Radiation and Nuclear Safety Authority 104/0002/2016
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Guide YVL B.3, Deterministic safety analyses for a nuclear power plant

1 Scope of application

Guide YVL B.3 shall be applied to the preparation of deterministic safety analyses in the licensing of new nuclear power plants, plant modifications in existing nuclear power plants as well as periodic safety assessments of nuclear power plants.

2 Justifications of the requirements

2.1 Nuclear Energy Decree (161/1988) and the Radiation and Nuclear Safety Authority Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2018)

The justifications of the guide consist of the requirements for the safety analyses of nuclear power plants, set forth in the Radiation and Nuclear Safety Authority's (STUK) regulation Y/1/2018. Requirements for the scope and methods of safety analyses include:

1. *The safety of a nuclear facility shall be assessed when applying for a construction license and operating license, in connection with plant modifications, and at Periodic Safety Reviews during the operation of the plant. It shall be demonstrated in connection with the safety assessment that the nuclear facility has been designed and implemented in a manner that meets the safety requirements. The safety assessment shall cover the operational states and accidents of the plant. The safety of a nuclear facility shall also be assessed after accidents and, whenever necessary, on the basis of the safety research results.*
2. *The nuclear facility's safety and the technical solutions of its safety systems shall be assessed and substantiated analytically and, if necessary, experimentally.*
3. *The analyses shall be maintained and revised as necessary, taking into account operating experience from the plant itself and from other nuclear facilities, the results of safety research, plant modifications, and the advancement of calculation methods.*
4. *The analytical methods employed to demonstrate compliance with the safety requirements shall be reliable, verified and validated for the purpose. The analyses shall demonstrate the conformity with the safety requirements with high certainty. Any uncertainty in the results shall be considered when assessing the meeting of the safety requirements.*
 (STUK Y/1/2018, Section 3)

Acceptance criteria set for safety analyses have been laid down in Section 22 b, Subsection 2–6 of the Nuclear Energy Decree:

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.

2.2 Justifications of the requirements by topic

The explanatory memorandum provides clarifications facilitating the use of the Guide and references to other Guides.

2.2.1 Chapter 3 Events to be analysed

301. According to Guide YVL B.3, the events to be analysed shall be chosen in a manner that allows the nuclear power plant's behaviour during operational occurrences and accidents as well as the emissions and radiation doses caused by operational occurrences and accidents to be comprehensively analysed. For the analyses, events are usually selected in which

- heat transfer from the reactor cooling circuit is increased or reduced
- coolant flow is increased or reduced
- reactivity or reactor power distribution changes
- coolant inventory is increased or reduced
- radioactive substances are released.

Release and radiation dose analyses are not necessary for all initiating events of analyses concerning plant behaviour. However, they shall be performed for events that are limiting in terms of releases and radiation doses. Other events which may require assessment of releases and radiation doses include, e.g., leaks from a container that contains radioactive liquid.

The frequencies of the initiating events of accident classes are provided with the definitions.

303. The requirement entails a separate report for identifying any operational occurrences and accidents where the course of the event requires operator measures and where erroneous operator action could significantly impact plant behaviour or releases. The report shall assess the possibility of operator errors and the impact of the operator errors. Such events may include, e.g., primary-secondary leaks in pressurised water reactors or leaving the containment relief valve open in the boiling water reactor.

In the analysis of an individual operational occurrence or accident, it can be assumed that the operator acts according to the instructions with regard to the available times (requirement 413).

304. The start-up of a system performing safety functions shall be considered as an initiating event in the same way (under similar assumptions) as any other initiating events.

306. With regard to the analysis of low operating temperatures, Guide YVL B.3 partly overlaps with Guides YVL B.5 “The primary circuit of a nuclear power plant” and YVL E.4 “Strength analyses of nuclear power plant pressure equipment”. Guide YVL B.5 requires (308) that *the permitted loads on the main components of a nuclear power plant at high and low operating temperatures shall be determined and used as a basis for specifying the pressure and temperature ranges for the safe operation of the components during normal operation.*

Chapter 6 of Guide YVL E.4 sets forth the practices for brittle fracture analyses of ferritic steel.

308. A management strategy for severe accidents is required by Guide YVL A.6 “Conduct of operations at a nuclear power plant” (requirement 710): *Guidelines for managing severe accidents shall be prepared for nuclear power plants. The guidelines shall provide a description of the measures for mitigating the consequences of severe accidents.*

Analyses of severe accidents in accordance with requirement 308 shall demonstrate that the chosen strategy meets the acceptance criteria of the Nuclear Energy Decree (161/1988).

2.2.2 Chapter 4 Analyses of plant behaviour

402. The analyses shall demonstrate how the safe state can be achieved. The analysis concerning bringing the facility from a controlled state to a safe state does not necessarily need to employ the same methods as the analysis concerning bringing the facility to a controlled state following the initiating event. In simple cases, the demonstration may also be qualitative. The analysis according to paragraph 1 of requirement 414 can also be ended with a return to power operation if this is the actual state of the plant following the event. However, the analysis shall propose how to reach the safe state when necessary.

403. The analysis methods shall be validated. Validation (the method’s suitability for its purpose) is a broader concept than ensuring the accuracy of programmes.

404. The purpose of the requirement is to enable the review of the models’ correctness in relation to the design of the plant. The initial data (input files) of the analysis models are not required as such.

408. The options for deterministic safety analyses, approved by Guide YVL B.3, are

1. the conservative analysis method supplemented by sensitivity studies

2. the best estimate method supplemented by an uncertainty analysis that can be justified mathematically.

The conservative analysis method does not require the calculation programme to be conservative. Still, the conservativeness of the end result shall be justified by means such as parameter selections and parameter sensitivity studies.

Using the best estimate method does not always require an uncertainty analysis. For example, it is allowed to use the best estimate method to perform design extension conditions (requirement 422) and severe accident analyses (requirement 424) without an uncertainty analysis. An uncertainty analysis shall be performed when the conservative analysis method is replaced with the best estimate method.

409. Sensitivity studies are not required to undergo the kind of processing that can be statistically justified as the best estimate and uncertainty methods do. The aim is to demonstrate the sensitivity of the final result in terms of analysis methods, initial values and the initial state. The number of necessary sensitivity studies depends on the case to be analysed and the margin of the calculation result in relation to the acceptance criteria in the base case.

The thermal conductivity of the fuel gas gap is an example: If the calculation result is close to the acceptance criterion with the thermal conductivity of a gas gap used in a base case, the impact of this assumption shall be clarified by means of its expected variation range. The thermal conductivity of the gas gap depends on, for example, the size of the gas gap and the assumed composition of the gas in the gas gap. Sensitivity studies would, in this case, require several analyses pertaining to the various assumptions.

A sensitivity study shall also be provided in relation to the calculation model (e.g. division into calculation nodes) if this can be expected to significantly affect the end result.

410. The IAEA report: “Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation, IAEA SRS 52” (reference 6 of Guide YVL B.3) lists some generally used best estimate and uncertainty analysis methods. Requirement 408 refers to option 3 of table 1 of report IAEA SRS 52 (the method according to the best estimate, uncertainty evaluation is applied to model uncertainties but not to the failure criterion). The applicable method does not need to be limited to the methods presented in the IAEA SRS 52 report. The important thing is that, with the method, the distribution of the result under review can be determined at the required level of certainty.

412. When the conservative analysis method is replaced with the best estimate method, the failure criterion shall be selected in the same way as when using the conservative analysis method. Option 4 of Table 1 of IAEA SRS 52, in which the operability of systems is also selected on a statistical basis, is not acceptable.

414. The requirement sets forth two analysis methods for anticipated operational occurrences. It is not necessary to analyse all initiating events in both ways. The initiating event analysed with method 2 may cover a less severe event of a similar type, which has been analysed with method 1. For example, method 1 can be used to analyse the stoppage of one feedwater pump and method 2 to analyse the stoppage of all feedwater pumps.

The aim of analyses using method 1 is to gain a more realistic understanding of the plant's behaviour during operational occurrences when systems with no safety classification are used in the analysis in the same way as they would operate during an actual operational occurrence. The purpose of the analysis method is, for its own part, to demonstrate the fulfilment of requirement 432 of Guide YVL B.1. A part of the failures referred to in requirement 432 may be such that their consequences can be assessed with failure analyses and there is no need to analyse them according to method 1.

The acceptance criteria for anticipated operational occurrences are provided in chapter 6.2 of the Guide.

417. Requirement 417 of YVL B.3 establishes how deterministic safety analyses take into account requirement 312 of Guide YVL B.2 "Classification of systems, structures and components of a nuclear facility": *Systems accomplishing safety functions and their necessary support systems shall be assigned to Safety Class 2 if they are designed to provide against postulated accidents to bring the facility to a controlled state and to maintain this state.*

418. Since an off-site grid is not a system in safety class 2, requirement 418 is derived from requirement 417. However, it has been added to Guide YVL B.3 for the sake of clarity. The assumption of the loss of the grid at the worst possible point of time corresponds with the wording of the updated revision of IAEA SSG-2. It shall not be necessary to perform analyses involving multiple grid loss times to indicate the worst point of time if the worst point of time with regard to the analysed event can be qualitatively justified believably.

419. The requirement is based on requirements 446 and 449 of Guide YVL B.1, which require single-failure tolerance from systems used in DEC A situations. DEC A -accident refers to an accident where an anticipated operational occurrence or class 1 postulated accident involves a common cause failure in a system required to execute a safety function.

419–422. Guide YVL B.3 requires that ATWS is analysed in the same way as the other DEC A cases:

- an anticipated operational occurrence where the scram fails (common cause failure)
- the most restrictive single failure in terms of the end result in the system required to execute a safety function

In case of ATWS analyses, analysis assumptions required of DEC A events mean that the following assumptions are made:

- The reactor scram is assumed to fail because of a fault in the protection system that hinders the initiation of the reactor scram function, or because of a mechanical common cause failure in the reactor scram system, or in the control rods, that prevents the insertion of the control rods into the reactor core.

- A single failure of relief and safety valves is assumed.
- Normal operational systems and operators are assumed to act in the most probable way.

420. The loss of off-site power shall be assumed to occur in DEC A accident analyses when the system is a Class EYT system. In DEC B and DEC C accidents, there is no need to assume the loss of off-site power unless it is a likely consequence of the initiating event. The assumption may be necessary due to the failure of shifting to house load operation on account of a turbine scram, for example, or as a result of a rare external threat. Therefore, a small primary circuit leak (DBC3) and emergency diesel common cause failure shall be analysed as DEC A cases, and the loss of off-site power shall be assumed in this case.

422. Sensitivity studies required for DEC A accident analyses "when necessary" mean that if, according to a base case analysed with the best estimate method, the acceptance requirements related to the case are met with a wide margin, it shall not be necessary to conduct sensitivity studies. The more closely the acceptance requirements are met, the more comprehensive sensitivity studies are required.

Sensitivity studies are limited to DEC A accidents because they are used to demonstrate that the dimensioning of the safety systems fulfilling the diversity principle meets the set requirements. DEC B and C accidents are, in nature, "cliff edge" type events as such and, therefore, do not require particular sensitivity studies. However, the parameter choices used in accordance with requirement 411 shall be justified for them as well.

425. The requirement is derived from requirement 456b of Guide YVL B.1: *Systems intended for achieving the controlled state after a severe reactor accident and maintaining this state shall meet the (N+1) failure criterion in terms of active components.*

425a. The assumption of off-site power in severe reactor accidents has been added for the sake of comprehensiveness, because it has been mentioned in all other chapters concerning event classes.

427. Scenarios referred to in requirement 427 include, e.g., feeding coolant from pressure accumulators or the low head safety injection system into the damaged reactor core after primary circuit pressure reduction.

The maximum amount of hydrogen released in a severe accident is one of the dimensioning criteria for the containment. The requirement for it is available in Guide YVL B.6 (309): *The containment shall be dimensioned so as to ensure that it retains its leaktightness in a severe reactor accident even if 100 % of the easily oxidised reactor core materials react with water.*

The requirement for the 100 % reaction of the easily oxidised materials is not an actual analysis requirement because it is trivial as a calculation case. It is not necessary to link the assumption concerning the 100 % reaction of the easily oxidised materials to the scenario referred to in requirement 427.

428. In pressure control analyses, it is not necessary to assume that there is a single failure in the discharge valves, but the valve control system shall meet the (N+1) failure criterion in accordance with requirement 456 of Guide YVL B.1.

430. Guide YVL B.3 requires similar assumptions to be used in analyses concerning primary and secondary circuit overpressure protection.

If the nuclear power plant involves several items to be protected (in a pressurised water reactor plant, the primary circuit and steam generators), the overpressure protection analyses according to Section 430 shall be performed separately for each item to be protected. Therefore, it is not necessary to inspect the secondary circuit as an entity.

2.2.3 Chapter 5 Analyses concerning emissions and radiation doses

501. It is not necessary to analyse the emissions and radiation doses of all initiating events for which analyses concerning plant behaviour are carried out. However, they shall be performed for events that are limiting in terms of emissions or radiation doses.

502. The requirement refers to an additional clarification which helps verify that the containment building's retention capability is acceptable for the largest number of damaged fuel rods allowed by the acceptance criteria for fuel in Class 2 accidents. In the additional analysis, the emission is assumed to occur in the containment building, and containment systems can be assumed to operate in the manner required by Class 2 accidents.

509. Guide YVL B.3 requires more generally that the increasing iodine and caesium concentrations (iodine spiking) are taken into account in the analyses. In line with other requirements for analysis assumptions, the assumptions used shall be justified when preparing the analysis.

2.2.4 Chapter 6 Acceptance criteria for the results

601. Requirements for systems needed to achieve and maintain a controlled and safe state have been provided in Guide YVL B.1, in requirements 444–455.

602. No separate acceptance limits have been provided for the best estimate and uncertainty analysis methods. Instead, a 95 % reliability level (95 % confidence level) is required in order to ensure that 95 % of the distribution of the calculated variable meets the acceptance criterion.

608. Requirement 608 of Guide YVL B.3 derives from requirement 432 of Guide YVL B.1: *No single anticipated failure or spurious action of an active component taking place during normal plant operation shall lead to a situation requiring intervention by systems designed to manage postulated accidents.*

609. The acceptance criterion derives from requirement 404 of Guide YVL B.5: *Primary circuit pressure control shall be so designed as to ensure that pressure can be maintained within the limits required for the normal cooling of the reactor during normal operation and anticipated operational occurrences;* and requirement 405 of

Guide YVL B.5: *Provisions shall be made for normal operational conditions and anticipated operational occurrences by means of systems intended for pressure control to ensure that it will not be necessary to use safety valves to restrict pressure increase in the primary circuit.*

611. Approval requirements are also set forth in Guide YVL B.5.

YVL B.5, requirement 404: *Primary circuit pressure control shall be so designed as to ensure that pressure can be maintained within the limits required for the normal cooling of the reactor during normal operation and anticipated operational occurrences.*

YVL B.5, requirement 405: *Provisions shall be made for normal operational conditions and anticipated operational occurrences by means of systems intended for pressure control to ensure that it will not be necessary to use safety valves to restrict pressure increase in the primary circuit.*

YVL B.5, requirement 406: *Systems related to pressure control shall be so designed as to ensure that it will not be necessary – during normal operation and anticipated operational occurrences – to remove coolant outside primary circuit, with the exception of a potential brief discharge to manage an operational occurrence.*

618. ASME 2011a Section III, Division 1 NB-6221 requires that systems or components are tested under pressure which is at least 1.25 times the design pressure. RCC-M (2000) requires that the primary circuit testing pressure is at least 1.25 times the highest design pressure of a primary circuit component. In the approval requirement used in the guide: *the pressure of the protected item shall not exceed pressure that is 1.2 times the design pressure of the item* the 0.05 margin is subtracted from the testing pressure used in the primary circuit.

2.2.5 Chapter 7 Documents to be submitted to STUK

702. In the decision-in-principle phase, it shall be demonstrated that the organization performing the analyses has the sufficient prerequisites to complete the analyses in the manner required by Finnish regulations. However, submission of the complete validation documentation concerning the calculation methods is not required in the decision-in-principle phase.

710. In this context, a system modification shall refer to a modification as a result of which the updated system does not correspond with the valid system description.

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 Rev. 1, requirement 10 entails that a comprehensive safety analysis is carried out for the nuclear power plant during its design process, ensuring that the safety requirements are met at all stages of the plant’s life cycle. Furthermore, requirement 10 of SSR-2/1 entails that safety analyses are started at an early stage of the design process and that their scope and amount of detail increase as the process moves forward. Interaction between design and verifying analyses shall be enabled during the design process. According to requirement 10 of SSR-2/1, safety analyses shall be documented in a manner that enables independent verifying analyses.

Guide YVL B.3 covers all requirements entailed by IAEA SSR-2/1 Rev. 1 (2016).

3.2 IAEA Specific Safety Guide SSG-2, Deterministic Safety Analysis for Nuclear Power Plants.

It is possible to make a more detailed comparison to IAEA's guide Deterministic Safety Analysis for Nuclear Power Plant (2009). Generally speaking, it can be stated that IAEA SSG-2 is a significantly more detailed guide on preparing deterministic safety analyses in comparison to requirements presented in YVL Guide B.3.

IAEA Specific Safety Guide SSG-2	YVL B.3
Introduction Background Objective Scope Structure	IAEA SSG-2 only contains deterministic analyses. Similarly, YVL B.3 only concerns deterministic analyses. Section "Background" of IAEA SSG-2 sets forth that the guide specifies requirements for safety assessment presented in IAEA SSR 2/1. In this regard, YVL B.3 refers to the requirements of SSR 2/1 as well as STUK regulation Y/1/2018. Section "Background" of IAEA SSG-2 presents various analysis methods (conservative/best estimate method), and both are stated to have their purpose, but the best estimate method is deemed to be particularly useful in situations where margins relative to the acceptance criteria are small. YVL B.3 allows the use of both the conservative and the best estimate method. Section "Scope" of IAEA SSG-2 sets forth that the guide is suitable for analyses performed to demonstrate the durability of a nuclear power plant's physical barriers, i.e. for demonstrating the application of the defence-in-depth principle. In addition, the implementation of the safety functions shall be demonstrated with analyses according to the guide. The

	scope of application of Guide YVL B.3 is the same.
Grouping of initiating events and associated transients relating to plant states	The chapter presents the grouping of initiating events in accordance with IAEA SSR 2/1. The classification corresponds with the division used by YVL Guides with the difference that the YVL Guides group any initiating events exceeding the design bases into three more specific design extension condition classes (DEC A, B, C) and severe accidents. IAEA SSG-2 divides any events exceeding the design bases into two groups based on whether severe fuel damage occurs or not. Neither guide provides explicit lists containing examples of the analysed cases, but describes generic practices which should ensure that initiating events essential for the plant are identified in all event classes.
Deterministic safety analysis and acceptance criteria	This chapter specifies the initiating event classification of the previous chapter by setting forth an example of the quantitative event frequencies defined for various initiating event groups as well as the qualitative acceptance criteria required of different event classes. The event classification and acceptance criteria are, for the most part, consistent in IAEA's guides and YVL guides. Guide YVL B.3 does not set out event category frequencies because they are presented in STUK regulation Y/1/2018. In addition to Guide YVL B.3, acceptance criteria are presented in Guides YVL B.4, B.5, B.6.
Conservative deterministic safety analyses	This chapter specifies at a general level how conservative assumptions affect analysis boundary conditions, the definition of available systems, operator actions and the nodalisation of the analysis model. IAEA's guides concerning conservative analysis methods are consistent with the requirements of Guide YVL B.3.

Best estimate plus uncertainty analysis	This chapter specifies how the best estimate analysis method should be applied in consideration of the used calculation code, sensitivity or uncertainty analysis, boundary conditions, available systems and the nodalisation of the analysis model. IAEA's guides concerning best estimate analysis methods are consistent with requirements presented in Guide YVL B.3.
Verification and validation of computer codes	This chapter provides guidance on how to validate calculation programmes used in safety analyses. The instructions provided are clearly more detailed than the requirements presented in Guide YVL B.3.
Relation of deterministic safety analysis to engineering aspects of safety and probabilistic safety analysis	This chapter specifies how deterministic safety analyses are utilised to demonstrate the acceptability of a plant's safety design as well as the connection to probabilistic risk analyses. Analyses required in STUK's regulation and YVL Guides are according to the instruction.
Application of deterministic safety analysis	This chapter sets forth the application areas of deterministic safety analyses, including assessing the safety design of a nuclear power plant during the licensing process, assessing plant modifications and operational events as well as validating emergency, transient and operating instructions. Guide YVL B.3 shall be applied to drawing up deterministic safety analyses in connection with licensing processes for new nuclear power plants and plant modifications of operating nuclear power plants as well as the periodic safety assessments of nuclear power plants.
Source term evaluation for operational states and accident conditions	This chapter presents the principles according to which the radiation source term released during plant operation or an accident shall be assessed. The instructions are more detailed than the requirements of YVL Guides, but the content is uniform with regard to the goals and principles.

3.3 WENRA (2014) reference levels

WENRA Reactor Safety Reference Levels (January 2014) sets forth some requirements for deterministic safety analyses in Section E “Design Basis Envelope for Existing Reactors” and F ”Design Extension of Existing Reactors”.

E 4.2 requires that the lists of initiating events shall cover all events which may affect the safety of the plant. Design basis accidents shall be selected of the initiating events; using them, plant structures, systems and components are designed in order to implement all the necessary safety functions.

E 4.3 requires that the design basis shall describe the realised plant.

E 5.1 requires that internal initiating events shall be taken into account in the design of the plant. Section E is appended with a list of example cases.

E 6.1 requires that believable event combinations, including internal and external threats that may lead to operational occurrences or accidents, shall be taken into account in plant design and that deterministic methods are used to identify event combinations.

E 7.1 requires that the initiating events shall be grouped according to the probability of the event and that acceptance limits are defined for them to ensure that the consequences of common events remain minor. Events that may have major consequences shall be highly unlikely.

E 7.2–7.5 require that acceptance criteria shall be determined for fuel and the primary and secondary circuit as well as the containment building.

E 8.1 requires that the initial state and boundary conditions of the analyses shall be selected conservatively.

E 8.2 requires that the most significant single failure in terms of consequences shall be assumed in design basis analyses.

E 8.3 requires that only safety systems may be taken into account in the execution of a safety function.

E 8.4 requires that a stuck control rod shall be assumed as an additional fault in design basis analyses.

E 8.5 requires that safety systems shall be assumed to operate in a manner that is the most conservative in terms of consequences.

E 8.6 requires that the consequences of an initiating event shall be included as part of the initiating event.

E 8.7 requires that: a) analyses are made using methods and assumptions that can be justified and are conservative, b) demonstrate that the impact of uncertainties has been assessed in the analyses, c) demonstrate that defining the design basis covers all design basis events, d) analyses are verifiable and repeatable.

F 2.1 requires that deterministic analysis methods shall be used in defining design basis extension cases.

F 3.1 requires the following from deterministic DEC analyses:

- methods and assumptions shall be justified, avoiding excessive conservatism
- methods shall be verifiable particularly in terms of expert assessments
- practical measures for preventing and controlling severe accidents shall be identified by means of analyses
- radiological consequences of DEC cases shall be assessed
- with analyses, an adequate margin shall be demonstrated for “cliff edge” consequences.

Guide YVL B.3, together with Guides YVL B.1 and YVL B.5, meets all the requirements of WENRA reference levels. When assessing the fulfilment of the requirements, it shall be taken into account that WENRA defines DEC A and B accidents in a very different manner from the Finnish regulations.

4 Impacts of the Tepco Fukushima Dai-ichi accident

The Fukushima accident does not impact the scope of application of Guide YVL B.3. Guide YVL B.3 describes what kind of deterministic analyses shall be made and how they are done acceptably.

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.

Change needs for Guide YVL B.3 were minor. The updates mainly involved correcting spelling errors, updating and harmonising references and clarifying requirements without changes to the requirement level. A few requirements have been removed due to overlaps with Guide YVL E.4; one has been removed due to an internal overlap in YVL B.3; and one requirement has been moved to Guide YVL B.1. Any possibilities for administrative burden reduction have not been identified in the requirements of the Guide.

The aim has been to clarify requirement 414 in order to make the two analysis methods more easily understandable. The earlier requirement version was missing a reference in connection with method 1 analysis, concerning the assumption of single failures in systems which are used to limit the development of operational occurrences into accidents. This is not a change in the requirement level but a more transparent way to present it, because the failure criterion provided in YVL B.1 has not been changed in this regard.

Requirement 418 entails an assumption of the loss of the grid at the worst possible point of time in terms of managing the situation. Typically, the loss of the grid has been assumed to occur simultaneously with the initiating event. The purpose of the requirement is to require the licence applicant or holder to provide, in connection with the analysis, a justified assessment of what kind of impact the timing of the grid loss shall have on the analysed event. In situations where the timing of the grid loss and its impact on the consequences of the analysed event cannot be clearly assessed, it may be necessary to perform analyses using several assumed points of time when the grid is lost.

Requirement 419a specifies that, in the analyses of DEC A events, it is not allowed to use systems which have not been safety-classified. The requirement is not stricter than before; however, previously, making the correct interpretation required combining requirements from Guides YVL B.1 and B.2 with requirements from Guide YVL B.3.

A requirement has been added to requirement 422, concerning sensitivity studies carried out during DEC A events if a base case analysed with the best estimate method is very close to meeting the acceptance criteria. The need for sensitivity studies shall be assessed and justified on a case-by-case basis.