels of reliability and a testing capacity reflecting the safety significance of the removal of decay heat and of residual heat in general.
5. In the case of storage under water, capacity to prevent a significant reduction of the fuel storage coolant inventory under accident conditions, such that the fuel assemblies remain covered.

Criterion 71. Prevention of criticality during fuel storage and handling

71.1 Criticality shall be prevented during fuel storage and handling through the use of physical processes or systems, preferably through the use of safe geometric configurations.

71.2 The calculated value of the effective multiplication constant (K effective) of the fresh fuel storage racks, loaded with fuel assemblies of maximum reactivity and covered with pure water shall not exceed 0.95, with a probability of 95% and a level of confidence of 95%.

71.3 In the event of the optimum moderation of the fresh fuel storage racks occurring under conditions of low moderation (for reduced equivalent water densities), the K effective calculated under these conditions for racks loaded with fuel assemblies of maximum reactivity shall not exceed 0.98, with a probability of 95% and a level of confidence of 95%.

71.4 The design of the racks for the storage of spent fuel under water shall fulfil the following conditions:

71.5 If no credit is given to soluble boron in storage, the calculated K effective of the racks, loaded with fuel assemblies of maximum reactivity and covered with pure water shall not exceed 0.95, with a probability of 95% and a level of confidence of 95%.

71.6 If credit is given to the reduction of storage reactivity produced by soluble boron, the calculated K effective of the racks, loaded with fuel assemblies of maximum reactivity shall not exceed 0.95, with a probability of 95% and a level of confidence of 95%, when covered with borated water, and shall be lower than 1.0, with a probability of 95% and a level of confidence of 95%, when covered with pure water.

These conditions shall not apply to fuel loaded in spent fuel storage and/or transport casks when these are located in the spent fuel pool.

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Criterion 72. Surveillance of fuel and waste storage

Fuel and radioactive waste storage systems and associated handling areas shall be equipped with systems appropriate for the following:

1. Detection of the presence of conditions that might lead to loss of the residual heat removal capacity or to excessive radiation levels.
2. Initiation of adequate safety actions.

Four. *Infringements and sanctions*

The present Nuclear Safety Council Instruction is binding, pursuant to the provisions of article 2.a) of Law 15/1980, of April 22nd, creating the Nuclear Safety Council, as a result of which any case of non-compliance therewith shall be penalised as established in chapter XIV (articles 85 to 93) of the Nuclear Energy Act, Law 25/1964, of April 29th.

Five. *Exemptions and equivalent measures*

Exemptions: The CSN may issue temporary exemption from compliance with certain of the requirements of this Instruction, as long as the licensee justifies both the difficulties involved in meeting these requirements in the established manner, which prevent compliance, and the compensatory measures proposed for exemption therefrom.

Equivalent measures: The CSN may report favourably on equivalent measures for compliance with the requirements of this Instruction, in response to proposals from the licensee, as long as the latter accredits such compliance adequately through justification of the equivalent measures proposed.

Single derogatory provision

Instruction IS-27, of June 16th 2010, on general nuclear power plant design criteria, and any other standard of equal or lower standing opposing the present Instruction, is hereby made expressly null and void.

First final provision

The licensees of the nuclear power plants shall incorporate the criteria contained in this Instruction in the first ordinary review of their Safety Studies following the publication of this Nuclear Safety Council Instruction.

The rest of the plant documents shall be updated in accordance with the provisions of this Instruction when reviewed in accordance with the facility's existing documentation maintenance programmes.

Second final provision

An adaptation period of three years is established as from the publication of this Nuclear Safety Council Instruction for the entry into force of what is set out in criterion 19.4.4 thereof; in the event that the information on this point were to consider alternatives that were not finally favourably accepted by the Nuclear Safety Council, the said period would be extended by six months.

Third final provision

The present Instruction shall enter into force on the day following its publication in the «Official State Gazette».

Madrid, June 14th 2017.– The President of the Nuclear Safety Council, Fernando Marti Scharfhausen.

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