Radiation and Nuclear Safety Authority 107/0002/2016
15.6.2019

Guide YVL B.6, Containment of a nuclear power plant

1 Scope of application

Guide YVL B.6 sets out the detailed requirements and acceptance criteria for the design and leaktightness testing of the nuclear power plant containment, by which compliance with the provisions of Nuclear Energy Decree 161/1988 and requirements of the Radiation and Nuclear Safety Authority Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2018) is ensured and demonstrated. The requirements set out in Guide YVL B.6 supplement the requirements set out in Guide YVL B.1 "Safety design of a nuclear power plant".

2 Justifications of the requirements

The justifications of the guide consist of the limit values for operational occurrences and accidents set out in Nuclear Energy Decree 161/1988 and requirements for the nuclear power plant containment set out in the Radiation and Nuclear Safety Authority Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2018).

Section 22(b) of Nuclear Energy Decree 161/1988 states:

> *The constraint for the annual dose of a member of the public, arising from the normal operation of a nuclear power plant or another type of nuclear facility equipped with a nuclear reactor, shall be 0.1 millisievert. The constraint for the annual dose of a member of the public, arising from the planned decommissioning of a nuclear power plant or another type of nuclear facility equipped with a nuclear reactor, shall be 0.01 millisievert.*
>
> *The limit for the annual dose of an individual in the population, arising as the result of an anticipated operational occurrence, shall be 0.1 millisievert.*
>
> *The limit for the annual dose of an individual in the population shall be 1 millisievert for class 1 postulated accidents, 5 millisievert for class 2 postulated accidents, and 20 millisievert for a design extension condition.*
>
> *The release of radioactive substances caused by a severe reactor accident or a severe accident at a nuclear power plant may not result in the need for large-scale population protection measures or prolonged restrictions on the use of large areas of land and water.*

In order to limit the long term effects, the limit for atmospheric releases of cesium-137 is 100 terabecquerel. The possibility of exceeding the set limit shall be extremely small.

The possibility of a release in the early stages of an accident requiring measures to protect members of the public shall be extremely small.

The Radiation and Nuclear Safety Authority Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2018, Section 10–11) presents the engineered barriers for preventing the dispersion of radioactive materials and the requirements for the containment and safety functions:

1. In order to prevent the dispersion of radioactive substances, the structural defence-in-depth safety principle shall be implemented.

2. Structural defence-in-depth design shall prevent dispersion of radioactive substances into the environment by means of successive barriers which are the nuclear fuel and its cladding, the reactor cooling circuit (primary circuit) and the containment.

3. The nuclear fuel, the reactor, the primary circuit, the cooling circuit (secondary circuit) of a pressurised water reactor removing heat from the primary circuit, the water chemistry of the primary and secondary circuit, the containment and the safety functions shall be designed so as to meet the safety objectives laid down below.

c) In order to ensure containment building integrity,

i. the containment shall be designed to maintain its integrity during anticipated operational occurrences and, with a high degree of certainty, during all accident conditions;

ii. pressure, radiation and temperature loads, radiation levels on plant premises, combustible gases, impacts of missiles and short-term high energy phenomena resulting from an accident shall be considered in the design of the containment; and

iii. the possibility of containment leaktightness becoming endangered as a result of reactor pressure vessel fracturing shall be extremely low.

4. A nuclear power plant shall be equipped with systems to ensure the stabilisation and cooling of molten core material generated during a severe accident. The possibility of direct interaction of molten core material with the load bearing containment structure shall be extremely low.

Section 11 Safety functions and provisions for ensuring them

3. In order to prevent accidents and mitigate the consequences thereof, a nuclear power plant shall be provided with systems for shutting down the reactor and maintaining it in a sub-critical state, for removing decay heat generated in the reactor, and for retaining radioactive materials within the plant. Design of such *systems shall apply redundancy, separation and diversity principles that ensure implementation of a safety function even in the event of a malfunction.*

8. The systems needed for reaching and maintaining a controlled state and the monitoring of the progress of an accident and the plant's status in severe reactor accidents in a nuclear power plant shall be independent of the systems designed for normal operation, anticipated operational occurrences and postulated accidents. The leaktightness of the containment during a severe reactor accident shall be reliably ensured.

9. The nuclear power plant shall be designed so that it can be reliably brought into a safe state after a severe reactor accident.

2.1 Justifications of the requirements by topic

IAEA Safety Standards Series No. NS-G-1.10, "Design of Reactor Containment Systems for Nuclear Power Plants" has been used as a model for the structure of Guide YVL B.6. However, the IAEA requirements have not been directly adopted in Guide YVL B.6. Other sources of the requirements include IAEA Safety Standards Series No. SSR-2/1, "Safety of Nuclear Power Plants: Design", WENRA reference levels and EUR requirements. Guide YVL B.1 sets out the requirements for the design of the nuclear power plant safety systems. Guide YVL B.6 complements the requirements regarding containment set out in Guide YVL B.1.

Justifications for the requirements and references to other guides that clarify the background of the requirements and make it easier to use the guide are presented below.

301. The requirement is based on Requirement 54 of IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), which is also set out in IAEA Safety Standards Series No. NS-G-1.10. Requirements regarding provisions for internal and external hazards are presented in Guide YVL B.7 "Provisions for internal and external hazards at a nuclear facility".

302 and 303. It is allowed to have only the primary containment as protection against external threats (such as an aeroplane crash). However, the capability of the primary containment to fulfil the required function shall be demonstrated. Guide YVL A.11 "Security of a nuclear facility" states detailed requirements for the containment with regard to aeroplane crashes. Requirements for other external events are given in Guide YVL B.7.

305. The requirement refers, for example, to containment flooding or other means whose use shall be taken into account already when designing the containment.

308. The requirement corresponds to the requirement presented in Section 4.49 of IAEA Safety Standards Series No. NS-G-1.10: *The design pressure should be determined by increasing by at least 10% the peak pressure that would be generated by the design basis accident with the most severe release of mass of material and energy. The calculated peak pressure should be determined on the basis of conservative assumptions in relation to the thermohydraulic characteristics.*

309. The containment shall retain its leaktightness even in a case where 100% of the reactor core materials that are easily oxidized react with water. The best-estimate method can be applied in determining leaktightness.

311. The purpose of the 50% safety margin is to take uncertainty related to severe-accident calculation cases into consideration. A severe accident may originate from a very large number of accident sequences with a very small probability. A restrictive case may not have been discovered. The 50% safety margin is not applied in combination with requirement 309, in which 100% oxidation shall be presumed.

When applying requirement 311, the severe accident management systems can be assumed to operate in accordance with the failure criterion presented in Guide YVL B.1. The calculation method is as follows:

1. The pressure behaviour of the containment and the amount of hydrogen in the containment during an accident is modelled according to the calculation method in Guide YVL B.3 “Deterministic safety analyses for a nuclear power plant”.
2. The 50% safety margin is added to the containment pressure curve (overpressure).
3. The pressure increase calculated with the AICC (Adiabatic Isochoric Complete Combustion) principle due to combustion of hydrogen in the containment at any given moment is added to the calculated pressure curve.
4. The maximum pressure is identified from the resulting pressure curve. Uncertainties in the resulting numerical values and times of maximum pressure are taken into account.

Because the margin is added to the loads, the best-estimate method (median level of fragility) can be applied in determining the containment burst pressure. A safety margin does not need to be added to containment temperature resulting from severe reactor accident analyses but the temperature rise resulting from AICC combustion shall be taken into account.

312. Requirement 312 requires the preservation of containment structural integrity. A momentary loss of leaktightness is allowed in the situation described in requirement 312 provided that the dose limit requirements set for an accident situation can be met.

315. Guide YVL B.6 does not require a secondary containment as protection against external threats (requirement 302). Requirement 315 requires a secondary containment function in those containment sections where a leak from the primary containment gas space is possible (e.g. penetrations).

321. During shutdowns, exceptions can be made from the requirement to keep both doors of the air lock closed (if necessary) while observing containment leaktightness requirements 353 and 355.

322a. Requirements for the isolations of pipelines and personnel access hatches and equipment hatches are presented in Sections 3.5 and 3.6. The fulfilment of these requirements ensures sufficient isolation of the containment and no separate general requirement on the single-failure tolerance of isolation is needed. However, it is possible that there are openings and penetrations in the containment that are not considered as pipelines or personnel access hatches and equipment hatches. The objective of requirement 322a is that also such openings are isolated according to the (N+1) failure criterion.

327. The requirement concerning the closure time shall not, however, weaken the reliability of leaktightness of the closure (the primary requirement of the isolation valve is leaktightness).

329. The reason behind the requirement is a requirement presented in the NRC Regulations according to which a simple check valve may not be used as a containment-external automatic isolation valve (“NRC Regulations. Appendix A to Part 50 – General Design Criteria for Nuclear Power Plants”). The requirement is based on possible deficiencies in the leaktightness of such valves. In addition, it is required that the consecutive isolation valves be different, in which case it makes sense for the passive check valve to be located inside the containment and the valve utilising an actuator be located outside where it is more easily accessible. For a justified reason, it is possible to deviate from the requirement, if the containment isolation is reliably implemented regardless of the valve type.

329a. Besides mechanical components, the diversity principle of the containment isolation valves also applies to the control function of the valves. In addition, it is possible that safety system pipelines are isolated later in the accident, at a point where the situation has escalated into a severe accident. Ensuring the control function of isolation valves in such systems shall be single-failure tolerant, as are the other active components used to manage severe reactor accidents.

331. Requirement 456e of Guide YVL B.1 states a fault criterion: *The containment isolation function shall satisfy the (N+1) failure criterion in postulated accidents in spite of possible maintenance, repair or testing operations on the I&C or other auxiliary systems needed to perform the isolation function. In design extension conditions, the I&C and auxiliary systems needed for the DEC A containment isolation shall fulfil the (N+1) fault criterion. Design extension conditions DEC B and C do not require fault postulation*.

337a. This is a design requirement which requires that complex controls in the management of severe reactor accidents (especially plant operator actions) and dependence on external electricity or other external power supplies shall be avoided. The requirement also applies to I&C which is required for monitoring the plant status and planning emergency response activities. The general requirement for the utilisation of components which do not require external power supply in safety functions has been stated in Section 11(2) of Regulation STUK Y/1/2018.

339. Guide YVL B.6 does not require a filtered venting system. It shall be possible to reduce overpressure but the implementation method is not specified. The functioning of the solution used for pressure reduction shall be demonstrated.

340a, 340b and 340c. Section 11(8) of Regulation STUK Y/1/2018 has been updated so that the requirement regarding the independence of severe reactor accident systems from other systems has been specified to apply to systems needed for reaching a controlled state following a severe reactor accident. In the case of the systems used to reach a safe state (in practise, pressure and temperature management) as referred to in Section 11(9) of Regulation STUK Y/1/2018, a milder requirement level can be applied wherein the systems do not need to be independent of the systems designed for normal operation, anticipated operational occurrences and postulated accidents. When applying a milder requirement level, the systems shall, however, fulfil the criteria to demonstrate the reliability of the system operations in a situation following a severe reactor accident.

It is possible that the system(s) to be applied have lost their operability in an earlier stage of a severe accident. Usually, extensive core damage and a severe reactor accident do not take place, if at least a part of the safety systems is working. For this reason, when applying systems not fulfilling the independence requirement, it shall be demonstrated that (1) the system can be repaired if necessary and (2) its operability will remain in the environmental conditions following an accident. The system capacity shall be sufficient and the repair shall be possible sufficiently quickly in view of the operational need.

If repairing is not possible when the system or component is located in a facility with no access in conditions following an accident, single-failure tolerance shall be required. It should be noted that the single-failure tolerance requirement applies to the function and the active components needed to implement it. If the function, for example pressure reduction, can be implemented using more than one system, there is no need to duplicate components of different systems. It is sufficient that the function as a whole has been ensured according to the (N+1) failure criterion.

343. Regulation STUK Y/1/2018 requires that the possibility of direct interaction of molten core material with the load bearing containment structure shall be extremely low, which in practise means either retaining the molten core inside the pressure vessel or placing a core catcher in the containment. Requirement 343 of Guide YVL B.6 on restricting the release of radioactive substances complements the requirement of Regulation STUK Y/1/2018.

344. The removal of radioactive substances from the gas space does not need to fulfil the single-failure criterion, unless the same system is used also for ensuring leaktightness of the containment.

345. The requirements regarding coatings are presented in requirement 514 of Guide YVL E.6 “Buildings and structures of a nuclear facility”.

353. Requirement 353 on containment leaktightness during shutdown applies to nuclear power plants equipped with a pressurised water reactor. In nuclear power plants equipped with a boiling water reactor, it also applies to the secondary containment if, owing to plant design, primary containment leaktightness cannot be maintained during refuelling.

402. The requirement for informing the pressure and leak tests schedule (in connection with the testing programme) is presented in requirement 352a. Test results shall be reported to STUK according to requirement 330 of Guide YVL A.9 “Regular reporting on the operation of a nuclear facility”. Regulatory oversight of the periodic leak test of the containment at the plant site shall be carried out at STUK’s discretion.

2.2 Definitions

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

The definition of a safe state used in Guide YVL B.6 is based on the EUR definition (European Utility Requirements for LWR Nuclear Power Plants, Vol 1, Appendix B):

> *Severe Accident Safe State (SASS)*
> *In case of Severe Accidents* the plant achieves a Safe State if the following conditions are ensured:*
> - *core debris has solidified and temperature is stable or decreasing,*
> - *core debris heat is being removed and transferred to Heat Sink*,*
> - *debris configuration is such that* K_{eff} *is well below 1,*
> - *the containment pressure is so low that, in case of a containment opening, the Criterion for Limited Impact* (CLI) would be met,*
> - *the evolution of fission products to the containment ceased.*

In the EUR document, CLI has been specified as follows:

> *Criterion for Limited Impact is an acceptance criterion, given by a comparison of a linear combination of families of isotope releases, versus a maximum value. Each criterion is associated with a specific kind of limited consequence to the public.*

A leak from the containment can be considered minor, if a non-leaktight containment does not generate releases hindering working at the plant site. Iodine and caesium releases possibly leaking from a non-leaktight containment have to be minor in comparison to the release limits of severe accidents presented in Guide YVL C.3 “Limitation and monitoring of radioactive releases from a nuclear facility”.

3 International provisions concerning the scope of the Guide

3.1 IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design

IAEA SSR-2/1 sets out general-level requirements for the containment of a nuclear power plant. Requirements 54, 56 and 57 of SSR-2/1 are practically the same as the corresponding requirements in Guide YVL B.6. Requirement 55 of IAEA SSR-2/1 regards the containment leaktightness and leakage tests. Corresponding requirements are presented in Sections 3.2 and 3.3 of Guide YVL B.6. Requirement 58 of SSR-2/1 regards the management of the containment pressure and temperature in accident situations. All the requirements presented here have a counterpart in Guide YVL B.6.

3.2 IAEA Safety Standards Series No NS-G-1.10, Design of Reactor Containment Systems for Nuclear Power Plants (2004)

IAEA NS-G-1.10 is more extensive and detailed than Guide YVL B.6. Safety Guide IAEA NS-G-1.10 presents many requirements that are presented somewhere else in the YVL Guides than in Guide YVL B.6. Such requirements include, for example, the safety classification and requirements regarding the design organisation of the containment. The table of contents of IAEA NS-G-1.10 is provided below:

1. INTRODUCTION
2. CONTAINMENT SYSTEMS AND THEIR SAFETY FUNCTIONS
3. GENERAL DESIGN BASIS OF CONTAINMENT SYSTEMS
4. DESIGN OF CONTAINMENT SYSTEMS FOR OPERATIONAL STATES AND FOR DESIGN BASIS ACCIDENTS
5. TESTS AND INSPECTIONS
6. DESIGN CONSIDERATIONS FOR SEVERE ACCIDENTS

The guide was used as a model for the structure of Guide YVL B.6.

3.3 WENRA (2014) reference levels

WENRA reference levels include some requirements concerning the containment design. All WENRA requirements have been observed in Guide YVL B.6.

WENRA E3.1: During normal operation, anticipated operational occurrences and design basis accidents, the plant shall be able to fulfil the fundamental safety functions:

- control of reactivity
- removal of heat from the reactor core
- confinement of radioactive material.

WENRA E7.5: Criteria shall be specified for protection of containment, including temperatures, pressures and leak rates.

WENRA E9.10: A containment system shall be provided in order to ensure that any release of radioactive material to the environment in a design basis accident would be below prescribed limits. This system shall include:

- leaktight structures covering all essential parts of the primary system
- associated systems for control of pressures and temperatures
- features for isolation
- features for the management and removal of fission products, hydrogen, oxygen and other substances that could be released into the containment atmosphere.

WENRA E9.11: Each line that penetrates the containment as part of the reactor coolant pressure boundary or that is connected directly to the containment atmosphere shall be automatically and reliably sealable in the event of a design basis accident. These lines shall be fitted with at least two containment isolation valves arranged in series. Isolation valves shall be located as close to the containment as is practicable.

WENRA E9.12: Each line that penetrates the containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve. This valve shall be outside the containment and located as close to the containment as practicable.

3.4 WENRA Safety of new NPP designs (2013)

WENRA's safety objectives for new plants state an objective in case of O3 accidents with core melt. According to the objective, early or large releases have to be practically eliminated. In terms of other severe accident releases, the objective is to limit protective measures in area and time needed for the public and to ensure sufficient time to implement the protective measures. In the Finnish regulations, the requirement corresponding to the objective is presented in Section 22(b) of the Nuclear Energy Decree (161/1988).

In order to achieve Objective O3, WENRA Safety of new NPP designs (2013) presents requirements regarding the containment:

- Complementary safety features (DiD level 4) specifically designed for fulfilling safety functions required in postulated core melt accidents shall be independent to the extent reasonably practicable from the systems, structures and components of the other levels of DiD.
- Complementary safety features specifically designed for fulfilling safety functions required in postulated core melt accidents shall be safety classified and adequately qualified for the core melt accident environmental conditions for the time frame for which they are required to operate.
- The systems and components necessary for ensuring the containment function in a core melt accident shall have reliability commensurate with the function that they are required to fulfil. This may require redundancy of the active parts.
- It shall be possible to reduce containment pressure in a controlled manner in a long term taking into account the impact of non-condensable gases.
- If a containment venting system is included in the design, the safety margins in containment dimensioning shall be such that it should not be needed in the early phases of the core melt accident, to deal with the containment pressure due to the non-condensable gases accumulating in the containment.
- Containment heat removal during core melt accidents shall be ensured. If included in the design, the containment venting system shall not be designed as the principal means of removing the decay heat from the containment.
- The strength of the containment shall be high enough to withstand static and dynamic loads during core melt accidents that have not been practically eliminated (pressure, temperature, radiation, missile impacts, reaction forces). There shall be appropriate provisions to prevent the damage of the containment due to combustion of hydrogen.
- There shall be provisions to reduce the amount of fission products in the containment atmosphere in case of the core melt accident.
- There shall be provisions to reduce the pressure inside the containment.
- If a containment venting system is included in the design to reduce the containment pressure in a core melt accident, it shall have a filtering capability.
- The containment penetrations should be surrounded by secondary structures to collect the potential leakages from the containment.

In addition, instrumentation is required which is safety classified and adequately qualified for environmental conditions, and based on which countermeasures of an accident can be implemented. WENRA Safety of new NPP designs (2013) presents general-level requirements for the deterministic analyses of severe accidents and their complementary level 2 PSA analyses that are applied to demonstrate the fulfilment of the safety objectives.

Guides YVL B.6, YVL B.1, YVL B.3 and YVL A.7 “Probabilistic risk assessment and risk management of a nuclear power plant” include requirements corresponding to WENRA’s requirements for new plants, but most of the YVL requirements are more detailed. Section 9 of Regulation STUK Y/1/2018 requires the greatest possible independence of the levels of defence required under the defence-in-depth principle.

4 Impacts of the Tepco Fukushima Dai-ichi accident

The Fukushima accident and current understanding of the contributing factors have not caused any major modification needs to the guide.

In the Fukushima accident, hydrogen leaked from the containment into the reactor building and exploded there. The hydrogen explosion destroyed the reactor building but apparently did not damage the containment. Requirement 341 of Guide YVL B.6 “The containment structure and systems used for managing accidents shall prevent such gas burns, gas explosions or other energetic phenomena that may jeopardise containment integrity or leaktightness, or the operability of the components needed for accident management” presents a sufficient requirement in this respect.

Guide YVL B.6 requires that following a severe accident, it must be possible to decrease the pressure difference across the containment pressure boundary to a level consistent with the safe state following a severe accident. However, filtered venting of the containment shall be used for pressure reduction only, if no other means designed for this purpose are available. This means that at new plants decay heat shall be removed from the containment in some other manner than via a filtered venting system.

Requirement 337 includes a self-sufficiency period for containment heat removal in a severe reactor accident during which the heat removal function shall be implemented without any material replenishments external to the plant site. The requirement observes the possibility that operations within the site have become more difficult due to an accident.

5 Needs for changes taken into account in the update

The needs for changes due to changes made to international and national laws/regulations and the change proposals made in connection with the preparation of the YVL Guide implementation decisions (SYLVI) together with others recorded in STUK’s change proposal database have been considered when updating the requirements. In addition, the possibilities to reduce the so-called administrative burden have been considered.

5.1 Safe state following a severe reactor accident

Based on Regulation STUK Y/1/2018, three new requirements 340a, 340b and 340c were added to Guide YVL B.6 that present the minimum requirements for the systems that are needed in achieving a safe state following a severe reactor accident. The requirements concern systems and situations where the systems that are used in achieving a safe state following a severe reactor accident are not independent from the systems designed for the normal operation, anticipated operational occurrences and postulated accidents of the plant.

The new requirements do not change or replace the valid requirements stated for severe reactor accident management systems independent of the other defence-in-depth levels. These dedicated SAM systems have been designed to maintain their operability in the conditions of severe reactor accidents, unlike the EYT-class systems or systems designed for the normal operation, anticipated operational occurrences and postulated accidents, as referred to in the new requirements.

Requirement 338 has been clarified because the previous wording enabled an erroneous interpretation, according to which containment venting would be an alternative option for the other pressure reduction arrangements. Venting can only be used at a late stage of an accident (in pressure reduction to a safe state), if requirement 339 is not otherwise fulfilled. In IAEA Safety Guide “Design of the Reactor Containment and Associated Systems for Nuclear Power Plants” (Draft Safety Guide DS 482, 2017), the matter is expressed in Section 3.43 as follows: *Multiple means to control the pressure build up in accident conditions inside the containment should be implemented, and venting (if any) should be used only as a last resort*. In the same connection, the wording of the requirement was changed so that it makes is clear that this is a design requirement of the plant, not to be interpreted as an operative procedure related to emergency preparedness arrangements or accident management.

5.2 Containment isolation

The guide includes a new requirement 329a. The requirement requires ensuring the containment isolation function (having a backup system) with electrical and I&C systems designed for severe accident management. The requirement applies to the control function of isolation valves. In practice, such ensuring has been required also previously but the requirement, as such, has been missing from the guides.

In requirement 325, “primary circuit” was replaced with the term “primary coolant”. This corresponds to the international practise and better describes the objective of the requirement, which is to isolate the primary circuit water and any fission products carried by it. This way, the requirement cannot be erroneously interpreted to only apply to the pipes that are directly connected to the primary circuit pipelines. For consistency, the same term is used in requirement 323 instead “reactor coolant”.

Requirement 330 regarding single-failure tolerance of the containment penetration isolation was moved within the guide to the end of Section 3.5 (now requirement 322a). In the same connection, the requirement was specified to target such containment penetrations whose sufficient isolation has not been covered by other requirements. At the same time, the title of Section 3.6 was changed to “Containment isolation valves", because after the replacement all requirements presented in the section regard isolation valves. In addition, the previous title was possibly misleading, because the section does not include all containment isolation-related requirements.

5.3 Pressure management and temperature management in accident conditions

Requirement 336 was completed with a mention of the 72-hour self-sufficiency criterion to match content-wise requirement 448a of Guide YVL B.1, which regards residual heat removal from the reactor and containment in a postulated accident. The single-failure tolerance required by the requirement regards a function whose implementation can involve one or several systems. Self-sufficiency is required from containment heat removal also in the case of a severe reactor accident. In this case, the applied systems shall be able to operate without any material replenishments external to the plant site, such as water or fuel, for 72 hours, because brining replenishments to the site and operating at the site during a severe accident may be extremely difficult. Requirement 337 was changed so that it includes this self-sufficiency period applying to the site.

Requirement 337a presents a general objective for the management of severe reactor accidents, which has been previously missing from the guides. The requirement supports, for example, requirement 5444 of Guide YVL B.1 regarding battery dimensioning and the above-mentioned self-sufficiency period of requirement 337. The objective is to avoid unnecessarily extensive dependency on external power supplies and complicated operator or automatic actions in the management and monitoring of severe reactor accidents. In practise, the requirement is applied in designing new plants. The requirement is presented in the form of an objective so as not to make it technically unreasonably limiting. The use of automatic and manual controls is still allowed, as is presented in requirement 5235a of Guide YVL B.1.

5.4 Containment instrumentation

Before the guide update, requirements 347 and 348 overlapped requirements 5218 and 5219 of Guide YVL B.1. Requirements 347 and 348 were removed from Guide YVL B.6. In the same connection, requirement 346, which concerns the same topic, was moved to Guide YVL B.1. A reference to the guides presenting the requirements for containment instrumentation was left in Section 3.13 of Guide YVL B.6 (description 349).

5.5 Other changes

Requirement 311 was clarified by replacing "maximum pressure" with just "pressure" in connection with the calculation method. The change was necessary because the maximum pressure of the containment during an accident is not necessarily reached in the moment when the hydrogen content and the resulting AICC pressure rise are significant. In that case, the "actual" maximum pressure with the AICC combustion assumption would remain unidentified. In order to facilitate the interpretation of the requirement, a description of the calculation method, as unambiguous as possible, was added to this explanatory memorandum for the pressure and temperature at which the containment leaktightness (avoidance of uncontrolled leak) shall be demonstrated in a severe accident.

A reference to requirement 318 was added to the end of requirement 319 in order to make it clear that the requirements link to each other and that the load-bearing capability of penetrations required in requirement 319 refers to maintaining operability and leaktightness during accidents.

The requirement regarding informing STUK of the containment pressure tests and leakage tests schedule was transferred from requirement 402 in Chapter 4 to form its own requirement (352a) in Section 3.14 “Containment pressure tests and leak tests”, wherein it fits better than in Chapter 4 “Regulatory oversight by the Radiation and Nuclear Safety Authority”. In the same connection, the requirement was completed by requiring the submittal of the test programme, in which the test schedule is usually reported. As a result of this change, requirement 403 became unnecessary and it was removed, because the requirement would have merely restated what the divided and updated requirement 402 sets out.

The other updates concerned references to regulations, correction of minor spelling errors, formulation of references and clarifying and unifying the linguistic style.