IAEA SAFETY STANDARDS

for protecting people and the environment

DESIGN OF REACTOR CONTAINMENT STRUCTURE AND SYSTEMS FOR NUCLEAR POWER PLANTS

DRAFT SAFETY GUIDE

DS 482

STATUS: STEP 11 **Submission to Review Committees**

Date: 2017-04-15

IAEA
International Atomic Energy Agency

FOREWORD

Later

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide was prepared under the IAEA programme for safety standards for nuclear power plants. It is a revision of the Safety Guide on Design of Reactor Containment Systems for Nuclear Power Plants (Safety Series No. NS-G-1.10, 2004), which is now superseded by this safety guide.

1.2. The confinement of radioactive material in a nuclear power plant, including the control of discharges and the minimization of radioactive releases into the environment, is a fundamental safety function to be ensured in any operational condition and accident condition. In accordance with the concept of defense in depth, this fundamental safety function is achieved by means of several barriers and levels of defense. For nuclear power plants, a strong structure surrounding the reactor termed ‘containment’ in this publication is designed for the confinement of radioactive material, notably in accident conditions. Moreover, taking into account energy and combustible gases released in case of accident, systems designed to preserve the integrity of the containment or to avoid a bypass of the containment are necessary. Systems necessary for the normal operation, or to minimize radioactive releases, to remove energy or to preserve the structural integrity of the containment in accident conditions are termed ‘associated systems’ or ‘systems’ in this publication.

1.3. Issues related to the confinement of spent fuel are addressed in [12] and [24]

OBJECTIVE

1.4. The objective of this Safety Guide is to make recommendations on the implementation and fulfillment of SSR-2/1 Revision 1 requirements [2] relevant for the containment structures and systems It is expected that this publication will be used primarily for land based, stationary nuclear power plants with water cooled reactors designed for electricity generation or for other heat generating applications (such as for district heating or desalination). It is recognized that for other reactor types, including future plant systems featuring innovative developments, some of the recommendations may not be appropriate or may need some judgment in their interpretation.

SCOPE

1.5. Recommendations given in this Safety Guide are targeted primarily to new nuclear power plants. For plants designed with earlier standards, comprehensive safety assessments are to be carried out considering these recommendations in order to identify safety improvements that are oriented to prevent accidents with radiological consequences and mitigate such consequences should they occur.

Reasonably practicable or achievable safety improvements are to be implemented in a timely manner [16] **[16]**. For their identification general guidance is provided in Appendix 1.

1.6. This Safety Guide addresses the functional aspects of the containment and major systems associated to the containment for the management of energy, radionuclides and combustible gases for the plant states considered in the design envelope. In particular, guidance and recommendations relevant for the design of equipment and systems necessary for the mitigation of design extension conditions without significant fuel degradation and for accidents with core melting have been added. Consideration is also given to the definition of the design bases for the containment structure and systems, in particular to aspects affecting the structural design, the reliability and the independence of systems that do not belong to the same level of defense.

1.7. Recommendations are provided on the tests and inspections that are necessary to ensure that the containment structure and systems are capable to accomplish their intended functions throughout the operating lifetime of the nuclear power plant.

1.8. Design limits and engineering criteria, together with the system parameters that should be used to verify them, are specific to the design and to the individual Member State, and are therefore outside the scope of this Safety Guide. However, general recommendations are provided.

STRUCTURE

1.9. In Section 2 the safety functions related to the containment structure and systems are indicated, and the application of the main requirements of SSR 2/1 rev 1 to be considered are addressed. Section 3 provides recommendations to the design basis of the containment structure and systems. Section 4 provides specific recommendations for the design of the containment structures and systems. Section 5 covers tests and inspections, and provides recommendations for commissioning tests and for in-service tests and inspections.

2. CONTAINMENT SAFETY FUNCTIONS AND DESIGN APPROACH

2.1. This section addresses the application of the principal technical requirements given in [2] for the design of the containment structures and systems.

SAFETY FUNCTIONS

2.2. The containment and its associated systems are designed to perform together with other design provisions the following safety functions (see [2], Requirements 4 and 54) :

- Confinement of radioactive material in operational states and in accident conditions,

- Protection of the reactor against external natural and human induced events,
- Radiation shielding in operational states and in accident conditions.

2.3. Conditions under which those functions need to be accomplished are identified and characterized to define the different elements of the design bases of the relevant structures, systems and components (see [2], Requirement 14).

CONFINEMENT OF RADIOACTIVE MATERIAL

2.4. The containment structure and its associated systems is primarily designed to ensure that any radioactive release from the nuclear power plant to the environment is as low as reasonably achievable, to comply with the authorized limits on discharges in operational states and the dose limits accepted by the regulatory requirements in accident conditions to achieve the required level of protection of the people and the environment (see [2], Requirement 55):

- For operational states, the cumulative annual effective dose received by people living in the vicinity of a nuclear site is expected to be comparable to the effective dose due to natural exposition originally existing at the site (an increase of up to about 1 mSv over the dose received in a year from exposure due to naturally occurring radiation sources is recommended by [21];
- Radiological releases in accident conditions are to be dealt with as follows:
  - ✓ For design basis accidents and design extension conditions without significant fuel degradation, the releases are minimized such that off-site protective actions (e.g. evacuation, sheltering, iodine thyroid blocking) are not necessary (see [2], Requirement 19 item 5.25);
  - ✓ For design extension conditions with core melting, the releases are minimized such that only off-site protective actions limited in terms of areas and times are necessary (see [2], Requirement 20 item 5.31A), and sufficient time shall be available to take such measures;
- Sequences which might lead to an early radioactive release or a large radioactive release are 'practically eliminated' by appropriate design provisions (see [2], item 2.13/4).

2.5. Moreover the containment and its associated systems are designed so that any radioactive release is as low as reasonably achievable, is below the authorized limits on discharges in operational states, and is below acceptable limits in accident conditions (see [2], Requirement 55).

2.6. Leak tightness of the containment is an essential element to confine radioactive material and to minimize radiological releases. Leak tightness is generally characterized by specified maximum leak rates (overall leak rate, specific leak rates for containment penetrations, hatches and containment isolation valves) which are not expected to be exceeded under accident conditions. Equipment ensuring the role of confinement barrier is designed and qualified to keep integrity and leak tightness, prior and during conditions for which they are necessary (see [2], Requirement 55).

2.7. A containment isolation system is necessary to confine radiological releases into the containment atmosphere caused by accident conditions (see [2], Requirement 56).

2.8. To preserve the containment structural integrity in accident conditions, systems designed to ensure that the design limits of the containment structure (e.g. pressure, temperature, combustible gases) not be exceeded are implemented as necessary. Multiple means are implemented for the energy management in accident conditions, and those specifically dedicated for conditions with core melting are functionally separated and independent as far as practicable from other systems. (see [2], Requirements 58 and 7).

2.9. Stresses in the civil structures due to loads or combinations of loads caused by operating conditions, accident conditions and hazards are such that the structural integrity of the containment and of the systems required for the mitigation of the accident conditions is maintained with appropriate margins (see [2], Requirement 42).

2.10. Regardless of the multiplicity of design provisions taken to prevent accident from escalating to significant core damage, a set of the most likely representative core melting conditions is postulated. For such conditions, additional safety features are implemented to minimize the radiological releases (see [2], Requirement 20).

2.11. Additionally to measures implemented to mitigate the consequences of the postulated conditions, the use of non-permanent equipment is considered, and adequate connection points and interfaces with the plant are installed with the objective to avoid large release and unacceptable off-site contamination in case of accidents exceeding those considered in the design (see [2], Requirement 58).

2.12. In accident conditions, high energetic phenomena that could jeopardize the structural integrity and the leak tightness of the containment are dealt with adequately by incorporating features for their practical elimination (see [2], Requirements 20 and 58).

PROTECTION AGAINST INTERNAL AND EXTERNAL HAZARDS

2.13. The containment, or the shielding structure is designed to protect structures, systems and components (SSCs) housed inside the containment against the effects of natural and human induced external hazards identified by the site hazard evaluation, and against the effects of hazards originated by equipment installed at the site. For both of them, causation and likelihood of hazard combination is considered (see [2], Requirement 17).

2.14. The containment, or the shielding structure is also designed to provide protection against the effects of possible malicious acts directed against the facility, but these aspects are outside the scope of this Safety Guide.

RADIATION SHIELDING

2.15. In operational states and in accident conditions, the containment contributes to the protection of plant personnel and the public from undue exposure due to direct radiation from radioactive material contained within the containment. The composition and thickness of the concrete, steel and other materials is such that the dose limits and dose constraints for operators and the public remain below acceptable limits and as low as reasonably achievable in, and following accident conditions (see [2], Requirement 5).

3. DESIGN BASIS OF CONTAINMENT STRUCTURES SYSTEMS AND COMPONENTS

GENERAL

3.1. The design of the containment structure and systems important to safety should be conducted taking into account the recommendations of GS-G-3.1 [14] **[14**and GS-G-3.5 [15] to meet the requirements 1 to 3 of SSR-2/1 Rev.1 [2] and GSR Part 2 requirements [13].

3.2. The design of the containment structure and systems important to safety should be conducted taking into account design recommendations for safety and security in an integrated manner in such way that safety and security measures do not compromise each other. Recommendations for security are detailed in [17].

3.3. The design basis for the containment and its associated systems should include any condition created by normal operation, anticipated operational occurrences, accident conditions (design basis accidents and design extension conditions). Load combinations created by internal and external hazards should also be included in the design basis of the structures, systems and components.

3.4. Design conditions and design loads should be derived from combinations of bounding conditions determined for the relevant plant states or hazards.

3.5. The performances of structures and systems necessary for operational states should be derived from the following needs:

- To confine the radioactive material,

- To minimize radiological releases,
- To contribute to biological shielding,
- To maintain pressure and temperature within the range specified for operational states,
- To establish and maintain adequate environmental conditions in the working areas,
- To provide for the necessary access and egress of personnel and materials,
- To perform containment structural and leak tightness tests,
- To accommodate the loads occurring during operational transients (e.g. loads due to differential thermal expansion, variation of outside environmental temperature).
- Other factors, including security considerations, also need to be taken into account.

POSTULATED INITIATING EVENTS

3.6. The following recommendations provide guidance to fulfill Requirement 16 [2]

3.7. The postulated initiating events (PIEs) relevant for the containment structures and systems should include equipment failures and errors potentially leading to accident conditions with a significant release of radioactive material or with a significant release of energy inside the containment. PIEs occurring in shutdown modes, with an open containment or when systems may be disabled for maintenance, should also be considered.

3.8. Typical postulated initiating events that should be relevant for the design of the containment and its associated systems are:

- Large, medium and small breaks in the reactor coolant system;
- Large, medium and small breaks in the main steam/feedwater system;
- Equipment failure in systems carrying radioactive liquid or gas within the containment;
- Fuel handling accidents in the containment.

INTERNAL HAZARDS

3.9. The following recommendations provide guidance to fulfill Requirement 17 [2] with its associated requirements relevant for 'Internal Hazards'.

3.10. Internal hazards that should be considered for design are those hazards of internal origin that may jeopardize the performance of the containment and its associated systems. A list of typical internal hazards usually considered is given for guidance; nevertheless, this list should be supplemented as needed to include specific hazards relevant for the design:

- Breaks in high energy systems located inside or outside the containment;
- Breaks in systems or components containing radioactive material located in the containment;
- Failure of fuel handling equipment;
- Heavy load drop;
- Internal missiles;
- Fires and explosions;
- Flooding.

3.11. Layout and design provisions should be taken to protect the containment structure and its associated systems against the effects of the internal hazards:

- The containment structure and the systems should be protected against impacts of high energy (internal missiles, pipe whipping, jet impingement, heavy loads), or designed to withstand their loads and the loads caused by explosions as well;
- The redundancies of the systems should be segregated to the extent possible or adequately separated, and protected as necessary to prevent the loss of the safety function performed by the system (prevention of Common Cause Failure initiated by the effects of the internal hazards);
- The implemented segregation, separation and protection should also be adequate to ensure that the modeling of the system response described in the analysis of the PIEs be not compromised by the effects of the hazard;
- A single hazard should not have the potential for a common cause failure between safety systems designed to control design basis accidents and safety features required for design extension conditions with core melting.

3.12. Methods, design and construction codes used should provide adequate margins to justify that cliff edge effects would not occur in the event of a slight increase of the severity of the internal hazards (see [2], Requirements 9 and 11).

3.13. More detailed recommendations are provided in [3].

EXTERNAL HAZARDS

3.14. The following recommendations provide guidance to fulfill Requirement 17 [2] with its associated requirements relevant for 'External Hazards'.

3.15. A list of typical external hazards, and their combination as appropriate, usually considered is given for guidance in [9] and [11] but should be adapted or supplemented as needed to include the site specific hazards:

3.16. Containment structures and buildings housing systems required to mitigate the consequences of accident conditions should be designed to withstand the loads imposed by external hazards and protected against the effects caused by the neighboring buildings not designed to withstand external hazard loads.

3.17. Systems required for energy management, the control of radionuclides and of combustible gases in accident conditions should be protected against the effects of external hazards or designed to withstand the loads caused by the external hazards. For each hazard, components whose operability or integrity is required during or after should be identified and specified in the design basis of the component.

3.18. Design methodologies should contain measures to verify that adequate margins exist to avoid cliff edge effects in the event of a small increase of the severity of the external hazards (see [2], Requirements 9 and 11).

3.19. Short term actions necessary to meet the dose limits and engineering criteria established for the containment in the event of design basis accidents or design extension conditions should be accomplished by permanent systems (see [2], Requirement. 5.17).

3.20. The autonomy of systems designed for energy management, the control of radionuclides and the management of combustible gases inside the containment during accident conditions should be longer than the time necessary prior to crediting off-site support services. The autonomy can be achieved crediting the provisions taken at the unit and at the site provided that the potential for some specific hazards to give rise to impacts on several or even all units on the site simultaneously has been considered (see Requirement 5.15B [2]).

3.21. The following recommendations provide guidance to prevent an early radioactive release or a large radioactive release in the event of levels of natural hazards exceeding those considered for design, derived from the hazard evaluation for the site (Requirement 5.21A [2]).

3.22. In the event of levels of natural hazards exceeding those derived from the site hazard, the structures, systems and components (SSCs) which are ultimately necessary to prevent an early radioactive release or a large radioactive release from the containment should refer in particular to the SSCs necessary to mitigate the consequences of accidents with core melting, and to prevent conditions not considered in the design of the containment structures. A detailed list of these SSCs is design dependent. The below list provides typical examples of SSCs which could be considered :

- Containment structure;
- Equipment or structure necessary to contain the molten core;
- Systems necessary to remove heat from the molten core;
- Systems necessary to remove heat from the containment and transfer heat to the ultimate heat sink in design extension conditions;
- Systems to prevent hydrogen fast deflagration or detonation;
- Containment venting system (if it exists);
- Containment isolation.

3.23. For external flooding this would mean that either all the structures hosting the above mentioned systems are located at an elevation higher than the one derived from the site hazard evaluation , or adequate engineered safety features (such as water tight doors etc.) should be in place to protect these structures and ensure that mitigating actions can be maintained.

3.24. Margins provided by the design of the containment structure should be adequate so that its integrity is preserved in case of natural hazards exceeding those resulting from the site hazard evaluation.

3.25. Margins provided by the design of the associated systems ultimately necessary to avoid an early radioactive release or a large radioactive release should be adequate so that the integrity and operability of those systems would be preserved in case of natural hazards causing loads exceeding those resulting from the site hazard evaluation.

3.26. More detailed recommendations are provided in [9 **[9**].

ACCIDENT CONDITIONS

3.27. Accident conditions relevant for the design of the containment and of the associated systems should be those having the potential to cause excessive mechanical loads or to jeopardize the capability to limit releases of radioactive substances to the environment.

3.28. Accident conditions should be used for determining capabilities, loads and environmental conditions in the design of the containment structures and systems.

3.29. The determination of capabilities, loads and environmental conditions should be based on, but not necessarily limited to the following:

- The mass and energy of releases inside the containment as a whole and as a function of time;
- The heat transfer to the containment structures and those to and from components;
- The mechanical loading, both static and dynamic, on the containment structure and its sub compartments;
- The releases of radionuclides inside the containment;
- The amount of radionuclides released to the environment;
- Cooling and stabilization of the molten core;
- Localization of the corium (for ex-vessel corium retention strategy);
- The rate of generation and amount of combustible gases released inside the containment.

3.30. The following recommendations provide guidance to fulfill Requirement 18 [2].

3.31. To the extent practicable, codes and engineering rules that are used for design should be documented, validated and, in the case of new codes, developed according to up to date knowledge and recognized standards for quality assurance. Users of the codes should be qualified and trained with respect to the operation and limits of the code and with respect to the assumptions made in the design.

3.32. Calculation of boundary conditions for design basis accidents and design extension conditions should be documented, indicating the relevant assumptions for the evaluation of parameters, the engineering criteria and the computer codes that are used.

3.33. Computer codes should not be used beyond their identified and documented domain of validation.

Design basis accidents

3.34. The following recommendations provide guidance to fulfill Requirement 19 [2].

3.35. For the performance of the containment structures and systems, conditions retained as design basis accident conditions should be calculated taking into account the less favorable initial conditions and equipment performance, and the single failure[1] which has the largest impact on the performance of the safety systems. Care should be taken when introducing adequate conservatism, since:

- For the same event, an approach considered conservative for designing one specific system could be non-conservative for another;
- The adoption of excessively conservative assumptions could lead to unrepresentative analysis and consideration of undue stresses on components and structures.

3.36. Containment structure and systems should be designed such that venting is not necessary during DBAs.

Design extension conditions

3.37. The following recommendations provide guidance to fulfill Requirement 20 [2].

3.38. Design extension conditions should be identified and used to establish the design bases of containment structure and of systems necessary to meet the objectives established for that category of accidents. For design extension conditions without significant fuel degradation and radiological consequences should be comparable to those established for design basis accidents, and for accident with core melting, the radiological release should be such that the necessary off-site protective actions remain limited in terms of times and areas.

3.39. Calculation performed to assess conditions imposed by DECs may be less conservative than those imposed by design basis accidents provided that margins needed to avoid cliff edge effects be still sufficient to cover uncertainties. Performing sensitivity analyses could also be useful to identify the key parameters.

3.40. DECs relevant for the design of the containment structure and systems should be identified on the basis of a deterministic approach supplemented by probabilistic analyses and engineering judgment.

3.41. For design extension conditions without significant fuel degradation, in general, the following three types should be considered:

- Equipment failure(s) leading to a release of mass and energy higher than the one postulated for design basis accidents (e.g. by a LOCA, main steam line break, etc.);
- Multiple failures (e.g. common cause failures in redundant trains) in the containment systems that prevent the safety systems from performing their intended function when requested;

1 IAEA glossary, 2016Revision [25]: A single failure is a failure which results in the loss of capability of a single system or component to perform its intended safety function(s), and any consequential failure(s) which result from it.

- Multiple failures that cause the loss of a safety system while this system is used to fulfill the fundamental safety functions in normal operation (e.g. residual heat transport systems).

3.42. As multiple failures are likely caused by the occurrence of dependent failures that may lead to the failure of the safety systems, an analysis of dependences between redundant trains of systems installed to control the pressure build up or to remove energy from the containment in the event of a DBA should be conducted to identify relevant possibilities for DECs.

3.43. Following conditions should be considered as generic candidates for design extension conditions relevant for the design of the containment structures and systems:

- Station black out;
- Loss of the means designed to controlling the pressure build up in the event of a design basis accident;
- Loss of the heat transfer chain to the ultimate heat sink removing heat from the containment in the event of a design basis accident;
- Loss of wet well / heat sink (BWR);
- Loss of the ultimate heat sink.

3.44. The following recommendations provide guidance to fulfill Requirements 20 and 68 of [2] for design extension condition with core melting.

3.45. A set of the most likely representative conditions in case of an accident with core melting should be considered to provide inputs to the design of the containment and of the safety features necessary to mitigate the consequences of an accident with core melting. Conditions with core melting, retained as boundary conditions for the design of the containment structures and for the associated systems, should be justified on the basis of PSA level 2 analyses supplemented by engineering judgement in order to allow the selection of appropriate conditions that are more probable and representative ones.

3.46. For establishing the design extension conditions with core melting that need to be considered as bounding condition in the design of safety features for DEC, the following aspects affecting accident progression that will influence containment response and fission product source term should be taken into account:

- Containment status (containment open or by passed);
- Radioactive material initially released to the containment;
- Pressure at onset of core damage;

- Timing of core damage: early emergency core cooling system failure (injection phase) vs. long term cooling failure;
- Status of containment safety features (spray, fan coolers, suppression pool);
- Status of AC or DC power sources;
- Status of instrument air systems;
- Status of the spent fuel pool systems if they are in the containment.

3.47. Design provisions should be implemented to prevent a containment failure in case of DEC. These provisions should aim to prevent a significant over pressurization of the containment structure, to stabilize the molten core, to remove the heat from the containment and to avoid fast deflagration and detonation of combustible gases.

3.48. Different 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 the last resort mean.

3.49. For DECs for which venting the containment would be necessary to preserve the integrity of the containment, its use should not lead to an early or a large radioactive release (see [2], Requirement 6.28A). In that case:

- The containment venting system should be equipped with filters of adequate capacity and high efficacy;
- The containment venting system should be designed to withstand loads from external hazards and the static and dynamic pressure loads existing when the containment venting system is operated;
- It should be possible to reliably open and close the vent line valves;
- Provisions should be taken to prevent from exceeding the design limit with regard to sub atmospheric containment pressure.

DESIGN LIMITS

3.50. The following recommendations provide guidance to fulfill Requirements 15 and 28 of [2].

3.51. The performance of the containment and of the systems should be assessed against a well-defined and accepted[2] set of design limits and criteria.

[2] 'Well defined and accepted' generally means either widely accepted by Member State regulatory bodies or proposed by international organizations.

3.52. A set of primary design limits for the containment and for the systems should be established ensuring the achievement of the overall safety functions of the containment. These primary design limits are usually expressed in terms of:

- Overall containment leak rate at design pressure;
- Unfiltered leakages;
- Limits on radioactive releases, dose limits or dose constraints for the public, specified for operational states and accident conditions;
- Dose limits or dose rate limits, and dose constraints for personnel specified for the biological shielding function.

3.53. Furthermore, design limits should be specified for each containment structure and component. Limits should be applied to operating parameters (e.g. maximum coolant temperature and minimum flow rate for air coolers) and performance indicators (e.g. maximum closing time for isolation valves and penetration air leakage).

RELIABILITY

3.54. The following recommendations provide guidance to fulfill Requirements 21, 22, 23, 24, 25, 26, 29 and 30 of [2].

3.55. The following factors should be considered to achieve the adequate reliability of the systems necessary to control energy, radioactive materials and combustible gases released inside the containment:

- Safety classification and the associated engineered requirements for design and manufacturing;
- Design criteria relevant for the systems (number of redundant trains, seismic qualification, qualification to harsh environmental conditions, power supplies,);
- Consideration of vulnerabilities for common cause failures (diversity, separation, independence);
- Layout provisions to protect the system against the effects of internal and external hazards;
- Periodic testing and inspection;
- Maintenance;
- Use of equipment designed to fail-in a safe direction.

Systems designed to mitigate design basis accidents

3.56. Energy management and control of radionuclides in the event of design basis accidents should be possible despite the consequential failures caused by the postulated initiating event and a single failure postulated in any system needed to accomplish the function. Unavailability for maintenance or repair should be considered in addition.

3.57. The emergency power source should have adequate capability to supply power to electrical equipment necessary for the energy management and control of radionuclides in the event of design basis accidents.

3.58. Vulnerabilities for common cause failures between the redundancies of the safety systems should be identified, and design or layout provisions be implemented to make the redundancies independent to the extent practical.

3.59. Recommendations related to the reliability of the system with regard to the effects of internal or external hazards and environmental conditions are addressed in paragraphs “Internal hazards”, “External hazards” and “Environmental conditions”, respectively.

Safety features for design extension conditions without significant fuel degradation

3.60. A reliability analysis of the safety systems designed for the energy management should be conducted to identify needs for additional safety features to preserve the containment integrity.

3.61. The more likely combinations of PIEs and common cause failure (CCF) between the redundancies of the safety systems should be analyzed. If consequences exceed the limits given for DBAs, the vulnerabilities for CCF should be removed or additional design features should be implemented to cope with such situations. The additional features for the energy management should be designed and installed such that they should be unlikely to fail for the same common cause.

3.62. Needs for additional safety features are reactor technology and design dependent.

3.63. Additional safety features should have an adequate reliability to contribute to the practical elimination of conditions that could lead to an early radioactive release or to a large radioactive release.

3.64. Similar recommendations as indicated for systems designed to mitigate the consequences of design basis accidents should be applied, taking into account that meeting the single failure criterion is not required and that the relevant additional features are expected to be unlikely to fail for the same CCF leading to the failure of the systems designed for design basis accidents.

3.65. Additional safety features for such DECs should be supplied from a different and diverse power source (e.g. by the alternate power source installed at the unit).

Safety features implemented to mitigate the consequences of an accident with core melting

3.66. The following recommendations provide guidance to fulfill Requirement 20 of [2].

3.67. Components necessary to mitigate the consequences of an accident with core melting should be capable of being supplied by any of the available power sources.

3.68. Recommendations related to the reliability of the system with regard to the effects of internal or external hazards and environmental conditions are addressed in paragraphs “Internal hazards”, “External hazards” and “Environmental conditions”, respectively.

DEFENSE IN DEPTH

3.69. The following recommendations provide guidance to fulfill Requirement 7 of [2].

3.70. Different systems should be implemented for the energy management (for pressure and temperature control, and for containment heat removal) in the different plant states.

3.71. Following recommendations contribute to implement independence:

- Successive items, belonging to different levels of defense, necessary to control the pressure inside the containment or to remove energy from the containment should be identified;
- Vulnerabilities for CCF between those items should be identified and the consequences assessed. The vulnerabilities for CCF should be removed to the extent possible where the consequences for the integrity of the containment structure and for radioactive releases are judged not acceptable. In particular, dedicated safety features designed to mitigate the consequences of accidents with core melting should be independent from equipment designed to mitigate the conditions inside the containment caused by design basis accidents;
- Independence implemented between systems should not be compromised by vulnerabilities for CCF in I&C systems necessary for the safety actuation of the systems or the monitoring of the containment conditions (see paragraph “Instrumentation” for more recommendations for I&C systems and Instrumentation).

PRACTICAL ELIMINATION OF CONDITIONS THAT COULD LEAD TO AN EARLY RADIOACTIVE RELEASE OR A LARGE RADIOACTIVE RELEASE

3.72. Conditions arising that could lead to an early radioactive release or a large radioactive release are required to be practically eliminated by design (see [2], Requirement 5.31).

3.73. Regarding the scope of this Safety Guide, such possibilities should include:

- Conditions involving high energetic phenomena the consequences of which could not be mitigated with implementation of reasonable technical means,
- Core melt accident combined with a containment bypass.

3.74. Typical examples of conditions to be practically eliminated are:

- Severe accident conditions that could damage the containment in an early phase as a result of a direct containment heating, steam explosion or hydrogen detonation;
- Severe accident conditions that could damage the containment in a late phase as a result of a basemat melt-through[3];
- Severe accident conditions with an open containment, notably in shutdown modes;
- Severe accident conditions with unintentional containment bypass.

3.75. Core melting accidents should be postulated as Design Extension Conditions regardless design provisions taken to prevent such conditions and of the estimate of their probability to occur.

SAFETY CLASSIFICATION

3.76. The following recommendations provide guidance to fulfill Requirement 22 of [2].

3.77. Consequences of the failure of a SSC should be considered both on the accomplishment of the function, and on the level of the radioactive release. For items to which both effects are relevant, the safety class and the associated quality requirements needed to achieve the expected reliability are defined with due account taken of those two effects. For items which do not contain radioactive material the safety class and the quality requirements are directly derived from the consequences assuming the function is not accomplished.

3.78. Engineering requirements applicable to a whole system (e.g. single failure criterion, physical and electrical separation, functional independence, emergency power supply, periodic tests, etc.)

[3] These conditions should be analyzed during the identification of situations to practically eliminate. Nevertheless, their consequences could generally be mitigated with implementation of reasonable technical means.

should be derived from the safety class assigned to the system, primarily assuming the function would not be accomplished.

3.79. The classification should be established in a consistent manner such that all systems (front line system and the associated service support systems) necessary for the accomplishment of one function are assigned to the same class.

3.80. More detailed guidance is given in the Safety Guide SSG-30 [4] **[4**.

3.81. Safety classified pressure retaining equipment should be designed and manufactured according to requirements established by proven codes and standards widely used by the nuclear industry. For each individual component, the requirements to be applied should be selected with due account taken of the two effects resulting from its failure (function not accomplished and radioactive release).[4]

3.82. Following the above recommendations:

- In the event of a design basis accident, systems necessary for the containment isolation, for the control of the pressure build up inside the containment (e.g. containment spray system), or to remove heat from the containment and transport heat from the containment to the ultimate heat sink should be assigned in SSG-30 safety class 1;
- Systems implemented as a back-up of SSG-30 safety class 1systems for design extension conditions should be assigned at least in SSG-30 safety class 2;
- Systems necessary to preserve the containment integrity in the event of an accident with core melting should be assigned at least in SSG-30 safety class 3 (e.g. ex-vessel core cooling/corium cooling system, reactor coolant system depressurization system, containment spray system, venting and filtering system, systems to prevent hydrogen detonation, heat transport chain);
- The containment structure designed as the last physical barrier against releases should be SSG-30 safety class 1.

[4] However, according to international practices, the pressure retaining boundary of components which are part of the reactor coolant pressure boundary should be designed and manufactured in compliance with the highest standards defined by the industry for nuclear application[4] (e.g. ASME Section III, Division 1, subsection NB [5], RCC-M1 code [6], JSME SNC1 [20], or similar standards), except parts of the reactor coolant pressure boundary whose failure would result in leakage that is compensable by the normal water make-up system

ENVIRONMENTAL QUALIFICATION

3.83. The following recommendations provide guidance to fulfill Requirement 30 of [2].

3.84. The structures, systems and components of the containment should be qualified to perform their functions in the entire range of environmental conditions that might prevail prior to or during their operation until their mission time be completed, or should otherwise be adequately protected from those environmental conditions.

3.85. The relevant environmental and seismic conditions that may prevail prior to, during and following an accident, the ageing of structures, systems and components throughout the lifetime of the plant, synergistic effects, and margins should all be taken into consideration in the environmental qualification [7], [11**[11**].

3.86. Environmental qualification should be carried out by means of testing, analysis and the use of engineering judgement, or by a combination of these.

3.87. Environmental qualification should include the consideration of such factors as temperature, pressure, humidity, radiation levels, and local accumulation of radioactive aerosols, vibration, water spray, steam impingement, flooding and contact with chemicals. Margins and synergistic effects (in which the damage due to the superposition or combination of effects may exceed the total damage due to the effects separately) should also be considered. In cases where synergistic effects are possible, materials should be qualified for the most severe effect, or the most severe combination or sequence of effects.

3.88. Techniques to accelerate the testing for ageing and qualification may be used, provided that there is an adequate justification.

3.89. For components subject to the effects of ageing by various mechanisms, a design life and, if necessary, the replacement frequency should be established. In the qualification process for such components, samples should be aged to simulate the end of their design lives before being tested under design basis accident conditions.

3.90. Components that have been used for qualification testing should generally not be used unless it can be shown that the conditions and methods of testing do not themselves lead to an unacceptable degradation of safety performance.

3.91. Qualification data and results should be documented as part of the design documentation.

CODES AND STANDARDS

3.92. The following recommendations provide guidance to fulfill the requirement 4.15 of Requirement 9 of [2].

3.93. For the design of the structures and systems of the containment, national or international codes and standards could be used provided that applicability and suitability of the code and standard is demonstrated. The selected codes and standards:

- Should be applicable to the particular concept of the design;
- Should form an integrated and comprehensive set of standards and criteria;
- For design and construction the newest edition of the codes/standards should be preferably considered. However, another edition might be used with justification.

3.94. Codes and standards have been developed by various national and inter- national organizations, covering areas such as:

- Quality assurance;
- Materials;
- Structural design of pressurized components;
- Civil structures;
- Instrumentation and control;
- Environmental and seismic qualification;
- Pre-service and in-service inspection and testing;
- Fire protection.

USE OF PROBABILISTIC ANALYSES IN DESIGN

3.95. Probabilistic analyses should complement the deterministic approach by identifying additional features (to achieve a balanced design). The use of probabilistic analyses should not be considered as a substitute to a design approach based on deterministic requirements, but as a part of the process to identify potential safety enhancements and to judge their effectiveness.

3.96. Probabilistic safety analysis should be used taking into accounts its limitations in support of demonstrating the practical elimination of conditions that could lead to an early radioactive release or a large radioactive release. In particular PSA may be used for analyzing the containment isolation provisions, for preventing containment by-pass and the total failure of the energy management systems.

3.97. As a complement to a number of investigations related to fabrication, testing, inspection, evaluation of the operating experience, PSA should be used to confirm the very low probability of the failure of the means implemented for an appropriate mitigation of the design extension conditions with core melting. This should include inter alia the analysis of the reliability of containment systems, e.g. containment cooling system, containment filtered venting, etc. and other aspects that have traditionally been considered in level 2 PSA.

4. DESIGN OF CONTAINMENT STRUCTURE AND ASSOCIATED SYSTEMS

GENERAL

4.1. A number of systems are design dependent and may be different in their design principles (e.g. use of active or passive systems for the energy management, in-vessel or ex-vessel core cooling in the event of an accident with core melting, etc.). Nevertheless, structures or systems having to accomplish the same safety function in different technologies should be designed in compliance with the same design requirements.

4.2. Regardless of permanent design provisions for DBAs and for DECs, features enabling the safe use of non-permanent equipment for restoring the capability to remove heat from and to depressurize the containment should be included (see [2], Requirement 6.28B).

Layout and configuration of containment systems

4.3. The layout and configuration of the containment structure and systems is design dependent and is significantly different for reactor technologies relying on containment with a large dry volume or on containment with a suppression pool.

4.4. The layout of the containment should be defined with account taken of several factors that are dealt with in this Safety Guide and that are summarized below:

- The layout and configuration should be such to facilitate the enrergy management (se recommendations in paragraph “Energy management”;
- Provision of adequate separation between divisions of safety systems and between redundant safety features for DEC where relevant;
- Location and provisions to protect items important to safety against the effects of internal hazards;

- Provision for sufficient space and shielding to ensure that maintenance and operations can be carried out without causing undue radiation exposure of personnel;
- Provision of the necessary space for personnel access and the monitoring, testing, control, maintenance and movement of equipment;
- Placement of the equipment and structures so as to optimize biological shielding;
- Optimization of the number of and location of containment penetrations so as to prevent unfiltered leaks and to ensure accessibility for inspection and testing;
- Allow the replacement of equipment that maybe foreseen during the plant life time;
- Minimization of water retention to facilitate water and condensates flowing back to the containment sumps;
- The lower part of the containment should be designed to facilitate the collection and identification of liquids leaked.

Maintenance and Accessibility

4.5. The following recommendations provide guidance to fulfill the relevant requirements of Requirements 6, 32, 81 and the requirement 5.15 of [2]. Recommendations [17] to prevent non authorized persons from accessing the containment and the buildings that housed the systems important to safety should also be implemented in an integrated manner with the recommendations for safety.

4.6. The design should take into account the potential exposure to radiation associated with operations such as maintenance or repair that would be needed to be conducted after an accident, or with operations that it may be necessary to conduct following the emergency procedures as well as with the recovery actions following an accident. Evaluations should include the consideration of access paths. If the doses due to such exposures exceed the applicable dose limits additional shielding or even the repositioning of components should be considered. The use of other connecting points should also be considered [22].

4.7. Maintenance related factors considered in the containment design should include the provision of adequate working space, shielding, lighting, air for breathing, and working and access platforms; the provision and control of environmental conditions; the identification of equipment; the provision of hazard signs; the provision of visual and audio alarms; and the provision of communication systems.

4.8. The accessibility of both the containment and the systems should be considered for all operational states. The ability to ensure that radiation doses to operators remain within the acceptable dose constraints will determine whether access can be allowed to the primary and/or the secondary containment (if applicable) during power operation, or whether plant shutdown is required for permitting such access.

4.9. If entry into the containment during power operation for the purposes of unplanned maintenance or even for routine (planned) maintenance is envisaged, provision should be made to ensure the necessary radiological protection and adequate working conditions of plant staff. This provision should include the application of the principle of keeping exposure as low as reasonably achievable, the provision of the necessary communication systems and alarms, and monitoring of the containment atmosphere, especially in the case of inert containments or containments at sub atmospheric pressure.

4.10. At least one emergency escape route from the containment should be provided that has the ability to be used while maintaining the integrity of the containment. In addition, security provisions for controlling access to the containment should be considered.

Operator actions

4.11. In the event of an accident, there should be no need for any action to be taken by the operator within a certain grace period. For necessary manual intervention, the operator should have adequate information available as well as sufficient time to diagnose and assess the conditions in the plant before taking any manual action.

Sharing of parts of the containment system between units

4.12. The following recommendation provides guidance to fulfill Requirement 33 of [2].

4.13. Each unit of a multiple unit nuclear power plant should have its own safety systems and its own safety features for design extension conditions.

4.14. Means allowing interconnections between units should be installed to facilitate the management of accidents not considered in the design of the NPP (e.g. connections to refill containment water storage tanks.

Ageing effects

4.15. The following recommendation provides guidance to fulfill Requirement 31 of [2].

4.16. The containment may be subject to several ageing phenomena such as the corrosion of metallic components, the creep of tendons and the reduction of pre-stressing (in pre-stressed containment), the reduction of resilience in elastomeric seals, the shrinkage and cracking of concrete and carbonization of concrete. Ageing mechanisms should be identified, taken into account in the design and incorporated into an ageing management programme. Provision should be made for controlling the ageing of the containment, for testing and inspection of components where possible, and for periodically replacing items whose safety characteristics are susceptible to age-related degradation. More detailed guidance is provided in [7].

Decommissioning

4.17. Provisions should be taken to facilitat. the decommissioning and dismantling of equipment, and to minimize the production of contaminated wastes (see [2], Requirement 12). Guidance on these aspects is given in [8]. **[8**

STRUCTURAL DESIGN OF CONTAINMENT

General design process

4.18. Pressures and temperatures which can be reached at all the different plant states are fundamental parameters used for the design of the containment and the associated systems. Those parameters result from assessing the containment conditions for each of them taking into account the relevant methodologies and rules.

4.19. The design pressure should be higher than the value of the peak pressure that would be generated by the design basis accident with the most severe release of mass and energy (DBA peak pressure + margin).

4.20. The design temperature should be defined as the value of the highest temperature that would be generated by the design basis accident with the most severe release of mass and energy calculated with conservatism.

4.21. Pressure and temperature used in the load combinations should be determined with adequate margins to avoid cliff edge effects and to take into account:

- Uncertainties in the amounts of fluids released and in the release rates in terms of both mass and energy, including chemical energy from metal– water reactions;
- Structural tolerances;
- Uncertainties in relation to the decay heat;
- The heat stored in components;
- The heat transferred in heat exchangers;
- Uncertainties in the correlations of heat transfer rates;
- Conservative initial conditions.

4.22. Design for a specific maximum leak rate is not a straightforward or purely quantitative process. A number of factors should be taken into account, including the limitation of stresses in accident conditions, the adequate choice of components (e.g. isolation valves), the adequate choice of sealing materials, limitation of the number of containment penetrations and control of the construction quality. Existing operational data, experience and practices should be used to the extent possible.

4.23. The mechanical behavior (stresses and deformations) of the containment structure, as tentatively determined on the basis of the design pressure and the design temperature, should be verified for all load combinations and should comply with the corresponding engineering criteria for the integrity and leak tightness of the containment.

4.24. The mechanical resistance of the containment and of the associated systems should be assessed in relation to the expected range of events and their anticipated probability over the plant lifetime, including the effects of periodic tests.

4.25. In steel containments the load bearing and leak tightness functions are fulfilled by the steel structure. The metallic structure should be protected against fires and missiles generated inside and outside the containment as a result of internal and external hazards that affect the plant.

Loads and load combinations

4.26. Loads (static and dynamic) that are foreseen to occur should be quantified and grouped according to their probability of occurrence, on the basis of operating experience and engineering judgment.

4.27. Loads should be combined with account taken of:

- Load type (i.e. static or dynamic, global or local);
- Whether loads are consequential or simultaneous (e.g. LOCA pressure and temperature loads);
- Physical barriers that protect equipment from hazard effects;
- Time history of each load (to avoid the unrealistic superposition of load peaks if they cannot occur coincidently).

4.28. At the end of the analysis the number of load combinations may be reduced by grouping them appropriately. The analysis will be performed only for the most demanding cases.

4.29. The steel liner of the containment (where applicable) should be able to withstand the effects of imposed loads and to accommodate relative movements of the liner and the concrete of the containment without jeopardizing its leak-tightness. The liner should not be credited in the structural evaluation for the resistance of the containment.

4.30. Additional pressure load on the concrete containment due to instantaneous temperature rise of the liner during an accident condition should be considered.

4.31. The metallic liner, structures, penetrations and isolation valves of the containment should be protected against the effects of the internal hazards or if not should be designed to withstand the loads.

4.32. For containment design with double walls, the pressurization of the space between the two containment walls caused by a high energy piping break should be considered, unless such a break is precluded by the design.

TABLE 1. TYPICAL SET OF LOADS ON THE CONTAINMENT TO BE CONSIDERED AT THE DESIGN STAGE

Load category	Load	Remarks
Pre-service loads	Dead	Loads associated with the masses of structures or components including the effects of shrinkage and creep (for concrete containment)
	Live	Loads associated for example with component restraints
	Pre-stressing, creep effects	For pre-stressed concrete structures only
	Loads in construction, maintenance	Temporary loads due to construction equipment or the storage of major components
	Test pressure	See Section 5
	Test temperature	See Section 5
Normal or service loads	Actuation of safety relief valve	Boiling water reactors only
	Lifting of relief valve	Boiling water reactors only
	Air cleaning of safety relief valve	Boiling water reactors only
	Operating pressure	In normal operation, including transient conditions and shutdown
	Operating temperature	In normal operation, including transient conditions and shutdown
	Pipe reactions	In normal operation, including transient conditions and shutdown
	Wind	Maximum wind speed assumed to occur over plant operating lifetime see [9]
	Environmental and site related loads	For example, snow load, buoyant forces due to the water table and extremes in atmospheric temperature
	External pressure	Loads resulting from pressure variations both inside and outside the primary containment
	Extreme wind speeds	Loads generated by extreme wind speeds, i.e. maximum wind speed that may be associated with the site
Loads due to extreme external events	Design basis earthquake	See [11]
	Loads associated with extreme wind speeds	Associated missiles to be considered
	Aircraft crash	See [9]
	External explosion	See [9]
Loads due to accidents	DBA pressure	Calculated peak pressure in DBA
	DBA temperature	Calculated peak temperature in DBA
	Design pressure	DBA pressure + margins
	Design temperature	DBA temperature + margins (to be applied as a uniform value)
	DBA pipe reactions	See [3]
	Jet impingement and/or pipe whip	See [3]
	Local effects consequential to a DBA	See [3]
	Dynamic loads associated with a DBA	Loads are design dependent (e.g. for a boiling water reactor design: discharge line clearing loads, pool swell, condensation oscillation and discharge line ‘chugging’)
	DEC pressure	Calculated peak pressure in the most penalizing condition (peak and time dependent profile)
	DEC temperature	Calculated peak temperature in the most penalizing condition(peak and time dependent profile)

Load category	Load	Remarks
	Actuation of the depressurization system	Depressurization of the primary circuit (where applicable)
	Internal flooding	See [3]

Engineering criteria

4.33. Engineering criteria for leak-tightness and integrity of the containment and appurtenances (penetrations, isolation systems, doors and hatches), as proposed in 4.35 and 4.36 should be established on the basis of stress and deformation limits for different load combinations. Meeting the criteria given by codes and standards internationally recognized provides reasonable assurance that structures and components are capable of performing their intended functions.

4.34. It should be demonstrated in the analysis that the engineering criteria for structural integrity and leak tightness are met with an adequate margin to cover uncertainties and to avoid 'cliff edge' effects. Margins should generally be provided by the used methodologies to calculating the design basis and design extension conditions, and the use of proven codes for limiting stresses in the structures.

4.35. In any case, design limits should be defined according to the expected performance (see paragraph "Design Limits"). Design margins should be provided by either or both:

- Limiting stresses and deformations to some fraction of the ultimate limit for that material; or
- Use of the load factor approach (i.e. increasing the applied loads by a certain factor).

4.36. For the design of the structural integrity of the containment, the following levels should be considered:

- Level I: elastic range. No permanent deformation of, or damage to, the containment structure occurs. Structural integrity is ensured with large margins;
- Level II: small permanent deformations. Local permanent deformations are possible. Structural integrity is ensured, although with margins smaller than those for Level I.

4.37. For the design of the leak tightness, the following levels should be considered:

- Level I: leak tight structure. Leakages from the containment are below the design value[5] and can be correlated with the internal pressure on the basis of analysis, experience and test results;
- Level II: possible limited increase of leak rate. The leak rate may exceed the design value, but the leak tightness can be adequately estimated on the basis of analysis, experience and test results.

4.38. The detailed load combinations are design dependent. Table 2 presents a minimum set of recommended load combinations and engineering criteria for a typical containment of pressurized water reactor.

[5] In this context 'design value' has the meaning of the value of the leakage rate established as a target of the design and used in the safety analysis to determine the radiological releases under design pressure and design temperature.

TABLE 2. LOAD COMBINATIONS AND ENGINEERING CRITERIA

Load description	Design	Test	Normal operation	Normal operating plus extreme wind speed	SL-2 Earthquake	External pressure	DBA	SL-2 plus DBA	Aircraft crash	Fire	External explosion	DEC wo significant fuel damage	DEC w core melting
Dead	x	x	x	x	x	x	x	x	x	x	x	x	x
Live	x	x	x	x	x	x	x	x	x	x	x	x	x
Pre-stressing (if applicable)	x	x	x	x	x	x	x	x	x	x	x	x	x
Test pressure		x											
Test temperature		x											
Sustained loads		x	x	x	x	x	x	x	x	x	x	x	x
Operating loads			x	x	x	x	x	x	x	x	x	x	x
Operating temperature			x	x	x	x			x		x		
Pipe reactions			x	x	x	x			x	x	x		
Extreme wind				x									
External pressure						x							
SL-2[6] earthquake					x			x					
Design pressure	x												
Design temperature	x												
DBA[7] pressure							x	x					
DBA temperature							x	x					
DBA pipe reactions							x	x					
Aircraft crash									x				
Fire										x			
External explosion											x		
DEC w core melting pressure													x
DEC w core melting (temperature)													x
DEC wo significant fuel damage (pressure)												x	

[6] SL-2 : Level of ground motion associated to the maximum earthquake to be considered for design often denoted as the safe shutdown earthquake.

[7] DBA : design basis accident

Load description	Design	Test	Normal operation	Normal operating plus extreme wind speed	SL-2 Earthquake	External pressure	DBA	SL-2 plus DBA	Aircraft crash	Fire	External explosion	DEC wo significant fuel damage	DEC w core melting
DEC wo significant fuel damage (temperature)												x	
Engineering criteria for steel containment:													
Structural integrity	I	I	I	I	II	II	I	II	N/A	II	N/A	II	II
Leak tightness	I	I	I	I	N/A	II	I	II	N/A	II	N/A	I	II
Engineering criteria for the pre-stressed containment													
Structural integrity	I	I	I	I	II	N/A	I	II	II	II	II	II	II
Leak tightness	I	I	I	I	N/A	N/A	I	N/A	N/A	N/A	N/A	I	II
Engineering criteria for a liner on pre-stressed concrete wall	I	I	I	I	II	N/A	I	N/A	N/A	II	N/A	II	II

4.39. To provide margins, loads resulting from earthquake level SL2 and design basis accidents should be combined using an adequate statistical combination of the loads although one cannot realistically be a consequence of the other since the pressure boundary is designed to withstand seismic loads caused by earthquake level SL2 [11].

Local stresses and fatigue

4.40. Localized stress, including those at welding regions, near supports and regions with changing geometry, and their effects on the mechanical performance of structures, including leak rates, should be evaluated.

4.41. For pre-stressed concrete containments, particular attention should be paid to identifying areas of low pre-stressing (such as areas surrounding large penetrations and transition zones between cylinder and basemat), to concentrate stresses near penetrations and near anchorages of the tendons and to tensioning sequence during the construction.

4.42. For containments provided with a metallic liner, the zones of anchorage of the liner to the concrete and the connections of the liner to other metallic structures such as penetrations are also critical areas. Local effects of stress in these zones should be analyzed and taken into account.

Ultimate capability and failure mode

4.43. To determine the ultimate load bearing capacity and confinement capacity, it should be considered to make a global evaluation of the structural behavior of the containment under static (pressure, temperature and actions of pipes) and dynamic (seismic) loads and to identify the most limiting parts so as to evaluate margins, and to study the failure mode of the structure. Local effects, thermal gradients and details of design should also be considered so as to identify possible mechanisms for large leaks. In this regard, special attention should be paid to the behavior of piping penetrations, soft sealing materials, electrical penetrations and structural singularities.

4.44. Failure modes such as liner tearing, penetration failures, rebar failure, local concrete failure and tendon failures should be analyzed. To the extent possible, a failure should not be catastrophic and should not cause additional damage to systems and components for retaining radioactive material.

STRUCTURAL DESIGN OF STRUCTURES WITHIN THE CONTAINMENT

4.45. Consideration should be given to the possibility of large releases of mass and energy inside the containment, and the need for the internal structures to withstand the pressure differentials that could arise between different compartments so as to prevent any collapse. For each compartment, the most unfavorable location for a break should be considered. Openings between compartments should be considered by means of a conservative approach at the design stage and should be verified to be free of unintended obstructions after construction has been completed.

4.46. Consideration should be given to the need for the internal structures to withstand the loadings associated with accident conditions, and so to withstand the dynamic loads that are caused by high energy discharges or pipe breaks, water flowing from the discharge line of the safety valves and the relief valves into the suppression pool, the swelling of the pool water, the oscillation of condensate water, chugging and any other relevant hydraulic phenomena.

4.47. The loads on the structures inside the containment, in case of design extension conditions, depend on the strategy to cope with the molten core adopted in the specific design.

4.48. Acceptance criteria for leak-tightness and integrity given by Table 3 should be met in the event of accident conditions with significant core degradation, and conditions for a basemat melt through should be practically eliminated for both of the design options retained for the core molten retention (In Vessel Retention or Ex Vessel Retention).

In-vessel retention strategy

4.49. In this strategy, the heat from the molten core is removed through the wall of the reactor pressure vessel. This requires the reactor cavity to be flooded to enable external cooling of the reactor pressure vessel. Mechanical and thermal loads in the walls of the cavity should be considered. Features should be included to remove the heat from the cavity and to avoid pressurization of the cavity and the containment.

4.50. Considering 4.47, the structures of the cavity should be considered as items ultimately necessary to avoid large releases and consequently they should be such that design margins are adequate to deal with seismic loads exceeding SL-2.

Ex-vessel retention strategy

4.51. In this strategy, the containment should be equipped with an ex-vessel retention structure dedicated to contain and cool the molten core outside of the reactor pressure vessel.

4.52. The ex-vessel retention structure should be designed to minimize the production of combustible gases due to molten corium concrete inter-action.

4.53. The structures and materials used for the design of the ex-vessel retention structure should be appropriate to withstand the different loads and effects caused by the ingress of the corium in the different elements of ex-vessel retention structure.

4.54. The structures and the cooling system of the ex-vessel retention structure should be appropriate and designed for stabilizing and confining the corium inside.

4.55. The ex-vessel retention structure should be considered as items ultimately necessary to avoid large releases and consequently, it should be such that design margins are adequate to deal with seismic loads exceeding SL-2.

STRUCTURAL DESIGN OF SYSTEMS

4.56. For containment systems, a set of representative loads and load combinations, as well as a set of adequate engineering criteria, should be established by a similar procedure as for the containment structures, with account taken of all the relevant accident conditions.

ENERGY MANAGEMENT

4.57. Energy management is a term used to describe the management of those design features of the containment that affect the energy balance within the containment and thereby play a part in maintaining pressure and temperature within acceptable limits.

Control of pressure and temperature during plant operation

4.58. During normal plant operation, a ventilation system should be operated to maintain the pressure and temperature in the containment within the limits specified for normal operation. More detailed recommendations are given in [23].

Control of pressure and temperature in accident conditions

4.59. The design performance of the systems for energy management should be established so as to be able in the event of an accident to control pressure and temperature within the specified limits and to reach a stable state, with the containment depressurized, within a reasonable period of time (typically a few days) after its onset.

4.60. Strategies for the pressure and temperature control in accident conditions rely on the use of inherent safety features, active or passive safety systems or safety features, or a combination of them. The design of those structures, systems and components should be compliant with the requirements relevant for the plant state category for which they are required to operate. Typical design options are described underneath.

Inherent energy management features (containment with a large dry space)

4.61. The free volume of the containment envelope[8] is the primary physical parameter determining peak pressures after postulated pipe rupture events. It can thus be used as an inherent and safe design

[8] The containment envelope includes all components of the reactor coolant pressure boundary, and those connected to the reactor coolant pressure boundary, that cannot be isolated from the reactor core in the event of an accident.

feature to accommodate with a large release of energy and mass inside the containment. If the volume of the containment is subdivided into compartments that are provided with collapsing panels or louvres that open in the event of energy released, these collapsing panels or louvres should be designed to open quickly at the predetermined pressure so as to achieve fast equalization of the pressures in the various compartments and to utilize the full free volume of the containment.

4.62. The containment structure and its internal structures, as well as the water stored within the containment, act as a passive heat sink. In the postulated conditions of a pipe rupture accident, the heat transfer rate and heat capacity of structures and components of structures are important parameters in determining pressures and temperatures. The primary mechanism for heat transfer is the condensation of steam on exposed surfaces, and the thermal conductivity of the structure plays an important part in determining the rate of heat transfer. All conditions that could affect the transfer of heat to the structures, such as the effects of coatings or gaps, should be considered in a conservative manner in the design, and adequate margins should be applied.

Spray systems

4.63. With regard to energy management, a pray system should be designed to:

- Limit the peak pressure and the time duration of the high pressure inside the containment in accident conditions, for containment with a large dry space;
- Limit the time duration of the high pressure inside the drywell and wetwell for containment with a supression pool system;
- Control of the temperature inside the drywell for containment with a supression pool system.

4.64. For containment with a large dry space, a spray system should be designed so that a major fraction of the free volume of the containment envelope into which the steam may escape in an accident, can be sprayed with water.

4.65. The spray headers and nozzles should be designed to provide an even distribution of water droplets, which should be small enough to reach thermal equilibrium with the atmosphere quickly during their fall.

4.66. The initial source of water for the containment spray system after a pipe rupture is usually a large storage tank or the suppression pool. Later the spray system may operate in a recirculation mode and take water from appropriate collection points in the containment sump or the suppression pool.

4.67. For a spray system designed to operate in a recirculation mode, the spray nozzles should be designed against clogging by the largest postulated pieces of debris that can reach them through the intake screens.

4.68. The layout of the containment with a large dry space should be such as to ensure an adequate single free volume in the upper part of the containment to improve the efficiency of the containment spray.

Pressure suppression pool systems

4.69. Containments of a design with a suppression pool system are divided into two separate compartments called the drywell and the wetwell. The two compartments are normally isolated from one another. When the pressure in the drywell is sufficiently higher than the pressure in the wetwell, steam and gases flow from the drywell to the wetwell and the steam condenses into the pool of water. In some designs, communication between the drywell and the wetwell can also occur if the pressure in the wetwell is higher than the pressure in the drywell. In the containments of some designs the suppression pools are also used to collect the steam discharged from the safety valves or the relief valves, or to provide water for recirculation in the emergency core cooling system, decay heat removal system and containment spray system. Complex hydraulic and pressure transients occur when steam and gases are vented into the suppression pool water.

4.70. With regard to energy management, the suppression pool should be designed such that the design pressure of both the drywell and the wetwell is not exceeded in case of a design basis accident:

- The vent flow area between the drywell and the suppression pool must be sized to limit the maximum pressure during blowdown;
- The amount of water in the suppression pool should be such to condensate steam released during design basis accidents (e.g. in the event of a LOCA);

4.71. The design of the drywell and wetwell and connection features should be such that the hydraulic responses and the dynamic loads can be reliably determined by analysis and tests.

4.72. The hydraulic response of and loads imposed to the pressure suppression pool in the different plant states should be determined and considered for design.

4.73. The structural design of the pressure suppression pool system should be such as to ensure that the pool, as well as the containment system as a whole and other safety systems, remains functional in all operational states and/or all postulated accident conditions.

4.74. The pressure suppression pool system should be designed in such a way that the pathway for steam and gases from the dry well to enter the wetwell following a postulated accident condition is through submerged vents in the wetwell water pool.

4.75. The leakage between the drywell and the wetwell that bypasses the submerged venting lines should be minimized and should be taken into account in the design.

4.76. The use of the water inventory in the pressure suppression pool system for other functions should not impair the performance of its main function of providing a means of controlling the pressure in the dry well in case of accident conditions.

4.77. The drywell should be designed to withstand, or should be protected from (e.g. by automatic vacuum breaker valves), excessive under pressure caused by steam condensation inside the drywell by operation of the spray system either on purpose or inadvertently.

Containment heat removal system

4.78. Containment heat removal systems should be designed to remove heat from the containment and to transfer heat to the atmosphere, or to sea/river.

4.79. Piping crossing the containment wall should be considered as an extension of the containment and should be subject to specifications for structural integrity and leak tightness comparable with those applied to containment structure itself.

Systems operating in a recirculation mode

4.80. A minimum and adequate net pump suction head should be available to the recirculation pumps under any accident conditions for which the operation of the pumps is necessary. The minimum net pump suction should be calculated taking into account the potential accumulation of debris on the surface of the strainer filters.

4.81. Devices at which suction takes place should be designed to minimize cavitation and to prevent the ingress of foreign material (such as thermal insulation), which could block or damage the recirculation system.

4.82. To avoid the clogging of sump screens or strainer filters, special care should be taken in the design of piping, component insulation and the intake sump screens or strainer filters themselves, and consideration should be given to the chemical effects as determined by the sump or suppression pool water chemistry and temperature, and to corrosion/erosion of some metallic components and their interaction with the debris. In addition, the material used inside the containment (thermal insulation material, paints, etc.) should be carefully considered. The design should also avoid certain combination of these materials which may worsen the issue of clogging at sump screens or strainer filters (see paragraph “Covering, thermal insulation and coating materials”).

4.83. Special consideration should be also given to the effects of debris by-passing the sump screens or strainer filters on the potential for blockage of flow channels in fuel assemblies.

4.84. Piping crossing the containment walls should be equipped with containment isolation devices and devices necessary to isolate leaks in the external recirculation loops to maintain a sufficient water inventory for cooling. Non-isolable leakage (e.g. between the containment penetration and the isolation valve) should be prevented by design (e.g. by means of the provision of a guard pipe).

Passive containment heat removal systems

4.85. For containment with a steel shell, heat released in the containment under accident conditions can be removed passively through the steel shell. A secondary and outside envelope is needed and is designed to remove heat by providing a natural circulation path for air (the chimney effect). Containment spray is implemented by spraying of the outside of the steel shell.

4.86. Heat can also be removed from the containment by the installation of a number of heat exchangers on the inner walls of the containment and transferred outside the containment by natural circulation to passive cooling condensers.

4.87. Where passive containment cooling is adopted, the following aspects should be considered:

- The area of the cooling surface should be sufficient to transfer the heat generated in the containment and to cool down the atmosphere and the structures inside the containment. The heat transfer coefficient should be conservatively determined.
- The necessary natural circulation within the containment and that to the outside heat sink should be ensured for all relevant plant states for which such passive transfer is necessary.
- The possibility for freezing outside conditions should be considered for normal as well as accident conditions.
- A thorough search should be conducted for possible harmful effects and failure modes, in order to achieve a high degree of confidence that the safety functions will be fulfilled.

CONTROL OF RADIONUCLIDES

4.88. Containment structure and systems should be designed to meet the objectives for preventing and limiting the radiological release specified for the different plant states as indicated in 2.4.

4.89. Compliance of the radiological release with the relevant limit should be demonstrated by crediting the provisions designed for the relevant plant state only. The demonstration should be conducted according to models and analysis rules applicable to the plant state category.

4.90. Design provisions necessary to minimize the doses and the radiological releases should consider that the source terms are specific to each plant state (in terms of magnitudes and physicochemical forms).

4.91. An assessment of potential radioactive releases from the containment should be made for the design basis accidents and design extension conditions in order to identify any potential weaknesses with regard to the leak tightness of the containment and to determine ways to eliminate them.

Containment source term

4.92. To design the overall containment performance and in particular the measures for radionuclide management, the amount and isotopic composition of the radionuclides postulated to be released from the containment (i.e. the source term) should be estimated for the various accidents to be considered.

4.93. For design basis accidents, this should be done by means of a conservative analysis of the expected behavior of the core and of the safety systems. Initial conditions for the relevant parameters (e.g. for the inventory of radionuclides in systems and for leak rates) should be the less favorable values authorized by the limits and conditions of the plant.

4.94. The anticipated evolution of the physicochemical forms of the radionuclides in the containment should be assessed, with account taken of the latest knowledge (e.g. it is known that certain paints enhance the production of organic iodine).

4.95. Once iodine is trapped in water pools inside the containment, it may volatilize again in the medium to long term if appropriate pH conditions are not maintained. Therefore, all conditions should be assessed that could change the pH of the water pools during an accident and, if necessary, provide the necessary means to keep the pH of the water pools alkaline.

Leak tightness of the containment

4.96. An effective way to restrict radioactive releases to the environment is to maintain the leak rate below conservative specified limits throughout the plant operating lifetime[9]. As a minimum, leak rates should be small enough to ensure that the relevant dose limits are not exceeded during normal operation or in accident conditions.

4.97. At the design stage, a target leak rate should be set that is well below the safety limit leak rate, i.e. well below the leak rate assumed in the assessment of possible radioactive releases arising from accidents. This margin should be established to reduce the likelihood that unforeseen modifications made at the stage of design or construction cause an actual leak rate to approach the safety limit leak rate.

---

[9] Examples of such limits that are applied in Member States are: 0.25–0.5% overall leakage of the contained mass of free gas and steam per day at design pressure for steel containments or concrete containments with a steel liner; 1.0–1.5% per day overall leakage for pre-stressed concrete containments without a steel liner.

4.98. To limit the number of leak paths, the number of penetrations should be optimized as indicated by the recommendation 4.4. The external extensions of the penetrations should be installed in a confined building, at least until the first isolation valve, in order to collect and filter any leaks before a radioactive release occurs.

4.99. Leak rates of isolation devices, air locks and penetrations should be specified with account taken of their importance to safety and the integral leak tightness of the containment.

4.100. A reliable design of and actuation for containment isolation system should be incorporated, as described in paragraph "Provisions for containment isolation of piping and ducting systems" to ensure the leak tightness of the containment in the event of an accident.

4.101. Containment structure and systems should be designed to limit leaks and to avoid, to the extent possible, the creation of unfiltered leak paths to the environment.

Secondary confinement

4.102. Secondary confinement is in some designs an arrangement, in which the primary containment is completely or partially enclosed within a secondary envelope. The purpose of the secondary envelope in such designs is not to take over the functions of the primary containment should it fail but to allow for the potential collection of leaks from the primary containment and for a filtered release via the vent stack. When such a design option is implemented, the secondary confinement structure is also often designed as the shielding structure of the containment.

4.103. When a secondary confinement is provided, direct leaks (i.e. leak paths from the containment directly to the outside) should be prevented to the extent possible.

4.104. When employing a partial secondary confinement (i.e. one which does not completely enclose the primary containment), the envelope should enclose the more leakage prone areas of the primary containment (such as the penetration areas).

4.105. Criteria should be set for the control of direct leaks and for the leak tightness of the secondary confinement envelope. It should be verified periodically by means of testing that these criteria are being met.

4.106. Systems associated with the secondary confinement should be designed to collect, filter and discharge gases and liquids containing radionuclides that have leaked from the containment in accident conditions, or to pump leaked liquids back into the containment.

4.107. To maximize the efficiency of the secondary confinement, a filtered ventilation system should be provided and designed to maintain a negative gauge pressure in DBAs. If a negative gauge pressure cannot be achieved and maintained in the confinement volume, account should be taken in the calculations of the radiological consequences of the unfiltered leakage to the environment that will result.

4.108. The confinement volume should be kept at negative gauge pressure in normal operation, to enable the leak tightness of the secondary containment to be monitored.

Containment bypass

4.109. Containment bypass events arise when primary coolant and any accompanying fission products escape to the outside atmosphere without being processed by containment systems for the control of radionuclides.

4.110. Appropriate design provisions should be taken to demonstrate that conditions involving a containment bypass and leading to an early radioactive release or a large radioactive release have been practically eliminated.

4.111. Design and quality of piping circulating outside the containment high contaminated liquids or high airborne activity in the event of an accident with core melting should be adequate to be leak tight under the conditions they are operated. Loads and process conditions should be properly considered and combined.

4.112. Conditions for the opening of the containment (equipment hatch, fuel transfer tube, etc.) should be specified and adequate to prevent from arising of accidents with a release of activity to the atmosphere of the containment, or to close the containment shortly.

4.113. Possible paths for interfacing system LOCAs should be eliminated as far as possible, either by relocating the system in the containment or by increasing the design pressure of the low pressure system above the pressure of the reactor coolant system. For any remaining possible paths for interfacing system LOCAs, reliable provisions for preventing or stopping the leak outside the containment should be implemented.

4.114. In pressurized water reactors, a steam generator tube rupture is a containment bypass event that could lead to significant releases of radioactive material. Preventive design features should be installed in steam generators to reduce the frequency of such events to a very low value. The design of the plant should allow a fast isolation of the affected steam generator in order to minimize the radiological release and not to exceed the radiological limit defined for the relevant plant state category.

4.115. Many containment designs include systems to recirculate water from collection points inside the containment envelope, either through heat exchangers or directly, for reinjection into the reactor vessel or into the containment spray system in an accident. Parts of these recirculation systems may be located outside the containment envelope, giving rise to a potential for leakage of radionuclides from pumps, valves or heat exchangers outside the containment envelope. Where a design of this type is used, provisions should be made to minimize the uncontrolled release of radionuclides to the environment resulting from such leakage, to test the leak rate periodically, and to detect and isolate accidental leaks by qualified means.

Reduction in airborne radionuclides

General

4.116. As an application of the defense in depth concept, and in addition to the measures taken to ensure the leak tightness of the containment, measures should be taken to reduce the inventory of radionuclides in the containment atmosphere.

4.117. In general, a single system is not sufficient for reducing the concentrations of radionuclides, and multiple systems should be employed. Methods used for the reduction of airborne radionuclides in water cooled reactors of extant and new designs are:

- Deposition on surfaces,
- Spray systems,
- Pressure suppression pools,
- Ventilation and venting systems.

4.118. As long as active systems for the reduction of the concentrations of airborne radionuclides are in the standby mode in normal operation, they should be testable.

Deposition on surfaces

4.119. The containment structure and its internals provide the first mechanisms for the removal of airborne radioactive material, since they present a large surface area for deposition. The plate-out and desorption factors ascribed to the containment structure should be conservatively based on the best available knowledge of deposition of radionuclides on surfaces. The surfaces of the containment and its internal structures should be decontaminable to the extent possible.

Containment spray system

4.120. The radionuclide management function of the containment spray system is intended to reduce amounts of airborne radioactive substances by removing them from the containment atmosphere and retaining them in the water of the containment sump or the suppression pool. This serves to limit any radiological consequences resulting from leakage of radioactive material from the containment to the atmosphere.

4.121. Important parameters and factors that should be considered in the design of the containment spray system include spray coverage, spray drop size, drop residence time and the chemical composition of the spray medium:

- Chemicals should typically be added to the spray water to enhance the removal of radionuclides from the atmosphere. Radioiodine is of particular importance, because of its potential consequences in terms of high specific doses. The chemical additive system should be designed to maximize the dissolution of radioiodine and to maintain the sump chemistry or the suppression pool chemistry such that radioiodine will not be released from solution in the long term following an accident.
- Any chemicals added should be non-corrosive for the materials present in the containment, both in the short term and in the long term after an accident. Corrosion might not only reduce the strength of vital structural components and impair the operation of safety systems but might also generate combustible gases and other undesirable compounds.

Pressure suppression pool

4.122. Water pools or tanks through which the containment atmosphere is bubbled for steam condensation should be considered a valuable means for the removal of radioactive products. However, care should be taken in evaluating the efficiency of such a process, since it is dependent on the thermodynamic conditions of water and steam. For example, the sub-cooling degree of the water and the consequent efficiency of steam condensation have significant effects on the scrubbing efficiency of a suppression pool.

Ventilation and venting systems

4.123. Where ventilation systems are used for cleaning exhaust air to mitigate the consequences of an accident, filters should be so designed and maintained as to preclude any loading of the filters with pollutants beyond authorized limits prior to their use in relation to an accident.

4.124. The ventilation system should, if necessary, be provided with equipment (such as moisture separators and preheaters before the filters) to prevent the temperature from dropping below the dew point at the air filter inlet.

4.125. The efficiency of the absorption material in iodine filters should be demonstrated in laboratory tests under simulated accident conditions as deemed appropriate. Provisions should be made to test periodically the filter system in situ.

4.126. Ventilation systems are often used to collect, filter and discharge air from a secondary confinement, which may become contaminated with airborne radionuclides in accident conditions as a result of leakage from the primary containment. For such cases the recommendations in paragraph “Provisions for containment isolation of piping and ducting systems” apply.

4.127. Where containment venting systems are installed, the system should be designed to minimize the release of radionuclides to the environment The system design could include a filtering system such as sand, multi-venturi scrubber systems, high-efficiency particulate air (HEPA) or charcoal filters, or a combination of these. HEPA, sand or charcoal filters may not be necessary if the released gas flow is scrubbed in a water pool.

4.128. Noble gases cannot be filtered out, but consideration should be given to the use of systems to delay their release until further radioactive decay has occurred.

MANAGEMENT OF COMBUSTIBLE GASES

Generation of combustible gases

4.129. Sources for potential release of combustible gases and the associated threats should be identified for the different plant states.

4.130. The effects of the combustion of gases should be evaluated to be prevented to the extent possible, or limited, or practically eliminated when their mitigation would be recognized as not achievable.

4.131. The sources of combustible gases should be identified with account taken of the following phenomena :

- Radiolysis of the water in the core,
- Radiolysis of the water in the sump or the suppression pool,
- Metal–water reactions of core components and reactor pressure vessel internals,
- Chemical reactions with materials in the containment,
- Degassing of hydrogen dissolved in the primary coolant,
- Releases from the hydrogen tanks used for control of the primary coolant chemistry,
- Molten core concrete interaction producing hydrogen and carbon monoxide.

4.132. The generation of combustible gases and the generated volume as a function of time should be calculated for design basis accidents and design extension conditions. The uncertainties in the various mechanisms for generation should be taken into account by the use of adequate margins for each of those mechanisms. During accidents with core melting, uncertainties of hydrogen production are essentially linked to such phenomena as flooding of a partially damaged core at high temperatures, the late phase of core degradation, the slumping of molten core material into residual water in the lower head of the reactor pressure vessel, and the long term interactions between molten core material and concrete.

Threats due to combustible gases in accident conditions with core melting

4.133. Threats to the containment structures are reactor technology and design dependent, but usually are caused by high pressure and thermal loads originated by a large production of non-condensable gases, and by various regimes of combustion of the combustible gases. Both should be considered, and their effects assessed. Even if it can be demonstrated that conditions for the gas mixture flammability are not met (e.g. in case of a low hydrogen concentration, or a high steam concentration or a low oxygen concentration), an over pressurization due to non-condensable gases is nevertheless relevant (e.g for inert containment the probability of hydrogen combustion is low due to the presence of inert gas and the absence of oxygen in normal power operation, and so for such a type of containment, the primarily threat is the fast over pressurization caused by a large production of non-condensable gases in a small volume).

4.134. The global and local effects of combustion (static pressure loads, dynamic pressure loads and thermal loads) on the containment structures and the safety features necessary for the mitigation of the consequences of accidents with core melting should be considered.

4.135. The general approach for designing performances and efficiency of the various means necessary for the management of combustible gases should be defined on the basis of gas concentration limits which should not be exceeded at any time of kinetics of gas releases:

- Hydrogen combustion should be postulated when flammability is exceeded (e.g. for hydrogen concentration higher than 4% in volume in dry air);
- As long as conditions for flame acceleration phenomena and for high dynamic pressure loads are not reached, the Adiabatic Isochoric Complete Combustion (AICC) pressure curve calculated for all the hydrogen combustions at a slow flame regime should be retained to define the global and local pressure bounding loads;
- Conditions for flame acceleration phenomena which could lead to deflagration to detonation transition (DDT) or to a detonation should not be reached to the extent possible in areas where hydrogen accumulation is possible. Areas where those conditions cannot be met, detailed analyses and calculations should be conducted with the aim at demonstrating that detonation, DDT or fast combustion regime would not lead to challenge the structural integrity of the containment structures and systems;
- To reach safe conditions inside the containment, performance and efficiency of the means to remove combustible gases should be designed to reduce their concentration in average in the

free volume of the containment below the gas flammability limit in dry air (e.g. below than 4% for hydrogen).

4.136. Calculations and analyses should cover gas generation, gas production time history, and gas concentration distribution to assess the possibility for the various regimes of combustion to occur (combustion at a slow flame regime, fast combustion regime with flame acceleration, or DDT regime).

4.137. The threat for hydrogen combustion while steam concentration is decreasing should be understood and considered with regard to the operation of the containment heat removal system.

4.138. Leaks and releases of combustible gases from the containment should also be taken into account when evaluating the threats.

Measures for the mitigation of hydrogen combustion and for the prevention of hydrogen combustion challenging the containment integrity

4.139. A set of various measures (selection of materials, free space inside the containment, removal, transport and mixing, venting) should be taken to minimize hydrogen production, to mitigate hydrogen combustion and to practically eliminate combustion regimes challenging the containment integrity.

4.140. Performance and efficiency of those measures should be such that the concentration limits indicated in clause 4.135 can be met.

4.141. Performance and the layout of those measures should be such that the containment integrity and leak tightness are maintained within the limits considered in the safety demonstration.

Removal

4.142. An adequate number of passive autocatalytic recombiners and/or active means such as igniters should be provided and suitably distributed inside the containment for burning/removing combustible gases.

4.143. The number and positioning of recombiners or igniters should be justified on the basis of detailed combustible gas distribution analyses resulting from different scenarios of an accident with core melting.

4.144. Removal means should be located in the neighborhood of the release location, near expected convection flow paths between inner containment rooms, the dome area as well as the containment periphery and at different heights in large rooms.

4.145. Layout provisions should be implemented so that thermal loads from combustion flame or due to hot off gas from recombiners could not damage containment liner (or containment steel shell), containment penetrations, components and cables necessary for the mitigation and monitoring of accidents with core melting.

Homogenization

4.146. The containment design either should incorporate active means (such as sprays and mixing fans qualified for operation in a combustible gas mixture) or should facilitate the action of mechanisms (such as large volume dispersion or natural circulation) to enhance the uniform mixing of the containment atmosphere within and between compartments, in particular owing to the implementation of openings and/or preventing to the extent possible dead-end zones.

Inerting

4.147. One possible way to avoid combustion is to inert the containment atmosphere during reactor operation (usually with nitrogen). This is mainly applicable to a small containment.

4.148. Ingress of oxygen into inert containment should be prevented for example by maintaining an overpressure in the containment by limiting depressurization or by additional nitrogen supply.

MECHANICAL FEATURES OF THE CONTAINMENT

4.149. The mechanical features of the containment comprise the mechanical components of the outermost barrier and the mechanical parts of the extensions of this barrier (i.e. piping, valves, ducts and penetrations). Together with the containment structure, these features comprise the containment envelope.

4.150. The leak tightness criteria for mechanical features of the containment and its extensions should be consistent with the assumptions used in the radiological analyses for accidents.

Provisions for containment isolation of piping and ducting systems

4.151. The following recommendations provide guidance to fulfill Requirement 56 of [2].

4.152. Each line penetrating the containment that is not part of a closed loop[10] and that either (a) directly communicates with the reactor coolant during normal operation or in accident conditions or (b) directly communicates with the containment atmosphere during normal operation or in accident conditions should be provided with two isolation valves in series. Each valve either should be normally closed or should have provisions to close automatically. Where the line communicates directly with the reactor coolant or the containment atmosphere, one valve should be provided inside

10 A closed loop is a piping or ducting system that penetrates the containment envelope and that is designed to form a closed circuit either inside or outside, or inside and outside the containment in operational states and in accident conditions.

the containment and one valve outside. Each valve should be reliably and independently actuated. Isolation valves should be located as close as practicable to the containment.

4.153. Loops that are closed either inside or outside the containment should have at least one isolation valve outside the containment at each penetration. This valve should be an automatic valve, a normally closed valve or a remotely operated valve11. Where the failure of a closed loop is assumed as a postulated initiating event or as a consequence of a postulated initiating event, the previous recommendation will apply to each line of the closed loop.

4.154. Loops that are closed both inside and outside the containment envelope should have at least one isolation valve, an automatic valve, a normally closed valve or a remotely operated valve located outside and as close as practicable to the containment at each penetration.

4.155. Small dead-ended instrumentation lines that penetrate the containment should have at least one isolation valve outside the containment.

4.156. Containment isolation valves for instrumentation lines that are closed (i.e. not in communication with the atmosphere) are not required provided that they are designed to withstand accident conditions for which the confinement is necessary. The rooms where these lines emerge should be equipped with a filtration–ventilation system to maintain sub atmospheric pressure. Such rooms and the equipment within them should be designed to withstand increased levels of temperature and humidity due to possible leakage from these lines.

4.157. The need for an automatic isolation of the containment in accident conditions should not prevent the systems necessary to mitigate those accidents from accomplishing their intended functions.

4.158. Overpressure protection should be provided for closed systems that penetrate the containment and for isolated parts of piping that might be over pressurized by a raise of the temperature inside the containment atmosphere in accident conditions.

4.159. The extensions of the containment envelope should be designed and constructed to levels of performance that are at least equivalent to those for the containment itself.

4.160. For specific operational conditions (e.g. conditions with an open containment, or inhibited containment automatic isolation), the risk to safety should be assessed and temporary provisions

11 An automatic valve is a valve or damper that can be actuated either by the protection system signals or by other instrumentation and control circuits without action by the operator or by the process medium itself. For example, certain types of check valves are considered automatic valves. A normally closed valve is a valve that is closed under active administrative control (such as being locked closed or continuously monitored to show that the valve is in the closed position) except for intermittent opening for specific purposes such as monitoring, testing or sampling. A remotely operated valve is a valve or damper that can be actuated by an operator from the control room and in some cases also from the supplementary control points.

should be implemented as needed to ensure the containment isolation function could be accomplished in a timely manner.

4.161. Particular consideration should be given to the containment isolation features of the following systems which potentially could create a by-pass of the confinement:

- Systems designed for removing heat from the core, the core debris or from the containment that can transport radionuclides outside the containment in accident conditions;
- Systems that can transport airborne radionuclides from the containment atmosphere to outside the containment in design basis accidents (i.e. systems used in some designs to mix the atmosphere inside the containment in order to prevent the ignition of hydrogen);
- Supporting systems or auxiliary systems (inside the containment) for which, in the event of leakage, fluids with a high activity might be released outside the containment (i.e. in some designs the component cooling water system, the containment sump purge system or the sampling systems).

4.162. Systems connected to the primary circuit in normal operations (i.e. primary circuit filtration systems or in some designs the chemical and volume control system) and systems connected to the containment atmosphere should be automatically isolated in accident conditions when they are not necessary for safety.

Isolation valves

4.163. To achieve the objective of limiting any radioactive release outside the containment, the isolation devices should be designed with a specified leak tightness and closure time.

4.164. Design provisions for leakage tests (such as nozzles and instrumentation test lines) should be made such that each isolation valve may be tested.

Penetrations

4.165. The following recommendations provide guidance to fulfill the requirement 6.21 of [2].

4.166. Containment penetrations should be designed to withstand at least the same loads and load combinations as the containment structure.

4.167. Containment penetrations should be accessible so that leaks from individual penetrations can be detected in the leak tightness tests.

Piping penetrations

4.168. In the mechanical design of piping penetrations, including isolation valves, the loads originating from the piping system as well as loads originating from the containment should be taken into account.

Electrical penetrations

4.169. Penetrations through the containment for electrical power cables and instrument cables should be leak tight. Means for ensuring the leak tightness of these penetrations may be based on the following:

- Pressure glass penetrations. The pressure glass design consists of studs embedded in a pressurized disc of glass flanged to the containment. Cables are connected to the studs, which extend on both sides of the glass disc and provide continuity for the electric power. The glass ensures electrical isolation between the studs and acts as a sealant. The design should include double seals on the flange to ensure the leak tightness of the assembly. These penetrations should be removable and individually testable for leak tightness at the design pressure;
- Pressurized and continuously pressure monitored penetrations. For pressurized penetrations, the pressurization should normally be higher than the internal pressure in the containment that could occur in accident conditions. so that leak tightness can be tested continuously. In any case, the pressure should not be lower than the pressure used in the containment leak rate test. The effects of increase in temperature on the design pressure of the fluid inside the penetrations should be assessed and taken into account in the design of the penetrations;
- Injected sealant penetrations. Penetrations of this type should be leak testable in integrated leak tests.

4.170. Preference should be given to designs of electrical penetrations that allow each penetration to be tested individually.

4.171. Heat produced by the electrical cables should be taken into account in selecting the materials for electrical penetrations. The materials used should be heat resistant and non-flammable. Penetrations using sealant injection should be at least flame retardant.

Air locks, doors and hatches

4.172. The following recommendations provide guidance to fulfill Requirement 57 of [2].

4.173. Penetrations for access by personnel or equipment to the containment (containment air locks) should have air locks equipped with doors that are interlocked to ensure that at least one of the doors is closed during reactor operations and accidents. In addition, they should be designed to prevent any undue exposure of operators to radiation in operational states of the plant.

4.174. The two air lock doors should be designed to withstand the same plant conditions as the containment. Local transient internal effects, such as exposure to open flames caused by hydrogen burning, need not be considered for the outer door.

4.175. The chamber between the two air lock doors should be so sized as to allow the passage of necessary maintenance equipment and a sufficient number of personnel, so as to avoid having to open the air lock too frequently during plant shutdown and maintenance.

4.176. The inner door of the air locks should be of a pressure sealing type. Double seals should be provided on each door and there should be provisions for testing the leak tightness of the doors and the inter-seal space. Low pressure alarms should be provided if inflatable seals are used.

4.177. Equipment hatches are large openings in the containment structure that are normally closed. They are usually designed with a bolted flange, whose leak tightness is ensured by means of soft elastomeric seals. Loads and deformations due to pressure and temperature effects should be taken into account in the design of equipment hatches.

4.178. Containment openings (i.e. penetrations, air locks and hatches) should normally be closed. Exceptions are allowed if they are necessary for operational reasons and provided that the openings can be closed quickly and reliably to comply with established engineering criteria that apply for the accident. Conditions allowing the opening of the equipment hatches should be specified and met prior to the opening.

MATERIALS

Concrete

4.179. Concrete should have characteristics of quality and performance (strength, density porosity) consistent with its use. The quality of the concrete used for containment structures should be correspondingly high, consistent with the safety function of the containment. Design considerations will depend on the containment concept. For example, a pre-stressed concrete containment can provide both structural support and leak tightness, whereas a reinforced concrete containment structure provides structural support but relies on a steel liner for leak tightness.

4.180. Consideration should be given to the design capacity of the concrete to cope with the loads (pressure loads and thermal loads) and environmental conditions (of heat, moisture and radiation) generated by design basis accidents. This should lead to strict specifications for the concrete in terms of strength and leak tightness.

4.181. Concrete specifications should also ensure that measures are taken to avoid material vulnerabilities that may lead to ageing effects (e.g. chloride attack, alkali-aggregate reaction and delayed ettringite formation).

4.182. Concrete with appropriate stiffness, thermal expansion and resistance to compression should be used for all electrical penetrations, large penetrations such as equipment hatches and the joint with the basemat.

4.183. In a pre-stressed containment not sealed with a metallic liner, the concrete should remain in a pre-stressed condition even in accident conditions. Concrete materials that would limit creep or shrinkage over the years and with low porosity should be used. The possible loss of pre-stress of the containment tendons over the operating lifetime of the plant should be evaluated and considered in the design.

4.184. Sleeve concrete interfaces should be designed to minimize leaks by avoiding direct paths through the interface.

4.185. Design and construction processes should be such as to prevent the development of cracks or high leak zones.

4.186. Ageing effects should be evaluated in the selection and design of types of concrete, and a programme for monitoring the effects of ageing over the time should be developed **[7**[7].

Metallic materials

4.187. Metallic materials used for containment structures and systems, including welds, should be of high quality; qualified and certified materials that meet national safety standards should be used.

4.188. In the selection of metallic materials, the following considerations should be taken into account:

- Thermal and mechanical loads;
- Chemical interactions, including those with chemicals used in containment spray systems;
- Sensitivity to ageing effects;
- Resistance to brittle fracture;
- Resistance to corrosion.

4.189. Metallic materials such as zinc and aluminum that have the potential to generate hydrogen on contact with water or steam should not be used inside the containment. If such materials are essential to the design, their use should be limited and the effects of hydrogen generation should be analyzed.

Soft sealing materials

4.190. Soft sealing materials are commonly used in multiple containment applications, such as in the sealing of ventilation valves or the inflatable sealing of air locks. Although these materials contribute to a very high leak tightness of the containment under normal conditions, their behavior in accidents should be properly demonstrated. Potentially damaging effects for soft sealing materials include embrittlement and cracking due to high temperatures and irradiation, dissolution due to moisture and steam, and swelling or shrinkage due to temperature fluctuations. Specific consideration should be given to the protection of these materials from the direct effects of hydrogen burning and/or the accumulation of radioactive aerosols. In extreme conditions such materials may degrade to the extent that their mechanical properties are altered.

4.191. The anticipated lifetimes of soft sealing materials and the ageing mechanisms that affect their performance should be assessed, and appropriate replacement intervals should be established. Sealing components should be designed to be easily inspectable and replaceable.

Covering, cushioning, thermal insulation and coating materials

4.192. The following recommendations provide guidance to fulfill the requirement 6.30 of [2].

4.193. Covering, cushioning, thermal insulation and coatings materials should not compromise any safety functions in the event of their deterioration. They should be installed and affixed to prevent loosening and the possible clogging of sieves and valves as a consequence.

4.194. In particular, materials used to insulate pipes and tanks inside the containment should be selected and designed to achieve the following:

- To minimize the production of debris that can accumulate on containment floors and clog the sump screen or strainer filter or damage recirculation pumps;
- To ensure easy decontamination if the need arises;
- To avoid giving rise to fire hazards;
- To minimize the release of toxic gases during their heating at the plant start up.

4.195. The amount of debris generated in the event of high energy pipe breaks and transported to the sumps should be assessed and the surface of filters sized accordingly in order not to compromise the normal operation of the pumps necessary to mitigate the consequences of accidents.

4.196. Moreover, the installation of a cleaning system of the filters should be established taking into account the large uncertainties on the types and the amount of debris susceptible to clog the filters.

4.197. If organic liners are applied to increase the leak tightness of the containment structure, they should be selected to provide good adhesion and a low air/gas permeability, and to withstand the thermal loads and pressure loads, as well as the environmental conditions in the containment, without losing their safety function (e.g. the organic material should have good crack spanning ability and resistance to blistering after thermal ageing). Provision for managing the ageing of these organic liners should be made, including provision for maintenance and surveillance).

4.198. Painting and coating materials should be selected so as not to pose a fire hazard.

4.199. In the selection of painting and coating materials, consideration should be given to the effect of the dissolution of their solvents in the sump on the volatility of iodine.

4.200. Ageing mechanisms that affect covering, cushioning, thermal insulation and coating materials should be assessed and appropriate replacement intervals should be established.

INSTRUMENTATION

4.201. The following recommendations provide guidance to fulfill Requirement 59 of [2].

4.202. For a safe operation in accident conditions, instrumentation should be provided for the purposes of:

- Monitoring of the stability of the containment structure;
- Detection of deviations from normal operation;
- Periodic testing;
- Monitoring of the availability of the containment systems;
- Initiation of automatic operation of systems;
- Post-accident monitoring.

4.203. As those different purposes can need measurements of the same parameters for different levels of defense, the consequences of sharing of sensors for different purposes should be considered in order to preserve adequate independence between the different levels of defense in depth and the following recommendations should be implemented to the extent possible:

- Not sharing the same sensors for the automatic actuation of the operation of the systems and the accident monitoring of the plant;
- Not sharing the same sensors for the automatic actuation of the reactor scrams or operation of the safety systems and their back up implemented to reinforce the prevention of accidents with core melting;

- Implementing different and dedicated sensors for the mitigation of accidents with core melting.

4.204. Instrumentation should be qualified under seismic and environmental conditions that might prevail prior to or during its operation until its mission be completed.

4.205. Test sequences for equipment qualification should be consistent with well proven international practices. General principles and adequate practices are indicated in **[18** [18].

Monitoring the stability of the containment structure

4.206. Deformation (radial, vertical or circumferential) or movement of the containment structures or of the containment walls should be monitored (e.g. monitoring of settlement and differential settlement of the buildings) throughout the lifetime of the containment.

4.207. For pre-stressed concrete walls, means to detect loss of the pre-stressing should be provided. The concrete compression and stiffness parameters (such as Young's modulus) should be defined, and they should be verified by such means as acoustic measurements. The temperature in singular locations should also be measured to aid the interpretation of the results of proof pressure tests.

4.208. Measurements adequate to monitor the containment stability and deformations over time should be recorded to show trends.

4.209. Appropriate instrumentation for measurements relating to earthquakes should be installed at suitable places (e.g. on the basemat of the containment and at suitable floors).

Detection of deviations from normal operation

4.210. Appropriate instrumentation should be incorporated inside the containment for an early detection of deviations from normal operation such as but not limited to:

- Leaks of radioactive material;
- Abnormal radiation levels;
- High energy leaks;
- Leaks;
- Fire;
- Failure of components.

4.211. Instrumentation sensitivity and ranges necessary to detect a developing deviation should be estimated by appropriate analytical methods.

4.212. For an adequate detection of the different abnormal conditions, information provided by the instrumentation can be used alone or in combination with others. Parameters typically monitored include:

Containment atmosphere temperature

4.213. Monitoring of containment atmosphere temperatures is necessary to check whether temperatures are within the ranges specified for the normal operation.

- A sufficient number of temperature sensors should be installed to measure the containment atmosphere temperatures;
- In/out containment air coolers may be used to estimate temperatures inside the containment.

4.214. The containment atmosphere temperatures should be recorded to show trends.

Containment pressure

- Monitoring of containment pressure should be established to check whether the pressure is within the range specified for the normal operation (small variations of the pressure may be caused by the operation of the air operated valves, changes in the containment temperature or by leakages of fluids such as compressed air, nitrogen);
- For the secondary containment or primary containment with double walls, monitoring of the pressure inside the secondary containment or the annulus should be established to check whether the pressure is within the range specified for the normal operation (a small negative pressure should be maintained in the annulus).

4.215. The containment pressure should be recorded to show trends.

Containment atmosphere gas composition

4.216. The containment atmosphere gas composition should be monitored at locations of potential high concentration.

Humidity at different locations

4.217. Humidity is a highly significant factor for the detection of leaks from a water reactor cooling system in operational states. Different techniques can be used to measure the humidity such as:

- The dew point temperature of the containment atmosphere,
- Electrical parameters (such as impedance or resistance) of sensors,
- The amount of condensate in the air coolers of the containment building.

4.218. Measurements should be recorded to show trends.

Water levels in the drain storage tanks and sumps

4.219. Drain storage tanks and sumps of each safety system as well as the condensate collector of each air cooler should be provided with a water level indicator.

Radiation levels and radioactivity measurements

4.220. Radiation levels at different locations inside the containment should be measured for the radiation protection of the workers and for an early detection of any anomalies with harmful effects caused by an excess of radiation.

4.221. Airborne and water (drain storages and sumps) activity measurements should be implemented as a diverse mean for complementary detection of leaks.

Visible abnormalities

4.222. A video monitoring should be installed inside the containment to detect anomalies at the more relevant locations where leaks or other malfunctions can be expected and/or where personnel access is difficult (e.g. reactor coolant pumps, equipment hatch, personnel airlocks, reactor pools, etc.).

4.223. Mobile cameras should be available for use if and when the demand for them arises.

Noise and vibrations

4.224. The use of audio signals from the containment building for the detection of abnormalities should be considered (e.g. the use of spectral and Fourier transform analyses for acoustic noise signals).

Fire.

4.225. Smoke and flame detectors should be installed as additional means of an early detection of a fire in each compartment where there may be a risk of fire.

Periodic testing

4.226. Appropriate instrumentation should be incorporated inside the containment for conducting the periodic leak tests. Measurements of temperature, pressure and humidity and flow rates should be combined for the periodic calculation of the mass of the containment atmosphere and for the estimate of the leak rate.

4.227. For steel containments, the temperature of the steel should also be measured.

Monitoring of the availability of systems

4.228. Appropriate instrumentation should be used to monitor the availability of the containment systems used for energy management, management of combustible gases and for the control of radionuclides.

4.229. The availability of the systems should be verified by means of the following:

- By continuous monitoring and display in the main control room of the main parameters important to safety (a single integrated monitor for critical safety parameters is recommended);
- By conducting the periodic tests and inspections as required;
- For the systems for energy management, by monitoring the positions of valves, the status of components in operational states and flow rates;
- For the systems for radionuclide management, by monitoring the positions of isolation valves, air locks and doors, the pressure of inflatable airlock seals and water levels in the different water tanks necessary to the operation of those systems.

Initiation of automatic operation of systems

4.230. In the event of a significant release of energy or of radioactive material into the containment different information is necessary for a complete and effective management of the energy, radionuclides and combustible gases release inside the containment. The requested management can be initiated automatically or by the operator provided that the time for implementing operator actions is sufficient.

4.231. Information should be elaborated from the monitoring of parameters giving evidence that a large release of energy or a significant release of radioactive material has occurred inside the containment. Depending on the reactor technology or the design, the following parameters may be relevant:

- High pressure inside the containment;
- High radiation levels inside the containment atmosphere;
- Low pressure in the reactor coolant system;
- Small sub cooling margin in the reactor coolant system (PWR);
- Low water level in the reactor pressure vessel.

4.232. In addition to conditions which require a complete and effective management of energy, gases and radioactive material releases inside the containment, there are other events for which only the individual isolation of the affected lines is necessary to limit the release of radioactive material from the containment to the environment. This is the case for a break occurring outside of the containment of a pipe crossing the containment and carrying radioactive material, or for the failure of an interface between two associated systems (e.g. rupture of a heat exchanger tube of the component cooling water system) that leads to a release of radioactive material from a system inside the containment to a system outside. The actuation of the isolation devices should be derived from the values of appropriate parameters, such as:

- Levels of radiation or of airborne contamination;
- Pressure changes in the affected system;
- Temperature changes in the affected system;
- Water level in the affected system.

Accident and Post-accident monitoring

4.233. For the determination of the plant status in case of accidents and for management of accidents, appropriate instrumentation displays and records should be available in the Main Control Room and the Emergency Control Center to allow personnel to make a diagnosis and to decide and to take the manual protection actions specified in the Emergency Operating Procedures or in the Severe Accident Management Guideline. Such instrumentation should provide information about:

- Conditions and gas composition inside the containment (containment pressure and temperatures, radiation levels, airborne activity, steam, oxygen or hydrogen concentration if relevant);
- Process parameters to check that the required safety actions are in progress and to indicate the operations of the required safety systems and safety features for DECs (flow rates, water levels in tanks and sumps, operating pressures in the systems, etc.);
- Process parameters to indicate the potential for degradation or loss of the containment leak tightness (containment isolation valves position, status of hatches and doors, containment pressure, airborne activity in the surrounding buildings etc.);
- Process parameters to implement actions specified in the emergency procedures or severe accident management guidelines (process parameters to control the pressure and to maintain the conditions inside the containment below the specified limits);
- Instrumentation for assessing in a timely manner the radiological consequences and for assisting in decisions on long term actions for the protection of the population (off-site emergency measures). Measurements for assessing radiological consequences may include:

- ✓ Dose rate meters and airborne activity sensors in the containment and in peripheral buildings;
- ✓ Sensors for monitoring conditions in the containment sump water (e.g. temperature and pH);
- ✓ Activity monitors for noble gases, iodine and aerosols in the stack(s) and in the containment venting line;
- ✓ Position indicators of containment venting valves.

- Dedicated instrumentation should be implemented to allow personnel in the Main Control Room to initiate long term actions necessary to maintain the containment integrity in the event of an accident with core melting. Such instrumentation should provide information about:
  - ✓ Process parameters to initiate the fast depressurization of the reactor coolant system (before core melting) and to confirm the open position of the valves;
  - ✓ Process parameters to confirm the flooding of the reactor cavity (for in vessel strategy), or the flooding of the ex-vessel retention structure (for ex-vessel retention strategy);
  - ✓ Process parameters for the localization of the corium (for ex-vessel retention strategy);
  - ✓ Process parameters to initiate and confirm the operation of the containment spray;
  - ✓ Process parameters to initiate and confirm the operation of the containment heat removal system;
  - ✓ Process parameters to initiate the venting of the containment (if relevant);
  - ✓ Process parameters for the hydrogen risk management.

4.234. A monitoring or sampling system should be provided inside the containment to enable assessment of the risks of explosion from combustible gases. The design of the system should take into account the following factors:

- Possible sources of combustible gases such as interaction between clad material and water, or interaction between molten core and concrete, or due to radiolysis;
- Presence or absence of oxygen and inert gases;
- Presence of noble gases and aerosols;
- Presence of devices aimed at recombining hydrogen as produced, and type of these devices (passive or active)
- Sufficient mixing or not of the containment atmosphere to avoid local possibilities of hydrogen accumulation.

4.235. The monitoring can be achieved by direct gas concentration measurement or sampling. An alternative possibility is to measure the recombining activity of the recombiners by temperature measurement.

4.236. Provisions should be made in the design for sampling of the containment atmosphere and the sump water at suitable locations. The sampling devices used should be qualified for the expected containment conditions and should be installed so as to avoid a containment bypass in the event of their rupture. They should be designed to ensure that occupational radiation dose limits are not exceeded for the personnel who operate them.

4.237. Monitoring or sampling lines that could transport radionuclides outside the containment should be considered as extensions of the containment and should be subject to specifications for structural integrity and leak tightness comparable with those applied to containment structure itself.

5. TESTS AND INSPECTIONS

5.1. In order to demonstrate that the containment and the associated systems meet design and safety requirements, construction, commissioning and in-service tests and inspections should be conducted taking into account underneath guidance / recommendations and be conducted according to more detailed recommendations provided by proven codes and standards.

5.2. The following recommendations provide guidance to fulfill Requirement 29 of [2] Recommendations given by [10] should also be considered.

INSPECTION AND CONTROLS DURING CONSTRUCTION

5.3. Inspection and controls should be performed at different stages of the construction of the containment structure to ensure the conformity to design and construction specifications. Deficiencies, deviations from standards and non-conformances should be tracked and reported. Typical examples of controls and inspections performed during construction are:

- Vertical tendon anchorage area;
- Basemat rebar installation and concrete work;
- Horizontal tendon anchorage area;
- Tendon duct arrangement;
- Liner plate work;
- Rebar arrangement around large opening.

5.4. Construction works, inspection and controls should be performed by qualified personnel.

COMMISSIONING TESTS

5.5. Commissioning tests for the containment should be carried out prior to the first criticality of the reactor to demonstrate the containment structural integrity, to determine the leak rate of the containment envelope and to confirm the performances of systems and equipment.

Structural integrity test

5.6. A pressure test should be conducted to demonstrate the structural integrity of the containment envelope (including extensions and penetrations) and of the pressure retaining boundary of systems.

5.7. The pressure test should be conducted at a specified pressure for which account is taken of the applicable codes for the material used, and which is at least the design pressure. The value of the test temperature should not be close to the ductile brittle transition temperature for the metallic material.

Integrated Leak Rate tests (of the containment envelope)

5.8. An integrated leak rate test should be conducted to demonstrate that the leak rate of the containment envelope does not exceed the specified maximum leak rate. The test should be conducted with the components of the containment in a state representative (to the extent practicable) of the conditions that would prevail following an accident, to demonstrate that the specified leak rate would not be exceeded under such conditions.

5.9. To establish a point of reference for future in-service leak tests, the leak rate test performed during commissioning should be conducted at a test pressure or pressures consistent with the pressure selected for in-service leak tests:

- At values of pressures between the pressure selected for in-service leak testing and the positive design pressure, if the in-service tests are to be conducted at a pressure lower than the design pressure; or
- At the design pressure of the containment, if the in-service tests are to be conducted at this pressure.

5.10. The need to validate reliably the leak rate assumed in the safety analysis over the entire plant operating lifetime for the entire range of pressures calculated should be taken into consideration in the choice of test pressure(s). The validation can be made as follows:

- Absolute method: the leak rate can be validated by measuring the decrease in pressure or the dry air mass as a function of time. In this method, the temperature and pressure of the containment atmosphere, the external atmospheric temperature and pressure, and the humidity of the containment atmosphere should be measured continuously and factored into the evaluation. Means should be provided to ensure that the temperature and humidity of the containment atmosphere are uniform.

- Reference vessel method: The reference vessel method determines the air mass from the pressure differential between the containment atmosphere and the reference vessel atmosphere. The pressure differential is determined from a manometer, one leg of which is open to the pressurized and leaking containment, while the other leg is connected to a leaktight pressurized system of tubing placed throughout the containment. The reference vessel temperature and the containment temperature are assumed to be equal.

5.11. The need for initial and periodic testing should be considered in the design, and all the components that might be damaged during testing should be identified. The necessary means to pressurize and depressurize the containment and appropriate instrumentation for testing should be included in the design.

5.12. Appropriate instrumentation should be provided in the containment, appropriately positioned and installed either permanently or as needed, to determine representative atmospheric conditions in the different zones of the containment.

5.13. For double wall containments, one way to determine the direct leak rate from the containment to the environment (i.e. if the leaked water or gas does not collect in the annular space between both) is by calculation. This calculation should determine the difference between (a) the total leak rate from the containment as determined by the leak test for the inner containment (this consists of both flow from the primary containment into the annulus and flow from the primary containment to the atmosphere) and (b) the leak rate from the primary containment wall to the annulus, obtained after ventilation of the annulus has been stopped (this is typically calculated by subtracting the normal flow out of the annulus vent from the flow out of the annulus vent during the leak test).

Local leak Rate tests of isolation devices, air locks and penetrations

5.14. Local leak rate tests should be performed to establish a baseline leakage measurement for each isolation device, air lock and penetration. The following components are the most sensitive parts of the containment envelope, and special attention should be paid to them:

- Isolation devices in systems open to the containment atmosphere;
- Isolation devices in fluid system lines penetrating the containment;
- Penetrations that have resilient or inflatable seals and expansion bellows, such as:
  - ✓ personnel air locks,
  - ✓ equipment air locks,
  - ✓ equipment hatches,

- ✓ fuel transfer tube
- ✓ spare penetrations with bolted closures,
- ✓ cable penetrations with resilient seals,
- ✓ pipe penetrations with flexible expansion bellows in the connections to the containment.

5.15. The design should permit leak rate tests of isolation devices, air locks, penetrations and containment extensions.

5.16. The design should facilitate local testing by providing access to penetrations and incorporating necessary connections and isolation valves.

5.17. To permit greater precision in measuring the leak rate and to improve the detection of leaking valves, a capability for testing individual valves should be provided.

Functional tests of equipment and wiring in the containment

5.18. Tests should be carried out to verify that the equipment in all associated systems is functional unless the tests would have a detrimental effect on safety.

5.19. Tests should be carried out on all electrical wiring associated with the containment systems to demonstrate that there are no deviations from the design and that all connections are in accordance with the design.

IN-SERVICE TESTS AND INSPECTIONS

5.20. Both in-service integrated leak rate and local leak rate tests, and inspections should be periodically performed to demonstrate that the containment systems continue to meet the requirements for design and safety throughout the operating lifetime of the plant.

5.21. The test methods and intervals for in-service tests should be specified so as to reflect the importance to safety of the items concerned. In devising test methods and determining the frequency of testing, consideration should be given to the necessary levels of performance and reliability of the containment systems individually and as a whole.

5.22. Appropriate features should be provided for performing commissioning and in-service testing for containment pressure and leak tightness, and the correlated loads should be considered for the purposes of structural design.

5.23. General guidance on in-service inspection is provided in [10].

Structural integrity tests

5.24. Periodic structural tests should be conducted to demonstrate that the containment structure continues to perform as intended in the design. The test pressure should be the same as in the pre-operational test and as required by the applicable design codes. In the design, attention should be paid to the additional stresses imposed by the tests, and test pressures should be established to prevent the tests from causing excessive stresses to the containment structure. A leak test should be performed during any structural integrity test.

Integrated leak tests (of the containment envelope)

5.25. The design should provide the capability for periodic in-service testing of the leak rate to prove that the leak rate assumed in the safety analysis is maintained throughout the operating lifetime of the plant. The in-service leak rate tests may be made at either:

- A pressure that permits a sufficiently accurate extrapolation of the measured leak rate to the leak rates at the accident pressures considered in the safety analysis; or
- The containment design pressure.

5.26. There are also methods available to provide a continuous estimate of the overall containment leak rate during plant operation and to derive rough indications of containment leak rates in accident conditions. Such approaches are generally based on variations in the containment pressure or the mass balance during normal operation of the plant. In some cases, the use of these methods together with extensive local leak rate tests during shutdown for refueling may justify a reduction in the frequency of the global tests.

5.27. In containment with a pressure suppression pool, features should be provided for periodically assessing any leakage that might lead to bypassing of the pool, so as to ensure that the bypass rate of the pool is consistent with the value considered in the safety analysis.

5.28. The testing method of the containment integrated leak test should be conducted according to proven codes and standards.

Visual inspection

5.29. Visual inspections are important for the monitoring and detection of ageing effects, detection and evolution of cracks and may augment the results from structural monitoring and instrumentation.

5.30. Where it is technically feasible, the design should provide for a complete visual inspection of containment structures (including the tendons for pre-stressed concrete containments), penetrations and isolation devices.

5.31. Visual inspection of the containment envelope, including appurtenances and penetrations, should be made in conjunction with each of the tests specified in paras 5.18–5.23. A visual inspection technique that is specifically qualified for detecting the type and size of cracks/ defects which are determined to be important for leakage and structural integrity should be employed.

Periodic tests

5.32. The design should provide a capability for testing safety systems and systems implemented to cope with DECs at intervals that reflect their importance to safety, or for otherwise demonstrating the necessary reliability for the containment systems individually or as a whole.

APPENDIX I. PLANTS DESIGNED WITH EARLIER STANDARDS

A.1 As stated in the IAEA Safety Fundamentals [1], the fundamental safety objective which applies for all facilities and activities is to protect people and the environment from harmful effects of ionizing radiation. For nuclear power plants this overall objective was confirmed by the 2015 Vienna Declaration [16] as follows:

Article 1: New nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible releases of radionuclides causing long-term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions.

Article 2 : Comprehensive and systematic safety assessments are to be carried out periodically and regularly for existing installations throughout their lifetime in order to identify safety improvements that are oriented to meet the above objective. Reasonably practicable or achievable safety improvements are to be implemented in a timely manner.

A.2 The sense of the Article 2 is already reflected in Article 1.3 of SSR-2/1:

It might not be practicable to apply all the requirements of this Safety Requirement publication to nuclear power plants that are already in operation or under construction; in addition, it might not be feasible to modify designs that have already been approved by regulatory bodies. For the safety analysis of such designs, it is expected that a comparison will be made with the current standards, for example as part of the periodic safety review for the plant, to determine whether the safe operation of the plant could be further enhanced by means of reasonably practicable safety improvements.

A.3 This implies for existing plants to deal with accident conditions not considered in their original design basis and that their capability to accommodate those new conditions should be systematically assessed with the further objective to improve the current level of safety and, in particular, the overall efficiency of the containment structures and systems.

A.4 Most of the containment systems of the existing plants were designed for DBAs (e.g. Large LOCA), without account taken of the possibility for severe accidents to occur. However, safety assessments showed that the conservative deterministic approach followed for the design gave the capability to withstand situations more severe than those originally included in the design basis for existing plants.

A.5 The assessment should be conducted on the basis of a set of design extension conditions whose consequences should be analyzed with the purpose of further improving the safety of the nuclear power plant by:

- Enhancing the plant capability to withstand more challenging events, conditions and hazards than those considered in the design basis,
- Minimizing radioactive releases harmful to the public and the environment as far as reasonably practicable, in such events or conditions.

A.6 Although design extension conditions to be assessed are reactor technology and design dependent, the selected set of design extension conditions should systematically include core melt situations, phenomena that could lead to the loss of the containment integrity and external events that exceed the original design basis.

A.7 The assessment for potential back fitting should utilize a holistic approach which considers the safety contributions of installed equipment, non-permanent equipment, and emergency preparedness planning measures to protect the public.

A.8 The assessment should aim at justifying with a reasonable level of confidence that the relevant equipment would be available to perform the expected function. The assessment may use realistic models and assumptions, relaxed acceptance criteria provided that the absence of a cliff edge effect can be still justified.

A.9 The assessment of the robustness of a structure or mechanical equipment can be performed applying deterministic methods, probabilistic methods or a combination of the two.

A.10 The assessment of leak tightness, integrity or operability of structures and components should be performed with account taken of reasonable uncertainties in the loads and in the response of the structure or component.

A.11 The back fitting measures for preventing early releases or for implementing actions in the short term should not rely on the use of off-site mobile equipment.

A12. Although the use of permanent equipment for avoiding large releases should be preferred (as for new plants) a more relaxed approach on the use of non-permanent equipment may be acceptable provided the plant is provided with adequate connection features.

A.13 Internal and external hazards:

- All internal and external hazards that are addressed in the design basis should be re-evaluated on the basis of up to date methodologies meteorological and geological data;
- Hazards not yet evaluated in the design basis that could have an impact on the containment should be considered and their effects evaluated.;
- The design of containment structures and systems that may be needed in beyond original design basis conditions should be assessed to show they would be capable of performing their function with adequate margins under the new conditions;
- Margins justifying that structures and components necessary to avoid releases which would require long-term protective measures and actions should be evaluated.

A.14 Energy management:

- Conditions leading to a direct containment heating should be prevented by different means;
- Possibilities for steam explosion arising should be identified and their effects evaluated;
- Different and diverse means should be implemented to control the pressure build up inside the containment in the different plant states;
- Multiple means should be implemented to remove heat from the containment in the different plant states;
- If a containment venting system is needed for certain beyond original design basis events, it should be reliable, robust to withstand loads from hazards (e.g. earthquake), accident conditions, and to withstand the dynamic and static pressure loads existing when the containment venting line is operated;
- Specific safety features and systems should be implemented to ensure the cooling and stabilization of the molten core.

A.15 Control of radionuclides:

- All piping penetrating the containment should be isolated but systems necessary for the mitigation of the accident conditions;
- The containment should be kept leak tight to the extent possible under severe accident conditions (no significant aggravation of the specified leak rate);
- Different means should be implemented to reduce the radionuclides in the containment atmosphere in the different plant states;
- Mechanisms and potential paths for unintentional containment bypass should be evaluated and identified;

- If venting of the containment atmosphere is necessary, it should be possible to close the containment venting line(s) reliably;
- Intentional release (e.g. containment venting) in the event of a severe accident should consider filtration through filters of high efficiency prior to being discharged to the environment.

A.16 Management of combustible gases:

- Risks for hydrogen deflagration and detonation should be evaluated and adequate provisions should be implemented, if necessary, to prevent hydrogen combustions challenging the containment integrity and to control the concentration of combustible gases inside the containment.

A.17 Instrumentation:

- Operability, reliability and adequacy of instrumentation should be evaluated (e.g. measurement ranges, environmental qualification, power supply) to ensure operators obtain essential and reliable information about the containment status in the different plant states;
- The containment shall be equipped with measuring and monitoring instrumentation that provides sufficient information on the progress of core melt accidents and threats to containment integrity and by which the operator can do the necessary SAMG actions. That instrumentation should be to the extent possible independent from the instrumentation used for the mitigation of DBAs;
- The new instrumentation for monitoring progression of severe accident should be qualified for accident conditions with core melting.

A.18 Non-permanent equipment:

- Non-permanent equipment that is relied upon to mitigate beyond original design basis events should be stored and protected to ensure its timely availability when needed taking into account restricted access due to external events (e.g. flooding, damaged roads etc);
- Relying on non-permanent equipment may be adequate provided justification that coping time to avoid the containment failure is long enough to make use of the equipment.

REFERENCES

[1] EUROPEAN ATOMIC ENERGY COMMUNITY, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1, IAEA, Vienna (2006).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev.1), IAEA, Vienna (2016).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Hazards in the Design of Nuclear Power Plants (DS494).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Classification of Structures, Systems and Components in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-30, IAEA, Vienna (2014).

[5] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, ASME Boiler and Pressure Vessel Code, Section III, Division 1, Rules for Construction of Pressure Vessels, ASME, New York, NY (2013).

[6] ASSOCIATION FRANÇAISE POUR LES REGLES DE CONCEPTION, DE CONSTRUCTION ET DE SURVEILLANCE EN EXPLOITATION DES MATERIELS DES CHAUDIÈRES ELECTRO-NUCLEAIRES: Design, Construction and In-Service Inspection Rules for Nuclear Island Components Design and Conception Rules for Mechanical Components of PWR Nuclear Islands (RCC-M), Paris (2012).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants (DS485).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities (DS452).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, External Events Excluding Earthquakes in the Design of Nuclear Power Plants IAEA Safety Standards Series No. NS-G-1.5, IAEA, Vienna (2003).

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-2.6, IAEA, Vienna (2002).

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Design and Qualification for Nuclear Power Plants, Safety Standards Series No. NS-G-1.6, IAEA, Vienna (2003).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Fuel Handling and Storage Systems for Nuclear Power Plants (DS487).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Nuclear Installations, IAEA Safety Standards Series No GS-G-3.5, IAEA, Vienna (2009).

[16] Vienna Declaration on Nuclear Safety: On principles for the implementation of the objective of the Convention on Nuclear Safety to prevent accidents and mitigate radiological consequences, Vienna (February 2015).

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Security Recommendation on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5), IAEA Nuclear Security Series No. 13, IAEA, Vienna (2011).

[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Instrumentation and Control Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-39, IAEA, Vienna (2016).

[19] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. NS-R-3 (Rev.1), IAEA, Vienna (2016).

[20] JAPAN SOCIETY OF MECHANICAL ENGINEERS, Codes for Nuclear Power Generation Facilities - Rules on Design and Construction for Nuclear Power Plants.

[21] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3, IAEA, Vienna (2014).

[22] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection Aspects of Design for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.13, IAEA, Vienna (2005).

[23] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Auxiliary Systems for Nuclear Power Plants (DS440).

[24] INTERNATIONAL ATOMIC ENERGY AGENCY, Storage of Spent Nuclear Fuel (DS489).

[25] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection, 2016 Revision, IAEA, Vienna (2016).

ABREVIATIONS

AC	Alternating current
AICC	Adiabatic Isochoric Complete Combustion
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CCF	Common cause failure
DBA	Design basis accident
DC	Direct current
DEC	Design extension condition
EVR	Ex Vessel Retention
IVR	In Vessel Retention
I&C	Instrumentation and control
LOCA	Loss of coolant accident
MCR	Main control room
NPP	Nuclear power plant
PIE	Postulated initiating event
PSA	Probabilistic safety assessment
PWR	Pressurized water reactor
RCPB	Reactor coolant pressure boundary
RCS	Reactor coolant system
SSCs	Structures, systems and components
SG	Steam generator
SFP	Spent fuel Pool

CONTRIBUTORS TO DRAFTING AND REVIEW

Azarian, G.	Consultant, France
Barbaud, J.	Electricité de France, France
Bettle, J.	Nuclear Regulatory Commission, United States of America
Fukazawa, M.	Nuclear Regulation Authority, NRA, Japan
Gasparini, M.	Consultant, Italy
Koski, S.	Teollisuuden Voima Oy, Finland
Poulat, B.	International Atomic Energy Agency
Sairanen, R.	Säteilyturvakeskus (STUK), Radiation and Nuclear Safety Authority, Finland
Takii, T.	Hitachi-General Electric, Nuclear Energy Ltd., Japan
Tarallo, F.	Institut de Radioprotection et de Surete Nucléaire, France
Tardivel, J.P.	Institut de Radioprotection et de Surete Nucléaire, France
Titus, B.	Nuclear Regulatory Commission, United States of America
Uhrig, E	Areva, Germany