The Swedish Nuclear Power Inspectorate Regulatory Code

ISSN 1400-1187
Publisher: Ingvar Persson

SKIFS 2004:2

Date of printing
November 18, 2004

The Swedish Nuclear Power Inspectorate's Regulations concerning the Design and Construction of Nuclear Power Reactors

Decided on October 7, 2004.

On the basis of 20 and 21 §§ of the Ordinance (1984:14) on Nuclear Activities, the Swedish Nuclear Power Inspectorate has issued the following regulations and general recommendations.

Applicability and definitions

1 § These regulations apply to measures required to maintain and develop safety in the design and construction of nuclear power reactors with the aim of, as far as reasonably achievable, taking into account the best available technology, preventing nuclear accidents. The regulations comprise provisions on technical and administrative measures.

These regulations supplement, for application to nuclear power reactors, what has been said about design and construction as well as safety analysis in Chapters 2, 3 and 4 of the SKI's Regulations (SKIFS 2004:1) concerning Safety in Nuclear Facilities.

2 § In these regulations, a nuclear power reactor is the same as the definition provided in 2 § of the Act (1984:3) on Nuclear Activities.

In these regulations, barrier, defence in depth, nuclear accident and safety function are the same as the definitions provided in SKI's Regulations (2004:1) concerning Safety in Nuclear Facilities.

The following terms and definitions are also used in these regulations

Diversification: two or more alternative systems or components that independently of each other perform the same safety task, but in essentially different ways or through having different characteristics.

Single failure: a failure which means that a component cannot fulfil its intended safety task, as well as any consequential failure that arise.

Common cause failure: a failure which simultaneously occurs in two or more systems or components due to one specific event or cause.

Functional separation: systems or components that do not affect each other's function unintentionally.

Physical separation: systems or components that are physically separated, through distance or barriers or a combination of these.

Event class: classification of events conducted in connection with safety analysis and which reflects an expected probability of an event occurring and affecting reactor performance. The following event classes are used in these regulations:

Normal operation (H1)
Includes disturbances that are successfully managed by regular operations and control systems, without interrupted operation.

Anticipated events (H2)
Events that can be expected to occur during the lifetime of a nuclear power reactor.

Unanticipated events (H3)
Events that are not expected to occur during the lifetime of a nuclear power reactor, but which can be expected to occur if several reactors are taken into account.

Improbable events (H4)
Events that are not expected to occur. This also includes a number of overall events that are analyzed to verify reactor robustness, independently of the event frequency. These events are often called design basis events.

Highly improbable events (H5)
Events that cannot be expected to occur. If the event should nevertheless occur, it can result in major core damage. These events are the basis of the nuclear power reactor's mitigating systems for severe accidents.

Extremely improbable events (residual risks)
Events that are so improbable that they do not need to be taken into account as initiating events in connection with safety analysis.

Nuclear fuel bundle: nuclear fuel pins with accessories for load-bearing structures, as well as with such boxes that in boiling water reactors surround the fuel pins and load-bearing structure components[1].

Reactor core: part of the reactor where nuclear fission occurs and which includes the nuclear fuel bundles, control rods and neutron detectors.

Reactor primary system: comprises the reactor pressure vessel and other pressure-bearing devices which are a part of the reactor coolant system or which are connected to the coolant system including

- the external isolation valve in a pipe penetrating the containment wall,
- the reactor pressure relief and blowdown valves,
- the second of two, during operation, normally closed valves in pipes which do not penetrate the containment wall,
- the second of two automatically closing valves which do not penetrate the containment wall.

Redundancy: two or more alternative – identical or different – systems or components that independently of each other perform the same safety task.

Safety system: systems that have the function of ensuring reactor shutdown and residual heat removal, as well as systems that are needed to mitigate con-

1 The term "fuel assembly" is used synonymously with "nuclear fuel bundle" in connection with both boiling water reactors and pressurized water reactors. However, one difference is that pressurized water reactors do not use fuel boxes.

sequences of events, to and including the event category improbable events.

Design principles for the defence-in-depth

3 § The nuclear power reactor shall be designed so that the safety functions reactivity control, protection of the primary system integrity, emergency core cooling, residual heat removal and the containment function[2] can be maintained, to the extent needed depending on the operational state, in all events to and including the event class improbable events.

The design shall take into account events in the event class, highly improbable events in accordance with 4-9 as well as 18-20 §§.

4 § The following design principles shall be taken into account in the design of the reactor's defence-in-depth to the extent reasonably practicable

(a) Simplicity and durability in the design of the safety systems
(b) Redundancy, including diversification as well as physical and functional separation in the design of the safety functions
(c) Automatic control or passive function in necessary activation and operational changeovers of the safety functions
(d) Failure in safety classified equipment leads to an acceptable safety level
(e) Failure in operations classified equipment may not affect the performance of equipment with safety function
(f) When sharing of safety systems between reactors, a failure in one of the reactors shall not affect the possibility to perform shutdown and residual heat removal in the other reactors

Manual measures in connection with necessary activation and operational changeover of the safety functions may only be applied if the personnel is given sufficient time – time for consideration – in order to conduct the measures in a safe manner.

2 The containment function is for boiling water reactors the containment leaktightness function and pressure suppression function, for pressurized water reactors, it refers to the leaktightness function.

5 § The reactor containment shall be designed taking into account phenomena and loads that can occur in connection with events in the event class highly improbable events, to the extent needed in order to limit the release of radioactive substances to the environment.

6 § Instrumentation shall exist which makes it possible to monitor the parameters that are essential for handling of all events to and including the event class highly improbable events.

7 § It shall be possible to cool the reactor core through spraying or sufficent water cover, in all types and sizes of loss of coolant that can result from breaks in connections to the reactor pressure vessel.

8 § It shall be possible in all events, to and including the event class highly improbable events, to achieve a stable end state with a water-covered core/core melt and established residual heat removal. It shall be possible to cool a molten core in a long-term sequence.

Withstanding of failures and other internal and external events

9 § The safety functions in accordance with 3 § shall be able to withstand single failures in all events to and including the event class improbable events. In connection with events in the event class highly improbable events, the active components that belong to the mitigating systems shall be able to withstand a single failure.

10 § Reasonable technical and administrative measures shall be taken in order to counteract common cause failures, in connection with design, manufacturing, installation, startup, operation and maintenance of safety systems.

11 § In order to counteract simultaneous failure of redundant parts of safety systems, the nuclear power reactor shall be designed so that the redundant parts and their support functions have sufficient physical and functional separation.

The degree of separation shall be determined based on the consequences in the facility of the initiating events, which result in the need to take the safety system into operation.

12 § The nuclear power reactor shall be able to withstand global and local loads and other effects, which can occur in connection with a pipe break.

The consequences of a pipe break as initiating event shall be analyzed and assessed with respect to how such effects affect barriers and those safety functions that are credited in connection with the pipe break.

13 § Local dynamic effects do not need to be taken into account in those parts of the facility where the pipe systems have been given such a design, such operating conditions and environmental conditions that the conditions for damage to the piping, as a result of known and identifiable degradation mechanisms, have been reduced as far as possible and where measures have been taken so that damage which, in spite of this, can arise leads to detectable leakage before pipe break occurs.

Further regulations concerning the design, manufacturing and control of pipe systems are stipulated in SKI's Regulations (SKIFS 2000:2) concerning Mechanical Components in Certain Nuclear Facilities.

14 § The nuclear reactor shall be dimensioned to withstand natural phenomena and other events that arise outside or inside the facility and which can lead to a nuclear accident. In the case of such natural phenomena and events, dimensioning values shall be established. Natural phenomena and events with such rapid sequences that there is no time to implement protective measures when they occur, shall also be assigned to an event class. For each type of natural phenomenon that can lead to a nuclear accident, an established action plan shall exist for the situations where the dimensioning values run the risk of being exceeded.

15 § Equipment with readiness for operation requirements may be taken off line for planned maintenance during operation, if the nuclear power reactor is designed so that the safety systems concerned can withstand a single failure in connection with the measures, and the applied diversification and separation of the safety function concerned can be maintained.

16 § Equipment with readiness for operation requirements may be taken off line for repair and testing during operation, if the nuclear power reactor is designed so that the safety functions, in accordance with 3 § can withstand single failure in connection with the measures. Such repair and testing may be applied, even if a safety function does not withstand a single failure in con-

nection with the measures, on condition that a safety analysis shows that the risk contribution that arises in such a way is very small.

Environmental durability and environmental impact[3]

17 § The barriers and equipment which belong to the safety systems of the nuclear power reactor, shall be designed so that they withstand the environmental conditions that the barriers and equipment can be subjected to, in the situations where their function is credited in the safety analysis of the reactor.

Equipment in the nuclear power reactor shall not make such an environmental impact that the performance of the safety functions of the reactor is reduced.

Provisions concerning control rooms

18 § It shall normally be possible to control and monitor the nuclear power reactor from the main control room during all operational states, and it shall be possible to take measures from the main control room to bring the reactor to a safe state, and to keep the reactor in this state, during all events to and including the event class improbable events.

19 § Events that can be a threat to continued activity in the main control room shall be identified and an established action plan shall exist for how these threats shall be handled with maintained reactor safety.

20 § In the case of events where the main control room is not available, an emergency control post shall exist with adequate instrumentation and maneuvering possibilities so that the reactor can be brought to hot shutdown, the residual heat removed and necessary safety parameters can be monitored. The emergency control post shall be physically and functionally separated from the main control room. Monitoring from the emergency control post shall be possible also in the event of a single failure in one of the systems that are necessary for the safe shutdown and cooling of the reactor.

3 Section 17 with general recommendations has been notified in accordance with the European Parliament's and European Council's Directive 98/34/EG.

When bringing the reactor to cold shutdown, other local maneuver posts besides the emergency control post may be used. However, it shall be possible to perform the supervision and monitoring of cold shutdown from the emergency control post.

Safety classification

21 § Structures, systems, components and devices of the nuclear power reactor shall be divided into safety classes. The detailed quality and functional reqluirements, resulting from this safety classification, shall be defined and controlled by specifying sub-classes, including mechanical quality class, electrical function class as well as classification with respect to seismics and environmental durability.

Further provisions concerning quality classification are stipulated in SKI's Regulations (SKIFS 2000:2) concerning Mechanical Components in Certain Nuclear Facilities.

Event classification

22 § In order to analyse safety, the initiating events included in the deterministic safety analysis, in accordance with Chapter 4. 1 § of SKI's Regulations (SKIFS 2004:1) concerning Safety in Nuclear Facilities, shall be divided into a limited number of event classes, with specified analysis assumptions and acceptance criteria.

These event classes shall cover normal operation, anticipated events, unanticipated events, improbable events and highly improbable events. When analysing events that have not been taken into account in the reactor design, realistic analysis assumptions and acceptance criteria may be applied.

Provisions concerning the reactor core

23 § The reactor core and connecting systems shall be designed so that
- design limits for the core can be met with adequate margins in all events to and including the event class <u>anticipated events</u>,
- power transients are not possible, or can reliably be detected and mitigated without exceeding the design limits of the nuclear fuel bundles.

24 § The reactor core and connecting cooling systems shall be designed so that the net impact of the core's immediate reactivity feedback counteracts a reactivity increase during power operation.

25 § The reactor core and reactivity control systems shall be designed in such a way that the reactivity addition is limited in all events to and including the event class <u>improbable events</u>, in order to prevent
- the design limits for the nuclear fuel bundle coolability from being exceeded,
- the reactor pressure vessel internals from being damaged so that core coolability is degraded,
- the acceptance limits in the design specifications for the pressure-bearing parts of the reactor's primary system from being exceeded.

26 § An established limit shall exist for the highest power output from the fuel bundles during <u>normal operation</u>.

In connection with the highest power output in accordance with the first paragraph, it shall be possible to cool the core in the event of a loss of coolant accident. The limit for the highest power output shall be determined so that
- overheating and embrittlement of the fuel cladding and hydrogen production from the bundles are limited in the event of a loss of coolant accident,
- the core geometry is not changed in such a way in the event of a loss of coolant accident that cooling is prevented,
- the residual heat from the nuclear fuel bundle can be removed.

27 § For each fuel design and configuration of the core, established operating limits and parameters shall exist which shall be monitored and followed up during the operation of the core, to the extent needed for the provisions in 23-26 §§ to be met.

The analyses of the design and operating limits for the reactor core shall be reported in the safety report of the nuclear power reactor, in accordance with Chapter 4. 2 § of SKI's Regulations (SKIFS 2004:1) concerning Safety in Nuclear Facilities.

Exceptions

28 § The Swedish Nuclear Power Inspectorate may grant exceptions from these regulations if particular grounds exist and if this can be done without neglecting the purpose of the regulations.

Entry into force and transitional regulations

These regulations enter into force on January 1, 2005.

Without any impediment from the first paragraph, measures for complying with the provisions in accordance with 3-17 and 20 §§ shall be taken no later than on the deadlines established by the Swedish Nuclear Power Inspectorate for each nuclear reactor. The same applies to 18 § with respect to the introduction of additional monitoring equipment, as well as 23 § with respect to the introduction of equipment for detection and automatic protective measures against power transients.

On behalf of the Swedish Nuclear Power Inspectorate

JUDITH MELIN

Erik Jende

GENERAL RECOMMENDATIONS

The Swedish Nuclear Power Inspectorate's General Recommendations concerning the Application of the Regulations (SKIFS 2004:2) concerning the Design and Construction of Nuclear Power Reactors

General Recommendations:

Such general recommendations on the application of regulations which specify how someone can or should act in a certain respect.

[1 § Regulatory Code Ordinance (1976:725)]

The Swedish Nuclear Power Inspectorate's General Recommendations concerning the Application of the Regulations (SKIFS 2004:2) concerning the Design and Construction of Nuclear Power Reactors

Comments on Certain Sections

3 §

This requirement means that the reactor pressure vessel internals, which are also important for maintaining the core geometry, are designed to withstand the loads that can arise during events to and including the event class improbable events.

4 §

The equipment included in safety systems should be designed and positioned in such a way that the probability of deficiencies and malfunctions is low, and that safety is adequate even if deficiencies and malfunctions should arise in the equipment. In connection with failures such as loss of power or with external environmental impact, the equipment should assume a fail-safe position.

The provision [b] on reasonably practicable separation in the design of the safety functions means for instance that safety functions should be independent at an initial stage, in connection with all events to and including the event class anticipated events, namely the execution of the function should not be dependent on the execution of other functions. In this analysis, realistic analysis assumptions and acceptance criteria can be applied. One example of initiating independence in boiling water reactors is that it should be possible for the reactor to be made sub-critical without reliance on pressure relief, and it should be possible for pressure relief to occur without reliance on scram.

The provision [b] also means that equipment, with the main task of functioning in order to limit radioactive releases in connection with severe accidents, shall not be affected by a malfunction in other equipment in the facility.

The provision [c] on automatic control or passive function means, as a rule, that necessary activation and changeover of the safety functions shall be automatic. If this is not possible or reasonable, prepared manual measures can be accepted. No initiating events that require activation of the reactor protection system should, however, result in demands on rapid operator action. Information and time should always be granted to the operator so that he/she can understand the event sequence, the facility status and have time for thought, before the design requires manual action to be taken. Measures required within the first thirty minutes after the initiating event, in order to bring the reactor to a safe state, should be automated for all events, to and including the event class improbable events.

Reasonable time for consideration should exist for operator action also in connection with anticipated and postulated events resulting from the initiating events.

The following time for consideration should apply in the event of severe accidents[4]:

- Manual measures should not be needed for the first 8 hours.
- The manual measures that may be needed after 8 hours should be well prepared and controlled by procedures.

Other measures, which are not prepared, should not be needed until after 24 hours.

If an automatic safety function should not be activated when needed, it should be possible to manually activate the function in the main control room. If an automatic function were to jeopardize safety, possibilities outside the control room should exist to interrupt or block the automatic function. Such an extraordinary measure should be thoroughly analyzed and controlled by procedures.

5 §
The design basis for the reactor containment is events, to and including the event class improbable events, as shown in 3 §. To meet the requirement in 5 §, a safety evaluation should be performed of events and phenomena which may be of importance for containment integrity in highly improbable events. Examples of such events and phenomena, which can result in need to take measures, include high pressure melt-through of the reactor pressure vessel, steam explosion, re-criticality, hydrogen fire and containment underpressure.

4 Included in the event class highly improbable events.

SKIFS 2004:2

8 §
The coolability of a molten core should be included in the safety evaluation mentioned in the general recommendation to 5 §.

9 §
A single failure should be postulated to occur in any component, at the most unfavourable time, in connection with the initiating event or thereafter. A single failure in passive components does not need to be assumed until 12 hours after the initiating event.

Certain components, such as check valves, as well as software and circuit card components have properties which should be subjected to safety assessment before they are considered to be active or passive components in individual cases. A check valve, which has to change position in order to fulfil its safety task, should primarily be considered to be an active component.

The requirement on the ability of consequence-mitigating systems to withstand a single failure can be considered to be fulfilled, if the ability to withstand a single failure exists for active components whose function may be needed within 8 hours after the initiating event, and for components which may be difficult to access for corrective measures when their function is demanded.

10 §
Technical measures are measures for diversification. A suitable and reasonable diversification should be applied to the design of the safety functions in accordance with 3 §, with realistic analysis assumptions and acceptance criteria for events to and including the event class unanticipated events, pipe breaks excluded. When designing such a diversification, all existing power supply to all plant systems can be credited.

The reactor protection system should, as far as reasonably practicable, be designed so that the need for protection is identified and so that protective measures are initiated through at least two different parameters, for example pressure and neutron flux, in connection with all events, to and including the event class unanticipated events. The various ways of detecting an event should be functionally separated.

12 §
Examples of global effects in connection with pipe break include pressure and temperature loads in the area where the pipe break occurs, as well as in

the adjacent areas to which pressure relief occurs, global vibrations due to condensation loads, loads due to flooding and steam release, including other environmental impact.

Examples of local dynamic effects are pipe whips, reaction forces and jets. The ability to withstand such events, especially in the case where a pipe break can result in the failure of an entire safety function, should be achieved through pipe whip restraints, missile shields or changes in pipe configurations.

When analyzing the measures that must be implemented, a pipe break should be assumed to occur where it is important for safety, as well as
- where there are basic conditions for such damage that can lead to pipe break, and
- in accordance with the criteria in SRP 3.6.1 and 3.6.2[5].

14 §
Examples of natural phenomena that should be taken into account are:
- extreme winds,
- extreme precipitation,
- extreme icing,
- extreme temperature,
- extreme sea waves,
- extreme seaweed growth or other biological conditions that can affect the cooling water intake,
- extreme water level,
- earthquake.

Examples of other events that should be taken into account are:
- fire,
- explosion,
- flooding,
- aeroplane crash,
- disturbances to or loss of the offsite grid.

5 US Nuclear Regulatory Commission Standard Review Plan: (SRP) 3.6.1 – Plant Design for Protection Aginst Postulated Piping Failures in Fluid Systems Outside Containment, NUREG 0800. SRP 3.6.2 – Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, NUREG 0800.

In these regulations, the events mentioned in the second paragraph are considered to be of the accident type and not intentionally initiated. Work is underway to issue regulations, on dimensioning and procedures to withstand terrorist attacks and sabotage, in special regulations concerning physical protection of nuclear facilities.

In the analysis of a fire in the facility, a fire that causes all equipment in a fire cell[6] to fail should be assumed to occur. If a fire hazards analysis can show that the probability of failure of an entire fire cell is low, through the fact that protective measures have been taken to prevent fire spreading, the burn out of the entire cell does not have to be assumed. Such a fire hazards analysis should encompass all measures necessary until the fire is extinguished. In the first instance, passive protective measures should be applied such as room dividers, encapsulation or shielding of equipment, minimized fire loads and distance separation between equipment.

If distance separation alone is counted as a protective measure between redundant equipment, this should apply to sufficiently large areas and on condition that the fire hazards analysis confirms that the separation is sufficient to prevent fire spreading.

Furthermore, fire should be taken into account in the following way when analyzing initiating events

- When analyzing fire as an initiating event, an additional fire does not have to be assumed in the facility.

- When analyzing other initiating events besides fire, which in turn can result in a fire, a fire should be assumed to occur as a possible consequential failure of the initiating event.

- When analyzing other events besides fire, which in turn cannot result in a fire, a fire should be assumed to occur no earlier than 12 hours after the initiating event. This event sequence does not have to be combined with a single failure. This applies to initiating events, to and including the event category unanticipated events, apart from pipe breaks.

6 Corresponds to "Fire Compartment" in accordance with IAEA Safety Guide NS-G-1.7: Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants. International Atomic Energy Agency. Vienna, 2004.

17 §
This requirement means that structures, systems, components and devices included in safety systems shall be environmentally qualified. Environments that can affect safety systems should be followed up as long as the systems are utilized for their purpose.

In environmental qualification of electrical equipment in safety systems, the principles for handling of ageing should be applied as specified in IEC 60780[7], Reg. Guide 1,89[8] or IEEE 323[9]. In connection with this, acceleration factors for thermal ageing exceeding 250 times, ionizing radiation lasting less than 10 days or a dose speed greater than 5 Gy/h should be avoided, or the applicability of the results should be justified.

In the case of fuel bundles and control rods, the requirement means that these should be able to withstand the irradiation and the environmental conditions in general, which can occur during all events, to and including the event class anticipated events.

Analyses of how equipment, from the environmental standpoint, can affect the reactor safety functions, should cover all events that are taken into account in the safety analysis of the reactor.

18 §
It should also be possible, from the main control room, to monitor the readiness of the safety functions to operate, namely that the equipment has assumed the correct position for operation. At events in the event category highly improbable events, it should be possible to perform an overall assessment of the facility's safety status.

The interface between the operator and the technical process of the facility should be designed so that the operator is given adequate, reliable and integrated information, which is sufficient to effectively monitor the reactor safety functions, make decisions within the available time, as well as receive feedback on

7 International Electrical Commission. Qualification of electrical equipment of the safety system for nuclear power plants.
8 US Nuclear Regulatory Commission Regulatory Guide. Environmental Qualification of certain Electric Equipment important to safety in Nuclear Power Plants.
9 The Institute of Electrical and Electronics Engineers Inc. Standard for qualifying class 1 E equipment for nuclear power generating stations.

automatic and manual measures. A suitable way of designing the annunciator presentation is pattern recognition.

The adequacy of the main control room and emergency control post should be evaluated and documented within the framework of the periodic safety review of the facility, in accordance with Chapter 4. 4 § of SKI's Regulations (SKIFS 2004:1) concerning Safety in Nuclear Facilities, as well as when operating experience shows that an evaluation is warranted.

An evaluation should comprise experience from the operation of the facility and similar facilities and simulator training, evaluations of the interfaces in relation to ergonomical requirements, as well as evaluations of how well the control room design supports the work of the operators. Local control rooms in the facility should be evaluated in connection with modifications, as well as when experience shows that an evaluation is warranted.

Ergonomical requirements and other conditions, that need to be taken into account in the interaction man-technology-organisation, should be specified at an early stage and taken into account in connection with such modifications to the main control room that relate to these conditions. Recurrent verification and validation of the new solutions should be conducted during the design process so that needed corrections can be made successively. Furthermore, verification and validation should be performed of the entire control room function, before modifications are introduced which essentially affect ergonomical or other conditions in the interaction between the operators and the technical process of the facility[10].

19 §
The threats against continued activity in the main control room, to which the regulations refer, are events like fire, steam release and flooding. A nuclear accident in another reactor at the same site should also be taken into account here. Requirements concerning procedures in connection with threats, such as armed intrusion and sabotage, will be stipulated in the special regulations on the physical protection of nuclear facilities, mentioned in the general recommendations to 14 §.

10 Examples of methodology for the evaluation of control room modifications are to be found in US Nuclear Regulatory Commission: Human Factors Engineering Program Review Model, NUREG 0711.

20 §
When designing the emergency control post, the events and conditions that result in the unavailability of the main control room should be taken into account. The personnel should be able to reach the emergency control post in a protected way. The interface should be designed to facilitate the transfer to working at the emergency control post.

Examples of other local maneuvering posts, besides the emergency control post, include relay rooms, switchgear rooms and local control rooms that do not include the emergency control and monitoring function.

21 §
The classification provides the basis of fulfilling the provisions of Chapter 3. 4 § of SKI's Regulations (SKIFS 2004:1) concerning Safety in Nuclear Facilities, through the design, manufacturing, installation and testing of structures, systems, components and devices with requirements that are adapted to their safety importance. The division into safety classes should be conducted in accordance with the principles provided in ANSI/ANS-51.1 for pressurized water reactors and ANSI/ANS-52.1 for boiling water reactors[11].

22 §
The selection of the initiating events to be included in each event class should be based on an analyzed probability with which the event is expected to occur. However, certain initiating events should be included as postulates, in order to verify the robustness of the facility, independent of the probability of these events occurring. An example of such an event is loss of coolant at a break of the largest pipe or connection to the reactor pressure vessel.

23 §
In the design of the core, the impact of changes in coolant temperature, coolant flow, reactor power and reactor pressure should be taken into account. In the case of pressurized water reactors, changes in the boron concentration of the coolant should also be taken into account.

In addition to design measures, boiling water reactors should have procedures for measures which need to be taken in the event of core instability. The procedures should state what characterizes instability, how it is detected and how it is mitigated. Concerned personnel should be well acquainted with the procedures and should be trained in handling instability.

The stability margins should be calculated for new core loadings.

25 §
In order to ensure nuclear fuel bundle cooling, the design limits stipulate that the nuclear fuel is not fragmented in connection with a reactivity accident. The reactivity value of the control rods should be limited, so that the energy accumulation in the fuel bundles will not become too high.

26 §
When analyzing the limit for the highest power output, the acceptance limits specified in 10 CFR 50.46[12] should be used.

27 §
In addition to limits for the highest power output, limitations should exist which provide margins for fuel bundle overheating and limits for conditions that can lead to stress corrosion cracking of fuel bundles. For pressurized water reactors, there should also be limits for asymmetrical power generation in the core.

In analysis of the limitations that provide a margin for overheating of the nuclear fuel bundles, acceptance criteria in accordance with SRP 4.4[13] should be used.

Further guidance for handling of nuclear fuel bundles at different stages and situations during operation and core configuration modifications, as well as analysis, monitoring, followup and documentation, is provided in the IAEA safety standard: Core Management and Fuel Handling for Nuclear Power Plants[14].

11 ANS-51.1: American National Standard: Nuclear Safety Criteria for the Design of Stationary Pressurised Water Reactor Plants. American Nuclear Society, 1983. ANS-52.1: American National Standard: Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants. American Nuclear Society, 1983.

12 Section 50.46 – Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors. US Code of Federal Regulation. Energy Parts 0 to 50.

13 US Nuclear Regulatory Commission Standard Review Plan (SRP) 4.4 – Thermal and Hydraulic Design, NUREG 0800.

14 Safety Guide NS-G-2.5: Core Management and Fuel Handling for Nuclear Power Plants. International Atomic Energy Agency, 2002.

AB Danagårds Grafiska, Ödeshög 2005