



ONR GUIDE			
CIVIL ENGINEERING CONTAINMENTS FOR REACTOR PLANTS			
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1. INTRODUCTION

ONR has established its Safety Assessment Principles (SAPs) [1] which apply to the assessment by ONR specialist inspectors of safety cases for nuclear facilities that may be operated by potential licensees, existing licensees, or other duty-holders. The principles presented in the SAPs are supported by a suite of guides to further assist ONR's inspectors in their technical assessment work in support of making regulatory judgements and decisions. This technical assessment guide is one of these guides.

The guidance is intended for use in the assessment of both existing and new design civil engineering containments for reactor plants. Any comments on this guide, and suggestions for future revisions should be recorded on HOW2 [2].

Further guidance on general aspects related to civil engineering is available in the interfacing TAGs listed in Appendix 1.

2. PURPOSE AND SCOPE

The purpose of this technical assessment guide (TAG) is to provide assessors with guidance on the interpretation and application of ONR's SAPs for Nuclear Facilities 2014 [1] that relate to civil engineering nuclear containments. This TAG contains guidance to advise and inform ONR staff in the exercise of their regulatory judgment.

For the purposes of this document, the term "containment" refers to civil engineering structures that fall broadly into four types:

- Pre-stressed concrete reactor pressure vessels (PCPV) where the concrete provides support to a thin primary containment liner against reactor operating pressure and provides biological shield functions to the nuclear reactor core. The PCPV provides the reactor pressure boundary
- Containments that house (pressurised water or boiling water type or hot gas) reactors and which are designed to contain leakage at pressure in the event of failure of the enclosed reactor pressure boundary. A secondary containment may also be included.
- Containments which are not pressurised such as radioactive waste stores, fuel ponds and dry stores containing nuclear material
- Historic containments that are not pressurised such as vaults and silos. These containments are not considered further in this TAG.

The components of nuclear containments can also include, isolating systems, penetrations, pipe work, and pressure relief valves that constitute the containment boundary; however, these items are not within the scope of this TAG. Freestanding steel tanks and containers, including the AP1000 primary containment, are outside the scope of this TAG, as they lie within the domain of ONR's Structural Integrity Specialism. Steel liners may be used to enhance the leak tightness of the containment and the design and construction of the liner and its anchors are assessed by civil engineers with assistance from structural integrity assessors.

The safety and leak tightness of civil engineering nuclear containments depends on a correct assessment of the loadings likely to be applied to the containment and on the design of the containment structure to resist these loadings. Additionally it depends on suitable in-service operational management arrangements to keep the containment within its designed service envelope and on adequate operational procedures being adopted for monitoring, inspection and testing throughout its service life.

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This TAG notes the relevant SAPs [1], provides a commentary on these principles, identifies areas that assessors should consider when reviewing a safety case and lists relevant supporting references. Although this guide specifically addresses the sub-sections of the SAPs concerned with containment, assessors are also referred to the SAPs that provide guidance on:

- Key Engineering Principles;
- Safety Categorisation;
- Special Case Procedures;
- Plant Ageing;
- Reliability; and,
- Structural Integrity.

3. RELATIONSHIP TO LICENCE AND OTHER RELEVANT LEGISLATION

Licence Conditions

The primary Licence Conditions [3] under which assessment of the design or integrity of containments and containment structures is carried out are:

- LC 6 (documents, records, authorities and certificates),
- LC 10 (training),
- LC 12 (duly authorised and other suitably qualified and experienced persons),
- LC 13 (nuclear safety committee),
- LC 14 (safety documentation),
- LC 15 (periodic review),
- LC 17 (management systems),
- LC 19 (construction or installation of new plant),
- LC 20 (modification to design of plant under construction),
- LC 22 (modification or experiment on existing plant),
- LC 23 (operating rules),
- LC 24 (operating instructions),
- LC 25 (operational records),
- LC 26 (control and supervision of operations),
- LC 28 (examination, inspection, maintenance and testing)
- LC 29 (duty to carry out tests, inspections and examinations),
- LC 30 (periodic shutdown), and

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LC 34 (leakage and escape of radioactive material and radioactive waste),

LC 35 (decommissioning).

Pressure Safety Systems

The Pressure Systems Safety Regulations (PSSR 2000) [4] require the owner or user of a pressure system to maintain and operate safe plant, to conduct regular examinations of the system and to keep suitable records. The primary purpose of the Regulations is to secure the safety of people at work i.e. to prevent serious injury from the hazard of stored energy as a result of failure of a pressure system or one of its component parts.

For the AGR, PWR and Magnox type concrete pressure vessels, EDF Energy NGL and Magnox Ltd. respectively are the Competent Person (as defined in the Regulations) for the PCPVs (and the PWR containment). The Company Officer designated as the Appointed Examiner (APEX) for a particular PCPV or containment undertakes the duties imposed by the Regulations on the Competent Person.

The Competent Person defined within the Regulations has two distinct functions. Firstly drawing up or certifying schemes of examination (Regulation 8) and secondly carrying out examinations under the scheme (Regulation 9). APEX's inspection regime including start-up statements and formal reporting on return to service after each periodic outage is undertaken and submitted in compliance with the reporting requirements of Regulation 9.

4. RELATIONSHIP TO SAPS, WENRA REFERENCE LEVELS AND IAEA SAFETY STANDARDS ADDRESSED

The principle SAPs [1] referred to in this TAG are listed in Appendix 1 and are referenced where relevant throughout the text in the appendices of this TAG. They precede the most relevant paragraphs and are often relevant to other paragraphs.

The relevant WENRA Reference Levels [5] (Issue E, Design Basis Envelope for Existing Reactors and Issue F, Design Extension of Existing Reactors) both apply to existing reactors. This TAG generally includes the provisions of these Reference Levels and more detailed information can be obtained from [5].

For new nuclear power plants, due account should be taken of the requirements of the WENRA guidance on Safety of new NPP designs in [6] and [7].

The IAEA Safety Standard SSR 2/1 entitled "Safety of Nuclear Power Plants: Design" [8] has been considered in this review of this TAG and more detailed information can be obtained from this reference.

Further guidance is available in IAEA Specific Safety Guide SSG-53 "Design of the Reactor Containment and Associated Systems for Nuclear Power Plants" [27] regarding implementation of SSR 2/1 in containment design.

The IAEA Safety Standard entitled "Storage of Spent Nuclear Fuel" [9] has been considered in this review of this TAG and more detailed information can be obtained from this reference.

The IAEA Nuclear Energy Series Construction Technologies for Nuclear Power Plants [10] has been considered in this review of this TAG and more detailed information can be obtained from this reference.

The IAEA have defined terms relevant to the design of civil engineering containments, these are reproduced below:

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OFFICIAL**Extract from IAEA SAFETY GLOSSARY 2016 [25]****Confinement**

Prevention or control of releases of radioactive material to the environment in operation or in accidents. Confinement is closely related in meaning to containment, but confinement is typically used to refer to the safety function of preventing the 'escape' of radioactive material, whereas containment refers to the means for achieving that function.

Confinement in nuclear safety is the safety function that is performed by the containment.

Containment

Methods or physical structures designed to prevent or control the release and the dispersion of radioactive substances.

Although related to confinement, containment is usually used to refer to methods or structures that perform a confinement function in facilities and activities, namely preventing or controlling the release of radioactive substances and their dispersion in the environment.

Further guidance is available in IAEA Specific Safety Guide SSG-53 "Design of the Reactor Containment and Associated Systems for Nuclear Power Plants" regarding implementation of SSR 2/1 in containment design [27].

5. ADVICE TO INSPECTORS

In Appendices 3 to 5 of this TAG, the key elements that an assessor should consider in a safety case submission from a Licensee are identified for the most relevant SAPs [1]. The topics identified for consideration in Appendices 3 to 5 should not be considered as a checklist but important areas that should be addressed when assessing a safety case. It is accepted there may be good reasons for a licensee not satisfying the requirements of a SAP. In these cases, the assessor should ascertain the validity of the arguments presented. Since the SAPs are not prescriptive, the assessor will need to judge the extent to which the safety submissions presented satisfy the principles. For many areas, this will rely on the skill and expertise of the assessor.

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- 1 Safety Assessment Principles for Nuclear Facilities. 2014 Edition Revision 1. November 2014. <http://www.onr.org.uk/saps/index.htm>.
- 2 ONR HOW2 Business Management System.
- 3 Licence condition handbook. Office for Nuclear Regulation. October 2014. <http://www.onr.org.uk/silicon.pdf>.
- 4 The Pressure Systems Safety Regulations (PSSR 2000). Approved Code of Practice and Guidance ISBN 978 0 7176 6644 7.
- 5 Western European Nuclear Regulators' Association. Reactor Harmonization Group. WENRA Reactor Reference Safety Levels. WENRA. September 2014. www.wenra.org.
- 6 WENRA Statement on Safety objectives for new nuclear power plants WENRA November 2010.
- 7 RHWG Report on Safety of new NPP designs WENRA March 2013 <http://www.wenra.org>.
- 8 IAEA Safety Standard. Safety of Nuclear Power Plants: Design. Site Specific Safety Requirement SSR-2/1.
- 9 IAEA Safety Standard. Storage of Spent Nuclear Fuel: Specific Safety Guide No. SSG-15.
- 10 IAEA Nuclear Energy Series No. NP-T-2.5 Construction Technologies for Nuclear Power Plants.
- 11 IMC Report - 'The Measurement of Tendon Loads in Pre-stressed Concrete Pressure Vessels and Containment Systems', Multi-Design Consultants, February 2000.
- 12 British Standard BS4975:1990. British Standard Specification for Pre-stressed concrete pressure vessels for nuclear engineering. British Standards Institution.
- 13 ONR Visit Report (Trim - 2009/102216) - Oldbury Periodic Outage 2009.
- 14 Assessment of the Pre-stressed Concrete Pressure Vessel Appointed Examiner's Start-up Statement. Hartlepool Power Station. Reactor 2 Statutory Outage 2013 ONR-CNRP-AR-13-019. (Trim 2013/137055).
- 15 NEA/CNRA/R (2012)2 – Nuclear Energy Agency, Committee on Nuclear Regulatory Activities, First Construction Experience Synthesis Report 2008-2011, Working Group on the Regulation of New Reactors (WGRNR), 03-May-2012.
- 16 ASME Section III, Nuclear Power Plant Components, ASME QAI-1, paragraph 3-3, Qualification for Authorized Nuclear Inspector Concrete.
- 17 The American National Board of Boiler and Pressure Vessel Inspectors, NB-263, Rules for Commissioned Inspectors.
- 18 The Royal Academy of Engineering: Engineering the Future- Nuclear Construction Lessons Learned Guidance on Best Practice: Concrete, 2012. ISBN: 1-903496-81-0.
- 19 ASME BPVC-XI-2017 ASME Boiler and Pressure Vessel Code, Section XI: Rules for In-service Inspection of Nuclear Power Plant Components.
- 20 ONR-CEEH-LTT-003, Fixings in Structural Concrete. (Trim 2017/78106).
- 21 USNRC guide, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, 10CFR50, Appendix J.
- 22 IAEA Safety Standard Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants. NS-G-2.6.
- 23 IAEA Guidebook on non-destructive testing of concrete structures. IAEA-TSC-17, September 2002.
- 24 EDF Energy NGL Civil Structures and Works LC 28 Arrangements Intervention Project Overview. Sir Robert McAlpine Design Group Report DG/S/66834/TR/10004 Revision A July 2013. Trim 2013/261782.
- 25 IAEA Safety Glossary 2018 Edition.
- 26 Civil Engineering Course - Prestressed Concrete Pressure Vessels and Containments - 02 CM9 2018/327155.
- 27 IAEA Specific Safety Guide SSG-53 Design of the Reactor Containment and Associated Systems for Nuclear Power Plants.

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- 28 Smith, LM, Brodt, GL, and Stafford, B “Performance Assessment and Reinstatement of Vibrating Wire Strain Gauges in Nuclear Power Plant Structures”, International Association for Structural Mechanics in Reactor Technology, 16th International Conference on Structural Mechanics in Reactor Technology, (SMiRT 16) , August 12-17 2001, Washington, DC, USA.
- 29 EDF NGL Wire Winding Inspection Phase 2 Report HYA R1 (Trim 2012/425545)
- 30 EDF NGL Wire Winding Inspection Phase 2 Report HYA R2 (Trim 2012/425558)
- 31 EDF NGL Wire Winding Inspection Phase 2 Report HRA R1 (Trim 2012/425571)
- 32 EDF NGL Wire Winding Inspection Phase 2 Report HRA R2 (Trim 2012/425577)
- 33 IAEA Nuclear Energy Series, NP-T-3.5, “Ageing Management of Concrete Structures in Nuclear Power Plants”, International Atomic Energy Agency, January 2016, Vienna, Austria.
- 34 New Nuclear Power Plants: Generic Design Assessment Guidance to Requesting Parties, ONR-GDA-GD-006.
- 35 New Nuclear Power Plants: Generic Design Assessment Technical Guidance, ONR-GDA-GD-007.
- 36 NEA/CSNI/R(2015)5, “Bonded or Unbonded Technologies for Nuclear Reactor Pre-stressed Concrete Containments”, OECD-NEA, Paris, France, June 2015

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ABWR	Advanced Boiling Water Reactor
AGR	Advanced Gas-cooled Reactor
ALARP	As low as reasonably practicable
AP1000	A reactor type
APEX	Appointed Examiner
ASME	American Society of Mechanical Engineers
BCU	Boiler Closure Unit
BOP	BCU Oversight Panel
DBA	Design Basis Accident
DBE	Design Basis Earthquake
EPR	Enhanced Pressurised (Water) Reactor
ESR	External steel restraint
ggbs	ground granulated blast furnace slag
GDA	Generic Design Assessment
HTGR	High Temperature Gas-cooled Reactor
IAEA	International Atomic Energy Agency
IoF	Incredibility of failure
LOCA	Loss of coolant accident
Magnox	Magnesium non-oxidising (a reactor type)
MDL	Minimum Design Load
PCPV	Pre-stressed Concrete Pressure Vessel
pfa	pulverised fuel ash
PSA	Probabilistic Safety Analysis
PSR	Periodic Safety Review
PVCW	Pressure Vessel Cooling Water
PWR	Pressurised Water Reactor
RGP	Relevant Good Practice
SAP	Safety Assessment Principle(s)
s/c	steel/concrete composite
SQEP	Suitably qualified and experienced person
SXB	Sizewell B power station
rebar	steel reinforcing bars for concrete
rc	reinforced concrete
TAG	Technical Assessment Guide(s)
USNRC	United States Nuclear Regulatory Commission
VWSG	Vibrating wire strain gauge
WENRA	Western European Nuclear Regulators' Association

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8. APPENDICES

APPENDIX 1: LIST OF RELATED TAGS

NS-TAST-GD-013	External Hazards
NS-TAST-GD-014	Internal Hazards
NS-TAST-GD-017	Civil Engineering
NS-TAST-GD-026	Decommissioning
NS-TAST-GD-076	Construction Assurance

APPENDIX 2: LIST OF SAPS REFERED TO IN THIS TAG.

Principle	Title	Subtitle
Civil Engineering SAPs within the Engineering Principles		
ECE.1	Engineering principles: civil engineering	Functional performance
ECE.2	Engineering principles: civil engineering	Independent arguments
ECE.3	Engineering principles: civil engineering	Defects
ECE.6	Engineering principles: civil engineering: design	Loadings
ECE.8	Engineering principles: civil engineering: design	Inspectability
ECE.12	Engineering principles: civil engineering: structural analysis and model testing	Structural analysis and model testing
ECE.13	Engineering principles: civil engineering: structural analysis and model testing	Use of data
ECE.14	Engineering principles: civil engineering: structural analysis and model testing	Sensitivity studies
ECE.15	Engineering principles: civil engineering: structural analysis and model testing	Validation of methods
ECE.16	Engineering principles: civil engineering: construction	Materials
ECE.17	Engineering principles: civil engineering: construction	Prevention of defects
ECE.18	Engineering principles: civil engineering: construction	Inspection during construction
ECE.19	Engineering principles: civil engineering: construction	Non-conformities
ECE.20	Engineering principles: civil engineering: in-service inspection and testing	In-service inspection and testing
ECE.21	Engineering principles: civil engineering: in-service inspection and testing	Proof pressure tests
Other SAPs within the Engineering Principles		

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EKP.3	Engineering principles: key principles	Defence in depth
ECS.3	Engineering principles: safety classification and standards	Codes and standards
ECS.4	Engineering principles: safety classification and standards	Absence of established codes and standards
EMT.3	Engineering principles: maintenance inspection and testing	Type testing
EMT.6	Engineering principles: maintenance inspection and testing	Reliability claims
EAD.2	Engineering principles: ageing and degradation	Lifetime margins
EAD.4	Engineering principles: ageing and degradation	Periodic measurement of parameters
ECV.2	Engineering principles: containment and ventilation: containment design	Minimisation of releases
ECV.3	Engineering principles: containment and ventilation: containment design	Means of confinement
NOTE: Additional SAPs may be required in specific circumstances		

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OFFICIAL**APPENDIX 3: GUIDANCE ON THE ASSESSMENT OF SAFETY CASES FOR EXISTING PRE-STRESSED CONCRETE PRESSURE VESSELS FOR GAS COOLED REACTORS****General**

- A3.1 Gas cooled reactors in the UK are uranium oxide fuelled within a graphite core moderator and pressurised carbon dioxide coolant that transfers the heat produced by the nuclear reaction to boilers that in turn provide steam for electricity generation. The reactor core, gas circulators and boilers are contained within a pre-stressed concrete pressure vessel (PCPV). The contained plant are not the subject of this TAG.
- A3.2 Typical dimensions of a cylindrical PCPV in the UK fleet are in the range of 30m internal diameter with 5m thick concrete walls and 7m thick top and bottom slabs. However, the Magnox type PCPVs at Wylfa NPS are formed by a vertical stepped cylinder with an internal diameter of 96 ft and minimum concrete thickness of 9ft. Both types have an internal insulated and water cooled mild steel liner approximately 20mm thick, anchored to the inner surface of the concrete.
- A3.3 Pre-stressing is provided by post tensioned steel tendons housed in un-grouted ducts in a variety of arrangements depending on the type of PCPV.
- A3.4 More details on the types of PCPV, tendon arrangements and including their monitoring and inspection, can be found within the ONR Civil Engineering Course, “Pre-stressed Concrete Pressure Vessels and Containments” [26].
- A3.5 The concrete and the liner are maintained below predetermined safe temperature limits by internal insulation a system of cooling water pipes fixed to the concreted face of the liner and the PCPV penetrations.
- A3.6 The PCPVs are free-standing structures supported on a system of flexible bearings on massive slab foundations that transfer the heavy PCPV loads down to rock.
- A3.7 This guidance relates to safety case revisions and the maintenance of existing PCPVs that were designed and constructed to historic standards. The PCPV provides the reactor pressure boundary during normal operation, fault and severe accident conditions of the reactor.
- A3.8 The various designs of the existing UK PCPVs have been able to withstand the test of time with only minor variations to the originally conceived safety cases. Under the Periodic Safety Review (PSR) process, the original designs have been assessed and confirmed using up to date finite element analysis techniques validated against original proof pressure test results and accounting for the operational time history loadings and instrumentation results. However, there remains a need for assessment criteria that can be applied at future PSRs and in assessing a vessel's suitability for return to service following Statutory Outage inspections, reactor fault excursions or the discovery of a defective or degraded strength component.
- A3.9 Due account of known and potential degradation should be taken into consideration during future assessments. More advice on degradation can be found in IAEA Nuclear Energy Series, NP-T-3.5, “Ageing Management of Concrete Structures in Nuclear Power Plants”, [33].
- A3.10 A detailed description of each PCPV pre-stressing system is given in Multi-Design Consultants’ Report ‘The Measurement of Tendon Loads in Pre-stressed Concrete Pressure Vessels and Containment Systems’ [11]. The report gives a detailed historical account of each pre-stressing system including the design and development work leading up to initial stressing operations. There is also an examination of each

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system during its operational life as revealed by periodic anchorage load checking operations carried out under the site licence conditions up until February 2000.

- A3.11 British Standard 4975:1990 [12] specifies the design, construction, inspection and testing of PCPVs for nuclear reactors, including components that are necessary to maintain the structural integrity and leak tightness of the vessels. Whilst the design of the Magnox type PCPVs at Wylfa and Oldbury, as well as some of the earlier AGR type PCPVs, predate BS 4975, this standard remains in use as the prime standard for the structural assessment of the operational PCPVs.
- A3.12 The safety of a PCPV depends on a correct assessment of the loadings likely to be applied during its operational life and on the proper design of the PCPV to resist these loadings. Safety is then maintained by the application of defined procedures being adopted for operation of the reactor with adequate monitoring and inspection of the PCPVs during their service life being carried out. In assessing a PCPV's suitability for continued operation assessors therefore need to take into account the operating history of the reactor in respect of vessel temperature, CO₂ gas leakage, operating pressure and PVCW leakage as well as the findings from inspections where the effects of ageing may be apparent. Any proposed changes to operating parameters should be analysed and assessed against BS 4975 taking into account the current, aged, structural capability of the PCPV.
- A3.13 BS 4975 notes that "The structural integrity of a PCPV depends on the joint action of the concrete, the pre-stressing system, and bonded reinforcement if present. The ultimate load is reached when the structure reaches a condition at which it can no longer transmit the forces required for equilibrium". In existing designs of AGR and Magnox PCPV types, the ultimate load capacity is primarily dependent on the integrity of the pre-stressing system. When this system consists of a large number of tendons, the premature failure of a small number of these is unlikely to significantly reduce the strength of the PCPV. The capacity of the tendons for additional extension beyond that imposed by pre-stressing also provides assurance that the transition from elastic behaviour to the collapse mechanism is progressive, gradual and detectable.
- A3.14 Guidance is given below on the application of the SAPs [1] to the assessment of the results from periodic inspections under station Maintenance Schedule arrangements and the assessment of potential changes to PCPV safety cases due to changes in operating or fault conditions, and to ageing effects.

Periodic measurement and inspection

Engineering principles: ageing and degradation	Periodic measurement of parameters	EAD.4
Where parameters relevant to the design of plant could change with time and affect safety, provision should be made for their periodic measurement.		
Engineering principles: civil engineering: in-service inspection and testing	Inspection and testing	ECE.20
Provision should be made for inspection, testing and monitoring during normal operations aimed at demonstrating that the structure continues to meet its safety functional requirements. Due account should be taken of the periodicity of the activities.		

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A3.15 To ensure the safe operation of PCPVs, ongoing assurance of its continued structural integrity is provided by the programme of inspections and measurements carried out by the Licensee's Appointed Examiner (APEX) and reported at each Statutory Outage. The scope of the examination is based on BS 4975 and is detailed in each Station's Maintenance Schedule, and normally covers:

- Condition of the accessible concrete surfaces of the vessel and support structure;
- Pre-stressing tendon anchorages;
- Tendon loads;
- Pre-stressing strand withdrawal, examination and tensile testing;
- Vessel settlement and tilt;
- Vibrating wire strain gauge monitoring;
- Vessel temperatures;
- Main reactor coolant loss;
- PCPV cooling water loss;
- PCPV top cap deflection;
- Review of operating history.
- ESR – but only for pod type boilers

A3.16 It is notable that PCPVs are not subject to periodic pressure/leak tests as per PWRs.

A3.17 The areas of interest that may influence the test regime include:

- The degree of relaxation of the pre-stress in the steel tendons.
- The influence of concrete creep on the pre-stress in the tendons and the changes to the stress regime in the concrete.
- The location of cooling water leaks and CO₂ coolant leaks, and their potential to degrade the concrete and corrode the steel tendons.
- Changes to the concrete temperatures during operation and the significance of failures to the thermocouples that monitor this.

A3.18 As a result of the above examinations and measurements, the APEX prepares a report on the Statutory Outage inspections and is usually able to make a recommendation that the PCPV is fit for a further specified period of service. In reviewing the APEX's Report the assessor should be satisfied that all the Maintenance Schedule requirements have been addressed, that any recommendations from previous APEX reports have been implemented and that the conclusions drawn are consistent with the findings from the inspections and measurements. If not all the previous recommendations can be closed out before the next start-up, it is important that they have been reviewed and fitness for purpose safety arguments made to justify start-up with the recommendation still outstanding.

A3.19 The passage of time and the consequent relaxation of tension in the pre-stressing tendons results in a diminishing margin of pre-stress above the originally specified Minimum Design Load (MDL). The MDL, a value used in the design process, is taken to be the tendon anchor load that should be exceeded by the arithmetic mean of all the residual tendon anchor loads in the PCPV. The use of the arithmetic mean value in the

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design process is considered acceptable where the relative stiffness of the tendon system and the concrete mass render minor variations in tendon load unimportant.

- A3.20 Residual tendon anchor loads are periodically measured by testing at least a 1% sample of the tendons on each PCPV and the mean value of each periodic sample is calculated for each PCPV. Often top and bottom tendon anchorage loads are considered separately and in the case of AGRs with helically wound tendons, the mean effective anchorage load at the top and bottom anchorages should both equal or exceed the MDL. The periodic test sample does not provide the true arithmetic mean of the population of tendon anchorage loads; it provides an estimate of it. The inaccuracy in the estimate will be insignificant to the assessment where the variance of the test load data is small, and where the margin between the MDL and the arithmetic mean of the test loads (as estimated by the sample) is large. Where the variance is large and the margin is small, the test data should be considered at an appropriate confidence level. An appropriate confidence level will depend on the consequences of the MDL being breached, for each PCPV.
- A3.21 It should also be considered that the MDL is not a specified design safety limit in BS 4975. The limiting design condition is the compressive stress in the vessel concrete, and the relationship between the MDL and the compressive stress is calculated by computer analysis. Whilst a large margin exists between the estimated mean tendon load and the MDL for a PCPV it is convenient and sufficient to use the MDL as a benchmark to demonstrate the presence of an adequate safety margin. At some sites, this margin is becoming relatively small and might, in the near future, reduce to zero. This would not necessarily mean that the PCPV has inadequate safety margins, rather that the safety case for continued operation should be made on the basis of the computer analysis and the calculated compressive stress in the concrete.
- A3.22 When assessing safety margins, consideration should be given to any uncertainty as to the condition of the tendon strands and the effectiveness of the corrosion protection grease applied to the tendons at the construction stage. This is particularly important where a tendon has been subsequently wetted by pressure vessel cooling water (PVCW) leakage or subjected to combined PVCW and CO₂ leakage. In addition, whilst the periodic inspection of a sample of tendons gives some reassurance of their integrity, it must be remembered that there will always be a proportion of tendons that may never be inspected due to access difficulties or obstructions that are too costly in time and effort to remove.
- A3.23 Generally, where operating environments that were not anticipated when the periodic measurement regimes were devised, are currently known or suspected to have occurred, the measurement regimes should be amended to suit. For example, where tendons are known to have been wetted by PVCW leakage, the sampling regimes should be extended and targeted to all known and suspected areas of wetting and known wetted tendons should be considered for replacement.
- A3.24 On PCPVs where the location and arrangement of tendon anchors facilitates safe access to test, inspect and if necessary replace suspected damaged or wetted tendons, it should be expected that these tendons are replaced promptly as a matter of routine.
- A3.25 On PCPVs that must be taken out of service to facilitate safe access to the tendons, pre-outage inspections of tendon anchors in known and potentially wetted areas of the PCPV should be undertaken to inform the location and scope of detailed inspections and tendon strand replacements to be undertaken during each outage. The locations of water and CO₂ leaks should be determined ahead of the outage and investigated prior to return to service of the reactor. Laboratory examinations of a sample of

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damaged tendon strands along with their protective grease layer should be undertaken to identify the cause of the damage. There is an expectation that the causes of tendon damage and of the degradation of their protective grease layer should be clearly understood and that wetted strands are replaced before return to service of the PCPV. Where this expectation is not met, the potential consequences should be fully understood and a programme of further mitigation prepared on an ALARP basis.

Functional performance and loading.

Engineering principles: civil engineering	Functional performance	ECE.1
The required safety functional performance of the civil engineering structures under normal operating and fault conditions should be specified.		

Engineering principles: civil engineering: design	Loadings	ECE.6
Load development and a schedule of load combinations, together with their frequencies, should be used as the basis for structural design. Loadings during normal operating, testing, design basis fault condition and accident conditions should be included.		

- A3.26 For the operational PCPVs the main concern is any change to the extant normal operating or claimed fault loading conditions that may be introduced to take account of potential changes in the safety case of other reactor components. For example, the potential for increased reactor over-pressure due to an increased likelihood of boiler tube or boiler spine failure; or increased over-pressure due to an increased risk of clogging of safety relief valves due to debris in the main reactor coolant. Similarly, the concrete temperatures expected during normal operation or fault conditions may be increased due to a revised safety case for undetected blockage or leakage of the PCPV cooling systems (PVCS).
- A3.27 In some key areas of the PCPV (such as the top-cap), the development of differential temperatures may be critical to the design of the concrete structure. Temperatures are monitored using thermocouples and vibrating wire strain gauges that were cast into key locations of the concrete.
- A3.28 Operational temperatures in the PCPV may also gradually rise due to deterioration of the vessel internal insulation; most likely in the upper parts. All these examples may effectively result in changes to the safety functional performance requirement of the existing civil engineering structure of the PCPV. The assessor should therefore be aware that apparently unrelated safety case revisions may have a combined effect on the PCPV safety case and should establish that the Licensee has fully taken this into account in his revised safety case, i.e. that the required safety functional performance of the civil engineering structure should be fully understood and specified.

Independent arguments.

Engineering principles: civil engineering	Independent arguments	ECE.2
For structures requiring the highest levels of reliability, multiple independent and diverse arguments should be provided in the safety case.		

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- A3.29 Where changes to a reactor component safety case may impose increased demands on the existing PCPV safety case the structural integrity of the vessel should be reassessed and shown by the Licensee to meet the requirements of SAP paragraphs 337 and 338 and to establish what safety margins exist.
- A3.30 If a reduction in the MDL is proposed as part of a revised safety case the assessor should establish that an adequate safety margin remains bearing in mind the aged condition of the reactor vessel strength components, the potential for future ageing and the inherent uncertainty in the knowledge of the actual aged condition of some of the pre-stressing tendons.
- A3.31 It should be borne in mind that most of the AGR type PCPVs were designed to allow for the future return to the original design condition by re-stressing of the pre-stressing tendons and the Licensee's arguments as to why the original MDL should not be restored by this means should be established by the assessor. Regardless of the justification provided for a reduced MDL it must be recognised that adoption of a reduced MDL automatically gives a real reduction in the safety margin that must be considered together with any other long term ageing effects that may also reduce margins.
- A3.32 Gradual long-term increases in the ambient concrete temperature of the upper half of the vessel are apparent in some PCPVs such that the average tendon load measured at the PCPV Top Cap is markedly lower than the average measured at the Bottom Cap. Where this is the case, the prevailing tendon anchor loads should be considered separately in the top anchorages and the bottom anchorages for comparison of each against the MDL.

Structural analysis for safety case changes.

- A3.33 Where changes to an extant safety cases are supported by structural analysis or model testing this should meet the requirements of SAP Engineering Principles ECE.1, ECE.12, ECE.13, ECE.14 and ECE.15 with respect to the demonstration that the PCPV can fulfil its safety functional requirement over the lifetime of the facility based upon analysis which meets these ECE requirements as to the use of data, sensitivity studies and validation methods.
- A3.34 A number of finite element analyses have been carried out by the licensees for the PCPVs to establish the effect on safety function of various types of faults or defects. For example, the effect of failure of groups of pre-stressing tendons; of asymmetric loss of tendon load; and of general loss of tendon load. These analyses give some understanding of the behaviour of the vessel in the event of unexpected findings at periodic inspections and are an aid to judgement for the assessor but specific case-by-case justifications should be requested from the Licensee.
- A3.35 When assessing a defect specific safety case the assessor should bear in mind the accumulated body of defects justified in previous defect specific safety cases. Where new finite element analysis is presented to justify a reduction in the pre-stressing MDL due to age related tendon relaxation it remains arguable whether the associated reduction in safety margin should be considered acceptable when the means are available to restore the original design margins by re-shimming. Re-shimming involves re-tensioning the tendons and re-installing the anchors, and it should be noted that given the large number of tendons in a typical PCPV such an exercise would be a major undertaking. As noted above, by definition, a reduced MDL results in a reduced safety margin during both normal operation and during fault conditions.

Defects and ageing.**OFFICIAL**

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Engineering principles: civil engineering	Defects	ECE.3
It should be demonstrated that safety-related structures are sufficiently free of defects so that their safety functions are not compromised, that identified defects can be tolerated, and that the existence of defects that could compromise safety function can be established through their life-cycle.		

Engineering principles: ageing and degradation	Lifetime margins	EAD.2
Adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on structures, systems and components.		

- A3.36 Where periodic outage inspections find the existence of defects in replaceable components of the pre-stressing system the default expectation should be that the component be replaced. Where defects are due to a common cause, for example, due to wetting from PVCW leakage resulting in corrosion of pre-stressing tendons, the assessor should request the removal of sufficient adjacent tendon strands for inspection and testing such as to establish to full extent of the potential defectiveness. The assessor should also have an expectation that the Licensee should seal the leak sources causing the corrosion problem to prevent further degradation.
- A3.37 It has been found that even some newly replaced tendons have been severely corroded after a relatively short period in a wetted environment, [13 & 14]. Therefore, in instances where leak sealing is found to be ineffective, tendons known to be subjected to wetting (along with adjacent tendons that could also be affected) should be scheduled for interim inspection between outages with particular attention being paid to the condition of their grease protection.
- A3.38 Defects such as cracks in the visible external concrete face of the vessel are subject to long term monitoring by the PCPV APEX. Gradual but permanent increases in concrete temperature due to changed ventilation or minor variations in operating conditions may cause increased concrete cracking due to drying shrinkage. In the event that significant changes in crack width or extent of cracking are found, or if new cracks are found, the assessor should be satisfied that the cause of the changes is understood and that potential adverse trends in crack development are acceptable for the next operating period.
- A3.39 Thermocouples embedded in key locations of the PCPV concrete record temperatures during the phases of operation of the reactor. Over time the thermocouples cease to provide reliable data.

Pressure vessel cooling systems

- A3.40 The AGR type PCPVs are generally provided with two 100% duty pressure vessel cooling systems (PVCS) cooling circuits of unlined small-bore steel pipe work either attached to the back of the vessel liner or feeding penetration cooling jackets. In some instances, the two PCPV Top Cap cooling circuits may not be truly independent of each other but have some interdependence. The objective of the PVCS is to keep concrete temperatures within prescribed limits and therefore to limit the development of adverse temperature gradients within the concrete and around penetrations. During periods of prolonged reactor shutdown the PVCS is used to circulate heated water around the vessel to maintain concrete temperatures within the specified range.

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- A3.41 To prevent corrosion of the internal surfaces of the PVCS pipe work the PVCW is demineralised, de-aerated and dosed with lithium hydroxide to achieve a pH of around 10 to 10.5. Notwithstanding these measures, leaks from the PVCS into pre-stressing tendon ducts are relatively common with the potential for corrosion of wetted pre-stressing tendons. At Hartlepool power station, the phenomenon of minor CO₂ pressurisation of some PVCW circuit legs has also been noted. This is due to ingress from a CO₂ leak site adjacent to a PVCW pipe leak site.
- A3.42 Each PCPV has a safety case for partial or full loss of PVCS which invokes the use of the Low Pressure Back-Up Cooling System, a once through flow arrangement which can provide a limited amount and duration of cooling. Safety cases are also in place for total loss of cooling with and without natural circulation of the primary CO₂ gas coolant and for longer-term post event cooling whereby in some situations elevated concrete temperatures may be maintained during depressurisation to prevent concrete cracking.
- A3.43 The Heysham 2, Torness and Hunterston B PCPVs have all been affected at various times by minor blockages of parts of the PVCS with the potential for increased concrete and penetration temperatures. Blockages have been caused by small sized debris in the system becoming trapped at the throttled back flow control valves. The potential exists for coincident blockage of both the 100% duty dual A and B circuits leading to elevated concrete temperatures and unacceptable thermal gradients.
- A3.44 There is the potential for loss of PCPV penetration restraint with subsequent penetration ejection or of failure due to elevated temperature of the penetrations and surrounding concrete known to be 'non-tolerant' to elevated temperature. Some extant safety cases for loss of PVCS require periodic flow measurements. At Statutory Outages (and at other periods required by the safety case) the assessor should ensure that any reductions in flow identified by flow checking are within the requirements of an extant safety case or that plant adjustments have been carried out to restore flows and where necessary, flow filtering is taking place at these sites to remove the cause of the blockages.

Vibrating wire strain gauges

- A3.45 During construction, vibrating wire strain gauges (VWSG) were installed at key positions within the concrete of each PCPV to monitor the performance of the PCPV during the proof pressure test. A full set of instrumentation generally being provided on the lead PCPV at each station with a reduced set installed on the second PCPV. This instrumentation is not a requirement of the long-term safety case but continues to be electronically monitored by the stations, reviewed by APEX and reported to ONR at each Statutory Outage for each PCPV.
- A3.46 With the passage of time, a proportion of the VWSGs have ceased to function or now give erratic results. However overall trends in compressive strain are still apparent and as such the instruments still give good evidence of PCPV performance. For some PCPVs, work has been carried out to compare the results of time history finite element analysis of PCPV behaviour, from proof pressure test onward, to the recorded strain variations with a good degree of correlation being found. The VWSG results show the long-term trend in PCPV performance and give an indication of the development of zones of low compressive or small tensile stresses as the pre-stressing relaxes with time. The assessor should maintain ONR's expectation of the Licensee that the strain gauge results continue to be collected, assessed by the APEX (including comparison with available finite element analysis results) and reported to ONR.
- A3.47 During the Boiler Closure Unit (BCU) recovery project at Heysham1 and Hartlepool power stations, it was found that it was possible to re-activate and re-baseline some of

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the VWSGs within the BCU concrete previously thought to have failed completely. It has therefore become possible to re-activate presently 'failed' VWSGs within the body of the PCPV concrete should this become necessary in the future. More recently, the use of modern data loggers has increased the number of active gauges at some stations. More guidance can be found in the paper "Performance Assessment and Reinstatement of Vibrating Wire Strain Gauges in Nuclear Power Plant Structures [28].

A3.48 It should be the expectation that recovery and reactivation of VWSG continues.

PCVCs with "Pod type" boilers

A3.49 The PCPVs at Hartlepool and Heysham 1 Power stations include "pod type" boilers within the thickness of the walls of the PCPVs. These preclude the installation of helical pre-stressing tendons within the walls and compressive pre-stress is provided via layers of tensioned steel wire wrapped around the external cylindrical surface of the PCPV, and vertical pre-stress tendons within the walls around the boilers.

A3.50 Routine inspection indicated corrosion damage to the tensioned steel wire at a number of locations around the external cylindrical surface of the PCPV and a major intrusive investigation was conducted. The outcome is recorded in four reports [29], [30], [31] and [32] by the licensee. The outcome has influenced future inspection requirements.

A3.51 The Boiler Closure Units (BCUs) within the PCPVs at Hartlepool and Heysham 1 Power stations have a concrete infill within a steel shell and contain the boiler inlet and outlet pipe work. They are also maintained in a state of compressive pre-stress by means of layers of tensioned steel wire wrapped around the external cylindrical surface in a wire-winding chamber. The original safety case was based upon the believed presence of a benign corrosion environment within the BCU wire winding chambers such that there was no degradation in the pre-stressing wires. However, inspections of the anchorages and windings enclosure found the presence of water from the pressure vessel cooling water system (PVCW) leakage and identified a number of instances of broken wire due to corrosion failure, corroded and thinned wires and instances of anchor slippage.

A3.52 In 2008 through 2009, a revised safety case based upon the modification of fitting lightly tensioned pre-stress lock-in bands over the top of the existing pre-stressing wire windings was prepared by the Licensee. This modification was accepted by ONR on the basis that there was sufficient confidence in the existing levels of pre-stress, due to the property of frictional restraint within the wire-winding bundle, such that even with a number of wire breaks the pre-stress level within the concrete would be maintained. The modification included the installation of a dewatering system to each BCU, a humidity controlled nitrogen environment and instrumentation to monitor the level of tension in the pre-stressing lock-in bands. The revised safety case specifies constant monitoring of the wire-winding environment and of the tension in the lock-in bands with phased inspection over Statutory Outages of the wire winding anchorages and lock in bands. The Licensee monitors the long-term condition of the BCUs, and adherence to the revised safety case, by means of the BCU Oversight Panel.

A3.53 Because further PVCW leaks into some wire winding chambers have occurred since the modification was carried out the assessor should ensure that the choice of and number of BCU enclosures chosen for inspection at Statutory Outages adequately reflects the likelihood of further corrosion degradation at each BCU. It should be noted that the Licensee's choice of BCUs for Statutory Outage inspections has a tendency to be dictated by the programme for PCPV penetration weld inspections, the timing of which may not be compatible with the requirement for BCU winding chamber inspection as indicated by the monitoring results from the previous operational period.

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Where excursions from the safety case have occurred, such as failure of the controlled environment in the winding chamber, the assessor shall seek confirmation that the controlled environment has been re-established in the shortest possible time and that any further degradation that may have occurred remains within the safety case envelope.

- A3.54 The Licensee's safety case for the modifications recognised that the BCU could no longer be claimed as an Incredibility Of Failure (IoF) component but that the fitting of an external steel restraint (ESR) to prevent ejection of the central section of the BCU concrete, brought the BCU back into the 'tolerable if ALARP' range. ONR recognises but do not use the term IoF but accepts that the ESR provides a secondary restraint system in the event of partial pre-stressing failure. The ESR is composed of frameworks of heavy cross section steel members bolted to the main BCU holding down studs and framed around the steam, feed and super-heater penetrations that pass through the BCU. To give a partial forewarning of failure of the BCU, the compressive strain in key members of the ESR is monitored during reactor operation.
- A3.55 A disadvantage of the ESR is that at periodic outages, where inspection of the penetration welds or of the BCU wire windings is required, the ESR must be completely dismantled and removed before inspections can take place. Since the ESR is fabricated to tight tolerances and the load bearing feet in contact with the BCU concrete are fitted with shims to guarantee load transfer under BCU failure conditions, (and to avoid the ESR loading the BCU during normal operation), the dismantling and re-installation of the ESR requires careful control. The assessor should establish that adequate checks have been carried out post re-installation to confirm that the ESR has been correctly reassembled to effectively fulfil its safety function.
- A3.56 The Licensee's safety case for the modifications recognised that the additional BCU condition monitoring required to meet the safety case would require oversight and instituted the BCU Oversight Panel (BOP). The work of the BOP is available for review by ONR and provides a forum for Hartlepool and Heysham 1 Power stations to discuss monitoring record data with the Licensee's Central Technical Organisation.

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OFFICIAL**APPENDIX 4: GUIDANCE ON THE ASSESSMENT OF SAFETY CASES FOR PRE-STRESSED AND REINFORCED CONCRETE CONTAINMENTS, FOR LIGHT WATER REACTORS****General**

- A4.1 This guidance relates to the design and construction of new containments to modern standards; some guidance on in-service maintenance is included. These containments do not provide the reactor pressure boundary but serve to provide a containment to confine escaped reactor coolant.
- A4.2 New designs may have undergone a Generic Design Assessment (GDA). Inspectors should become fully aware of the outcome of that GDA that may include issues to be taken forward to the design. GDA assessments of containments for EPR and ABWR reactors can be located on CM9 and were published on the ONR web site. The containment for an AP1000 type reactor is not a civil engineering structure and falls outside the scope of this guidance.
- A4.3 Reactor containments have several safety functions. Under normal operating conditions, the containment may provide support to the reactor vessel, primary circuit and associated plant. However, during fault and accident conditions which may result in reactor primary circuit leakage when the interior of the containment becomes pressurised the function is to contain any escaped coolant and reactor core inventory against release to the outside environment. Reactor containments do not require cooling.
- A4.4 Reactor containments may be constructed from steel plate, steel reinforced concrete, steel/concrete composite units or pre-stressed concrete. Steel plate containments are outside the scope of this guidance. They may act in combination with secondary containments or reactor building external envelopes to protect the reactor pressure vessel and associated safety systems from internal and external hazards (including aircraft impact). The reactor building may also provide a secondary containment. They may also provide biological shield functions and provide a degree of security from unauthorised access. Examples include:
- ABWR, reinforced concrete with a mild steel liner with secondary containment provided by a reinforced concrete Reactor Building. Sections of the structures may use steel/concrete composite units.
 - PWR and EPR, pre-stressed concrete with mild steel liner and a reinforced concrete secondary containment envelope.

Primary containment liners

- A4.5 The primary containment of the PWR and ABWR type reactor containments supports a steel liner that forms an impermeable and gas tight membrane to the inner faces of the containment concrete and is usually secured to the concrete by means of steel studs and steel channel sections. It also acts as an inner shutter when the concrete is poured. The floor of the liner is usually covered by several metres of mass concrete. At support points for plant and equipment and around access hatches and penetrations the liner thickness is increased locally. The transition from thin to thick steel plate is accommodated by tapered thickness plates, to avoid a stress-raising step-change in liner thickness.

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- A4.6 The primary containment steel liner to the ABWR (and other) type reactor containments may be formed from steel/concrete composite units acting to provide reinforcement to the concrete and/or permanent formwork.
- A4.7 Where a vented containment is proposed and the maximum containment design pressure and temperature have been reduced by virtue of the ability to vent to atmosphere via a filtered system, the assessor should ensure that the containment has the structural capacity to accept the full unvented pressure albeit that the structure may go into the inelastic range. This provides a safeguard against the possible blockage of filters and the consequent need for unfiltered venting and provides resilience against more severe accident conditions.
- A4.8 Due to its relatively small stiffness the liner's behaviour will be dictated almost entirely by the behaviour of the concrete structure. Strains and stresses will be induced in the liner by deformation of the concrete structure, along with stresses from restrained thermal expansion of the liner. Liner stresses are therefore strain controlled. Thus, gross cracking of the concrete or extensive overheating of the liner accompanied by high strain cycles would have to occur before any serious damage to the liner could develop.
- A4.9 The assessor should establish that the design of the steel liner, liner plate welds, embedded retention components and thickness transition plates is such that differential strain rates that will arise in fault conditions and at elevated temperature are accommodated without damage to the liner. It should also be established that the liner would not tear in the event of the stud anchors failing first.
- A4.10 Before construction of the liner begins, the assessor should ascertain that a design justification of the specified construction tolerances with respect to local alignment, bowing, dishing and ovality is in place to ensure that no unaccounted for strains are accumulated due to construction errors. Where erection of the liner will be significantly in advance of concreting a temporary wind brace structure may be required to control construction strains and ensure stability.
- A4.11 Where the liner is buried in concrete the assessor should establish that adequate means of pressure testing buried welds are provided such that it can be confirmed that these welds remain pressure tight after the possible disturbance due to concrete placing. Where novel forms of design are used such as use of s/c units or a 'floating liner base' it should be established by calculation to appropriate codes, mock up testing and post concreting pressure testing that the liner meets its safety functional requirements.
- A4.12 The assessor should establish that liner weld procedures, welder qualification, weld materials and the welding environment are all appropriate to the nuclear safety classification of the liner both for on-site and off-site fabrication.

Pre-stressing systems

- A4.13 PWR type concrete containments are constructed from pre-stressed concrete where the ultimate load capacity is primarily dependent on the integrity of the pre-stressing system. However, compared to the multitude of tendons provided in AGR and Magnox type PCPVs, a PWR type containment contains a relatively small number of tendons, the premature failure of which may not be readily apparent or detectable if the tendons are grouted in place.
- A4.14 The Sizewell B PWR type concrete containments is constructed from post-tensioned concrete with un-bonded tendons.

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- A4.15 The assessors expectation should be that the pre-stressing system be designed such that for all but the most extreme accident load cases the containment structure remains generally within the elastic range and has some degree of redundancy even at infrequent fault load conditions. The design should be robust to the undetected failure of several tendons, either together or distributed throughout the structure, such that their loss will not significantly reduce the PCC's ability to meet safety functional requirements. For design basis load conditions, the assessor should ascertain that there are no 'cliff edge effects' that result in structural instability in the event of a small further increase in loading. Some assurance is provided by the design capacity of the tendons for additional extension beyond that imposed by pre-stressing. This ensures that the transition from elastic behaviour to the collapse mechanism is progressive.
- A4.16 The assessor should establish that the values of modulus of elasticity and the stress/strain relationship of pre-stressing tendons to be used in the design have been evaluated by testing full-scale tendons of suitable length. This length should be sufficient to ensure a uniform loading of each of the bars, wires or strands forming the tendon.

Reinforced and pre-stressed concrete containments.

Engineering principles: containment and ventilation: containment design	Minimisation of releases	ECV.2
Containment and associated systems should be designed to minimise radioactive releases to the environment in normal operation, fault and accident conditions.		

Engineering principles: containment and ventilation: containment design	Means of confinement	ECV.3
The primary means of confining radioactive substance should be by the provision of passive sealed containment systems and intrinsic safety features, in preference to the use of active dynamic systems and components.		

- A4.17 Much of the guidance on the pre-stressing systems for PCPVs is relevant to the pre-stressing systems for containments.
- A4.18 In assessing the basic design parameters chosen by the designer for the structural analysis of a containment, the assessor should ensure that there is not an undue dependence for structural stability on the effectiveness of active pressure reduction systems in the determination of the peak internal design pressure and temperature. In the UK, it has been the design practice to ensure the containment structure remains essentially within the elastic range during the most likely fault and accident conditions.
- A4.19 PWR (light water) type reactors may also have a reinforced concrete secondary containment (Reactor Building) that should be able to withstand the possible pressurisation of the volume between the primary and secondary containments in the event of an accident or a malfunction of the interstitial annulus ventilation system, and should be able to withstand external loads either alone or in combination with the primary containment.
- A4.20 In the case of ABWR reactors, the secondary containment is provided by reactor building external envelope.

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Engineering principles: civil engineering	Functional performance	ECE.1
The required safety functions and structural performance of the civil engineering structures under normal operating and fault conditions should be specified.		

A4.21 Paragraph 337 of the SAPs requires that the containment design be based upon the use of sound design concepts and proven design features. Specific nuclear design standards should be used as appropriate, where such standards exist.

A4.22 The assessor should establish that a detailed schedule of loading for both serviceability and ultimate limit states such as:

- Normal operation.
- Plant transients
- Faults
- Internal and external hazards

has been prepared and that the design analysis covers potential failure modes for conditions arising from design basis faults and potential in-service degradation mechanisms.

A4.23 The assessor should be satisfied that sufficiently high safety margins are provided to ensure that for structure types that are inherently less ductile, failure would be extremely unlikely to occur for credible initiating events. Failure modes for severe loadings should be predictable, gradual and detectable in advance.

A4.24 The assessor should establish that only proven materials are used for construction and that high standards are applied, as verified by inspection of the construction work and of the materials used.

A4.25 The structural performance of the containment will be influenced by adjacent and adjoining structures and connected and supported plant. The containment around an ABWR type reactor is constructed integral with the reinforced concrete Reactor Building structure and this may be the case for other reactor types.

Independent arguments

Engineering principles: civil engineering	Independent arguments	ECE.2
For structures requiring the highest levels of reliability, multiple independent and diverse arguments should be provided in the safety case.		

A4.26 To support assessment of the design the assessor may seek evidence of design substantiation by alternative calculation and analysis routes. In this respect, the provision by the Licensee of Third Party endorsement of the design by means of independent analysis can provide the assessor with a high level of confidence in the design. Similarly, for assurance of construction compliance with specification and quality on site, the assessor may also seek endorsement of the construction process by a Third Party that is independent of the site-specific management and quality assurance arrangements.

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A4.27 It is worthy of note that the UK PWR reactor containment, at Sizewell B power station, was subject to the requirement that it be 'licensable in the country of origin'. To meet this requirement the integrity the structural design and construction of the containment had to be certified by the UK equivalent of an American 'Authorised Nuclear Inspector'. This was done by setting up an independent inspection agency function within the Client organisation to monitor the construction of the containment and to carry out an independent analysis of the containment design. Both design and construction were then able to be certified to American standards.

Codes and standards.

Engineering principles: safety classification and standards	Standards	ECS.3
Structures, systems and components that are important to safety should be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate standards.		
Engineering principles: safety classification and standards	Codes and standards	ECS.4
Where there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar equipment, in applications with similar safety significance, should be adopted.		

A4.28 The current (2019) UK nuclear new build programme is considering a variety of designs for the reactor containments. Some of these designs will have been subject to regulatory approval within the country of origin but the newer designs may still be the subject of an ongoing approval process. In considering a new containment design the assessor should establish the design standards and loading schedules on which the design is based and their degree of equivalence or otherwise to UK standards and Eurocodes. The assessor should also establish the degree of regulatory scrutiny that the design has undergone in the country of origin and whether there are any shortfalls or gaps in this assessment. A judgement can then be made as to where and how much further scrutiny should be applied. Much of this may have been considered during a GDA process. More information on the ONR GDA process can be found on the ONR web site in New Nuclear Power Plants: Generic Design Assessment Guidance to Requesting Parties, ONR-GDA-GD-006 [34] and New Nuclear Power Plants: Generic Design Assessment Technical Guidance, ONR-GDA-GD-007 [35].

A4.29 The applicability of non-UK design codes and material standards should be viewed with caution and assurance sought that the design intent of the codes can be met using available UK materials.

A4.30 Where novel forms of construction result in appropriate established codes or standards not being available, proposals to adopt other available codes and standards should be examined carefully and assurances sought that the adopted codes and standards provide a similar or higher degree of reliability to the design. The reliability provided by using alternative codes and standards should be quantified and compared against that provided by the use of established and accepted codes and standards.

A4.31 The combining of different codes and standards for a single aspect of a structure, system or component should be avoided. Where this cannot be avoided, the combining of the codes and standards should be justified and their mutual compatibility demonstrated.

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Design basis.

Engineering principles: civil engineering: design	Loadings	ECE.6
Load development and a schedule of load combinations,, together with their frequencies, should be used as the basis for structural design. Loadings during normal operating, testing, design basis fault and accident conditions should be included.		

- A4.32 The pressures and temperatures that apply within the primary and secondary containments under normal operating, fault, and accident conditions should be determined and validated. The safety function and safety categorisation of each part of the structure, plant and equipment that forms the containment boundaries to contain any nuclear matter arising should also be clearly delineated. The ultimate limit state design requirements for the containment boundaries will depend on the chosen combinations of loading.
- A4.33 For example, a loss of coolant accident (LOCA) may be considered coincident with an earthquake loading if it is deemed credible that an earthquake may induce, or occur at the same time as, a LOCA. Also for example, for the EPR type containment design, a load case combining rupture of the reactor pressuriser surge line, a LOCA and the design earthquake, results in a containment pressure combined with a design base earthquake (DBE). Additionally, for an EPR type containment severe accident with core melt, and including the loads due to local hydrogen deflagrations, the requirement is to demonstrate by calculation that the primary containment remains leak-tight at high pressure and temperature.
- A4.34 In addition, containments for ABWR type reactors are subject to the novel hydrodynamic loads in the Suppression Pool (a pool within the containment) due to condensation oscillation, pool swell, chugging and repeated operation of the safety relief valves.
- A4.35 The peak fault pressures and temperatures seen by some types of primary containment during a LOCA may be dependent on the reliability of containment emergency internal spray or heat removal systems activated to relieve or reduce peak pressure. The assessor should therefore establish that there is a consistent and validated set of arguments supporting the choice of design and extreme load combinations to which the containment must be designed. The forces on the containment induced by the activation of passive and/or active pressure and temperature reduction measures should be fully understood and quantified.
- A4.36 Special attention should be paid to the effects of temperature on the civil engineering containments during both normal operation and design basis accident (DBA) cases. It is not expected that the containment will see high temperatures, but if it does then some combination of insulation and cooling will be required to protect the inner surfaces of the containment to limit the temperature rises in the concrete. It should be noted that temperature induced stresses in the containment will influence the concrete reinforcement requirements and possibly result in cracking in the cooling phase.
- A4.37 For PWR and EPR type reactors it is expected that the secondary (outer) containment should be designed to withstand external loadings of human or natural origin. Serviceability should be demonstrated for loadings due to meteorological phenomena (snow, wind, extreme temperatures, etc.) Stability must be ensured for the DBE and for loadings corresponding to aircraft impact, external explosion and other internal hazard loads. To prevent releases of radioactivity in accident situations, the secondary (outer) containment design should be sufficiently leak-tight to enable a negative pressure to

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be maintained in the containment without the operation of the containment ventilation system for a specified grace period.

- A4.38 The purpose of the secondary containment is not to take over the functions of the primary containment should it fail. Its function is to collect potential leaks from the primary containment for filtered release. More detailed advice is available in 27 IAEA Specific Safety Guide SSG-53 Design of the Reactor Containment and Associated Systems for Nuclear Power Plants [26].

Structural analysis.

Engineering principles: civil engineering: structural analysis and model testing	Structural analysis and model testing	ECE.12
Structural analysis or model testing should be carried out to support the design and should demonstrate that the structure can fulfil its safety functional requirements over the lifetime of the facility.		
Engineering principles: civil engineering: structural analysis and model testing	Use of data	ECE.13
The data used in any analysis should be such that the analysis is demonstrably conservative.		
Engineering principles: civil engineering: structural analysis and model testing	Sensitivity studies	ECE.14
Studies should be carried out to determine the sensitivity of analytical results to the assumptions made, the data used, and the methods of calculation.		
Engineering principles: civil engineering: structural analysis and model testing	Validation of methods	ECE.15
Where analyses have been carried out on civil structures to derive static and dynamic structural loadings for the design, the methods used should be adequately validated and the data verified.		

- A4.39 The structural performance of a containment should be demonstrated to be conservative by a combination of analysis; design, model test and pressure test of the built containment. The design should generally meet the requirements of SAP paragraphs 337 to 339 to establish what safety margins exist.
- A4.40 Regardless of the work done by other regulators where an 'overseas' containment design is employed, the assessor should be satisfied that appropriate load combinations have been used; that sufficient numerical analysis has been carried out to demonstrate that the structure will meet the requirements of the SAPs; and that the computer codes used for design have been validated and the outputs verified. This may have been considered during a GDA process.
- A4.41 Validation of the computer codes used may be against model testing and/or against alternative computer codes. Validation may need to consider the limits of application of the calculation method, the structural representation in the model, comparison with

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other calculation methods, the level of quality assurance and user proficiency. The material properties used in the analysis shall also be shown to be validated by test.

- A4.42 In considering the design of a containment, it must be confirmed that all the component parts of the containment boundary have been identified: e.g. all structures that form part of the boundary, isolating systems, access penetrations, pipe work and cable penetrations. The safety functions of each element under normal, fault, and accident conditions should be defined together with appropriate criteria to measure their ability to fulfil these functions under the specified load combinations. In this respect the assessor should give particular attention to local membrane strains around large penetration groups, plant access hatches and personnel airlocks to establish that they remain within code allowables.
- A4.43 Structural interactions with non-civil components of the containment should be fully understood and considered in the analysis and design. For example, the top dome of the containment to an ABWR type reactor comprises is a metal plate fabrication, bolted to the upper perimeter of the reinforced concrete cylinder. Any subsequent changes to the design, detail or operation of these non-civil components should be considered by the reinforced concrete containment designers.
- A4.44 Calculations of beyond design basis conditions may involve the prediction of extreme physical behaviour and the calculation methods used are often not amenable to rigorous validation. In such cases, the results should be reviewed to ensure that they sensibly reflect the expected physical performance in broad terms. In addition, a containment ultimate capacity analysis is required.
- A4.45 The assessment should also consider whether the design has demonstrated that the containment's ability to meet its safety functional requirements is not impaired by the effects of internal and external hazards. Additionally SAP paragraph 525 Clauses a) to n) set out the general design requirements for any form of nuclear containment in the broadest sense. The requirement to establish a set of design safety limits and to define the performance requirements in the event of a severe accident including requirements for structural integrity and stability also apply in most part to containments. Much of this may have been considered during a GDA process.
- A4.46 Parallel structural analyses undertaken to support probabilistic safety analyses (PSA) should also be considered as they represent realistic margins to realistic failure mechanisms and they can provide guidance on which areas of the design are most critical.

Construction.

Engineering principles: safety classification and standards	Standards	ECS.3
Structures, systems and components that are important to safety should be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate codes and standards.		
Engineering principles: civil engineering	Defects	ECE.3
It should be demonstrated that safety-related structures are sufficiently free of defects so that their safety functions are not compromised, that identified defects can be tolerated, and that the existence of defects that could compromise safety functions can be established through their life-cycle.		

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Engineering principles: civil engineering: construction	Materials	ECE.16
Civil construction materials should comply with the design methodologies employed, and be shown to be suitable for the purpose of enabling the design to be constructed and then operated, inspected and maintained throughout the life of the facility.		
Engineering principles: civil engineering: construction	Prevention of defects	ECE.17
The construction should use appropriate materials, proven techniques and a quality management system to minimise defects that might affect the required integrity of structures.		
Engineering principles: civil engineering: construction	Inspection during construction	ECE.18
Provision should be made for inspection during construction to demonstrate that the required standard of workmanship has been achieved.		
Engineering principles: civil engineering: construction	Non-conformities	ECE.19
Where construction non-conformities or identified defects are judged to have a detrimental effect on integrity, remedial measures should be applied to ensure the original design intent is still achieved.		

- A4.47 During the recent (2014) round of containment constructions in France, Finland and China a number of construction problems have come to light which if not properly addressed could lead to shortfalls in safety functional performance during operation. OECD NEA Report, [15] analyses construction experience events at Flamanville 3, Olkiluoto-3 and Shin-Kori 1 and reports on the lessons learned from these events and proposes regulatory actions to aid in the prevention of safety events in the operating phase. The assessor should be aware of current containment design and construction issues (e.g. at Hinkley Point C) to ensure that these have been adequately addressed in the containment that is the subject of assessment.
- A4.48 In addition to the usual routine inspection of construction work carried out by the Licensee's Site Resident Engineer function it is also desirable to have additional Third Party inspection of the containment construction, and some Third Party testing of construction materials independent of the pressures of the site management function.
- A4.49 In the USA the construction of a containment has to be certified by an 'Authorised Nuclear Inspector', [16 & 17]. There is also a requirement for the design of the containment to be independently certified. For the construction of the UK's first Pressurised Water Reactor (PWR) containment an arrangement equivalent to the American Authorised Nuclear Inspector, known as the Independent Inspection Agency was set up and effectively carried out these functions. ONR's expectation is that appropriate inspection arrangements will be established for construction of each new containment.
- A4.50 The assessor should review the arrangements for categorising and sentencing non-conformances and be satisfied that sentencing is carried out at a design management level appropriate to the safety significance of the non-conformance.
- A4.51 Concrete. A number of reports are available detailing base slab and containment wall concreting problems where concrete cracking has occurred. To avoid problems with

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heat of hydration cracking, and the consequent need for remedial works to be agreed by the Regulator, the assessor should ensure that the Licensee carries out adequate concrete mix trials using the aggregates, cement, cement replacement, placing equipment and methods to be used for construction. Where it is proposed to pour a thick containment base slab in two or more layers the assessor should be satisfied that sufficient additional reinforcement is provided to ensure that the layers act monolithically.

- A4.52 More guidance on assessing the construction of thick concrete sections for containment can be found in Appendix 5.
- A4.53 Concrete. Following concrete mix trials the assessor should establish that an evaluation of the properties of the hardened concrete that are relevant to the behaviour of the primary containment has been carried out and the results are compatible with the values used in the containment design and analysis. An evaluation of the following properties would be expected:
- compressive strength
 - tensile strength
 - modulus of elasticity
 - poisson's ratio
 - creep
 - shrinkage/coefficient of thermal expansion
 - coefficient of thermal conductivity.

Where possible, the properties shall be determined under test conditions conforming to those expected in the containment under design, fault, and accident conditions.

- A4.54 Concrete. When the difficulties of placing and compacting concrete at a congested location may affect the resulting concrete properties or the behaviour of the containment structure under loading, appropriate concreting trials shall be carried out under conditions and configurations simulating the actual structure. For example, for a PWR type reactor, concreting trials may be appropriate to simulate the dome/ring beam and wall/base junctions and other highly reinforced areas. Further guidance is available in [18].
- A4.55 Steel plate. The AP1000 and other similar reactor types have a containment constructed from steel plate. The assessment of this type of reactor containment would be the responsibility of Structural Integrity assessors. In this case, an outer reinforced concrete and steel/concrete (s/c) composite unit structure provides a secondary containment function and protection from hazards including impact.
- A4.56 Steel plate. Where steel/concrete composite units, are used in the construction, sufficient full-scale construction trials should be conducted to demonstrate that the design intent would be met. Areas of the construction to be examined and/or demonstrated in the mock up should include:
- site steel fabrication details by non-destructive testing
 - method of concrete placement
 - monitoring of concrete temperatures
 - visual inspection of exposed concrete surfaces

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- preparation of construction joints
- preparation connection to reinforced concrete sections, including rebar coupler tests
- concrete strength and shrinkage measurement
- confirmation of concrete placement by cutting and/or coring of complex sections.

Pre-stressing system tests

Engineering principles: maintenance, inspection and testing	Reliability claims	EMT.6
Provision should be made for testing, maintaining, monitoring and inspecting structures, systems and components (including portable equipment) in service or at intervals throughout their life commensurate with the reliability required of each item.		

Engineering principles: civil engineering: design	Inspectability	ECE.8
Designs should allow key load bearing elements to be inspected and, where necessary, maintained.		

A4.57 Un-bonded tendons. In UK practice, it has been the norm to install un-bonded tendons firstly in the Magnox and then in AGR type PCPVs, and latterly in the Sizewell B PWR type containment such that it is possible to check the load in any tendon and, if necessary, re-tension it. Additionally, it is also possible to remove tendons for examination to assess their continuing integrity and, if necessary, to replace them in the event that corrosion or other defect is found. This facility gives continuing assurance of the serviceability of containments (and PCPVs) throughout their lives via a series of periodic checks (Statutory Examinations).

A4.58 Bonded tendons. Whilst pressure testing a containment structure with bonded tendons via a programme of integrated leak rate testing provides proof of overall structural integrity, there is little assurance of tendon integrity to be gained in the interim periods between tests. Therefore, where it is proposed to use bonded tendons the assessor should establish that a fully effective means of in-service monitoring of tendon behaviour is provided to give assurance of the tendons' continuing integrity throughout service life. The monitoring arrangements provided should be capable of giving a near equivalent level of in-service integrity assurance as would be obtained by the use of un-bonded tendons. The effectiveness of the long term monitoring instrumentation should be proven and validated as part of the pre-operational structural over-pressure test.

A4.59 Bonded tendons. The long-term structural integrity of a fully grouted (bonded) tendon is dependent on the completeness and quality of the grout body around the tendon within the tendon duct. The assessor should therefore be satisfied that the installation and grouting procedures have been fully tested to establish that the site construction

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methods will result in a fully contiguous grout body fully adhered to the tendon and duct. The effectiveness of the grouting method should be demonstrated at full scale using the actual grout material to be used in the structure.

- A4.60 Dome ring beam. One particularly difficult area of PWR reactor type containment design and construction occurs where the torispherical containment dome is connected to the cylindrical containment 'skirt' by means of a ring beam which may contain both vertical and horizontal pre-stressing and anchorages as well as bonded reinforcement. The assessor should be satisfied that sufficient trials have taken place to demonstrate that this complex junction can be properly constructed under site conditions without the inclusion of voids or poorly compacted areas in the concrete.
- A4.61 Further discussion on bonded and un-bonded (un-grouted) tendons can be found in - NEA/CSNI/R(2015)5, "Bonded or Un-bonded Technologies for Nuclear Reactor Pre-stressed Concrete Containments" [36].

Over-pressure test.

Engineering principles: civil engineering: in-service inspection and testing	Proof pressure tests	ECE.21
Pre-stressed concrete pressure vessels and containment structures should be subjected to a proof pressure test, which may be repeated during the life of the facility.		

- A4.62 ECE.21 also applies to reinforced concrete containments without pre-stress.
- A4.63 On completion of construction, an initial proof pressure test is carried out in order to demonstrate the elastic behaviour, strength and the leak tightness of the containment. The objective of the test is to verify that the containment can withstand the applied pressure with no visible damage; has acceptable maximum concrete crack widths at peak pressure; develops strains and deformations compatible with design calculations; and provides verification, after completion of the test, of the reversibility of deformations, (i.e. elastic behaviour).
- A4.64 The test only partly represents Loss of Coolant Accident (LOCA) accident conditions as it is performed by air pressure at ambient temperature. The outward thrust on the concrete resulting from thermal effects on the steel liner is not represented and this therefore justifies use of an over-pressure for the test.
- A4.65 The requirement for a structural over-pressure test is quite onerous in terms of the resources required to pressurise the containment, and in terms of the safety of site personnel. With the provision of suitable instrumentation, there is however much information to be gained on the performance of the structure but this must be balanced against the potential for local damage, cracking or weakening of the structure if the test is not carefully controlled. Pressure raising should be in discrete stages with pressure held at each stage for a period of time at each stage before the stage pressure is reduced slightly to allow personnel access to the external face of the structure for examination before progression to the next pressure stage.
- A4.66 It should be noted that undertaking a repeat pressure test at an operational site poses specific problems as any confined volume within the pressurised containment will experience the applied test pressure. Plant, equipment and instrumentation must therefore be adequately prepared (usually by removing them or venting them to the containment air).

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- A4.67 For the Sizewell B (SXB) PWR containment, the proof pressure test was undertaken in accordance with ASME Section III [19] Section CC Article CC-6000. The acceptance criteria being based on Sub-article CC-6400 at a test pressure of 1.15 times the design pressure.
- A4.68 At SXB, the over-pressure test was followed by a leak rate test in accordance with Appendix J to USNRC guide 10CFR50 [21] to ensure that the leakage through the primary containment (including penetrations) did not exceed the allowable leak.

Periodic measurement and inspection

- A4.69 Examination, maintenance, inspection and testing (EMIT) is required to ensure that the required safety and reliability of the containment will be maintained throughout the life of the plant. This should comply with LC28.

Engineering principles: ageing and degradation	Periodic measurement of parameters	EAD.4
Where parameters relevant to the design of plant could change with time and affect safety, provision should be made for their periodic measurement.		

Engineering principles: civil engineering: in-service inspection and testing	Inspection and testing	ECE.20
Provision should be made for inspection, testing and monitoring during normal operations aimed at demonstrating that the structure continues to meet its safety functional requirements. Due account should be taken of the periodicity of the activities.		

- A4.70 IAEA NS-G-2.6 [22] provides general guidance on frequency of inspections for PWR type reactor containments and IAEA SSG 53 Section 5 [27] gives guidance re inspections during construction, commissioning and in-service. Detailed guidance on the frequency and content of inspection of pre-stressed and reinforced concrete containments, for reactors, is provided in the ASME XI B&PV Code [19].
- A4.71 During the construction phase at site, a surveillance program should be produced that satisfies the requirements of ASME Code Section III [19] Division 2 Subsection CC, or similar.
- A4.72 During the operating phase, in-service visual inspections of the reinforced concrete components of the containment should be conducted at a frequency determined by the safety case and the outcome of previous inspections, but at intervals of not more than 5 years. The visual inspections should be undertaken by SQEPs and should consider the importance of each structural component, any changes to the loading and the local environmental conditions. Inspections should record concrete cracks and any change in their state since the last inspection, and any evidence of surface degradation such as spalling and the local environmental conditions. It may be appropriate to conduct tests for carbonation and rebar corrosion.
- A4.73 Pre-stressed and reinforced concrete containments for PWR type reactors include an internal integral mild steel liner that secures the pressure boundary in the event of a release of coolant from the reactor or the primary circuit. During the operation phase, the liner should be inspected and tested in conjunction with the concrete containment, at the refuelling outages. The inspections should include a visual inspection which for the SXB PWR is based on ASME XI [19] Subsections IWE and IWL.

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- A4.74 In addition, a containment leak test should be undertaken in accordance with Appendix J to USNRC guide 10CFR50 [21] to ensure that the leakage through the primary containment (including penetrations) does not exceed the allowable leak rate. Integrated Leak Rate Tests are repeated on the SXB containment currently every 15 years.
- A4.75 At SXB, inspections take place to a 10-year plan, with examinations and tests inside the containment being done during refuelling outages that occur every 18 months.

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OFFICIAL**APPENDIX 5: GUIDANCE ON THE ASSESSMENT OF SAFETY CASES FOR NEW CONSTRUCTION OF FUEL PONDS, DRY STORES AND WASTE STORES****General**

- A5.1. This guidance relates to the design and construction of new reinforced concrete fuel ponds and stores to modern standards. Some guidance on in-service maintenance is included.
- A5.2. New designs may have undergone a Generic Design Assessment (GDA). Inspectors should become fully aware of the outcome of that GDA that may include issues to be taken forward to the design.
- A5.3. IAEA Safety Guide No. SSG-15 [9] states that waste shall be stored in such a manner that it can be inspected, monitored, retrieved and preserved in a condition suitable for its subsequent management. Due account shall be taken of the expected period of storage, and, to the extent possible, passive safety features shall be applied. For long-term storage in particular, measures shall be taken to prevent degradation of the waste containment.
- A5.4. Short-term storage (conventional storage) is defined in IAEA Safety Guide No. SSG-15 [9] as storage that can last up to approximately 50 years. A short-term storage concept needs to include an end point that will be reached within this time period. If this is not possible, the safety considerations of long-term storage should be considered.
- A5.5. Long-term storage is considered in this IAEA Safety Guide No. SSG-15 [9] to be storage beyond approximately 50 years, and with a defined end point. The end point is important in determining the basis of the design life of the facility. Long-term storage is not expected to last more than 100 years, and this is based on technical experience with civil construction. The 100-year period is also judged adequate to allow enough time to determine future fuel management steps.

Waste Stores

- A5.6. Nuclear waste has a variety of solid, fluid and gaseous forms. It is generally classified as high, medium and low level waste depending on its nuclear inventory and radioactivity. The waste can be stored in its processed or unprocessed forms. The form of the wastes and their radioactivity determine the form, function and details of the waste stores.
- A5.7. Elements of the required support, security, shielding and containment functions are normally provided by the civil structure.

Fuel Ponds and Dry Stores

- A5.8. Spent fuel from nuclear power plants is usually stored in one of two types of facility:
- Wet storage in pools at or remote from a reactor site where shielding and elements of the containment are provided by the contained chemically treated water and the civil structures. The civil structure also contains the chemically treated water and may include one or more liners.
 - Dry storage in vault type facilities where shielding and elements of the containment is provided by the civil structures.

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A5.9. Spent fuel is not normally stored in mobile casks for long periods in the UK.

A5.10. Elements of the required support, security, shielding and containment functions are normally provided by the civil structure.

Design of fuel ponds, dry stores and waste stores

A5.11. A facility should provide a passive inherently safe containment for a period in excess of the design life of the store that will include allowance for decommissioning and clean-out. The design should demonstrate the required resistance to degradation of the construction materials and include a specification for future monitoring and tests. Consideration should be given to the potential for storage or operation beyond the specified design life.

Engineering principles: containment and ventilation: containment design	Minimisation of releases	ECV.2
Containment and associated systems should be designed to minimise radioactive releases to the environment in normal operation, fault and accident conditions.		

Engineering principles: containment and ventilation: containment design	Means of confinement	ECV.3
The primary means of confining radioactive substance should be by the provision of passive sealed containment systems and intrinsic safety features, in preference to the use of active dynamic systems and components.		

Structural performance.

Engineering principles: civil engineering	Functional performance	ECE.1
The required safety functions and structural performance of the civil engineering structures under normal operating and fault conditions should be specified.		

A5.12. The design should include multiple engineered containment barriers that are resistant to the specified design basis operating conditions, credible hazards and faults and extreme loads considered by the safety case.

Structural analysis.

Engineering principles: civil engineering: structural analysis and model testing	Structural analysis and model testing	ECE.12
Structural analysis or model testing should be carried out to support the design and should demonstrate that the structure can fulfil its safety functional requirements over the lifetime of the facility.		

Engineering principles: civil engineering: structural analysis and model testing	Use of data	ECE.13
The data used in any analysis should be such that the analysis is demonstrably conservative.		

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Engineering principles: civil engineering: structural analysis and model testing	Sensitivity studies	ECE.14
Studies should be carried out to determine the sensitivity of analytical results to the assumptions made, the data used, and the methods of calculation.		

Engineering principles: civil engineering: structural analysis and model testing	Validation of methods	ECE.15
Where analyses have been carried out on civil structures to derive static and dynamic structural loadings for the design, the methods used should be adequately validated and the data verified.		

A5.13. Appropriate analytical and design methods should be used that consider conservative but realistic combinations of design basis loads. Allowances should be made for uncertainties and the sensitivity of the design to known uncertainties should be quantified. All analyses should be verified and validated. The load combinations, design allowables and margins should be carefully documented and retained for future reference.

A5.14. The concept of defence in depth should be applied to the civil structure design features that provide the safety functions, normally shielding and containment. The use of multiple engineered containment barriers with provision for preventing, collecting and monitoring minor leakage represent relevant good practice. The required leak tightness of the civil structure containment barriers should take account of:

- The degree of radiological contamination of the contents, including any contained liquids.
- The chemical properties of the contained liquids.
- The potential for the degradation of civil structure containment barriers by action of the contents.
- The environmental consequences of leakage of the contents.
- The potential for harm to operators and the public from leakage of a radiological or chemically harmful inventory.

Codes and standards.

Engineering principles: safety classification and standards	Standards	ECS.3
Structures, systems and components that are important to safety should be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate standards.		

Engineering principles: safety classification and standards	Codes and standards	ECS.4
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Where there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar equipment, in applications with similar safety significance, should be adopted.

- A5.15. Appropriate design codes and materials should be selected to provide the required reliability throughout the design life of the civil structures. Where no standards apply, justification for the allowable design stresses should be provided. Also, to provide the required reliability, the fabrication and construction of the civil structures and their components should be undertaken to accepted industry and nuclear standards. The constructions and fabrications should be durable and include appropriate protection from degradation through the design life. Any required in-service monitoring, tests and maintenance should be clearly specified by the designer.
- A5.16. The construction materials should provide for easy decontamination of surfaces. In particular, consideration should be given to the provision of an impermeable barrier (liner) at surfaces regularly or permanently in contact with fuel pool water or waste, this is considered RGP. The migration over time of nuclear contamination into concrete and other permeable surfaces should be considered and prevented by design. The effects of dissolution of the construction material into the fuel pool or other contained fluid water should be considered. Where soluble boron may be used to control criticality, it is expected that protection to the inner surfaces of the containment will be provided.

Engineering principles: civil engineering: design	Loadings	ECE.6
Load development and a schedule of load combinations, together with their frequencies, should be used as the basis for structural design. Loadings during normal operating, testing, design basis fault and accident conditions should be included.		

- A5.17. The design basis should consider all loads and environmental conditions required by the safety case, including accidental overfilling of the containment. It should consider loads and conditions following the action of postulated accident events and consider continued storage of the nuclear inventory following such events. Cyclic thermal loads and their durations during construction, operation and accident conditions require special consideration as stresses induced by variations in thermal gradients, particularly through thick concrete sections, may exceed the tensile capacity of the construction materials (concrete) to result in cracking. The resultant leakage could include a loss of nuclear inventory and a potential for degradation of the construction materials.
- A5.18. The effects of potential fire should be explicitly considered in the design; where necessary fire protection should be provided and its effects on other design features considered.

Design basis.

- A5.19. Regardless of the work done by other regulators where an 'overseas' containment design is employed, the assessor should be satisfied that appropriate load combinations have been used; that sufficient numerical analysis has been carried out to demonstrate that the structure will meet the requirements of the SAPs; and that the computer codes used for design have been validated and the outputs verified. This may have been considered during a GDA process.

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- A5.20. The design should provide adequate and appropriate containment of radioactive materials to prevent release to the environment. Containment should be ensured by at least two independent static barriers and the design should make provision for the collection of any leakage. A reinforced concrete structure with a leak-tight liner is considered to provide a single barrier. Reinforced concrete structures are not leak tight but may provide bulk containment of fluids and may support a barrier such as a leak-tight membrane attached to inner surface of the concrete structure. Leakage of mobile radioactive material to ground may be prevented and collected by leak-tight membranes with a leak detection system constructed below the reinforced concrete structures, and their leak-tightness should be proven by appropriate tests during construction.
- A5.21. The necessary longevity of the barriers should be proven under the worst case operating conditions and their leak-tightness proven during construction. Provision should be made to monitor any leakage and the condition of the barriers in order to determine if corrective action is required to maintain or repair the barriers. Appropriate means for sampling groundwater should be provided around the containments; sampling boreholes are an appropriate means.
- A5.22. Leak-tight membranes should be designed to remain in an elastic state of stress under normal operating conditions. Their function and performance during and following fault and accident loads should be specified by the safety case. Performance testing and demonstration should be specified by the designers and should be in accordance with recommended nuclear industry guidance. Where leak-tight membranes are provided by plates of metal welded together, special consideration should be given to the leak-tightness and its demonstration by test at the welded joints between the plates. Leakage at the welded joints between the plates should be anticipated by the design and additional provisions made for collection of the leakage at those locations.
- A5.23. The design should provide adequate and appropriate radiation protection to workers and the public. Containment penetrations through the civil structures should be designed to maintain the required shielding. Penetrations should not be placed below the minimum water level required for adequate shielding, in fuel ponds.
- A5.24. Penetrations through the surrounding weather protection envelope should be designed to prevent the ingress of foreign matter that degrade the material or function of the containment structure. The penetrations should provide appropriate security measures.
- A5.25. Items cast into the structure of the containments should be demonstrated to be durable and resistant to degradation for the intended life of the plant. Expansive forces in the containment due to degradation of the cast-in items should be prevented by design. When specifying finishes, to provide the required durability of cast-in items and their required longevity and effectiveness over the operating life of the containment should be proved.
- A5.26. Structural anchors and plant fixings to the concrete containment structure should be specified at the design stage and should be cast into the concrete. Post-installed anchors and fixings that require drilling into the surface of the concrete should not be specified for containment structures. More guidance can be found in ONR-CEEH-LTT-003, Fixings in Structural Concrete [20].
- A5.27. Concrete reinforcement should be detailed, using appropriate nuclear codes and steel materials appropriate to ensure the structural ductility assumed in the

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analysis and design. Fuel ponds, dry stores and waste stores are often designed to resist seismic loads and the reinforcement should be carefully detailed to meet that requirement. The use of mechanical couplers should be considered in areas of high reinforcement congestion and where large diameter reinforcing bars are specified. Mechanical couplers should be specified and detailed during the design process, and not to facilitate site modifications.

- A5.28. Provision should be made in the layouts for inspection, decontamination and any necessary repairs to the internal and external surfaces of the containment. Provisions should be made for leak repairs.
- A5.29. Dropped loads and set-down loads present significant load cases to the design of the civil structures of the containments. The maximum credible dropped load impact cases should be considered. The required performance of the impacted and supporting civil structures should be clarified with the plant designers for the range of dropped loads and set-down loads determined by the safety case. Their required performance may include resistance to combinations of collapse, surface penetration, containment perforation, distal face material ejection (scabbing), cracking (surface and through thickness), damage to surface finishes (including liners) and deflection. A revision to the containment function of the civil structure due to failure of the containment function of the dropped load should also be considered.
- A5.30. Measures shall be taken to protect the containment structures from damage and degradation throughout construction and commissioning. These should include protection from low and high temperatures and solar radiation. Stresses due to construction should be considered; in particular, the stresses due to the heat of hydration when constructing thick concrete sections should be minimised by specifying an appropriate and proven concrete mix. The construction sequences, the location of movement and construction joints, minimum reinforcement quantities and timing of the construction of adjoining sections of concrete should be specified by specialist designers to reduce the occurrence of early thermal cracking. Provisions should be made in the construction details to repair cracks and leaks in the containment following leak testing. The leak test acceptance criteria should be determined based on the potential for consequential damage to the containment and supporting structures, in conjunction with the potential loss of nuclear material into the surrounding environment, from the leak.
- A5.31. Dry fuel stores provide weatherproofing to the stored material and should have provision for drainage and removal of unintended water ingress that does not compromise the containment and shielding functions of the store. The weather proofing of dry fuel stores should prevent the ingress of rainwater when operated under the decompression necessary to prevent the outward leakage of airborne contamination. Proprietary roofing systems often fail to provide the required reliability of leak tightness.

Construction of fuel ponds, dry stores and waste stores

- A5.32. Civil engineering structures that provide containment of nuclear material are normally constructed from reinforced concrete detailed to water retaining standards, to limit cracking. The structure may support an impermeable liner. Water retaining concrete is not watertight but can provide bulk containment and can be detailed and constructed to minimise leakage. Structural thicknesses are normally determined by radiation shielding requirements and this often results in thick concrete sections.

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A5.33. The type and strength of the concrete and reinforcing bars should be determined by the design to satisfy the strength demands of the load cases to meet safety case requirements. This part of the guidance relates to methodologies suitable to construct thick reinforced concrete structures to meet radiation shielding and containment requirements.

Engineering principles: safety classification and standards	Standards	ECS.3
Structures, systems and components that are important to safety should be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate codes and standards.		

Engineering principles: civil engineering	Defects	ECE.3
It should be demonstrated that safety-related structures are sufficiently free of defects so that their safety functions are not compromised, that identified defects can be tolerated, and that the existence of defects that could compromise safety functions can be established through their life-cycle.		

Engineering principles: civil engineering: construction	Materials	ECE.16
Civil construction materials should comply with the design methodologies employed, and be shown to be suitable for the purpose of enabling the design to be constructed and then operated, inspected and maintained throughout the life of the facility.		

Engineering principles: civil engineering: construction	Prevention of defects	ECE.17
The construction should use appropriate materials, proven techniques and a quality management system to minimise defects that might affect the required integrity of structures.		

Engineering principles: civil engineering: construction	Inspection during construction	ECE.18
Provision should be made for inspection during construction to demonstrate that the required standard of workmanship has been achieved.		

Engineering principles: civil engineering: construction	Non-conformities	ECE.19
Where construction non-conformities or identified defects are judged to have a detrimental effect on integrity, remedial measures should be applied to ensure the original design intent is still achieved.		

A5.34. The concrete mix should be designed and specified to an appropriate standard and placed to address the following requirements:

- Strength
- Workability
- Avoidance of cracking
- Durability

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- Longevity.

A5.35. The concrete mix design process involves the selection of suitable ingredients (aggregates, Portland cement, water and additives) and determining their relative proportions to achieve the required properties. The mix design should be undertaken by SQEPs and all the required properties of all mix designs should be proven by laboratory tests and field trials before installation into the site works.

A5.36. Concrete mixes for thick sections normally include pozzolanic cement replacement products such as pulverised fuel ash (pfa) or ground granulated blast furnace slag (ggbfs) to delay and reduce the heat of hydration of the Portland cement. This is to mitigate the heat of hydration of Portland cement in thick (>500mm) concrete sections that causes excessive thermal expansion of the concrete while still in a plastic state, soon after casting. The tensile stresses in the solid but weak concrete during subsequent cooling and contraction, lead to significant through thickness cracking if not controlled. The cracking is also greatly influenced by the external restraint applied to the concrete pour by an adjacent previously hardened concrete structure. The delay and reduction in the heat of hydration limits the thermal expansion and hence the cracking of the concrete during the cooling phase. Cement replacement using ggbfs also improves the durability and resistance of the concrete to chloride and sulphate attack. The influence of the external restraint on cracking can be mitigated by carefully sequencing and limiting the dimensions of each successive concrete pour. It is good practice to determine these concrete placing sequences and limits, during the design phase, when detailing the steel reinforcing bars to be cast into the concrete.

A5.37. The methods used to place and “cure” the concrete also affect the quality, durability, and hence the longevity of the final concrete. Fixing reinforcing bars and cast-in items and placing the concrete around them is a skilled operation and should only be undertaken by SQEPs.

A5.38. The methods of “curing” the concrete should include temperature control of the concrete that will require through thickness temperature monitoring and the facility to regulate the temperature of each concrete pour. Sacrificial sensors are normally cast into the concrete for this and may also be able to provide information on the maturity of each concrete pour. It may be necessary to modify the concrete mix designs and “curing” to prevent adverse early age temperature excursions. The sensors also provide real time information of the state of each concrete pour to facilitate other construction activities such as formwork removal. Records of adverse temperature excursions can provide an indication of concrete cracking that may require repair.

A5.39. The water retaining performance of reinforced concrete structures can be greatly enhanced by including water bars at construction joints. Water bars are also necessary at structural movement joints. Water bars are usually manufactured from extruded polyvinylchloride (pvc) based materials that are susceptible to radiation damage particularly in movement joints that may include a clear “line of sight” from the contained nuclear materials to the water bar. The water bar locations should be carefully designed to include radiation shielding to prevent radiation damage to the pvc materials. The longevity of the pvc in its proposed location should be proved by reference to accelerated radiation testing. The tests should account for the predicted hydraulic pressures and the range of movements to be applied during the design life of the containment. During the design phase, leakage should be anticipated at movement joints and the containment barriers and leakage collection provisions should be enhanced at these locations.

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- A5.40. The construction methods and details should anticipate the need to repair any cracked concrete that may lead to leakage of the contained water, or may cause durability issues. The need to repair should be considered and appropriate details included during the design phases.
- A5.41. Where a containment includes a water retaining liner, the requirement to limit concrete cracking remains to ensure the serviceability and durability of the concrete structure. Leakage at the joints in the liner should be anticipated by the design and additional provisions should be made for the monitoring and collection of the leakage at these locations.
- A5.42. The leak-tightness of all water retaining containments should be proved by full scale tests against criteria defined by the designers and in accordance with nuclear industry standards and safety case requirements. Provision should be made in the design and construction details to repair leakage to attain an effective containment of the nuclear inventory. Where practicable, provision should be made for in-service monitoring, inspection and repair of containments.
- A5.43. Post installed structural anchors and plant fixings should not be drilled into the surfaces of water retaining containment structures.
- A5.44. Concrete reinforcement should be fixed according to the details and specifications prepared by the designers. No site based detail changes should be permitted.
- A5.45. A civil engineering structure formed from steel-concrete (s/c) composite units can be constructed by placing concrete between two steel plates that serve as stay-in-place forms. The steel plates may also serve as reinforcement and, in this case, are connected to the concrete by steel shear studs welded to the internal faces of the plates. Unless only having the purpose of temporary formwork, this form of construction should only be used when specifically detailed for this purpose during the design process. The design should, in all cases, satisfy the requirements of the appropriate nuclear standards and the inner surface of the units should satisfy the requirements of a containment liner.

Ageing and periodic inspection of fuel ponds, dry stores and waste stores

Engineering principles: civil engineering: in-service inspection and testing	Inspection and testing	ECE.20
Provision should be made for inspection, testing and monitoring during normal operations aimed at demonstrating that the structure continues to meet its safety functional requirements. Due account should be taken of the periodicity of the activities.		
Engineering principles: civil engineering	Defects	ECE.3
It should be demonstrated that safety-related structures are sufficiently free of defects so that their safety functions are not compromised, that identified defects can be tolerated, and that the existence of defects that could compromise safety function can be established through their life-cycle.		
Engineering principles: ageing and degradation	Lifetime margins	EAD.2

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Adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on structures, systems and components.

- A5.46. Licence condition 28 [3] states “The licensee shall make and implement adequate arrangements for the regular and systematic examination, inspection, maintenance and testing of all plant which may affect safety”. Plant which may affect safety include civil structures and containments that support or confine nuclear material.
- A5.47. It is necessary to inspect, and sometimes to test, fuel ponds, dry stores and waste stores to ensure that they are able to continue to provide their nuclear safety functions up to and beyond the intended design life of the plant. External concrete surfaces can usually be accessed for inspection and testing from outside the containment, and this should ideally be done without damaging the concrete i.e. non-destructive testing. Non-destructive testing includes, but is not limited to [23]: visual inspection, Schmidt hammer rebound test, half-cell electrical measurement, carbonation depth measurement and permeability test.
- A5.48. All inspection programs should include a range of visual examinations to identify a variety of potential defects and degradation including cracks, spalling, disintegration, colour change, weathering, staining, surface blemishes and lack of uniformity. Visual inspections provide initial information on the condition of a structure and inform the development of a more specific test regime.
- A5.49. ONR expect that licensees will develop and implement appropriate inspections and tests and their arrangements for this should ensure compliance with LC28. The arrangements should be in written form and their implementation should be a mandatory requirement of the site management.
- A5.50. The licensee’s arrangements for inspection and tests should include the following provisions [24]:
- Visibility of safety case. The safety functions and plant life should be identified and made known to those undertaking the inspections.
 - Comprehensive procedures. Written schemes including a maintenance schedule, guidance documents, a database of known defects and causes, and a work control system should be in place and implemented.
 - Suitably qualified and experienced personnel should manage the provisions.
 - The inspections should be demonstrably complete and thorough while taking account of OPEX. A record of the relevant environmental conditions should be included in the inspection reports.
 - The significance to nuclear safety of any defects or degradation should be evaluated and remedial works sentenced on this basis.
- A5.51. It is important that the causes of any defects or degradation should be investigated and recorded; these are often due to adverse environmental conditions that may, in turn, be due to other building defects or lack of maintenance of the environmental protection. The remedy required to remove the cause of the defects or degradation may be essential in repairing the defects or degradation.

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