Form10/00

UK ABWR

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UK ABWR Generic Design Assessment

Preliminary Safety Report on Civil Engineering and External Hazards

Hitachi-GE Nuclear Energy, Ltd.

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Abbreviations and Acronyms

ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ALARP	As Low As Reasonably Practicable
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ASD	Allowable Stress Design
BAT	Best Available Techniques
BDB	Beyond-Design-Basis
BPVC	Boiler and Pressure Vessel Code
BSL	Basic Safety Levels
BSO	Basic Safety Objectives
B/B	Backup Building
CST	Condensate Storage Tank
CWS	Circulating Water System
C/B	Control Building
CRDH	Control Rod Drive Housing
CRGT	Control Rod Guide Tube
DB	Design Basis
DBA	Design Basis Accident
DBE	Design Basis Earthquake
D/F	Diaphragm Floor
D/W	Drywell
EDG	Emergency Diesel Generator
EMI	Electro-Magnetic Interference
EMIT	Examination, Maintenance, Inspection & Testing
EUR	European Utility Requirement
FE	Finite Element
GDA	Generic Design Assessment

HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
HPCF	High Pressure Core Flooder System
HVAC	Heating Ventilation and Cooling
Hx/B	Heat Exchanger Building
IAEA	International Atomic Energy Agency
IBA	Intermediate Break Accident
I&C	Instrumentation & Control
LBL	Large Break LOCA
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
L/D	Lower Drywell
MCR	Main Control Room
NPP	Nuclear Power Plant
OBE	Operational Basis Earthquake
PCSR	Pre-Construction Safety Report
PSA	Probabilistic Safety Assessment
RC	Reinforced Concrete
RCCV	Reinforced Concrete Containment Vessel
RCIC	Reactor Core Isolation Cooling System
RCW	Reactor Building Cooling Water System
RHR	Residual Heat Removal System
RPV	Reactor Pressure Vessel
RSW	Reactor Building Service Water System
Rw/B	Radwaste Building
R/B	Reactor Building
SBA	Small Break Accident
SGTS	Standby Gas Treatment System
SHA	Seismic Hazard Assessment
SIT	Structural Integrity Test
SMA	Seismic Margin Assessment
SFP	Spent Fuel Pool
SRP	Standard Review Pool
SRSS	Square Root of Sum of Squares

SRV	Safety/Relief Valve
SSI	Soil-Structure Interaction
S/B	Service Building
S/C	Suppression Chamber
SSCs	Systems, Structures and Components
T/B	Turbine Building
TCW	Turbine Building Cooling Water System
UHSp	Uniform Hazard Spectra
USNRC	The U.S. Nuclear Regulatory Commission

I. Civil Engineering

1 Introduction

This document describes the initial safety case on civil engineering and external hazards which is used for the design of the new build nuclear power plant in UK.

1.1 ABWR Generic Plant Scope

Following buildings are described in this document.

- Reactor Building (R/B)
- Control Building (C/B)
- Heat Exchanger Building (Hx/B)
- Turbine Building (T/B)
- Radwaste Building (Rw/B)
- Service Building (S/B)
- Backup Building (B/B)
- Others

1.2 Generic Site Envelope

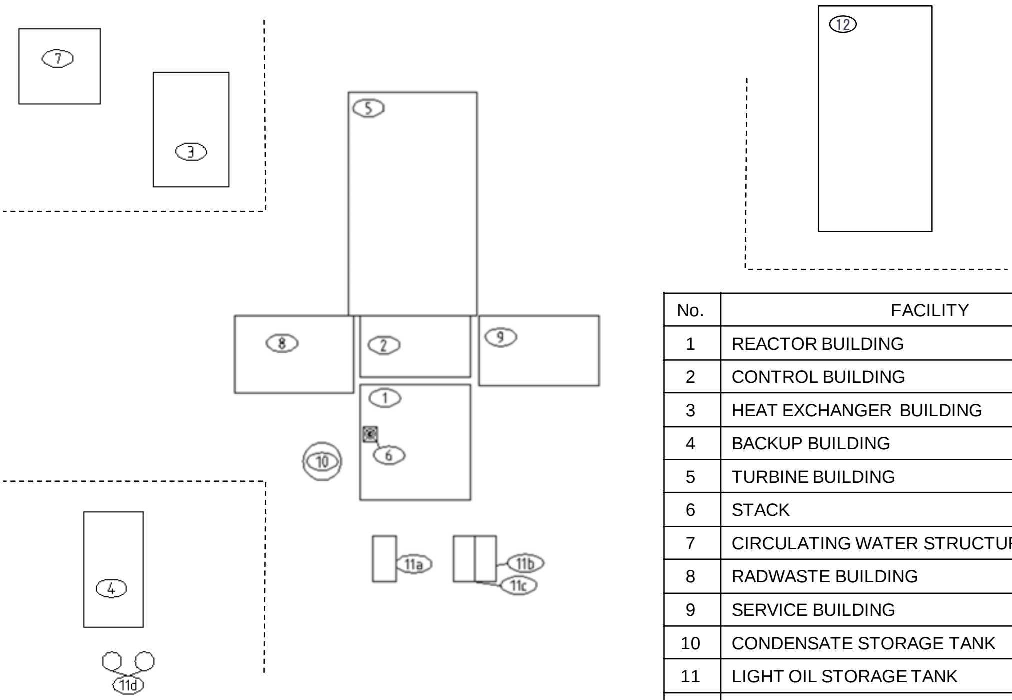
All the buildings and structures in the new build nuclear power plant in UK are designed based on the Generic Site Envelope of the ABWR. The Generic Site Envelope takes into account considerable naturally/not naturally occurring environmental conditions including external hazards. The Generic Site Envelope is described in GDA Step 1b document, “Generic Site Envelope” [RD 1]. Internal hazards are considered for design too and are described in GDA Step 1b document, “Internal Hazards Report” [RD 2].

2 General Plant Descriptions

2.1 Site and Layout of the Civil Structures

The standard UK ABWR plant is located on a site adjacent or close to a body of water with sufficient capacity for either once-through or recirculated cooling or a combination of both methods. The generic arrangement of the containment and buildings on the plant site is shown in Figure 2.1-1

The C/B that houses the main control room is located on the central location as the hub of the plant operating staff's activities. The T/B is oriented in such a way, that any plane perpendicular to the turbine generator axis does not intersect with the R/B and C/B in order to minimize the probability of a turbine missile striking any Safety Related Systems, Structures and Components (SSCs).



No.	FACILITY
1	REACTOR BUILDING
2	CONTROL BUILDING
3	HEAT EXCHANGER BUILDING
4	BACKUP BUILDING
5	TURBINE BUILDING
6	STACK
7	CIRCULATING WATER STRUCTURE
8	RADWASTE BUILDING
9	SERVICE BUILDING
10	CONDENSATE STORAGE TANK
11	LIGHT OIL STORAGE TANK
12	SPENT FUEL STORAGE FACILITY

Figure 2.1-1 Generic Site Plan

2.2 Functions and Features of the Civil Structures

2.2.1 Reactor Building

The Reactor Building includes the containment, drywell, major portions of the nuclear steam supply system, steam tunnel, refueling area, diesel generators, essential power, non-essential power, emergency core cooling systems, Heating Ventilation and Cooling (HVAC), as well as additional supporting systems. The secondary containment is a reinforced concrete building that forms the secondary containment boundary which surrounds the primary containment above the basemat. The secondary containment permits monitoring and treatment of any potential radioactive leakage from the primary containment. Treatment consists of utilizing High Efficiency Particulate Air (HEPA) and activated charcoal filtration systems. Safety functions in the R/B are protected against external hazards shown in Section10 including aircraft impact.

2.2.2 Control Building

The Control Building includes the control room, the computer facility, the cable tunnels, some of the plant essential switchgear, some of the essential power and the essential HVAC system. The main steam tunnel from the R/B to the T/B is located in the ground floor of the Control Building. Safety functions in the C/B are protected against external hazards shown in Section10 including aircraft impact.

2.2.3 Heat Exchanger Building

The Heat Exchanger Building is a structure which houses portions of the Reactor Building Cooling Water (RCW) System and Turbine Building Cooling Water (TCW) System. The Hx/B is located adjacent to the intake point of the plant cooling water.

2.2.4 Backup Building

The Backup Building provides an alternative safety management capacity for accident management. The building houses an alternative Alternating Current (AC) power source as well as Instrumentation & Control (I&C) facilities. An alternative water source is also available via a water tank adjacent to the building. The building also includes transportable RCW replacement, water injection pumps and Nitrogen supply facilities.

2.2.5 Turbine Building

The Turbine Building houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building.

2.2.6 Stack

The plant stack is located on the Reactor Building. The stack vents the R/B, T/B, Rw/B, and a small portion of the Control and Service buildings.

2.2.7 Circulating Water Structure

The Circulating Water Structures houses pumps and butterfly valves, these are a portion of the circulating water system. The Circulating Water System (CWS) provides cooling water for removal of the power cycle waste heat from the main condensers in the Turbine Building.

2.2.8 Radwaste Building

The Radwaste Building houses all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

2.2.9 Service Building

The Service Building is a structure which houses the main entrance of the plant, security and access control facilities, hot and cold laboratories, locker rooms, and operating support offices.

2.2.10 Condensate Storage Tank

The Condensate Storage Tank (CST) provides condensate quality water for both normal and emergency operations. CST is a water source for systems which require condensate makeup water. It also provides water to High Pressure Core Flooder System (HPCF) and Reactor Core Isolation Cooling System (RCIC).

2.2.11 Light Oil Storage Vault

The Light Oil Storage Vault is a structure which houses a light oil storage tank and fuel oil pump as a portion of oil storage and transfer system. The UK ABWR has three trains of the system with two individual structures. One vault is an independent structure for one train. And the other vault houses two trains with physical separation by concrete walls.

2.2.12 Spent Fuel Storage Facility

The Spent Fuel Storage Facility provides safe on-site storage for spent fuel casks.

2.2.13 Tunnels

The tunnels provide inter-buildings connection for piping, cables and personnel pathways.

3 Description of Safety Aspect of the Containment and Civil Structures

3.1 Design Requirements for Safety

This section describes general requirements on the containment and civil structures. Safety Requirements on structures, Design requirements on structures and Safety class for structures are described below.

3.1.1 Safety Requirements on Structures

This nuclear power plant facility consists of buildings including Reactor Building, Turbine Building, Main stack, and switchyard, etc, structures and components. These buildings, structures, and components are designed to have appropriate characteristics on operability, maintainability, structural integrity, confinement function, shielding function and related components.

General safety requirements/functions for buildings are shown below;

- To support Structures, Systems and Components (SSCs), which deliver safety functions, for Design Basis (DB) loads
- To maintain atmosphere appropriate for SSCs inside buildings
- To protect SSCs, which deliver safety functions, from Design Base natural phenomena and human-induced events for them, and
- To protect SSCs, which deliver safety functions, from Design Base condition by internal hazards, including the effects of missiles, pipe whipping, and discharging fluids etc.

In addition to the above, a building like the Reactor Building or some part of buildings has specific safety requirements/functions, such as;

- To contain radioactive material to limit a potential release to the environment, and/or
- To shield radiation to lower the acceptable level

Other requirements during Normal Operations and Fault Conditions are shown below.

a. Design Requirements for Safety during Normal Operations

Design requirements, which contain normal operation condition, are developed for maintaining its function at the level equivalent to non-nuclear-specific codes and standards, and applied for each structure, based on the following items;

- Protection against natural phenomena
- Human-induced events (man-made)
- Internal dynamic effects
- Fire protection
- Environmental conditions

b. Design Requirements for Safety during Fault Conditions

Design condition, which contain fault condition, are developed for maintaining its function at design basis fault and applied for each structure, based on following items;

- Protection against natural phenomena
- Human-induced events (man-made)
- Internal dynamic effects
- Fire protection
- Environmental conditions

3.1.2 Safety Class for Structures

Safety Classes for buildings are determined by the relation to those requirements/functions. If the buildings have only general safety requirements/functions, its Class depends on Class of SSCs housed. If the buildings have specific safety requirements/functions, its Class depends on its functions, which is confirmed by Fault Study.

Safety Categories of functions and safety classes for structures, which directly deliver safety function like above first one, are defined in “Categorisation and Classification of Structures, Systems, and Components” [RD 3], which shows three categories (Category A, B, and C) and three classes (Class 1, 2, and 3). These definitions are described below.

Category A safety functions play a principal role in ensuring nuclear safety in that they are associated with the removal of intolerable radiological risks from Design Base faults by either prevention of the risks or reduction of the risks to broadly acceptable levels.

Category B safety functions make a significant contribution to nuclear safety in that they are associated with the removal of radiological risks outside the Design Basis (DB) by either preventing the risks or reducing the risks to broadly acceptable levels for Foreseeable events and Beyond-Design-Basis (BDB) accidents, which are identified in fault studies.

Category C safety functions are those that do not fall into either of the above categories. They are mainly associated with the support of Category A or B safety functions or identified from ALARP or BAT analyses.

Class 1: Any SSC that provides a main or first line means of fulfilling a Category A safety function.

Class 2: Any SSC that provides a backup or second line means of fulfilling a Category A safety function, or provides the main means of fulfilling a Category B safety function.

Class 3: Any SSC that provides a Category C safety function. In exceptional circumstances, a third line of provision of a Category A safety function may be classified as Class 3.

Safety classes for the other structures are defined by following these rules, or alternatively, as noted below.

Auxiliary services that support components of a system important for the safety should be considered part of that system and should be classified accordingly, unless failure does not prejudice successful delivery of the safety function. These are treated as follows:

- Supporting systems directly needed for a competent system to fulfill its safety functions are considered to have the class equal to that of competent systems.
- Supporting systems needed for a competent system to maintain or assure its reliability but not directly needed to fulfill its safety functions are considered to have the importance that may be lower than that of competent systems. However, supporting systems for a competent system of Class 3 are a minimum of Class 3.

For example, containment, which has the function to contain radioactive material, is Class 1 in Category A. Class 1 component support function in reactor building is Class 1. Some part of reactor building, which has the function to contain radioactive material, is Class 2 in Category B.

3.1.3 Seismic Categorisation

SSCs are assigned to one of three seismic categories; Seismic Category 1, Seismic Category 2, and Seismic Category 3, which are defined using the following approach.

- The evaluation of SSCs shall be proportionate to the mitigated radiological hazard.

- Where the integrity of SSCs is evaluated to withstand seismic effects, the earthquake return period shall correspond to the radiological dose that is defined in the Fault Analysis. However, ‘the Initiating Event Frequency (per year)’ shall be adopted as ‘Earthquake Frequency’.

- Definition of Design Basis Earthquake (DBE) and Operating Basis Earthquake (OBE) is required for UK seismic design. In addition, it is required that the frequency of Design Basis environmental hazards is less than 10^{-4}/year.

- Normal industrial seismic design standards shall be applied if the radiological dose mitigated by the SSC is less than Basic Safety Objective (BSO).

Moreover, SSCs that are not classified as the highest seismic category (Seismic Category 1) but which satisfy the following conditions are classified as Seismic Category 1A.

- SSCs whose failure could lead to the failure of an adjacent Seismic Category 1 SSC.

- SSCs which are needed to perform a Beyond Design Basis Accident (BDBA) function after an earthquake or other hazard event.

The integrity of these SSCs must be maintained during and after the earthquake that applies to Seismic Category 1 SSCs.

Furthermore, the UK ABWR design principles require additional demonstration that the reactor design is robust against events more severe than that assumed for the Design Basis plant design, to demonstrate that no ‘cliff edges’ exist just beyond the Design Basis by completing a Seismic Margins Assessment (SMA).

SSCs are classified in Table 3.1.3-1 and Figure 3.1.3-1. As an example, seismic category of buildings is shown in Table 3.1.3-2

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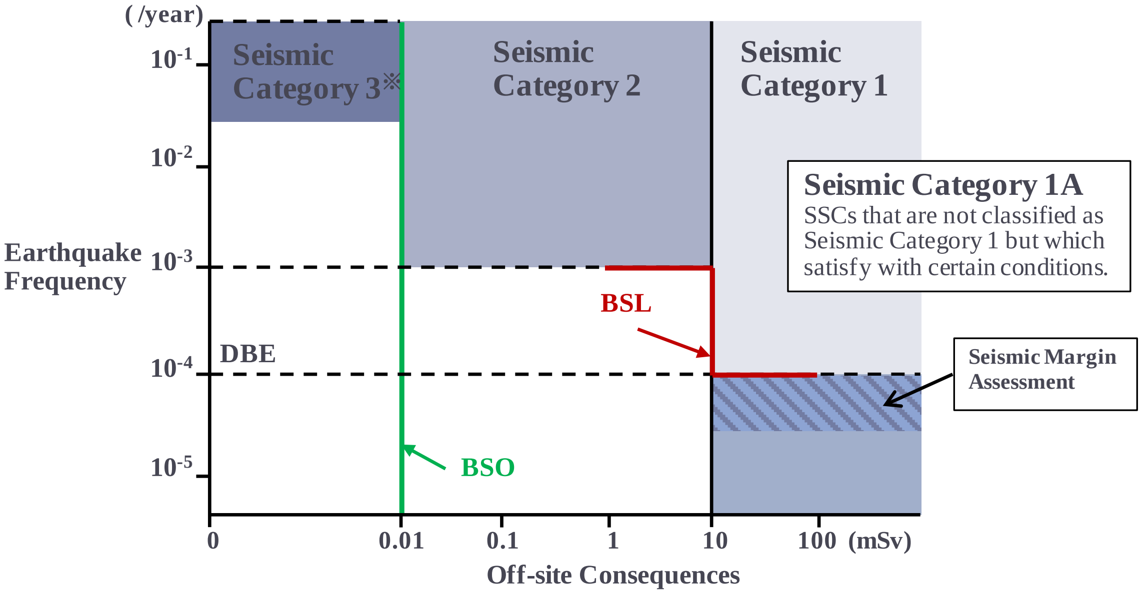
Table 3.1.3-1 Definition of Seismic Classification

Seismic Category	Off-site Consequences	Requirement
Seismic Category 1	Release > 10mSV	Confirm Integrity during and after DBE*2
Seismic Category 1A	- *1	Confirm Integrity during and after DBE*2
Seismic Category 2	10mSV> Release > 0.01mSV (BSO)	Confirm Integrity during and after 10^{-3}/year Earthquake*2
Seismic Category 3	Release < 0.01mSV (BSO)	Design by normal industrial standards. *3

*1 In case of SSCs satisfying certain condition
*2 See section 4.3.1.1.1.
*3 BS EN 1998-1:2004 [RD 4]



*: Refer to BS EN 1998-1:2004 [RD 4] for the return period applicable in the case of design to normal industrial standards.

Figure 3.1.3-1 Seismic Classification

Table 3.1.3-2 Seismic Category of Buildings (Example)

Seismic Category	Building
Seismic Category 1	R/B, C/B, Hx/B
Seismic Category 1A	T/B, B/B
Seismic Category 2	-
Seismic Category 3	S/B

3.2 Primary Containment Vessel

3.2.1 Structure Outline

The primary containment of the UK ABWR is a reinforced concrete containment vessel (RCCV) with an internal steel liner. The RCCV is integrated with the reinforced concrete reactor building. The structure includes various penetrations, equipment hatches and personnel access locks. This containment provides an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. The internal structures of the containment provide equipment support, radiation protection, and components for operation of the ABWR pressure suppression containment. The conceptual arrangement of the containment is shown in Figure 3.2-1.

The RCCV is constructed with the steel liner to the inside of the concrete. Metal Containment (MC) components such as the Drywell Head, hatches, personnel airlocks and penetrations also form part of the RCCV. The RCCV is a cylindrical vessel and consists of a top slab, shell and basemat. The RCCV is divided into a drywell (D/W) and a suppression chamber (S/C) by the diaphragm floor (D/F) and the Reactor Pressure Vessel (RPV) pedestal. The RPV pedestal is a double shell steel structure filled with concrete. The diaphragm floor is a reinforced concrete structure with a seal plate.

The D/W contains the RPV and the main steam pipes etc. The S/C contains a pool and an air space. The D/W and S/C are connected to each other via steel vent lines which are embedded into the RPV Pedestal. In the event of a Loss of Coolant Accident (LOCA), a mixture of steam and water is discharged in the D/W and is led to the S/C pool through the vent lines, where the steam is cooled and condensed. This can effectively suppress the increase of pressure within the RCCV.

The inside diameter of the RCCV is 29m, and the height from the upper surface of the foundation to the upper surface of the drywell head is 36m.

The thickness of the RCCV shell is 2m, which includes the steel liner thickness.

3.2.2 Design Requirements

The containment of the UK ABWR is designed to have the following functional capabilities:

(1) The containment structure has the capability to maintain its functional integrity during and following the peak transient pressures and temperatures which would occur following any postulated LOCA. A design basis accident (DBA) is defined as the worst LOCA pipe break (which leads to maximum containment and drywell pressure and/or temperature) and is further postulated to occur simultaneously with loss of offsite power and a design base earthquake (DBE).

The containment structure is designed for the full range of loading conditions consistent with normal plant operation and accident conditions, including the LOCA-related design loads in and above the suppression pool.

The containment structure is designed to accommodate the negative pressure difference between the drywell and wetwell and relative to the R/B surrounding.

(2) The containment structure and isolation system, with concurrent operation of other accident mitigation systems, is designed to limit fission product leakage, during and following the postulated DBA. Structural monitorings of the containment will be carried out. One of structural monitorings is the leak rate test of the RCCV. Other monitoring may include crack observation for the outer concrete surface of the RCCV and material test of the concrete.

(3) Capability for rapid closure or isolation of all pipes or ducts which penetrate the containment boundary is provided to maintain leakage within acceptable limits.

(4) The containment structure can withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.

(5) The containment structure is designed to accommodate flooding to a sufficient depth above the active fuel to permit safe removal of the fuel assemblies from the reactor core after the postulated DBA.

(6) The containment structure is protected from or designed to withstand hypothetical missiles from internal sources and uncontrolled motion of broken pipes which could endanger the integrity of the containment.

(7) The containment structure provides means to channel the flow from postulated pipe

ruptures in the drywell to the suppression pool.

(8) The containment system is designed to allow for periodic tests at the calculated peak or reduced test pressure to measure the leakage from individual penetrations, isolation valves and the integrated leakage rate from the structure to confirm the leaktight integrity of the containment.

The containment is classified as Safety Category A, Safety Class 1 and Seismic Category 1.

The Containment and the internal structures are designed and constructed to accommodate the dynamic and static load conditions and load combinations associated with the containment design basis accident;

(1) Live loads, dead loads, temperature effects and building vibration loads from normal plant operation.

(2) Earthquakes loads from DBE.

(3) Blowdown pressures and temperature from design basis loss-of-coolant accidents.

(4) Hydrodynamic loads and structural vibrations resulting from steam discharges into the suppression pool.

(5) Reaction forces on structures resulting from pipe break jets or fluid impacts.

This drawing is to illustrate the concept of the primary and secondary containment and not to intend to show the final detail design.

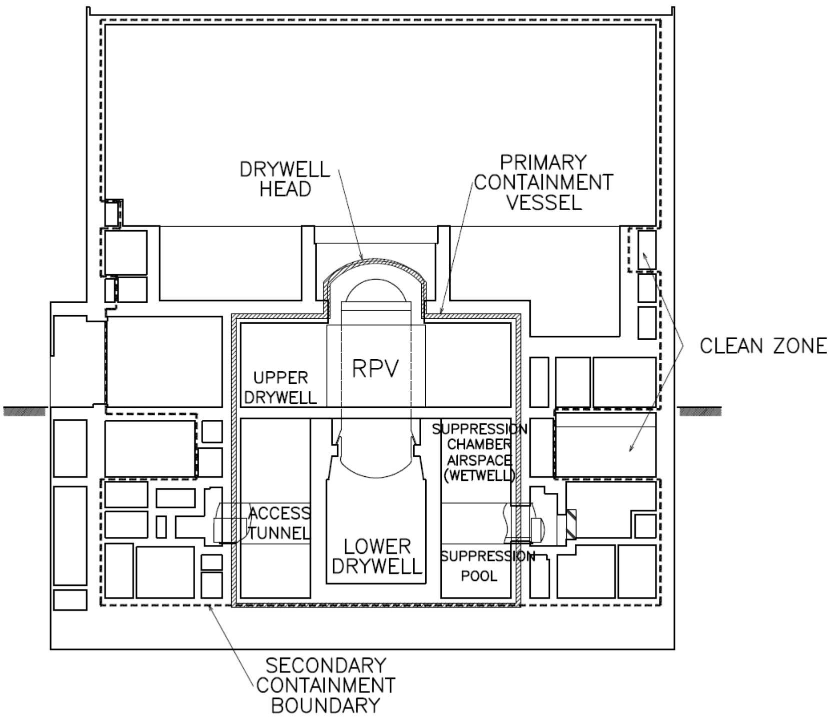


Figure 3.2-1 Conceptual Arrangement of the Containment

3.3 Secondary Containment

The secondary containment boundary completely surrounds the primary containment vessel (PCV) except for the basemat and, together with the clean zone, comprises the Reactor Building (R/B) as shown in Figure 3.2-1. The secondary containment encloses all penetrations through the primary containment and all those systems external to the primary containment that may become a potential source of radioactive release after an accident. During normal plant operation, the secondary containment areas are kept at a negative pressure with respect to the environment and clean zone by the HVAC System. Following an accident, the Standby Gas Treatment System (SGTS) provides this function.

Fission products that may leak from the primary to secondary containment are processed by the SGTS before being discharged to the environment. The HVAC exhaust systems and SGTS are located within the secondary containment to assure collection of any leakage. The secondary containment provides detection of the level of radioactivity released to the environment during abnormal and accident plant conditions. Personnel or material entrances to the secondary containment consist of airlocks with interlocked doors or hatches.

3.4 Reactor Building

3.4.1 Structure Outline

The R/B is constructed of reinforced concrete and structural steel with a steel frame and reinforced concrete roof. The R/B has four stories above the ground level and three stories below. The R/B encloses the primary containment Vessel. The R/B and the Secondary Containment share structural and barrier walls, and penetrations. The R/B slabs and fuel pool girders are constructed monolithically with the RCCV. The R/B, together with the RCCV and the RPV pedestal, are supported by a common basemat. The conceptual section of the R/B is pictured in Figure 3.4-1.

3.4.2 Design Requirements

The Reactor Building protects the equipment required safe and orderly shutdown equipment from adverse site-related environmental events (e.g., seismic, flood, storm, wind, snow, etc.). The Reactor Building provides environmental controls to safety related equipment during normal operation and plant transients. The Reactor Building also houses and provides spacial, physical and electrical separation to other divisional separation zones or compartments (e.g., Emergency DG Rooms, Emergency Electrical Equipment Rooms) to protect safety functions against internal hazards such as fire and flooding.

The R/B is classified as Safety Category A, Safety Class 1 and Seismic Category 1.

The R/B is designed and constructed to accommodate the dynamic and static loading conditions associated with the various loads and load combinations.

(1) Natural phenomena; wind, floods, tornados, earthquakes, rain and snow.

(2) Internal events; floods, pipe breaks and missiles.

(3) Normal plant operation; live loads, dead loads, temperature effects and building vibration loads.

3.4.3 Protection against Internal Hazards

The R/B is divided into three separate divisional areas for mechanical and electrical equipment and four divisional areas for instrumentation racks. Inter-divisional boundaries are designed to prevent fire and flooding in one division from propagating to other divisions.

Internal hazards are described in Section 9 of this document.

3.4.4 Protection against External Hazards

Seismic analysis is performed for the generic site conditions to ensure the seismic adequacy of the building.

External walls are provided with sufficient thickness and flood protection features such as watertight sealing of penetrations to prevent seepage of external flood

Safety functions in the R/B are protected against external hazards shown in Section10 including aircraft impact.

Aircraft impact assessment is performed for reactor core, spent fuel pool and these cooling systems are prevented against intentional and accidental aircraft impact.

External hazards are discussed in Section 10 of this document.

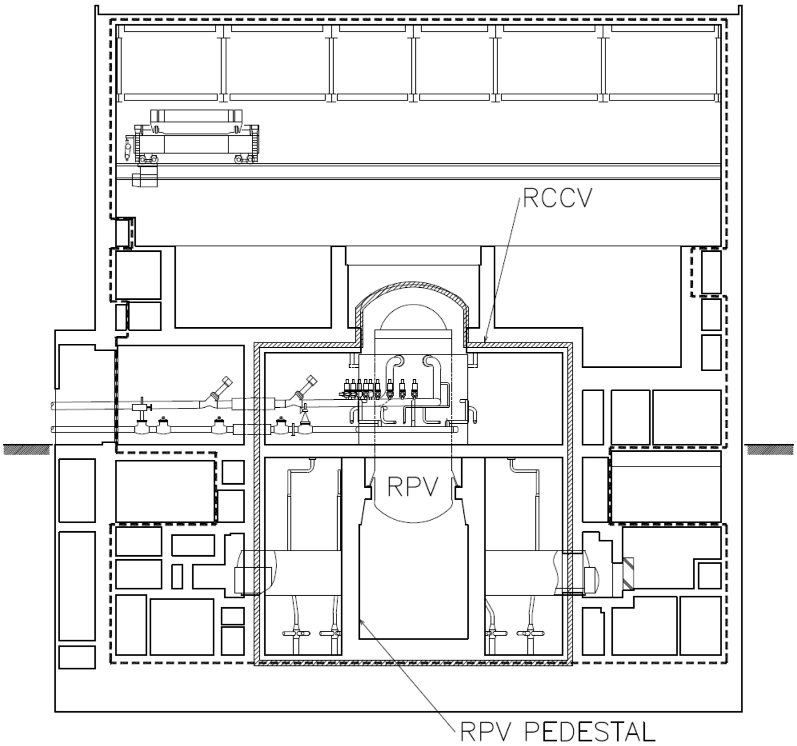


Figure 3.4-1 Reactor Building

3.5 Control Building

3.5.1 Structure Outline

The C/B is a structure which houses and provides protection and support for plant control and electrical equipment, batteries, and C/B heating, ventilating and air conditioning equipment. The C/B is located between the R/B and T/B. The C/B is constructed of reinforced concrete and structural steel.

3.5.2 Design Requirements

The C/B is classified as Safety Category A, Safety Class 1 and Seismic Category 1.

The C/B is designed and constructed to accommodate the dynamic and static loading conditions associated with the various loads and load combinations which form the structural design basis. The loads are those associated with;

(1) Natural phenomena; wind, floods, tornados, earthquakes, rain and snow.

(2) Internal events; floods, pipe breaks and missiles.

(3) Normal plant operation; live loads, dead loads and temperature effects.

Within the C/B, the steam tunnel has no penetrations from the steam tunnel into other areas of the C/B. The concrete thickness of the steam tunnel walls, floor and ceiling are designed to minimize dose rate to operators.

The steam tunnel is protected against pressurization effects that occur in the steam tunnel as a result of postulated rupture of pipes containing high energy fluid.

Neighbor buildings of the Control Building are designed such that superstructure of the building has adequate seismic capability or located away from the C/B with adequate spatial separation not to damage the safety function of the C/B under the DBE condition.

3.5.3 Protection against Internal Hazards

The C/B, except for the main control area envelope, is divided into three separate divisional areas for mechanical and electrical equipment and four divisional areas for instrumentation and control equipment (including batteries). Interdivisional boundaries are designed to prevent fire and flooding in one division from propagating to other divisions.

Other safety measures against internal hazard are described in the Section 9 of this document.

3.5.4 Protection against External Hazards

Seismic analysis is performed for the generic site conditions to ensure the seismic adequacy of the building. Safety functions in the C/B are protected against external hazards shown in Section10 including aircraft impact. Other safety measures against external hazard are detailed in the Section 10 of this document.

3.6 Heat Exchange Building

The Hx/B is constructed from reinforced concrete and structural steel. The Hx/B consists of four separate divisional areas, for mechanical and electrical equipment, for three RCW trains and one TCW train. Interdivisional boundaries are constructed from reinforced concrete and have no penetrations, to prevent flooding and fire in one division, from propagating to other divisions.

3.7 Safety -related Tunnels

The purpose of the safety-related tunnels is to provide protected and separated pathways for piping, power cables, and C&I cables. The safety-related tunnels will be used to route piping and cabling from the R/B and C/B, to the Emergency Diesel Generator fuel oil storage tanks, and to the Hx/B, including allowances for segregation and separation for resilience and diversity of services.

3.8 Backup Building

The Backup Building is a robust structure to protect alternative safety functions, such as water injection facilities, AC power source and I&C facilities, from adverse site-related environmental or human-induced events. This building is located away from the other safety related buildings in order to enhance redundancy of core and spent fuel pool cooling capability against potential risks of damage to the main buildings such as the R/B, C/B and Hx/B.

3.9 Turbine Building

3.9.1 Structure Outline

The Turbine Building (T/B) houses the main turbine generator and other power conversion cycle equipment and auxiliaries. The T/B is located adjacent to the Control Building.

There are some components which may cause initiating events to close main steam isolation valve. The requirements to the T/B structure will be appropriately evaluated. The T/B also houses various plant support systems and equipment such as non-divisional switchgear and chillers. A tunnel connects the Rw/B, T/B, C/B and R/B for the liquid radwaste system piping. The penetrations from the tunnel to the T/B are watertight and fire protected.

3.9.2 Design Requirements

Flood conditions in the T/B are prevented from propagating into the C/B via the S/B. This is achieved by locating the access from the T/B to the S/B at above grade level and providing a flood control doorway at the access location. The T/B will be designed such that its intended safety functions are not compromised during a DBE.

3.10 Radwaste Building

3.10.1 Structure Outline

The Radwaste Building (Rw/B) is a structure which houses the solid and liquid radwaste treatment systems.

3.10.2 Design Requirements

The Rw/B is designed such that its intended safety functions are not compromised for a DBE. Flood conditions in the Rw/B are prevented from propagating into the R/B and T/B by providing the penetrations in external walls below flood level with flood protection features. A tunnel connects the Rw/B, T/B, C/B and R/B for the liquid Radwaste piping system. The penetrations from the tunnel to the Rw/B will be watertight. The tunnel connecting the Rw/B, T/B, C/B and R/B is designed in such a way that potential damage to the penetration seals, at the interfaces with the safety-related structures, will not occur under the DBE.

4 Design Requirements, Criteria and Evaluation of Responses to Loads

4.1 Codes and Standards

This section provides the Codes and Standards, which are applied to the civil structural design of the safety-related structures of the UK ABWR. They are basically designed based on the US codes and standards, which are international recognized for nuclear facilities, considering the conformity with the UK requirements. The modern and most current year edition is applied. When the older edition of the code is applied, its technical justification is needed.

4.1.1 Seismic Analysis and Design

a. ASCE 4: ASCE Standard for Seismic Analysis of Safety Related Nuclear Structures

b. ASCE 43: Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities

c. NUREG-0800: USNRC Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants-LWR Edition;
Section 3.7.1, Seismic Design Parameters,
Section 3.7.2, Seismic System Analysis,
Section 3.7.3, Seismic Subsystem Analysis

d. RG 1.61: Damping Values for Seismic Design of Nuclear Power Plants

e. RG 1.92: Combining Modal Responses and Spatial Components in Seismic Response Analysis

f. EUR Volume 2: General Nuclear Island Requirements, Chapter 4 Design Basis

g. BS EN 1998-1:2004: UK National Annex to Eurocode 8: Design of structures for earthquake resistance

h. IAEA NS-G-3.6: Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants

4.1.2 RCCV Design

a. ASME Boiler & Pressure Vessel Code, Section III, Division 2, Subsection CC: Code for Concrete Containments

b. ASME Boiler & Pressure Vessel Code, Section III, Division 1, Subsection NE: Class MC Components

c. NUREG-0800: USNRC Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants-LWR Edition;
Section 3.8.1, Concrete Containment

d. RG 1.136: Material Construction, and Testing of Concrete Containments

e. RG 1.57: Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

4.1.3 RCCV Internal Structures Design

a. ACI 349: Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary

b. ANSI/AISC N690: Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities

c. ASME Boiler & Pressure Vessel Code, Section III, Division 2, Subsection CC: Code for Concrete Containments

4.1.4 Building Structural Design

a. ACI 349: Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary

b. ANSI/AISC N690: Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities

c. ASCE 7: ASCE Standard for Minimum Design Loads for Buildings and Other Structures

d. ACI 350.3: Seismic Design of Liquid-Containing Concrete Structures

e. NUREG-0800: USNRC Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants-LWR Edition;
Section 3.8.4: Other Seismic Category I Structures,
Section 3.8.5: Foundations

f. RG 1.142: Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)

4.1.5 Reference Standard for Material

a. EN 1992-1-1: Design of concrete structures - Part 1-1: General rules and rules for buildings.

b. EN 206-1: Concrete - Part 1: Specification, performance, production and conformity.

c. EN 10080: Steel for the reinforcement of concrete - Weldable reinforcing steel – General.

d. BS 4449: Specification for carbon steel bars for the reinforcement of concrete

e. EN 10025-2: Hot rolled products of structural steels - Part 2: Technical delivery conditions for non-alloy structural steels.

f. EN 10025-4: Hot rolled products of structural steels - Part 4: Technical delivery conditions for thermomechanical rolled weldable fine grain structural steels.

In principle, these material standards are used, as they are generally available in the UK supply chain. A full comparison matrix analysis of the differing requirements of the design codes to be applied, together with the American Standards and their European or UK equivalents, will be executed at a later stage of the GDA process.

Where American Standards impose more restrictions on material performance then these conditions are specified and imposed on the supply chain, and, if required, imported American or other appropriate materials will be used.

4.2 Design Loads and Load Combinations

This section contains the loads and load combinations for the design of UK ABWR.

4.2.1 Design Loads

4.2.1.1 Dead Load (D)

Dead loads are taken as the weight of the SSCs plus any other permanent loads including vertical and lateral pressure of liquids.

4.2.1.2 Live Load (L, Lo)

Live loads include floor area loads, laydown loads, nuclear fuel and fuel transfer casks, equipment handling loads and similar items.

4.2.1.3 Lateral Soil Pressure (H)

Lateral soil pressure on the outer walls shall be included as part of live load, including the effects of ground water pressure where appropriate.

4.2.1.4 Snow(S)/ Rain(R) Load

Snow and rain loads will be determined based on the generic site envelope.

4.2.1.5 Wind Load (W, Wt)

Design Wind Loads (W)

Design wind loads will be determined based on the generic site envelope.

Tornado Wind Loads (Wt)

Tornado loads will be determined based on the generic site envelope.

4.2.1.6 Seismic Loads (E')

Design Basis Earthquake, DBE (E') will be determined based on the generic site envelope. The seismic loads will be obtained taking into account soil-structure interaction (SSI) effect according to the ASCE 4 code [RD 5], as described in Section 4.3.1.

4.2.1.7 Thermal Loads (To, Ta)

The ambient thermal conditions and the stress free temperature for the design will be determined based on the generic site envelope. The evaluation method of temperature effect on the concrete design is based on ACI 349 [RD 6].

Two cases, winter and summer, are considered for the following two conditions.

Normal Operating Conditions (To)

Accident Conditions (Ta)

4.2.1.8 Pressure Loads (Po, Pa, Pt, SRV, LOCA)

Normal Operating Condition (Po)

Pressure load is the pressure difference between the interior and exterior of the containment, considering both interior pressure changes because of heating or cooling and exterior atmospheric pressure variations.

Accident Conditions (Pa)

Design accident containment pressure loads generated by the Loss of Coolant Accident (LOCA) are based upon the calculated peak pressure with an appropriate margin.

LOCAs include Design Basis Accident (DBA), Small Break Accident (SBA), and Intermediate Break Accident (IBA).

Pressure Test Loads (Pt)

Pressure test loads is the loads applied to the containment during the Structural Integrity Test (SIT).

Safety/Relief Valve Loads (SRV)

Safety/Relief Valve loads are oscillatory dynamic pressure loadings resulting from the discharge of Safety/Relief Valves into the Containment Suppression Pool.

LOCA Loads (LOCA)

LOCA loads are hydrodynamic pressure loads applied to the Suppression Pool boundary during the LOCA.

4.2.1.9 Pipe Reactions (Ro, Ra)

Normal Operating Condition (Ro)

Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition, are obtained from the pipe stress analysis.

Accident Conditions (Ra)

Pipe reactions from thermal conditions generated by a LOCA and from seismic load. These loads are provided from the results of the pipe stress analysis.

4.2.1.10 High Energy Line Break (Y)

The local effects due to a postulated high-energy line break include the following:

- Yr - Load on the structure generated by the reaction of a ruptured high-energy pipe.
- Yj - Load on the structure generated by the jet impingement from a ruptured high-energy pipe.
- Ym - Load on the structure resulting from the impact of a rupture high-energy pipe.

4.2.1.11 Construction Loads

Construction Loads are those that are applied during construction, and allow for the possible impact on the permanent works from temporary works.

4.2.1.12 Other External Hazard Loads

Other external hazards described in Part III are considered as design loads.

4.2.2 Load Combinations and Structural Acceptance Criteria

4.2.2.1 Load Combination for RCCV

The load combinations and associated load factors and acceptance criteria for the RCCV are in compliance with ASME BPVC Sec III [RD 7] and SRP 3.8.1 [RD 8].

4.2.2.2 Load Combination for RCCV Internals

The load combinations and associated load factors and acceptance criteria for the RCCV Internals are in compliance with ACI 349 [RD 6], ANSI/AISC N690 [RD 9] and SRP 3.8.3 [RD 8].

4.2.2.3 Load Combination for Reinforced Concrete Structures

The load combinations and associated load factors and acceptance criteria for concrete structures outside the containment are in compliance with ACI 349 [RD 6] and SRP 3.8.4 [RD 8].

4.2.2.4 Load Combination for Steel Structures

The load combinations and associated load factors and acceptance criteria for steel structures outside the containment are in compliance with ANSI/AISC N690 [RD 9] and SRP 3.8.4 [RD 8].

As for the external hazard loads which are not included in the above codes and standards, the adequate load combination cases are determined considering probability of occurrence of each event.

4.3 Design Method

This section covers the Design Method for all buildings. However, the R/B is specifically mentioned here where particular design criteria apply.

4.3.1 Seismic Design

This section describes the seismic design methodology. The seismic design of the NPP provides mitigation in relation to the radiological hazard by maintaining the integrity of SSCs during and after an earthquake. Therefore, the characteristics of the seismic design earthquakes must be defined, and the seismic evaluation of each SSC must be presented to demonstrate adequate seismic integrity. The seismic design philosophy presented for the UK ABWR standard plant is based on Hard Site characteristics.

4.3.1.1 Design Earthquake

a) Definition of design earthquake

The seismic design of each SSC within a NPP must mitigate the effect of seismic design earthquakes that correspond to the seismic categorization of the SSC. The seismic design earthquakes are defined as DBE and OBE, in the SAPs as follows;

- The DBE is determined to cover an earthquake with Peak Ground Acceleration (PGA) and response spectra corresponding to an earthquake with a return period of 10,000 years calculated using the seismology and geology of the area at and around the site.

- At OBE level, no SSCs important to safety should be impaired by the repeated occurrence of ground motions. Therefore, OBE corresponds to the Seismic Level 1 (SL-1) requirement of IAEA NS-G-3.6 [RD 10], and is an earthquake with a return period of 100 years.

Furthermore, 10^{-3}/year earthquake is defined as an earthquake with a return period of 1,000 years, which is used to confirm the integrity of Seismic Category 2 SSCs.

b) Soil properties and design earthquake

The soil properties and earthquake are set on the assumption that the plant is constructed on the hard site. Therefore, DBE and soil property of SSI analysis are defined as hard site conditions of EUR as a general one. The hard site earthquake of EUR is confirmed to cover candidate site UHSp.

Moreover, design earthquakes for seismic category 2 and 3 are determined as follows.

- 10^{-3}/year earthquake is assumed to be covered by one half of DBE. Therefore, one half of DBE is applied to design earthquake for Seismic Category 2.
- The earthquake that is defined in BS EN 1998-1:2004 [RD 4] is applied to design earthquake for Seismic Category 3 as mentioned in the seismic categorization.

The relationship between seismic categorization and seismic design earthquakes is shown in Table 4.3.1.1.2-1.

Table 4.3.1.1.2-1 Relationship between seismic categorization and seismic design earthquakes

Seismic category	Definition of earthquake	GDA
Seismic Category 1	DBE (10^{-4}/year earthquake)	DBE (EUR hard site spectra)
Seismic Category 1A	DBE (10^{-4}/year earthquake)	
Seismic Category 2	10^{-3}/year earthquake	1/2 DBE (EUR hard site spectra)
Seismic Category 3	BS EN 1998-1:2004	BS EN 1998-1:2004

4.3.1.2 Seismic Analysis

The seismic analysis considers two orthogonal horizontal and one vertical earthquake components as input and dynamic interaction between the soil and structure. Where the structure is supported on a flexible foundation, Soil-Structure Interaction (SSI) is taken into account by coupling the structural model with the soil model. The linear finite element computer program SASSI is used. The program uses finite elements with complex moduli for modeling the structure and foundation properties and is based on the modified subtraction method. Moreover, the frequency domain complex response method is used for analysis, and the cut-off frequency for seismic analysis is 33Hz.

The SSI analysis procedure is as follows:

- The SSI model includes horizontal soil layers and embedded structure; accordingly, wave propagation and scattering effects are simulated.
- The SSI analysis is partitioned into three substructures: foundation, structure and excavated soil.
- The SSI analysis is implemented using the model which is composed of the sum of foundation and structure minus excavated soil.
-
- Interaction between the structure and excavated soil is considered at surface nodes of the excavated soil.

Figure 4.3.1.2-1 shows the overview of SSI modeling.

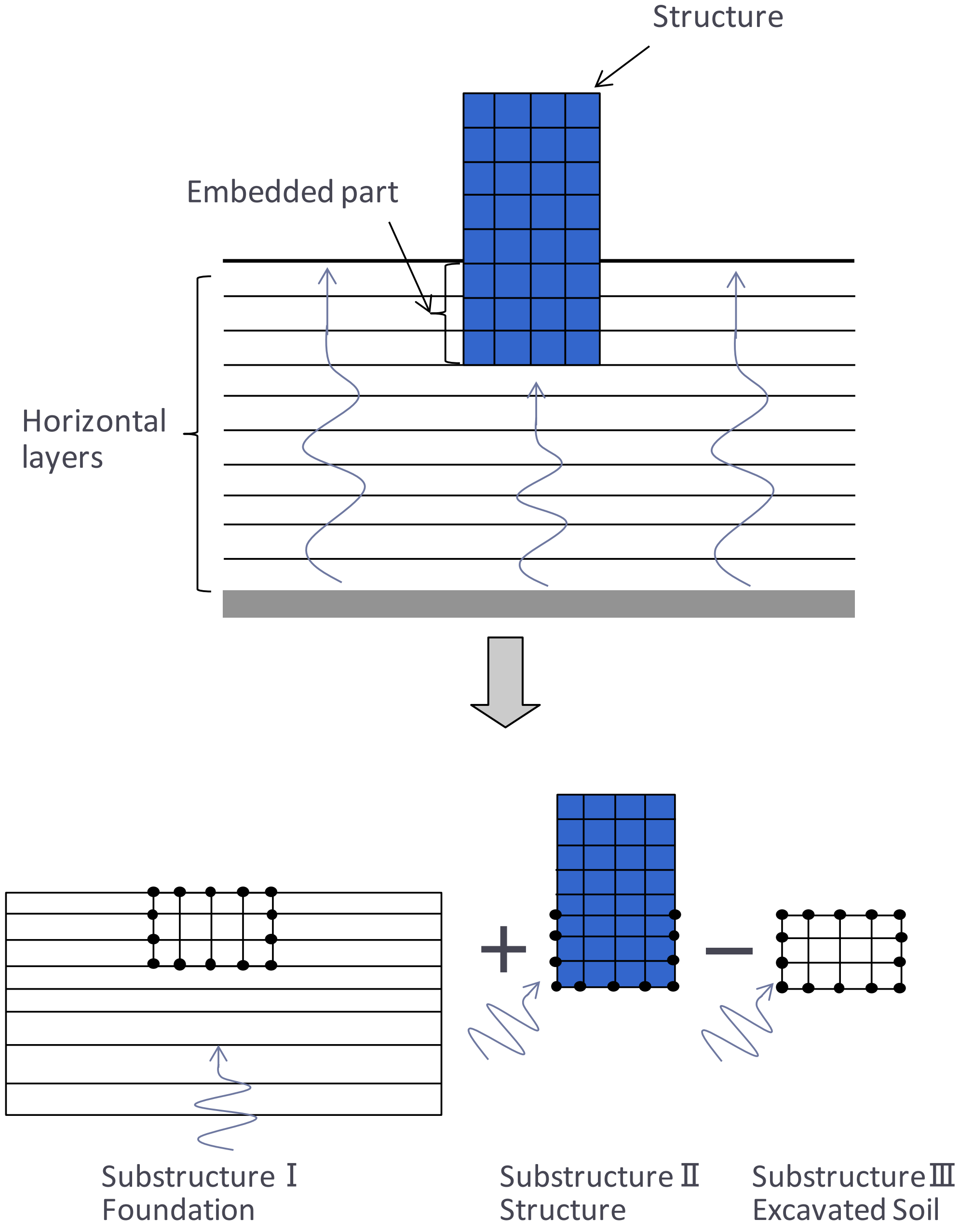


Figure 4.3.1.2-1 SSI modeling

4.3.1.3 Seismic Analysis Model

The ABWR building and coupled structure are mainly represented by three-dimensional lumped mass and stick model (3D-stick model) as shown in Figure 4.3.1.3-1 and 4.3.1.3-2 (Note: Supports and other boundary conditions are not shown). 3D-stick model is used to simulate 3D behavior including the effects of torsion, eccentricity and oscillation mass resulting from local response.

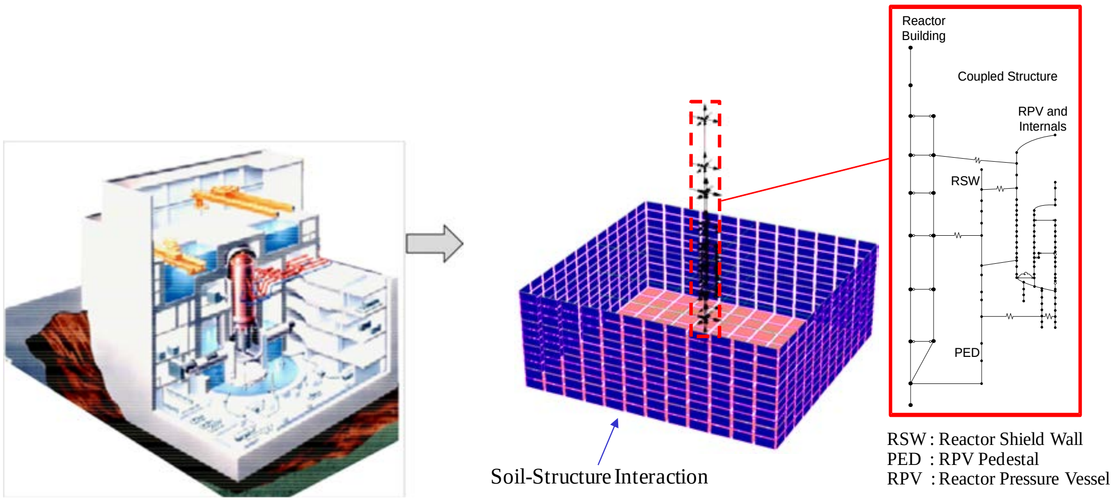


Figure 4.3.1.3-1 3D-Stick Model for ABWR Building and Coupled Structure

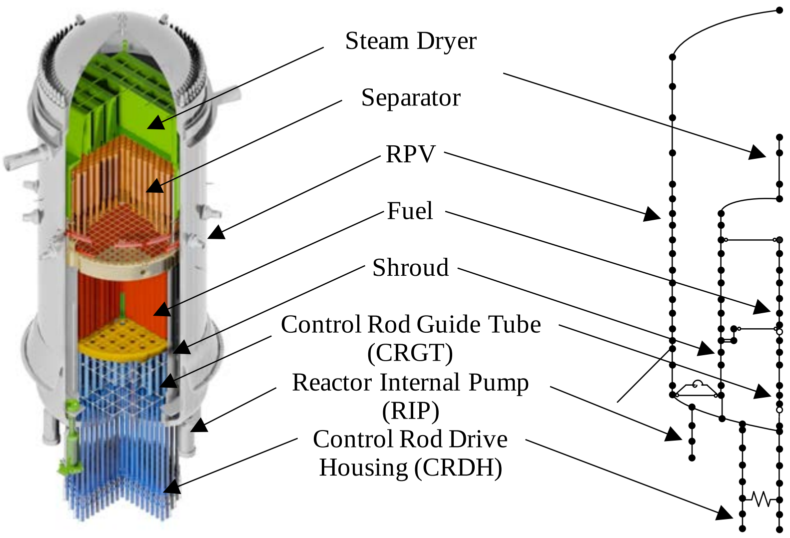


Figure 4.3.1.3-2 3D-Stick Model for RPV

4.3.1.4 Variation of Design Condition

The following variations of the design condition are considered in SSI analysis based on the assessment data, test data and design practice.

- Soil property variation: Soil property variation is considered as the lower and upper range of soil property which are based on hard type site condition of EUR and cover the variation of candidate site one.
- Concrete stiffness reduction
- Embedment effect: The embedment effect is considered by using hard type site condition of EUR and candidate site condition.
- Material effect due to EURO standard
- RCCV stiffness reduction
- Stiffness variation of supporting structure between coupled structures
- RPV pedestal stiffness reduction

4.3.1.5 Structure-Soil-Structure Interaction

The effects of Structure-Soil-Structure Interaction are taken into consideration in seismic design (e.g. R/B-C/B-T/B).

4.3.1.6 Damping Value

Energy dissipation within a structural system is represented by equivalent viscous dampers in the mathematical model. The applicable damping values are based on the materials, load conditions and types of construction used in the structural system. The damping values are based on RG 1.61 [RD12]. In addition, the damping values defined in the ABWR DCD [RD 11] are applied where specific damping values are not defined in RG 1.61.

Table 4.3.1.6-1 Damping Values

Items	Percent Critical Damping		
	DBE	Others	
Reinforced Concrete	7%	4%	RG1.61
Welded Steel Structure	4%	3%	
Equipment	3%	2%	
Piping Systems	4%	3%	
Fuel	6%	6%	ABWR design practice

4.3.2 RCCV Design

4.3.2.1 Materials

(a) Concrete and Rebar

The material properties associated with the concrete and reinforcement for the RCCV is discussed in Section 4.3.3.1.

(b) Liner

The materials used in the construction of the containment are in accordance with Subarticle CC-2500 of ASME Code Section III, Division 2 [RD 7], and are augmented by the requirements of RG 1.136 [RD 13].

The materials conform to the requirements of Subarticles CC-2500 to CC-2700 of ASME Code Section III, Division 2 [RD 7]. The liner plate shall be of the following type and grade:

・Carbon Steel : ASME SA-516 Gr.-70

・Carbon Steel with Stainless Clad: ASME SA-264 (SA-516 Gr.-70 + SA-240 tp 304L)

・Stainless Steel: ASME SA-240 Type 304L

(c) MC Components

The materials used in construction of the containment penetrations are in accordance with Article NE-2000 of ASME Code Section III, Division 1 [RD 14].

The Penetrations shall be of the following type and grade:

・ Plate (SA-240 type 304L, SA-516 grade 60 or 70).

・ Pipe (seamless SA-333 grade 1 or 6 or SA-106 grade B or SA-312 type 304L or welded SA-671 Gr CC70).

・ Forgings (SA-350 grade LFl or LF2 or SA-182F 304L/316L)

・ Bolts (SA-320-L43 or SA-193-B7 or SA-193-B8 or SA-437 grade B4B bolts with SA-194 nuts or to the requirements for nuts in the specification for the bolting material to be used).

・ Cladding (SA-240 type 304L).

(d) RCCV Internal Structures

The materials used in the construction of the Containment Internal Structures shall be in accordance with ANSI/AISC-N690 [RD 9] or equivalent standards for steel structures.

The materials used in the construction of the diaphragm floor are discussed in Section

4.3.3.1.

4.3.2.2 Design Analysis

(a) Reinforced Concrete

The structural design of the RCCV is performed according to the specifications of ASME Code Section III, Division 2, Subsection CC, Article CC-3000 [RD 7].

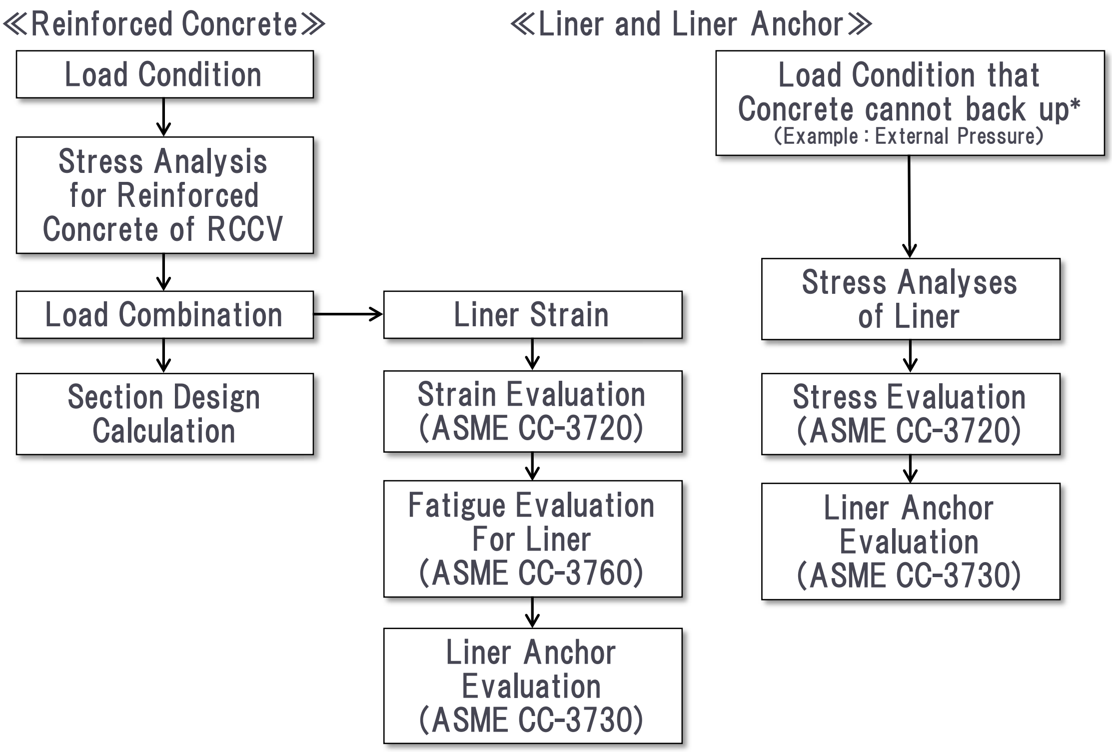
Since the RCCV is monolithic with the R/B, the design analysis of the RCCV is described in section 4.3.3.2.

(b) Liner

The design and analysis procedures for the liner conform to Sub-article CC-3600 of ASME Code, Section III, Division 2 [RD 7].

The design analyses are performed according to the procedure shown in Figure 4.3.2.2-1.

Evaluations of the liner and liner anchor are carried out using liner strain evaluated by the analysis of the reinforced concrete of the RCCV. For the load condition that concrete can not back up, evaluations of the liner and liner anchor are carried out respectively.



Note * : As for the external-pressure, the liner becomes the structural member, because there is not a structural member inside the liner.

Figure 4.3.2.2-1 Flow Path for Liner Design

(c) MC Components

The MC Components design analyses procedures conform to Article NE-3200 and / or NE 3300 of ASME Code Section III, Division 1 [RD 14]. The design analysis is performed according to the following procedure, as shown in Figure 4.3.2.2-2.

The design of the MC component is divided into two, one is “design by formula” another is “design by analysis”.

The Design by Formula is to confirm the basic figure of the MC components against the internal or external pressure. (based on ASME NE-3300)

The Design by Analysis is to conduct stress analyses for the MC components against the internal /external pressure, thermal load, jet force, seismic load etc. (based on ASME NE-3200)

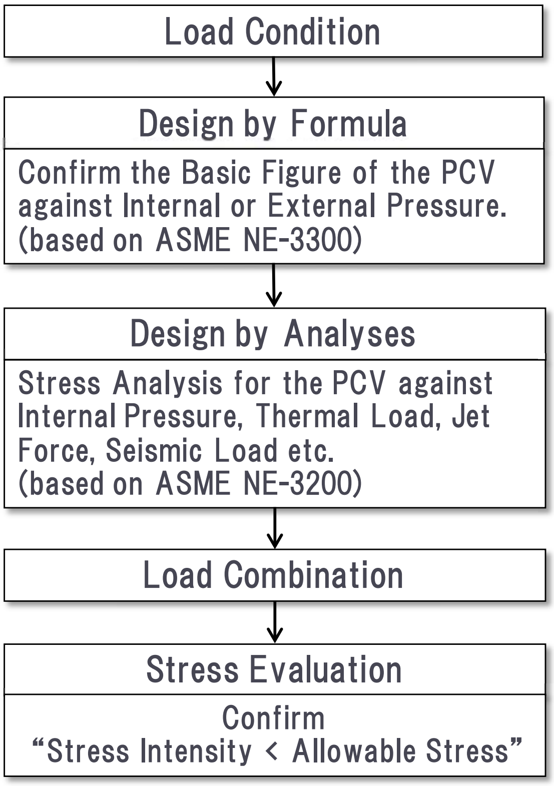


Figure 4.3.2.2-2 Flow Path for MC Components Design

(d) RCCV Internal Structures

The design and analysis procedures for the containment internal structures, including assumptions on boundary conditions and expected behavior under loads, are to be in accordance with the following:

- For steel structures, the procedures shall be in accordance with ANSI/AISC-N690 [RD 9].
- For Diaphragm Floor, the procedures shall be in accordance with Sub-article CC-3000 of ASME Code, Section III, Division 2 [RD 7].

Composite steel concrete effects, which include heat transfer and thermal stress, will be evaluated in the design.

D/F (Diaphragm Floor)

The D/F is composed of a seal plate and a reinforced concrete slab. The seal plate is considered as non-structural member conservatively in the design.

RPV Pedestal

The RPV pedestal consists of an inner and an outer cylinder with the vertical ribs. The annulus between the inner and outer cylinders is filled with concrete. Main structural members are the inner and outer cylinders and vertical ribs. The filled concrete is considered as non-structural member conservatively. The filled concrete is considered to be shielding material. The rigidity of the filled concrete is considered in the seismic analysis model. The applied value is surveyed parametrically in the seismic analyses.

Reactor Shield Wall

The Reactor Shield Wall consists of an inner and an outer cylinder with the vertical ribs. The annulus between the inner and outer cylinders is filled with grout. Main structural members are the inner and outer cylinders and vertical ribs. The filled grout is considered as non-structural member conservatively. The filled grout is considered to be only shielding material.

4.3.3 Building Design

4.3.3.1 Materials

The building materials will be selected by considering appropriate procurement routes within the UK and within the European market. The materials used in construction are in accordance with EN 206-1 [RD 15] for concrete, EN 10080 [RD 16] for steel for the reinforcement of concrete, and EN 10025-2 [RD 17] or EN 10025-4 [RD 18] for steel. However, if there is any material, which cannot be procured in the UK, the US or other appropriate standard material will be imported.

(a) Concrete

The compressive strength of concrete is as follows.

	Application	Strength Class	Characteristic Cylinder Strength f_{ck}
•	Basemat	C30/37	30.0 MPa
•	Other Structures	C35/45	35.0 MPa

All concrete ingredients are tested, batched and mixed in accordance with EN 206-1 [RD 15] and the placement, compaction and curing of concrete are in accordance with EN 13670 [RD 19].

Structural designs are carried out according to ACI 349 [RD 6] will make allowance for the use of British reinforcement steel work.

(b) Reinforcing Steel

Reinforcing steel for concrete is deformed bars meeting requirements of EN 1992-1-1 ANNEX C, Class C [RD 20], and EN 10080 [RD 16]. Characteristic yield strength, f_{yk} is 500MPa.

Structural designs are carried out according to ACI 349 [RD 6] will make allowance for the use of British reinforcement steel work.

(c) Structural Steel

Structural steel conforms to the standards shown in Section 4.1.5.

Structural designs are carried out according to AISC N690 [RD 9] with specified yield strength, F_y using yield strength to product standards, R_{eh}.

4.3.3.2 Design Analysis

(a) General Description

The structural analysis and design are performed according to the following procedure, as shown in Figure 4.3.3.2-1

1. Prepare a finite element (FE) model for stress analyses considering structural characteristics and materials, and appropriate articulation.
2. Perform stress analyses for the design loads described and calculate the section forces.
3. Select the basic and critical load combinations as the selected design load combinations.
4. Combine the section forces according to the selected design load combinations through the application of appropriate load factors.

5. Perform structural design calculations using the section forces for the selected design load combinations.

The design of reinforced concrete structures is performed according to the specifications of ACI 349 [RD 6].

The design of steel structures is performed in accordance with ANSI/AISC N690 [RD 9].

(b) Stress Analysis

The computer program used for the stress analysis calculations is NASTRAN. It is a general-purpose stress analysis program, which is technically based on the finite element method.

(c) Analysis Model

Major structural members of the buildings include the basemat, floor slabs, external walls, shear walls, and frame members. FE models are employed considering the complexity of the structure, load conditions, and boundary conditions.

The ground is modeled with spring elements. Three independent spring elements, one vertical and two horizontal, are attached to each of the basemat grid points. Spring constants are calculated using the soil properties.

(d) Method of Applying Loads

Load application methods in the global FE analyses are determined based on the characteristics of each load type described in Section 4.2.1.

(e) Analysis Results

The element forces and moments obtained by stress analyses are combined in accordance with the load combinations described in Section 4.2.2. These are used for the section design of the RC structures and Steel structures as discussed in the sections that follows.

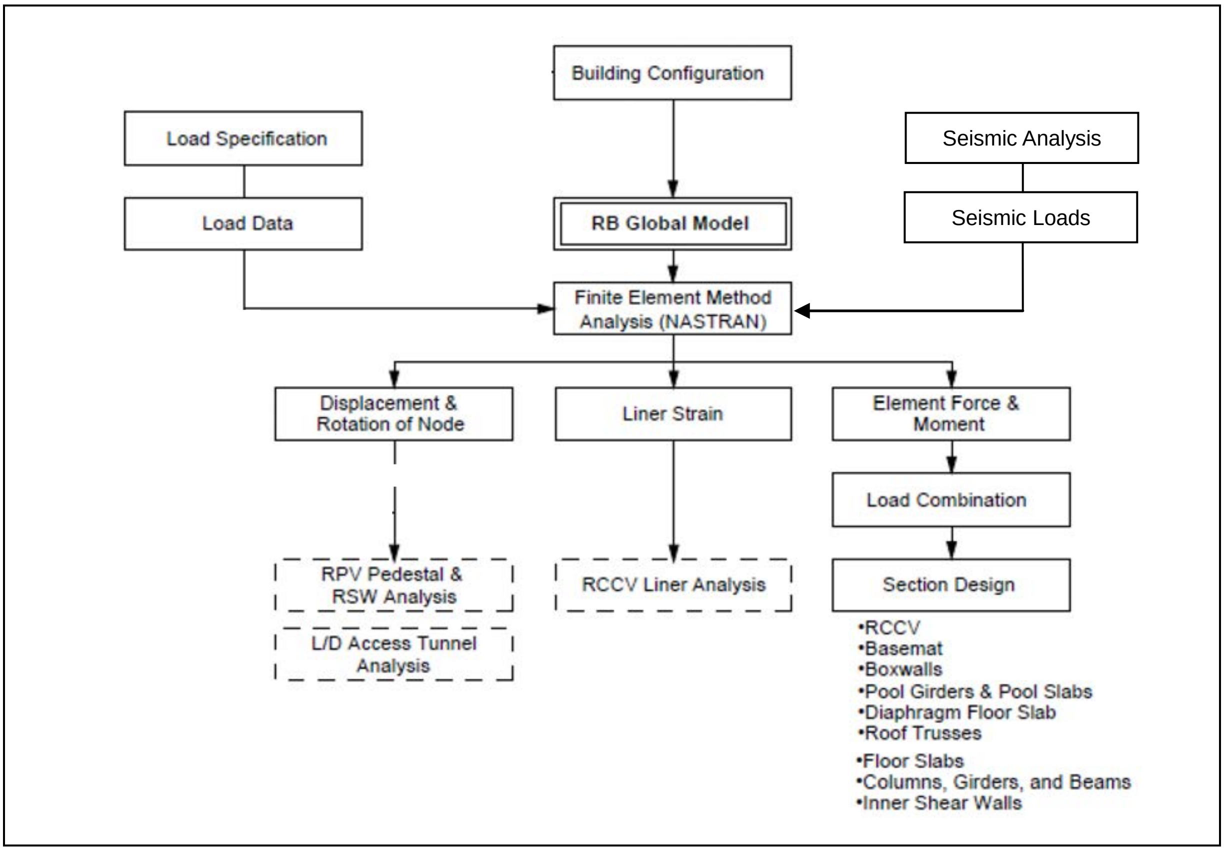


Figure 4.3.3.2-1 Flow Path for Reactor Building Design

4.3.3.3 Section Design of RC Structures

Structural design of the reinforced concrete structures is carried out according to the method contained within ACI 349 [RD 6]. The design flow chart is shown in Figure 4.3.3.3-1.

Design calculations will be carried out for the following section forces:

- Flexure and Membrane Forces
- Membrane Compressive Forces
- Transverse Shear

The evaluation method for each of the section forces is described in the following subsections.

(a) Section Design for Flexure and Membrane Forces

ACI 349 [RD 6] requires confirming that section forces at a section do not exceed the specified strength in the section design. The design calculations are requested to be carried out for flexure and axial forces and for in-plane shear forces. Axial forces and in-plane shear forces are combined into the membrane forces, and design calculations for flexure and membrane forces are carried out.

As for the thermal effects, section forces due to thermal loads, which are evaluated by FE analyses using uncracked concrete stiffness, are reduced considering the depth and direction of cracking in calculations. .

(b) Section Design for Membrane Compressive Forces

In the section design, membrane compressive forces in the RC members are also checked.

(c) Section Design for Transverse Shear

Section design calculations for transverse shear are performed according to ACI 349, Chapter 11 [RD 6].

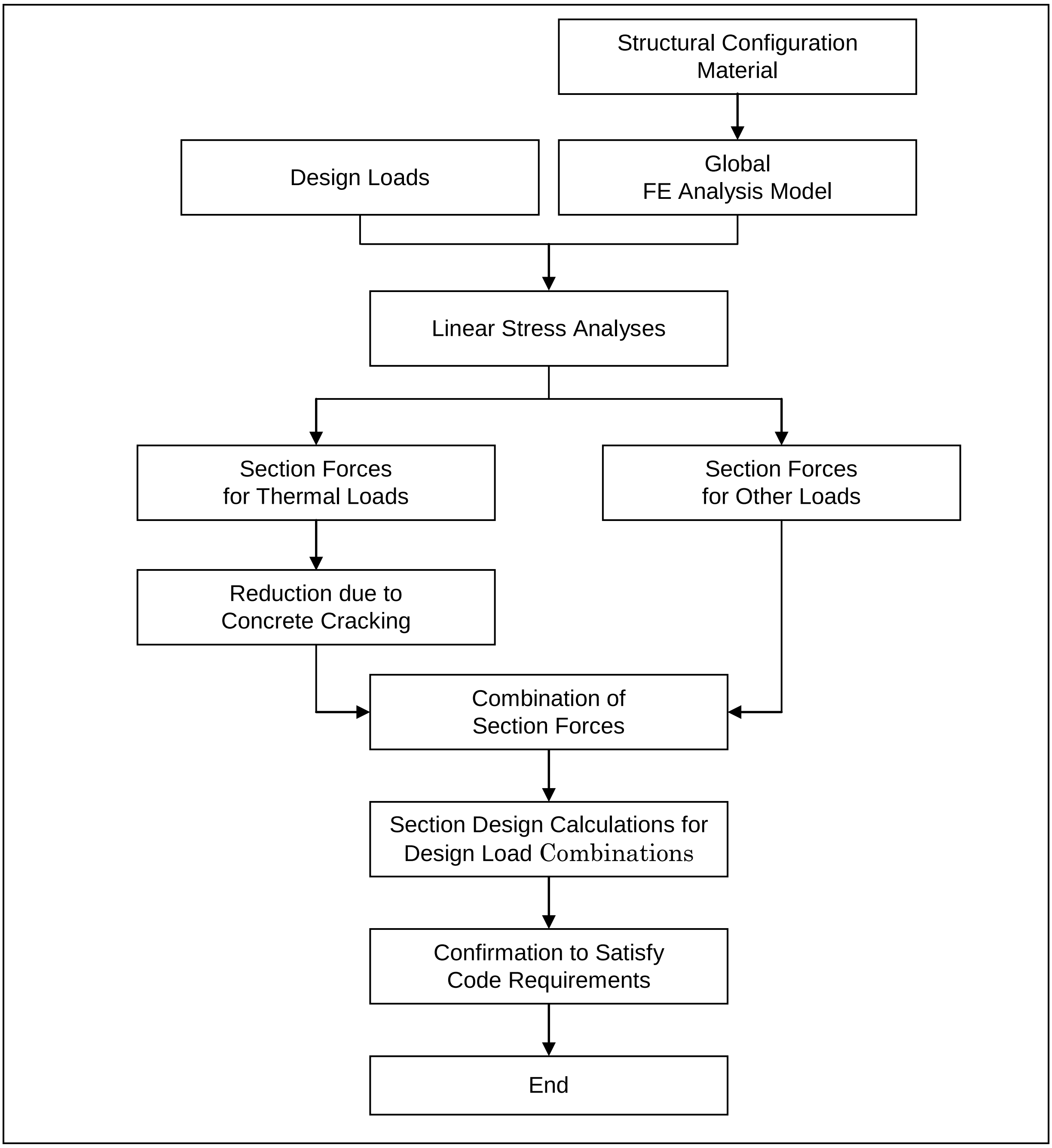


Figure 4.3.3.3-1 Flow Chart of Design for RC Structures

4.3.3.4 Section Design of Steel Structures

Section design of steel member is performed according to ANSI/AISC N690 [RD 9].

The design flow of steel structures is almost same as the flow of reinforced concrete structures. However, reductions of thermal stresses are not considered for the steel design.

Section design is performed for the following section forces.

- Section Design for Axial Compression and Bending
- Section Design for Axial Tension and Bending
- Section Design for Transverse Shear

The following allowable stresses are calculated according to ANSI/AISC N690 [RD 9] and used for the stress check.

- Allowable Axial Tensile Stress
- Allowable Axial Compressive Stress
- Allowable Bending Stress of H-shaped Members (Strong Axis Bending)
- Allowable Bending Stress of H-shaped Members (Weak Axis Bending)
- Allowable Shear Stress

5 Construction Works

A detailed description of the Construction Works will be provided in Step-2.

6 Decommissioning

6.1 Introduction

There is a regulatory requirement to explain how and when the reactor will eventually be decommissioned and how the resulting waste will be managed. Therefore, new nuclear power station design must be considered in such a way so as to facilitate future decommissioning in a safe and environmentally acceptable way at an early stage. This includes design principles and fulfillment of International Atomic Energy Agency (IAEA) requirements, UK CDM regulations and Japanese lessons learned related to decommissioning.

6.2 Objectives and Principles

The objective of decommissioning a UK ABWR is to transition the site from its operational state to an agreed end state. And, for the purpose of the site release of the UK ABWR after the cessation of operation, the withdrawal of the services from facilities, followed by its transformation into an out-of-service state and eventually its complete removal will be executed. In the consistency of this scenario, followings are considered.

1) For the purpose of prevention of radiation hazards, design measures will be appropriate to facilitate decommissioning.
2) Associated wastes (radioactive and non-radioactive) during decommissioning will be completely managed.
3) The process of decommissioning will be comprehensively consistent with the UK regulatory expectations.

6.3 Decommissioning Plan

Based on the Design Principles, a site decommissioning strategy will be established considering the assumed site condition. Based on the principles and management method to carry out the site decommissioning, a rational and defined plan assuming safety will be established with the assumed site situation. Based on the UK ABWR decommissioning strategy, scope and plant status assumptions will be established. Scopes of UK ABWR decommissioning defines pre-closure planning, defueling and spent fuel management, dismantling and demolition, waste management, maintenance and modification of the site infrastructure and services, de-licensing. Given the timescales before the strategy will be implemented, it is inevitable that a number of assumptions will have to be made. Assumptions of UK ABWR decommissioning defines, *inter-alia* start of reactor decommissioning, end of site decommissioning, availability of disposal, interim decommissioning milestones, de-licensing criteria, technologies, off-site storage, design for decommissioning, incidents, housekeeping, etc.

The principles of the UK ABWR decommissioning strategy includes, *inter-alia* compliance with UK policies and legislation, the safety and protection of the public, the workforce and the environment. This also encompasses nuclear safety goals, the best appropriate scientific and technical knowledge, focus on tasks and hazards, radioactive and non-radioactive waste management, In addition, the strategy will show that the proposals are economic and fit for purpose.

7 Inspection and Testing

A detailed description of the Examination, Maintenance, Inspection and Testing (EMIT) will be provided in Step-2. Such regular and systematic examination will help to inform the planned maintenance requirements for the life of the Plant.

8 Quality Assurance

Quality Management Systems are deployed based on IAEA GS-R-3 and ISO9001. During GDA, Quality Management Plan mainly consists of design, document, record, design change, SQEP and procurement control, are implemented to support to the achievement of the UK ABWR GDA Project. ASME NQA-1 is applied to design and construction of RCCV, which is designed by ASME BPVC Sec. III Div. 1 and 2. Qualification and Management of contractor are also subject to Quality Management System in accordance with the Quality Management Plan.

A detailed description of the overall Management System during design and construction periods will be provided in PCSR in Step-2.

II. Internal Hazards

9 Internal Hazards

Internal hazards may include the following:

- Fire
- Explosion / missiles
- Toxic and corrosive materials and gases
- Dropped loads
- Structual collapse
- Impacts
- Flooding / spray

Appropriate internal hazards shall be selected from all the possible internal hazards and shall be assessed. The Plant is designed taking into account those internal hazards as described in GDA Step 1b document, "Internal Hazards Report" [RD 2].

III. External Hazards

10 External Hazards

External Events which affect the design conditions are derived.

External hazard event list is based on all hazards listed in following reference sources.

No.	Reference Source
1	HSE, "Generic Design Assessment Guidance to Requesting Parties" (ONR-GDA-GD-001 Revision 0)
2	HSE, "Technical Assessment Guide" 013
3	OECD Nuclear Energy Agency (NEA), "Probabilistic Safety Analysis (PSA) of the Other External Events Than Earthquake, "
4	European Utilities Requirements (EUR), Volume 2, Chapter 2.4, "Generic Nuclear Island Requirements: Design Basis,"
5	IAEA, "External Events Excluding Earthquakes in the Design of Nuclear Power Plants" (IAEA Safety Guide, NS-G-1.5)
6	USNRC, "PRA Procedures Guide" (NUREG-CR-2300)
7	USNRC, "PRA Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," & "Evaluation of External Hazards to Nuclear Power Plants in the United States," (NUREG 1407&NUREG/CR-5042)
8	Western European Nuclear Regulator's Association (WENRA) – Study by Reactor Harmonization working Group, Safety of New NPP Designs, I October 2012.
9	Swedish Nuclear Inspectorate (SKI), "Guidance for External Events Analysis"
10	HNP, "Nuclear Safety Site Description Wylfa, Isle of Anglesey" (NSSD, HNP-RL-REP-11-16)
11	Pre-Construction Safety Report (Sizewell B PCSR)
12	AREVA & EDF, "The Pre-Construction Safety Report" (EPR PCSR)
13	Westinghouse, "Pre-Construction Safety Report" (AP1000 PCSR, UKP-GW-GL-732)

The screening process will determine appropriate and credible site-specific external hazards capable of initiating fault sequences. Such faults have the potential to challenge safety functions of the ABWR with potential faults that could lead to core melt.

The following definitions are provided:

- External hazards

External hazards are those natural or man-made hazards to a site and facilities that originate outside to both the facilities' site and its processes, where the duty holder has no control over the initiating event. Terrorist or other malicious acts are also assessed as external hazards.

Thus the assessment of these hazards requires detailed knowledge of the site surroundings, the natural processes, along with plant and site layout.

- Natural hazards

Those that take place at the site as a result of the geophysical location and prevailing meteorological conditions, e.g. flooding, extreme wind, ground motion.

- Man-made hazards

Hazards that may affect a plant settled in a particular location, as a result of human’s presence or utilisation of an area near or adjacent to the site, e.g. external explosions, fires, or aircraft impacts.

In general, it is more difficult to define the magnitude of a hazard and its consequences than for an internal plant fault. For some hazards, the information required for a probabilistic fault analysis is simply not available.

During the screening process of the external hazards, probabilistic approach will be taken to select and see the magnitude of the external hazards. Basically, ABWR is designed to tolerate external events with probability of X>10-4 /yr.

Events with probability of less than 10-4/yr are considered with following criteria.

	Criteria	Remarks
I	Frequency of occurrence	Events of which frequency is10^{-7}/yr are screened-out. (e.g. Meteorite impact)
II	Site significance	Events which could not occur close enough to the site are screened-out. (e.g. Volcanic activity)
III	Consequence	・ABWR is considered to tolerate events which have low impact outcome to the nuclear site. ・ABWR is considered to tolerate events in which sufficient warning is available to allow operators to take appropriate actions.
IV	Defense-in-depth	ABWR is considered to tolerate events if the nuclear safety component or structure has the mitigating capacity to deal with the events beyond the design basis in order to avoid “cliff-edge effects”.

“Beyond Design Bases” is considered through a process of “a priori” screening utilising world nuclear operation experience and by proper assessment avoid ‘cliff-edge’ effects. Hazards considered not credible are screened out.

10.1 Meteorological Phenomenon

This section summarises the character of the meteorological phenomenon surrounding the nuclear power station.

10.1.1 Air temperature (dry-bulb and wet-bulb temperatures which includes climate change)

This section describes the characteristics of the air temperature surrounding the nuclear power station. The generic design of the nuclear power station will consider the extreme air temperature conditions of the potential UK sites.

Extreme air temperatures will be derived from historical data relating to the potential UK sites. The effects of climate change will also be taken into consideration. The data will clearly define the maximum and minimum extreme air temperatures.

The HVAC of the nuclear power station will be designed based on this extreme air temperature.

10.1.2 Wind load

The generic UK ABWR plant is designed to accommodate standard wind speed (km/h) that is defined considering extreme wind condition of candidate site(s) within the UK. The standard wind speed consists of values to be utilized for design of non-safety-related structures and safety-related structures. For a specific site, the site licensee shall demonstrate that the site specific wind speed can be bounded within the standard condition.

10.1.3 Tornado

This section describes the characteristics of design consideration against tornado. The design of the nuclear power station considers tornado events.

Wind forces and loads resulting from wind borne missiles will be considered. The wind speeds are based on historical UK data.

The integrity of the R/B and all other nuclear safety related structures, will be assessed as part of the GDA process.

10.1.4 Precipitation

The generic UK ABWR plant is designed to accommodate maximum rainfall rate (cm/h) and maximum snow load (kPa) that is defined considering precipitation condition of candidate site(s) within the UK. For a specific site, the site licensee shall demonstrate that the site specific rainfall rate and snow load can be bounded within the standard condition.

10.1.5 Lightning

This section describes the characteristics of the lightning strikes. The design of the nuclear power station considers the effect of lightning strike of the possible site in UK.

The severity of lightning strike is defined as the frequency of lightning strikes on the ground expressed as the number of strikes/km^2/year.

The frequency of lightning strikes on the ground will be defined based on the meteorological data of the specific site. The meteorological data will be collected from site investigation report. Those data will be statistically evaluated and clearly defined as a density of lightning strikes on the ground.

Lightning protection related to electrical system for the UK ABWR will be designed based on this value of the frequency of lightning strikes.

10.1.6 Sand storm

The hazard of sand storm outside of a site boundary is so strongly site dependent that it should be considered as the site specific hazard and assessed on the Site Specific PCSR.

10.2 Hydrological Phenomenon

10.2.1 Sea water temperature (include Climate change)

This section describes the characteristics of the sea water temperature of sea as the ultimate heat sink for the nuclear power station. The generic safety system design of the nuclear power station will considers extreme sea water temperature conditions for potential UK sites.

The values of extreme sea water temperatures to be used in the design of a NPP for potential UK sites will be determined from historical data. The data will clearly define the extreme maximum and minimum sea water temperatures to be used. The influence of climate change will also be considered.

The cooling water systems such as RCW and RSW and the heat removal systems including RHR will be designed based on this range of sea water temperatures.

10.2.2 Sea water level (include Climate change)

The sea water levels for a site specific UK ABWR will be defined based on the evaluation of results from an external flooding assessment.

10.2.3 External flooding

This section describes the characteristics of design consideration against external flooding. The design of nuclear power station considers the external flooding and the design values and levels relating to external flooding will be defined based on historical data for the site. The influence of climate change will also be considered. The site license holder shall take into account the effects of external flooding and considered the following:

- The ground level of the site will be constructed above the level of the established external flooding level.
- Coastal sea level protection measures will be constructed around the nuclear power station as required.
- Watertight doors will be used where applicable for the protection of important SSCs.

The effects of external flooding from extreme rainfall, tidal effects, storm surge, seiche, tsunami, dam failure and water-course containment failure shall be considered in the GDA process.

10.3 Seismic Input and Soil Properties

10.3.1 Seismic Input

The seismic design of the UK ABWR considers the effect of the seismic design motion to allow for a Design Basis Earthquake (DBE) and Operational Basis Earthquake (OBE). DBE is assumed to cover the Uniform Hazard Spectra (UHSp) for a generic nuclear site in the UK and that obtained from Seismic Hazard Assessment (SHA) of candidate site.

10.3.2 Soil Properties

The generic UK ABWR plant is designed to accommodate standard soil parameters that are defined considering soil conditions of candidate site(s). The standard soil parameters include minimum static bearing capacity, minimum shear wave velocity and liquefaction potential. For a specific site, the site licensee shall demonstrate that the site specific soil properties can be bounded within the standard soil parameters.

10.4 Electrical

10.4.1 External natural EMI/External man-made EMI

This section describes the characteristics of the external natural/man-made EMI surrounding the nuclear power station. The electric equipment design of the nuclear power station considers external natural/man-made EMI condition of the possible site in UK.

The following have been considered as external natural EMI/external man-made EMI sources.

・ Natural EMI sources - Sources that are associated with natural phenomena. They include atmospheric charge/discharge phenomena such as lightning and extraterrestrial sources including radiation from the sun (solar flares).

- Man-made EMI sources - Sources associated with man-made devices such as power lines.
- Intentional radiating emitters - Emitters whose primary function depends on radiated emitters.

Class 1 electronic equipment within the UK ABWR design is resistant to External natural EMI/External man-made EMI. Even so, the complete loss of operability of electronic equipment within multiple cabinets would not compromise the reactivity control, the heat removal from the core, and radioactivity containment.

10.4.2 Electromagnetic pulse

An electromagnetic pulse is an instantaneous burst of electromagnetic radiation that can disrupt and overload electrical and electronic systems.

10.4.3 Frequency & Duration of Loss of Offsite Power

This section describes the characteristics of the design consideration against loss of offsite power. The design of the nuclear power station considers the loss of offsite power.

Main protection against a loss of off-site power will be the introduction of Alternating Current (AC) power supply via Emergency Diesel Generators (EDGs). The generators will be able to provide power for a maximum of 7days.

The reliability of on-site AC power supply and site(s) robustness against loss of off-site power will be ascertained by carrying out a Probabilistic Safety Analysis (PSA). Frequency and duration of loss of offsite power for PSA in GDA is taken as the generic data from UK, US, or Japan. In relation to loss of off-site power, the likely site specific frequency and duration of off-site power will be determined by reviewing the existing grid connection.

10.5 Geophysical, Landslide

10.5.1 Volcanic activity

This section describes the characteristics of the volcanic activity against the nuclear power station. The design of the nuclear power station considers the activity of volcano located around the possible site in UK.

No direct influence of the volcanic activity is considered because plant construction site is selected from the area where it is far from the volcano and is not affected directly from the volcano. However, the volcanic ash from volcano is considered because it gives influence on wider area. The measure against volcanic ash is described in Site License as it is bounded by that against sandstorm, which is considered in the Site License.

10.5.2 Meteorite Impact

This section describes characteristics of design consideration against meteorite impacts.

The natural phenomenon of a meteorite impact has the potential to cause serious damage to a NPP. Since the event has a significantly lower mean frequency of occurrence, the event is onerous than the consequences arising from other external hazard events [RD 21]. Therefore, this hazard is screened out.

10.5.3 Inland Erosion

Inland erosion is a site specific design condition. For a specific site, the site licensee will demonstrate that the structures important to maintain nuclear safety are not affected by inland erosion.

10.5.4 Coastal Erosion

Coastal erosion is a site specific design condition. For a specific site, the site licensee will demonstrate that the structures important to maintain nuclear safety are not affected by coastal erosion.

10.5.5 Sedimentation / Siltation

For a specific site, the site licensee will provide site-specific information related to sedimentation and siltation.

10.5.6 Landslide (impacts)

For a specific site, the site licensee will provide site-specific information about the static and dynamic stability of all soil and rock slopes, the failure of which could adversely affect the safety of the plant.

10.6 Man-made Hazards

10.6.1 Aircraft Impact (accident and intentional)

This section describes the characteristics of design consideration against the aircraft impact.

Intentional aircraft impact is assessed as a Beyond-Design-Basis (BDB) event in the GDA process, and an accidental aircraft impact is assessed as a design basis event in the site licensing.

The load conditions arising from intentional aircraft impact are determined according to UK security guidance or based on a general large type of commercial aircraft. Accidental impacts resulting from aircrafts are based on the site specific data. The basic functions of aircraft impact protection are:

- To maintain integrity of containment and SFP (Spent Fuel Pool), and
- To maintain core and spent fuel cooling.
- Preventing the release of material that would cause harm to workers, the public or the environment.

The safety assessment relating to aircraft impact includes structural integrity, fire propagation, and vibration effects on the safety related SSCs.

10.6.2 External Fire

This section describes the characteristics of the external fire against the nuclear power station. The design of the nuclear power station considers external fire from site boundary of the possible site in UK.

The hazard sources of external fire are considered to be external industrial installations including stockpiles of petroleum products and other flammable liquid and gaseous chemicals as well as flammable materials, natural sources including bushes and forests and transport sources such as road, railway and ships. The condition of external fire source which may pose a hazard to the plant is identified in site specific studies. Where such studies identify specific risks, they will be addressed. However, the hazards to be considered are as follows mainly.

- Heat flux from fire may result in high temperature damage to safety-related equipment.
- Ventilation system may be affected by smoke and toxic fumes. And also, a significant reduction in visibility and operability by smoke and toxic fumes may be observed.

“Heat flux” effect from outside of site boundary is so dependent on site condition that it is considered in site specific studies. The ventilation system for the Main Control Room (MCR) is designed to exclude smoke and toxic fumes from coming in by isolating the ventilation inlets and supplying 100% recirculation air to the MCR to maintain the function of MCR.

10.6.3 External Missiles

This section describes the characteristics of design consideration against external missiles. The design of the nuclear power station considers external missiles.

The GDA will assess the effect of wind-borne missiles. The source of external missiles will be addressed by the site license holder.

10.6.4 External Transport Impacts

This section describes the characteristics of the external transport impacts against the nuclear power station. The design of the nuclear power station considers external transport impacts from the possible site in UK.

The hazard sources of external transport impacts are considered to be road vehicles, railway trains, and marine ships outside a site boundary. The transport condition which may pose a hazard to the plant is identified in site specific studies. The hazards to be considered in the study are as follows mainly.

- Fire and smoke
- Explosion
- Direct impact from collision
- Oil spill from ship
- Release of toxic, corrosive or radioactive substances

The hazard of “fire and smoke”, “explosion” and “release of toxic, corrosive or radioactive substances” are bounded by other external hazards in consideration of the hazard influence against the plant. Therefore remaining effect, which are “direct impact from collision” and “UHSp blockage by oil spill from ship” are considered as the hazards of external transport impacts.

This external hazard depends on site specific condition so that it will be assessed on the Site Specific PCSR.

10.6.5 External Explosion

This section describes the characteristics of design consideration against external explosion. The design of the nuclear power station considers external explosion.

Standard load-time function derived from pressure waves resulting from explosions will be used as a design basis load for the generic design. The integrity of R/B and other important SSCs will be assessed against this hazard. The control of explosive materials on the site will be addressed in the site licensing.

10.6.6 Industrial Environment

This section describes the characteristics of design consideration against industrial environment. The design of the nuclear power station considers industrial environment. As part of the site licensing the effects from any industrial environment nearby the site boundary will be considered, such as:

- Corrosive, chemical, or toxic substances,
- Exhaust gases,
- Acoustic noise levels and frequencies, and
- Vibration amplitude and frequencies.

10.7 Biological Hazards

10.7.1 Water –based

This section describes the characteristics of the Water-based biological hazards.

The nuclear power plant needs protection against the entry of water-based biological agents such as fishes, shellfishes, marine growth, biological fouling and seaweeds.

Appropriate facilities will be installed in the nuclear power plant to prevent invasion of such water-based biological agent.

10.7.2 Land & Air-based

This section describes the characteristics of the Land & Air-based biological hazards.

The biological agents resulting from the invasion of wildlife considered are follows:

- Rodents or other pests
- Birds
- Insects (airborne swarms, infestation)
- Tree roots or other vegetation

The nuclear power plant requires protection against the potential infiltration of biological agents resulting from the invasion of land and air based wildlife as this could compromise nuclear site safety. For example, bird screens are fitted to protect the building from air inlets and smaller agents like insects are caught at the HVAC filters.

10.8 Others

10.8.1 Flotsam/Jetsam/Log jam

The influence of flotsam, jetsam and log jam outside of a site boundary (from sea) is considered to the design of cooling water intake. However, these hazards depend on site specific condition so that it will be assessed on the Site Specific PCSR.

10.8.2 Residual artifacts from previous use

The hazard of residual artifacts from previous use (i.e. munitions) outside of a site boundary is so strongly site dependent that it will be considered as the site specific hazard and assessed on the Site Specific PCSR.

10.8.3 General Resilience to Hazard

Most of important components which deliver safety functions are housed in the Reactor Building whose walls are made of concrete. Therefore, important components are protected by the barrier of the Reactor Building. Some part of important component is exposed outside. In this case, those components are designed considering the environmental condition.

If these components are simultaneously damaged by the extreme environmental condition, so called beyond design load, plant safety is maintained by easily access mobile equipment, since several connections or accesses are prepared with offset distance.

11 References

Reference Number	Document Title
[RD 1]	"Generic Site Envelope", GA91-9901-0010-00001, XE-GD-0106, 2013
[RD 2]	"Internal Hazards Report", GA91-9901-0002-00001, XE-GD-0108, 2013
[RD 3]	"Categorisation and Classification of Systems, Structures and Components", GA91-9901-0007-00001, XE-GD-0104, 2013
[RD 4]	BS EN 1998-1:2004: UK National Annex to Eurocode 8: Design of structures for earthquake resistance
[RD 5]	ASCE 4: ASCE Standard for Seismic Analysis of Safety Related Nuclear Structures
[RD 6]	ACI 349: Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary
[RD 7]	ASME Boiler & Pressure Vessel Code, Section III, Division 2, Subsection CC: Code for Concrete Containments
[RD 8]	NUREG-0800: USNRC Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants-LWR Edition; Section 3.8.1, Concrete Containment
[RD 9]	ANSI/AISC N690: Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities
[RD 10]	IAEA NS-G-3.6: Evaluation of Seismic Hazards for Nuclear Power Plants
[RD 11]	ABWR DCD： "ABWR Design Control Document", Rev.4 , GE Nuclear Energy, March 1997
[RD 12]	RG 1.61: Damping Values for Seismic Design of Nuclear Power Plants
[RD 13]	RG 1.136: Material Construction, and Testing of Concrete Containments
[RD 14]	ASME Boiler & Pressure Vessel Code, Section III, Division 1, Subsection NE: Class MC Components
[RD 15]	EN 206-1: Concrete - Part 1: Specification, performance, production and conformity.
[RD 16]	EN 10080: Steel for the reinforcement of concrete - Weldable reinforcing steel – General.
[RD 17]	EN 10025-2: Hot rolled products of structural steels - Part 2: Technical delivery conditions for non-alloy structural steels.
[RD 18]	EN 10025-4: Hot rolled products of structural steels - Part 4: Technical delivery conditions for thermomechanical rolled weldable fine grain structural steels.
[RD 19]	EN 13670: Execution of concrete structures.
[RD 20]	EN 1992-1-1: Design of concrete structures - Part 1-1: General rules and rules for buildings.
[RD 21]	ASME "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME/ANS RA-Sa-2009.