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Assessment and management of ageing of major nuclear power plant components important to safety:

Metal components of BWR containment systems

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

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FOREWORD

At present, there are over four hundred operational nuclear power plants (NPPs) in IAEA Member States. Operating experience has shown that ineffective control of the ageing degradation of the major NPP components (e.g. caused by unanticipated phenomena and by operating, maintenance or manufacturing errors) can jeopardize plant safety and also plant life. Ageing in these NPPs must therefore be effectively managed to ensure the availability of design functions throughout the plant service life. From the safety perspective, this means controlling within acceptable limits the ageing degradation and wear-out of plant components important to safety so that adequate safety margins remain, i.e. integrity and functional capability in excess of normal operating requirements.

This TECDOC is one in a series of reports on the assessment and management of ageing of the major NPP components important to safety. The reports are based on experience and practices of NPP operators, regulators, designers, manufacturers, and technical support organizations and a widely accepted Methodology for the Management of Ageing of NPP Components Important to Safety, which was issued by the IAEA in 1992.

The current practices for the assessment of safety margins (fitness for service) and the inspection, monitoring and mitigation of ageing degradation of selected components of Canada deuterium-uranium (CANDU) reactors, boiling water reactors (BWRs), pressurized water reactors (PWRs), and water moderated, water cooled energy reactors (WWERs) are documented in the reports. These practices are intended to help all involved directly and indirectly in ensuring the safe operation of NPPs, and to provide a common technical basis for dialogue between plant operators and regulators when dealing with age related licensing issues. The guidance reports are directed toward technical experts from NPPs and from regulatory, plant design, manufacturing and technical support organizations dealing with specific plant components addressed in the reports.

This report addresses the metal components of BWR containment systems. The primary ageing mechanisms that may potentially impact the structural capacity, leaktight integrity, or service life of BWR containments are corrosion of metal components and stress corrosion cracking of bellows. Other potential degradation mechanisms include fatigue and mechanical wear. Areas of concern are where surfaces are inaccessible for inspection (e.g areas adjacent to floors, where the containment vessel is embedded in concrete, and locations adjacent to equipment or other structures).

The work of all contributors to the drafting and review of this document, identified at the end, is greatly appreciated. In particular, the IAEA would like to acknowledge the contributions of D.J. Naus, J.P. Higgins, P. Krebs and J. Stejskal. The IAEA officer responsible for this publication was J. Pachner of the Division of Nuclear Installation Safety.

EDITORIAL NOTE

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INTRODUCTION

1.1. BACKGROUND

Managing the safety aspects of nuclear power plant (NPP) ageing requires implementation of effective programmes for the timely detection and mitigation of ageing degradation of plant systems, structures and components (SSCs) important to safety, so as to ensure their integrity and functional capability throughout plant service life. General guidance on NPP activities relevant to the management of ageing (i.e. maintenance, testing, examination and inspection of SSCs) is given in the IAEA Nuclear Safety Standards (NUSS) Code on the Safety of Nuclear Power Plants: Operation (Safety Services No. 50-C-O, Rev. 1) [1.1], and associated Safety Guides on in-service inspection (50-SG-02) [1.2], maintenance (50-SG-07, Rev. 1) [1.3] and surveillance (50-SG-08, Rev. 1) [1.4].

The operation code requires that NPP operating organisations prepare and carry out a programme of periodic maintenance, testing, examination and inspection of plant systems, structures and components important to safety to ensure that their level of reliability and effectiveness remains in accord with the design assumptions and intent, and that the safety status of the plant has not been adversely affected since the commencement of operation. This programme is to take into account the operational limits and conditions, any other applicable regulatory requirements, and be re-evaluated in light of operating experience. The associated safety guides provide further guidance on NPP programmes and activities that contribute to timely detection and mitigation of ageing degradation of SSCs important to safety.

The Safety Guide on In-Service Inspection (ISI) provides recommendations on methods, frequency and administrative measures for the ISI programme for critical systems and components of the primary reactor coolant system aimed at detecting possible deterioration due to the influences such as stress, temperature, and irradiation, and at determining whether they are acceptable for continued safe operation of the plant or whether remedial measures are needed. Organisational and procedural aspects of establishing and implementing a NPP programme of preventive and remedial maintenance to achieve design performance throughout the operational life of the plant are covered in the maintenance safety guide. Guidance and recommendations on surveillance activities, for SSCs important to safety, (i.e. monitoring plant parameters and systems status, checking and calibrating instrumentation, testing and inspecting SSCs, and evaluating results of these activities) are provided in the surveillance safety guide. The aim of the surveillance activities is to verify that the plant is operated within the prescribed operational limits and conditions, to detect in time any deterioration of SSCs as well as any adverse trends that could lead to an unsafe condition, and to supply data to be used for assessing the residual life of SSCs. The above safety guides provide general programmatic guidance, but do not give detailed technical advice for particular SSCs.

Programmatic guidance on ageing management is given in Technical Report No. 338 [1.5] and in a Safety Series No. 50-P-3 [1.6]. Guidance provided in these reports served as a basis for the development of component specific Technical Documents on Assessment and Management of Ageing of Major NPP Components Important to Safety. This publication on Metal Components of BWR Containment Systems is one of these TECDOCs. TECDOCs already issued address: Steam Generators [1.7], CANDU pressure tubes [1.9], Concrete containment buildings [1.8], PWR reactor pressure vessels [1.10], and PWR reactor vessel intends [1.11].

Information related to NPP containment system designs is provided in IAEA Safety Series N0. 50-SG-D12 [1.12]. Most of the operating BWRs are housed in pressure-suppression type primary containments . The BWR containment is designed to act as the final barrier to the release of fission products to the environment that may occur during a loss-of-coolant or other design-basis event. The management of accidents also requires additional measures to protect the environment in the case of an accident beyond the design-basis.

The metallic components of the BWR containments are susceptible to several different corrosion mechanisms. Other ageing mechanisms of importance include fatigue, wear and erosion, radiation embrittlement, and mechanical damage. Ageing of nonmetallic elements and coating degradation also are of importance. Operating experience has shown that ageing degradation caused by several of these ageing mechanisms has occurred. If not detected and mitigated, the degradation potentially could have progressed to the extent that it could affect the containment structural or leaktight integrity. As nuclear power plants age, degradation occurrences tend to increase. In order to maintain the fitness-for-service of the metal components of BWR containments it is necessary to control within defined acceptable limits any age-related degradation that might occur. This is effectively accomplished through application of a systematic ageing-management process that is an adaptation of Deming's "plan-do-check-act" cycle to ageing management [1.13]. This process is based on the understanding ageing and consists of the basic elements of programme coordination, operational activities, inspection and assessment activities, maintenance, and continuous improvement.

1.2. OBJECTIVE

The objective of this report is to document the current practices for the assessment and management of the ageing of the metal components employed in BWR nuclear power plant containments. Safety aspects are to be emphasized and information is to be provided on current inspection, nonitoring and maintenance practices for managing ageing of the metal components of BWR containments. Ageing degradation is defined as ageing effects that could impair the ability of a system, structure, or component (SSC) to function within acceptance criteria. Ageing degradation is influenced by the interactions between design, materials, and service conditions, and includes ageing effects such as loss of fracture toughness, strength, fatigue resistance, or material thickness.

The underlying objective of this report series is to ensure that the information on the current assessment methods and ageing management techniques is available to all involved, either directly or indirectly, in the operation of nuclear power plants in the IAEA Member States.

The target audience includes nuclear power plant operators, regulators, technical support organizations, designers, and manufacturers.

1.3. SCOPE

This report deals with the steel components of BWR pressure-suppression type containments. Designs of different types of BWR pressure suppression containments that have evolved over the year are described as well as the ageing stressors, potential degradation sites, and mechanisms that affect performance of metal components of BWR containments. In-service inspection requirements are discussed and recommendations provided on techniques for improved management and mitigation of any ageing degradation of the metal components of BWR containments. Although it is recognised that several BWR containments have been fabricated using reinforced or prestressed concrete as the primary construction material, these structures are not directly addressed as information related to these structures and ageing management practices is provided in the IAEA-TECDOC-1025 [1.14]. Also, ageing management and maintenance of piping systems penetrating the containment vessel and isolation valves belonging to the BWR containment system is not part of this report.

1.4. STRUCTURE

Chapter 2 of the report provides a general description of BWR containments, their design basis, and materials of construction. Chapters 3 and 4 support understanding of metal component ageing by providing information on BWR containment service conditions and potential ageing mechanisms. Chapter 5 gives an overview of the various techniques that may be used to detect ageing degradation and assess its significance. Chapter 6 presents methods for prevention and correction of ageing effects. Chapter 7 summarizes operating experience by describing several ageing degradation incidents. Chapter 8 shows how the key elements of ageing management of metal components of BWR containment systems are integrated within a systematic ageing management programme. Finally, Chapter 9 summarizes the report, and provides conclusions.

REFERENCES TO CHAPTER 1

[1.1] INTERNATIONAL ATOMIC ENERGY AGENCY, Code on the Safety of Nuclear Power Plants: Operation, Safety Series No. 50-C-0 (Rev. 1), IAEA, Vienna, (1988).

[1.2] INTERNATIONAL ATOMIC ENERGY AGENCY, In-Service Inspection for Nuclear Power Plants: A Safety Guide, Safety Series No. 50-SG-02, IAEA, Vienna, (1980).

[1.3] INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance of Nuclear Power Plants: A Safety Guide, Safety Series No. 50-SG-07 (Rev. 1), IAEA, Vienna, (1990).

[1.4] INTERNATIONAL ATOMIC ENERGY AGENCY, Surveillance of Items Important to Safety in Nuclear Power Plants: A Safety Guide, Safety Series No. 50-SG-08 (Rev. 1), IAEA, Vienna, Austria (1990).

[1.5] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety, Technical Report Series No. 338, IAEA, Vienna, (1992).

[1.6] INTERNATIONAL ATOMIC ENERGY AGENCY, Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing: A Safety Practice, Safety Series No. 50-P-3, IAEA, Vienna, (1991).

[1.7] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Steam Generators, IAEA-TECDOC-981, Vienna (1997).

[1.8] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Buildings, IAEA-TECDOC-1025, Vienna (1998).

[1.9] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: CANDU Pressure Tubes, IAEA-TECDOC-1037, Vienna (1998).
[1.10] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Improtant to Safety: PWR Pressure Vessels, IAEA-TECDOC-1120, Vienna (1999).
[1.11] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Vessel Internals, IAEA-TECDOC-1119, Vienna (1999).
[1.12] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of the Reactor Containment Systems in Nuclear Power Plants: A Safety Guide, Safety Series No. 50-SG-12, IAEA, Vienna, (1985).
[1.13] INTERNATIONAL ATOMIC ENERGY AGENCY, Implementation and Review of Nuclear Power Plant Ageing Management Programme, Safety Series No. 15, IAEA, Vienna, (1999).
[1.14] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Buildings, IAEA-TECDOC-1025, IAEA, Vienna, (June 1988).

Chapter 2

BWR CONTAINMENT DESCRIPTION AND DESIGN BASIS

2.1. GENERAL DESCRIPTION

The BWR containment is designed to act as the final barrier to the release of fission products to the environment that may occur as a result of a design-basis accident. Although the shapes and configurations of BWR containments vary significantly from plant-to-plant depending on the nuclear steam supply system vendor, architect-engineering firm, and owner preference, leaktightness is assured by a continuous pressure boundary consisting of nonmetallic seals and gaskets, and metallic components that are either welded or bolted together. Nonmetallic components are used to prevent leakage from pumps, pipes, valves, personnel airlocks, equipment hatches, manways, and mechanical and electrical penetration assemblies. The remaining pressure boundary consists primarily of steel components such as metal containment shells, concrete containment liners, penetration liners, heads, nozzles, structural and nonstructural attachments, embedment anchors, pipes, tubes, fittings, fastenings, and bolting items that are used to join the pressure-retaining components. Each containment type includes numerous access and process penetrations that complete the pressure boundary (e.g. large opening penetrations, control rod drive removal hatch, purge and vent system isolation valves, piping penetrations, and electrical penetration assemblies).

Several BWR containment designs have been developed that are in use at operating nuclear power plants. With one exception (i.e. Big Rock Point plant in the U.S., which is being decommissioned), all the designs are of the pressure-suppression type that consist of a drywell (reactor pressure vessel and recirculating and other associated piping) and a wetwell, or suppression chamber (contains a large volume of water). During normal operation, the BWR primary containment is closed, with cooling provided by a ventilation system. In the event of an accident, any steam that is released from the reactor primary coolant system into the drywell enters the suppression chamber through vents (downcomers) and condenses. The suppression pool is also a primary source of water for the emergency core cooling (ECCS), the residual heat removal (RHR), and the high-pressure coolant injection (HPCI) systems. All these systems take pump suction from the pressure suppression chamber water and are considered part of the containment pressure boundary because they penetrate the pressure suppression chamber.

The most severe "design-basis" accident assumed to calculate the containment design pressure is a "double-end" brake of a recirculation, main steam, or feed water pipe. As a consequence of this pipe rupture and the subsequent need for emergency cooling of the core, a significant amount of primary and emergency cooling water in the containment is converted to steam that results in a pressure rise. The steam generated, however, is channeled to the suppression pool, where it is condensed. This prevents pressure buildup that otherwise would occur in the primary containment (i.e. "pressure suppression"). BWR containment pressures, therefore, remain relatively moderate compared to the pressures that may occur in pressurized water reactor (PWR) containments during similar events. Passive, self-regulating systems for the filtered pressure relief have been included (especially for accident management) in designs of some reactor containments to keep the containment pressure below the critical value. The systems are passive in that neither water nor electrical energy has to be supplied from an external source, nor is operator intervention. Pressure relief is provided automatically by a rupture disc in the event of overpressure or, if staff provides, manually by the staff opening the valves. In newer containment designs, the volume below the reactor is waterfilled for cooling the core in the event an accident results in core melting and melt through of the reactor vessel. Table 2.1 presents information on several worldwide BWR containments. More detailed information on specific BWR containment designs to that presented in the table follows.

TABLE 2.1. INFORMATION ON WORLDWIDE BWR CONTAINMENTS

Type	Main Contractor	Material	Number of Units
MK I	GE	Steel	23
MK I	GE	Concrete	2
MK II	GE	Steel	1
MK II	GE	Concrete	7
MK III	GE	Steel	3
MK III	GE	Concrete	2
Baulinie 69	Siemens	Steel	6
Baulinie 72	Siemens	Prestressed Concrete	2
Type I	ABB	Prestressed Concrete	5
Type II	ABB	Prestressed Concrete	6
MK I	GE/Hitachi/Toshiba	Steel	10
Improved MK I	GE/Toshiba/Hitachi	Steel	5
MK II	GE/Hitachi/Toshiba	Steel	4
Improved MK II	Toshiba/Hitachi	Steel	7
MK III	Toshiba/GE/Hitachi	Concrete	2

2.2. DESIGN BASIS AND DESCRIPTIONS OF SELECTED BWR CONTAINMENTS

BWR containment designs developed by General Electric [2.1], Siemens-KWU [2.2-2.4], and ABB Atom are discussed below.

2.2.1. General Electric (GE) Mark I, II, and III designs

2.2.1.1. Design requirements

The basic laws that regulate the design (and construction) of NPPs in the U.S. are contained in Title 10, “Energy,” of the *Code of Federal Regulations* [2.5] that is clarified by documents such as U.S. Nuclear Regulatory Commission Regulatory Guides, NUREG reports, and Standard Review Plans. It is required, in part, that structures, systems, and components be designed, fabricated, erected, and tested to quality standards commensurate with the safety functions to be performed and that they be designed to withstand effects of postulated accidents and environmental conditions associated with normal operating conditions. A reactor containment and associated systems are to be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. The containment and associated systems are to be designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident (LOCA). Also, the containment must be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations; (2) an appropriate surveillance programme; and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows. The *CFR General Design Criteria* provide essential safety requirements for design and licensing basis. Basic rules for the design and construction of metal (as well as concrete) containments are prepared by the American Society of Mechanical Engineers (ASME) and published in the *ASME Boiler and Pressure Vessel Code* [2.6]. Current rules for construction of metal containments are provided in Section III, Division I, Subsection NE of the *Code*. Piping, pumps, and valves that are part of the containment system or that penetrate or are attached to the containment are classified as Class 1 or 2 components and are covered by rules in other subsections of Section III, Division 1 of the *ASME Code*.

Prior to 1963, metal containment for NPPs (e.g. GE Mark I) were designed according to rules for unfired pressure vessels that were contained in Section VIII of the *ASME Code*. Subsequent metal containments were designed either as Class B vessels or as Class MC components according to rules provided in Section III of the *ASME Code*. Almost every aspect of metal containment design is addressed by the *ASME Code*, including methods for calculating required minimum thickness of pressure retaining components. The *ASME Code* also recognizes that service-related degradation to pressure-retaining components is possible, but rules for material selection and in-service degradation are outside its scope. Two provisions of the *ASME Code* are pertinent with respect to degradation. First, according to the *ASME Code*, it is the owner's responsibility to select materials that are suitable for the service conditions, and to increase minimum required thickness of the base metal to offset material thinning due to corrosion, erosion, mechanical abrasion, or other environmental effects. Typically, however, there is no corrosion allowance for vessels that are coated and partially embedded in concrete. Second, criteria are provided in Section III of the *ASME Code* for conduct of a detailed vessel fatigue evaluation. The criteria limit the number of cycles of service pressure, temperature, and mechanical load, and are based on the *ASME Code* fatigue curves for metal containment materials. For vessels meeting these criteria, it can be assumed that fatigue stress intensities have been met by compliance with other code provisions. If the vessel does not meet the criteria, a fatigue usage factor must be calculated and compared with the allowable usage factor of 1.0. At the time of the design for many metal containment vessels detailed fatigue evaluations were not required, primarily due to the lack of developed criteria to account for any synergistic interaction between corrosion and fatigue.

2.2.1.2. Containment descriptions

The drywells and pressure suppression chambers of the GE BWR Mark I, II and III containment designs are completely enclosed in a reinforced concrete reactor building (i.e. secondary containment or shield building). Although both reinforced concrete with steel liner and steel containment designs have been developed, the majority of Mark I containments are steel pressure vessels. Prior to each startup, the primary containment is purged with pure nitrogen until the atmosphere contains less than 4% oxygen by volume. Nitrogen inerting is used to prevent ignition of any hydrogen and oxygen mixture that may occur following an accident. Typical design characteristics for the Mark I, II, and III containments are provided in Table 2.2. Additional information on each of these containment designs follows.

TABLE 2.2. SOME DESIGN PARAMETERS FOR GE CONTAINMENTS

	Mark I	Mark II	Mark III
Drywell design overpressure (bar)	3.94–4.36	3.16–3.73	1.05
Drywell design temperatur (°C)	139–171	139–171	166
Drywell air volume (m^3) × 1000	3.7–5.0	5.7–8.6	7.1–7.9
Suppression chamber design overpressure (bar)	3.94–4.36	3.16–3.73	1.05
Suppression chamber design temperatur (°C)	139–155	100–139	74
Suppression chamber volume (m^3) × 1000	4.9–7.2	6.1–9.8	23.6–39.6

Mark I containments

Mark I primary containments consist of an inverted lightbulb-shaped drywell vessel surrounded at the base by a torus-shaped pressure suppression chamber, as shown in Figure 2.1. Typically, the drywell and suppression chamber are connected by eight vent lines equally spaced around the base of the drywell. Presently there are over 20 NPPs with Mark 1 steel containments (e.g. Oyster Creek, Dresden Units 2 and 3, Peach Bottom Unit 2, Browns Ferry, and Mühleberg).

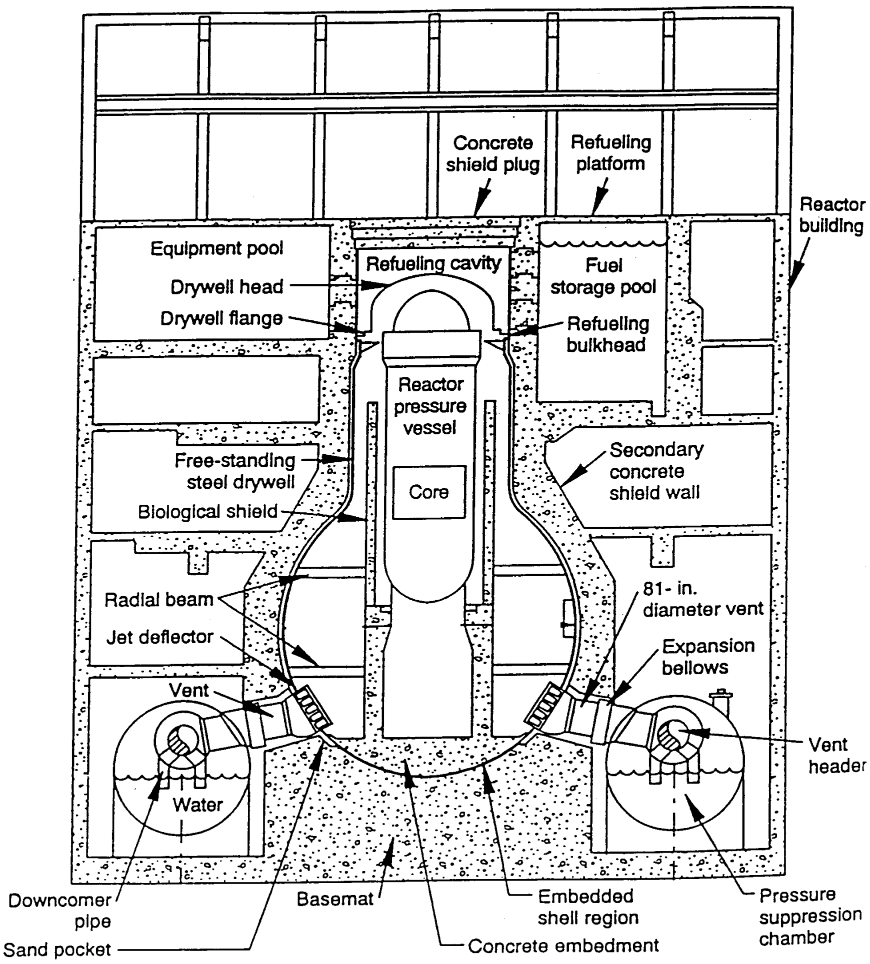


FIG. 2.1. GE-BWR Mark I design.

The drywell is a free-standing steel vessel supported at the base by embedding it in concrete. The bottom spherical-shaped portion of the drywells have diameters that range from about 18 to 21 m, depending on the size of the plant. Overall heights of the drywells range from about 30 to 35 m. The drywell head is a flanged, removable metal closure for access to the reactor pressure vessel during refueling. The area above the drywell flange is filled with water up to the refueling platform during refueling. There are typically six ventilation hatches in the drywell that are left open during operation to allow for forced ventilation, which helps to avoid excessive local heating of the drywell. Access to the drywells is provided by equipment hatches and a personnel airlock. A control rod drive removal hatch may also be present. The inside surfaces of most drywells are covered by protective coatings to minimize corrosion. A reinforced concrete wall, 123 to 185 cm thick, called the secondary concrete shield wall, surrounds the metal drywells to provide shielding. In areas where the secondary shield wall backs up the drywell, it provides additional resistance to deformation and buckling of the metal shell. A 50- to 75-mm gap between the drywell and the secondary concrete shield wall allows for thermal and pressure expansion and contraction during normal operation and during design-basis accidents. This gap is usually filled with a compressible fill material (e.g. polyurethane foam, ethafoam, or Firebar D and fiberglass) during construction of the concrete shield to maintain proper spacing. The fill material was removed after construction at some Mark I containments, but has been left in place at others. Moisture can be trapped in the filler material and may cause corrosion of the exterior surface of the drywell. A sand pocket is located at the bottom of the drywell-to-secondary concrete shield wall gap. The purpose of the sand pocket is to reduce the stress concentration at the point of embedment of the shell in concrete when the shell is subjected to a loss-of-coolant accident (pressure and thermal loadings) or a seismic event. Moisture may collect in the sand pocket and cause corrosion problems. However, the sand pocket in some Mark I containments is covered by a galvanized steel plate to prevent moisture from entering; drains remove any moisture that may collect on top of this plate. The concrete-to-metal interface where the drywell becomes embedded in the concrete is usually sealed (e.g. polysulfide seal) to prevent moisture ingress. Moisture may also collect in this region to cause corrosion if there has been a breakdown of the seal. Penetrations for high-energy pipelines have two- or multi-ply Type 304 stainless steel expansion bellows to accommodate thermal movements between the pipe and containment shell. These expansion bellows serve as part of the primary containment. A guard pipe is installed between the pipes and bellow to prevent damage to the bellows during an unlikely pipe-rupture event. Insert plates are used to reinforce the drywell near penetrations.

The pressure suppression chamber, or wetwell, is a carbon steel pressure vessel that is formed by 16 to 20 mitered cylindrical segments joined together to form the shape of the torus. Depending on the plant, the torus cross section has a major diameter between about 29 and 34 m and a minor diameter between about 8 and 10 m. The torus is located below and encircles the drywell, and is normally about half-full of water. Vertical support at each mitered joint is provided by ring girders. Original designs also provided a horizontal restraint system on the freestanding suppression chamber to transfer horizontal earthquake loads to the reactor building while allowing for overall thermal expansion of the wetwell. These support systems have been substantially modified and strengthened due to recently defined hydrodynamic loads so that they can transfer all postulated vertical loads to the reactor building foundation. Typically, eight to ten equally spaced vent pipes form a connection between the drywell and the pressure suppression chamber (see next paragraph). In addition, the pressure suppression chambers are equipped with access hatches having bolted flange connections, which may be up to 1.2 m in diameter, to permit personnel access and transfer of equipment and components. Also, there are several other penetrations having diameters up to about 0.6 m that relate to the suppression chamber's role as a heat sink or source of water for the emergency and hot standby core cooling system. The early Mark I plant designs have a carbon steel pipeline below the pressure suppression torus in direct connection with the suppression chamber. The emergency core cooling and the heat removal systems are connected to this line and take pump suction from the pressure suppression chamber water. Interiors of the most wetwells are painted with a zinc-rich primer coating to help resist corrosion. Red lead, modified phenolic, and epoxy coatings also have been employed.

The vent system connects the drywell to the suppression chamber with eight to ten vent pipes, equally spaced around the circumference of the containment. Diameters of the vent pipes range from about 1.7 to 2.1 m. The vent pipes have single- or double-ply, Type 304 stainless steel expansion bellows where they penetrate the wetwell to accommodate differential motion between the drywell and the suppression chamber. The typical thickness of one ply is 2.0 mm. Jet deflectors installed in the drywell at the entrance of each vent pipe prevent possible damage to the vent pipes from jet forces that might accompany a pipe break in the drywell. The vent pipes exhaust into a continuous vent header, about 1.2 m diameter, from which downcomer pipes extend into the suppression chamber pool about 0.9 to 1.2 m below the normal pool water surface level. Both the inside and outside surfaces of the vent system are generally covered with protective coatings.

Mark II containments

The Mark II containment design retained the basic pressure suppression function of the Mark I containment, but rearranged the drywell and the suppression chamber into an ”over/under“ single pressure vessel configuration as illustrated in Fig. 2.2. The vessel is supported by a concrete basemat in which the bottom of the vessel is embedded. The drywell and pressure suppression chamber are separated by a diaphragm slab. Vertical downcomers connect the two volumes. Steam released into the drywell during a LOCA is directed into the downcomers and is discharged below water level in the suppression chamber. Mark II containment design has been implemented using three different construction techniques: (1) free-standing steel (only one plant: Washington Public Power Unit 2), (2) reinforced concrete with steel liner (e.g. Nine Mile Point Unit 2, and Limerick Units 1, 2), (3) post-tensioned concrete with steel liner (two plants: LaSalle Units 1, 2). Only the free-standing steel design will be discussed below.

The inside diameters of the suppression chamber cylinders range from about 21 to 27 m with a height of about 18 m. The diaphragm floor slab is constructed of reinforced concrete and is supported at the interior and exterior edges, and by columns that extend from the floor of the pressure suppression chamber. Penetrating the diaphragm and protruding downward into the pressure suppression chamber pool water are downcomers, generally 90 to 120 in number and about 0.6 m diameter. The downcombers are typically braced structurally to increase their resistance to lateral loads. Each vent opening is shielded by a steel deflector plate to prevent overloading any single vent by direct flow from a pipe break to that particular vent. Two equipment hatches, about 3.6 m in diameter, are contained in the drywell wall to permit transfer of equipment and components. Also a control rod drive removal hatch, about 1 m in diameter, is contained in the drywell wall. Two access hatches, about 1.2 m in diameter, are contained in the pressure suppression chamber wall to permit personnel access and transfer of equipment and components. A number of process pipe and instrumentation penetrations are also contained in the containment wall. Inside steel surfaces are generally covered by protective coatings to minimize corrosion. A reinforced concrete reactor building surrounds the containment vessel. A gap of polyurethane foam material, sized to accommodate the vessel expansion from the design temperature in conjunction with the design internal pressure, is contained between the pressure vessel and the concrete reactor building. Drywell penetrations that extend from the steel pressure vessel through the concrete wall are surrounded with concentric pipe sleeves. Gaps are provided to permit free movement of the attached piping and instrumentation lines.

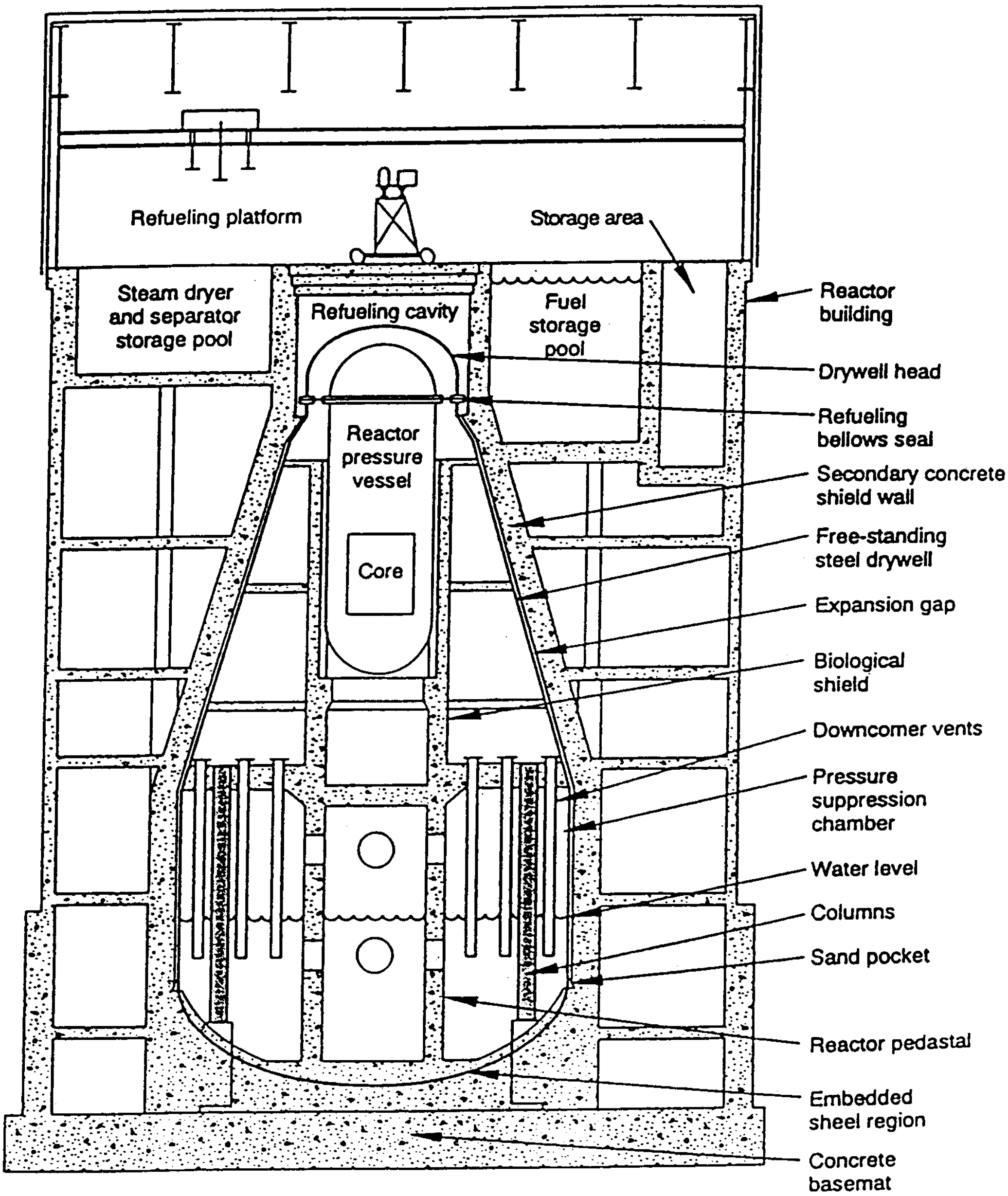


FIG. 2.2. GE-BWR Mark II design.

Mark III containments

The Mark III containment, shown in Figure 2.3, is substantially larger than either the Mark I or II vessels and houses nearly all the reactor building components. The Mark III containment consists of a drywell and pressure suppression chamber inside a primary containment shell that is surrounded by an enclosure or shield building and various equipment rooms that function as part of the secondary containment boundary. The Mark III containment design has been implemented using two construction techniques: (1) free-standing steel (e.g. Perry Unit 1, River Bend Unit 1, and Leibstadt), and (2) reinforced concrete with steel liner (e.g. Grand Gulf Unit 1 and Clinton 1). Only the free-standing steel design will be discussed below.

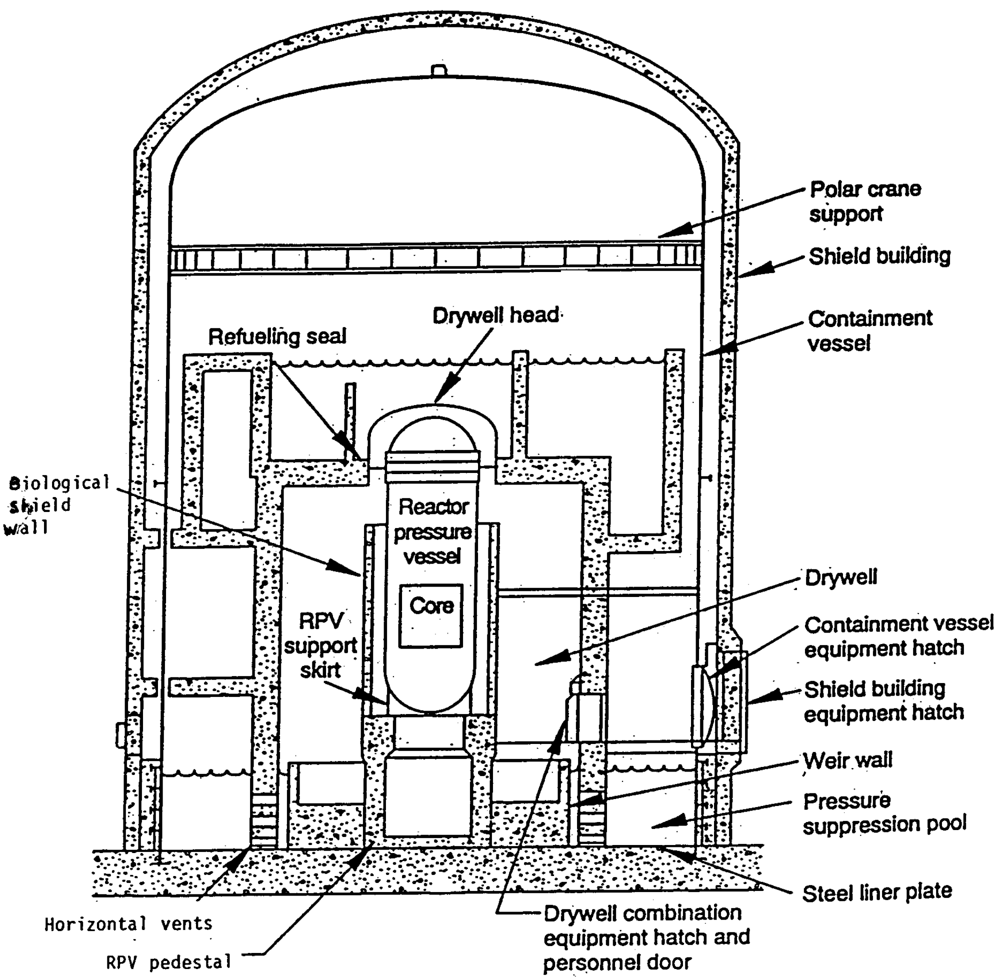


FIG. 2.3. GE-BWR Mark III design.

In all cases, the drywell is a reinforced concrete structure in which the reactor pressure vessel and much of the high-pressure piping are housed. In comparison to the Mark II containment, the Mark III containment is a simpler structural design. The downcomers have been replaced by a weir wall with horizontal vents (about 125) to separate the drywell from the suppression pool. The volume and arrangement of the Mark III containment provides lower containment design pressure and does not require nitrogen inerting. The fuel pool has been moved out of the reactor building. However, most of the support equipment placed in the Mark I and Mark II reactor buildings (e.g. the ECCS and RHR systems) are located inside the Mark III containment building, except at a few plants in Europe where there is an adjacent reactor building. The Mark III containments are provided with one large equipment hatch and two personnel airlocks of similar design to those in the Mark I and II containments. The basemat, typically about 3-m thick and lined by steel plate, provides a major portion of the load-carrying pressure boundary in that it is the bottom head of the pressure vessel. Similarly, portions of the drywell and weir wall that are submerged in the pressure suppression pool are provide with steel liner plates, primarily to prevent contact of the pool water with the concrete.

2.2.1.3. Materials

All containments include pipes, electrical penetration assemblies, equipment hatches, manways, airlocks, etc., as part of the pressure boundary. These components generally are either welded or bolted to the liners and shells and typically have compositions and properties that are different from those of the liner and shell materials. Leaktightness of the containment pressure boundary is provided by a combination of nonmetallic seals and gaskets.

The *ASME Code* permits the use of certain materials for fabrication of containment pressure boundary components. These materials must conform to ASME or American Society of Testing and Materials (ASTM) specifications. Section II, Parts A and D of the ASME Code provide specifications and property values for ferrous materials that are acceptable for use. Although the list of acceptable materials is fairly extensive, metal containments have been primarily fabricated of low- or unalloyed steels such as ASME SA-516 (Grades 60 or 70), ASTM A 212 (Grade B), and ASME SA-537 (Grade B) materials. Mark III free-standing steel containments primarily use ASME SA-516 (Grade 70) material for construction, with the shell plate in the pressure suppression pool clad with ASME SA-240 (Type 304) stainless steel to avoid contact of the carbon steel plate with water. Closure heads, access airlocks, and penetrations are also normally made of low-alloy steels with the exception of the expansion bellows of piping penetrations that are made of Type 304 stainless steel.

2.2.2. Siemens-KWU Baulinie “69” and “72” designs

2.2.2.1. Design requirements

The Siemens-KWU design specifications for containments are based on the ”RSK Leitlinie für Druckwasserreaktoren“ [2.7] (a corresponding document for BWR’s does not exist), the conventional *German Codes and Standards for Pressure Vessels* (AD-Merkblätter, DIN) [2.8], the *ASME Code* rules for stress and fatigue analysis, and (for newer plants) the special German containment rule KTA 3401 (No. 3401.1 - 3401.4) [2.9]. The design specifications are approved and accepted by the authorities and provide the code basis for design, manufacture and testing.

2.2.2.2. *Containment descriptions*

Two basic designs have been developed by Siemens-KWU: “Baulinie 69” and “Baulinie 72.” Basic attributes of these designs are provided in Table 2.3.

TABLE 2.3. ATTRIBUTES OF SIEMENS-KWU PLANTS

	Baulinie 69 900 MW	**Baulinie 69 1300 MW**	**Baulinie 72 1300 MW**
Drywell design overpressure (bar)	3.25–3.40	3.50	3.30
Drywell design temperature (°C)	135–146	146	146
Drywell air volume (m^3) × 1000	3.70–3.85	4.97	8.64
Suppression chamber design overpressure (bar)	3.25–3.40	3.50	3.30
Suppression chamber design temperature (°C)	90	90	90
Suppression chamber air volume (m^3) × 1000	2.17–2.28	2.72	6.10
Suppression chamber water volume (m^3) × 1000	2.23–2.60	3.70	3.13

Siemens -KWU “Baulinie 69”

The first Siemens-KWU design (S/KWU Baulinie 69) is shown in Figure 2.4 and has been utilized in several power stations in Germany (e.g. Brunsbüttel, and Philippsburg Unit I). The steel pressure vessel is in the shape of a ball with a diameter of 27 or 29.6 m and a connected cylindrical portion at the bottom. The closure at the bottom has a torospherical shape. The reactor pressure vessel is contained in the spherical portion of the containment. A flanged, removable closure is provided at the zenith of the sphere to provide access to the reactor vessel during refueling.

The interior of the spherical portion of the steel containment vessel is subdivided by steel structures into a drywell chamber (pressure chamber) and a suppression chamber (wetwell). The drywell chamber surrounds the reactor vessel and houses the feedwater and main steam lines up to the penetrations, the safety and relief valves, the control rod drives, and the motors for the internal recirculation pumps. The interior concrete structure also functions as a biological shield and a fragment catcher for the protection of the containment steel shell in the event of a LOCA. The suppression chamber is located in the equator region of the sphere and contains a large volume of demineralized water. Downcomer pipes penetrate the upper concrete floor for venting the drywell atmosphere into the suppression pool during a LOCA. The suppression chamber is part of the load-bearing shell. Support of the steel containment is provided by a reinforced concrete foundation. Flanged openings for installation and maintenance activities are located at the upper spherical chamber, below the suppression pool, and in the suppression chamber above the water level. Most of the pipe and cable penetrations are located in the upper spherical chamber or in the cylinder below the spherical steel portion of the containment. Pipe penetrations are either of the check valve design or have bellows.

Prior to each startup, the primary steel containment is purged and inerted with pure nitrogen to prevent ignition of any hydrogen and oxygen mixture that may occur following an accident. This type of containment has been used for nuclear power plant capacities of both 900 MW(e) and 1300 MW(e).

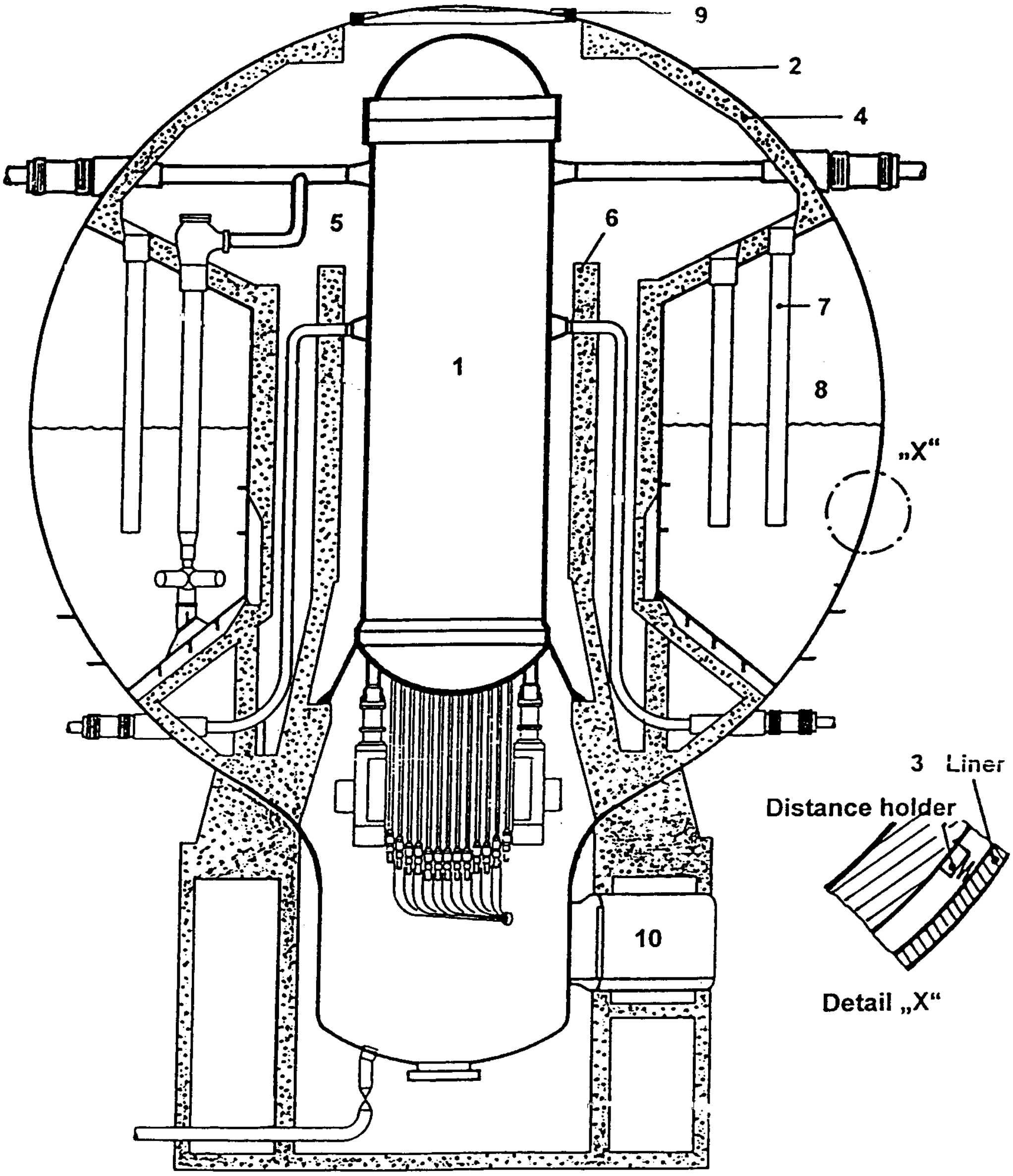
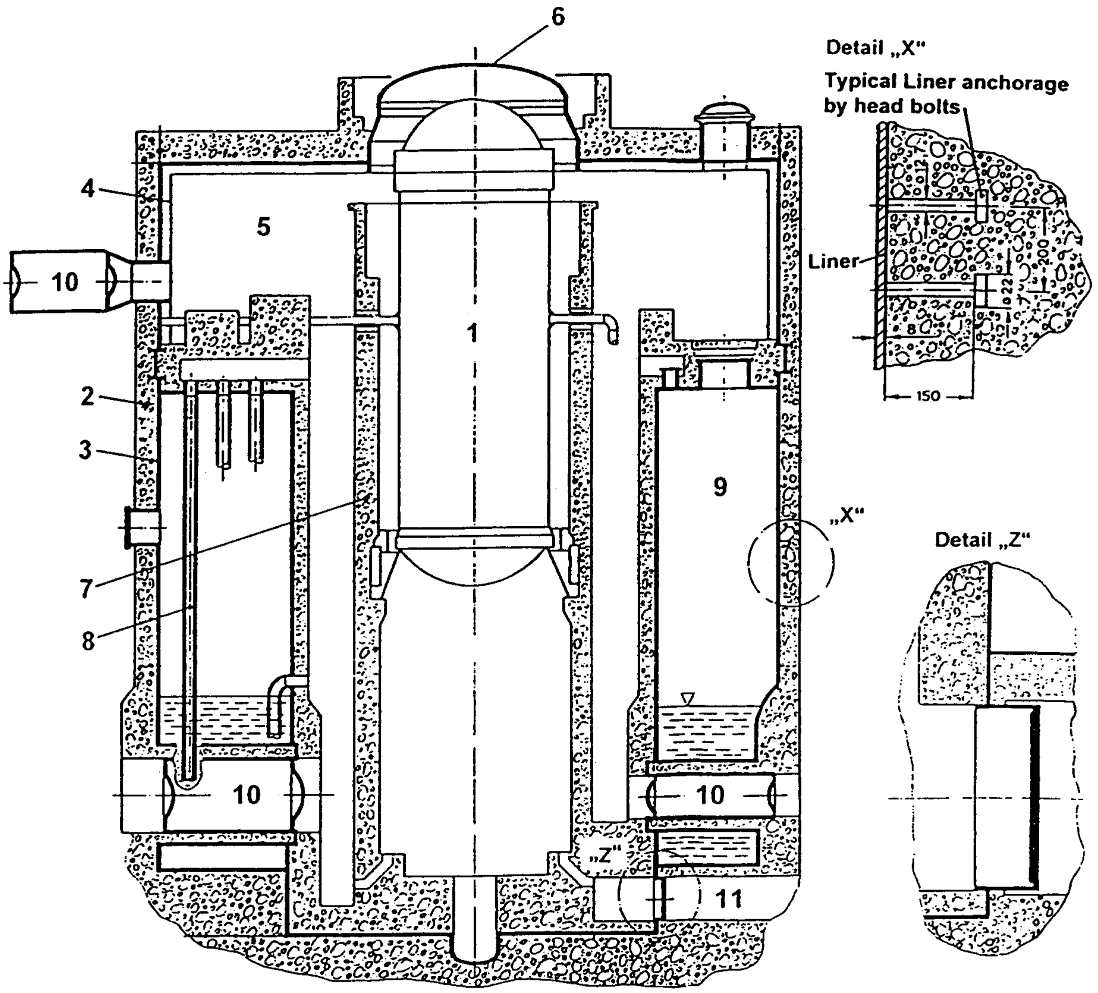


FIG. 2.4. Siemens-KWU Baulinie "69" design.

Siemens -KWU "Baulinie 72"

The Siemens-KWU design "Baulinie 72" (S/KWU Baulinie 72) has been utilized in the Gundremmingen B and C power plants. The pressure suppression system and nuclear steam supply system are contained within a reinforced concrete reactor building as shown in Figure 2.5. The containment consists of a cylindrical prestressed concrete structure having a removable drywell cover (drywell head) located above the reactor pressure vessel. Other concrete structures subdivide the inside volume of the containment into an annular condensation chamber (suppression chamber) and a drywell chamber. Both the containment and the condensation chamber are lined on the inside with steel to provide a leaktight barrier. The reactor pressure vessel is surrounded by a biological shield. Steam released into the drywell during a LOCA will flow through the condensation pipes (downcomer pipes) into the condensation pool (suppression pool). Access to the containment is provided by air looks (access hatches) in the lower part of the drywell chamber.



1 = Reactor Pressure Vessel
2 = Prestressed Safety Containment
3 = Steel Liner
4 = Protective Concrete Shield
5 = Drywell
6 = Drywell Hatch
7 = Biological Shield Wall
8 = Downcomer Vent Pipes
9 = Pressure Suppression Pool
10 = Locks
11 = Cable Cannels

FIG. 2.5. Siemens-KWU Baulinie "72" design.

2.2.2.3. Materials

Materials used for fabrication of the containment are primarily German fine-grained and boiler steels (e.g. WB25, BHW 25, Aldur 50 D, 15 MnNi6 3, and H II). Austenitic stainless steels (e.g. 1.4541 and 1.4550) have been used for some piping and bellows. The inner and outer surface of the containment, including areas covered by concrete, are coated with a primer based on zinc silicate (e.g. zinc epoxy material). Additionally, the exposed steel surfaces of the drywell, airlocks, and liners have a final decontamination coating based on an epoxy-resin material. The dry film thickness of primer and finishing coat are specified to be 50 and 100 μm, respectively. Corrosion protection of the wetwell is provided by coating with a coaltar-epoxy material. The coating system is specified to meet stringent requirements for both manufacture and application. Prior to coating application, the steel surface is properly conditioned (e.g. cleaned). Performance requirements for the coatings include:

- maximum temperature of 40°C in normal operation, with short time exposure to 110°C at LOCA;
- maximum radiation exposure of 1kGy within 40 years;
- mechanical loading capacity;
- capability in decontamination requirements; and
- testability of seam welds for ultrasonic in-service inspection.

The coating is applied in four layers (total dryfilm thickness of 800 μm) according to specifications related to workmanship and control of temperature and relative humidity. After application, the coating is inspected to:

- check the intermediate and total dryfilm thickness;
- check for surface pinholes using an electric-based detector;
- check the bond strength of production test coupons using the pull-off test method; and
- check the chemical and thermic resistance in demineralized water (max. 60°C).

2.2.3. ABB Atom designs

2.2.3.1. Design requirements

The first containments (Type I) were designed before Section III, Division 2 of the *ASME Code* was published and were based on relevant national codes (e.g. code for prestressed concrete bridges, accepted by national authorities). Currently, prestressed concrete containments are designed in accordance with Section III, Division 2, Article CC-3000 of the

ASME Code. The steel dome containment closure and other pressure-retaining steel structures are designed in accordance with Section III, Division 1, Article NE-3000 of the *ASME Code*. Design of structures or structural parts that are not pressure-retaining or of importance to the primary safety functions of the containment are designed in accordance with relevant Finnish or Swedish codes. All parts of the containment with penetrations and access openings are designed and constructed according to Safety Class 2, with the exception of pressurized process piping in the penetrations which are Class 1, if the main system is Class 1. The capability of the containment vessel to withstand postulated pressure loads is tested by pressurization up to 15 per cent in excess of the design overpressure. Leakage-rate testing of the containment structure is carried out in accordance with requirements contained in Appendix J of U.S. *Code of Federal Regulations* (Part 50). In addition to the general design basis, provisions are taken, in accordance with Swedish and Finnish codes, to manage severe accidents beyond the current design criteria. Thus a filtered containment venting system is installed to prevent containment failure due to overpressurization, and a water pool is located below the reactor for cooling of the core after a reactor vessel melt-through.

2.2.3.2. Containment description

The containment system of all ABB Atom reactors is based on pressure suppression and consists of a prestressed concrete structure with an embedded steel liner to provide leaktightness. Access to the containment interior is through metallic openings (hatches) and penetrations. The reactor pool is located above the containment dome and is filled with water during normal operation to act as a radiation shield. During removal of the reactor pressure vessel head, the pool is drained and then refilled during refueling. Basically, the pressure-retaining barriers of the vessel are provided by a cylindrical slip-form constructed wall, and a roof slab and bottom slab that are fixed-form constructed. The roof slab and the upper part of the cylindrical wall are rigidly connected to the pool walls as well as the bottom slab of the fuel pools. Thus all these parts interact structurally. The upper face of the roof slab also functions as part of the floor in the reactor pools. Internal containment structures are basically a reactor pedestal structure and a partition structure fabricated of conventionally reinforced concrete. The internal structures divide the containment vessel volume into three separate compartments, (i.e. drywell, wetwell and reactor shaft). In the upper drywell, a carbon steel liner is embedded about 200 to 300 mm into the concrete. The liner is embedded in concrete to protect it against mechanical impacts (missiles), temperature transients, and corrosion. The wetwell is divided into the condensation pool and the gas compression chamber above the water level. The blowdown pipes from the drywell and the pipes from the relief system discharge into the condensation pool. In the wetwell, missile protection is not necessary and temperature effects are moderate for the liner. Wetted surfaces of the liner are clad with stainless steel plate. The reactor pedestal consists of eight columns that support the biological shield wall which in turn supports the reactor pressure vessel. The partition structure between the drywell and wetwell is connected to the containment vessel in such a way that relative displacements are prevented in the horizontal and tangential directions, but not in the vertical direction. A coating is applied to the partition slab at its outer edge to seal the drywell from the wetwell. Vacuum breakers restrict the differential pressure across the partition structure, which may occur following an accident in the reactor containment. The reactor pressure vessel is positioned inside the biological shield with provisions for preventing lateral movement but permitting radial movement due to thermal expansion. In the more recent designs, the reactor pressure vessel is hanging on the shield, anchored to it with vertical prestressing tendons. The opening for the reactor is covered with a containment dome, a steel dome that is bolted to an upper flange of vertical steel cylinder that is anchored to the concrete roof structure and welded to the steel liner. The reactor vessel flange is permanently connected to the containment dome counterflange, and thus to the condensation pool bottom, with a flexible stainless steel structure with openings that are opened during reactor operation and closed during water filling for refueling. The containment vessel structure is provided with access penetrations as well as a large number of penetrations for pipes and electrical cables. The penetrations consist of pipes that are welded to the steel liner and anchored to the concrete structure. The penetrating devices (e.g. pipes, electrical cable assemblies, and access airlocks) are in turn welded to the penetration pipes. For piping systems, a short length of the process pipe is forged in one piece and welded to the penetration anchor pipe so that the containment penetrations are always anchor points without bellows. The process pipes of high-temperature power systems are insulated inside the anchor pipe. Expansion joints isolate the containment structure from the reactor building.

The basic design specification for the plant is done by ABB Atom with detailed design and construction performed by the construction company. There are two basic containment designs dependent of whether the plant has external circulation loops (Type I, Figure 2.6) or internal pumps (Type II, Figure 2.7.). Design developments are aimed at a configuration with the reactor vessel situated lower in the containment to decrease the bending moment resulting from dynamic forces. Also, the newer design has increased the rigidity of the containment to better accommodate dynamic loads to provide improved seismic and wet-well dynamic behavior. Table 2.4 provides some basic design parameters for the Type I and Type II plants.

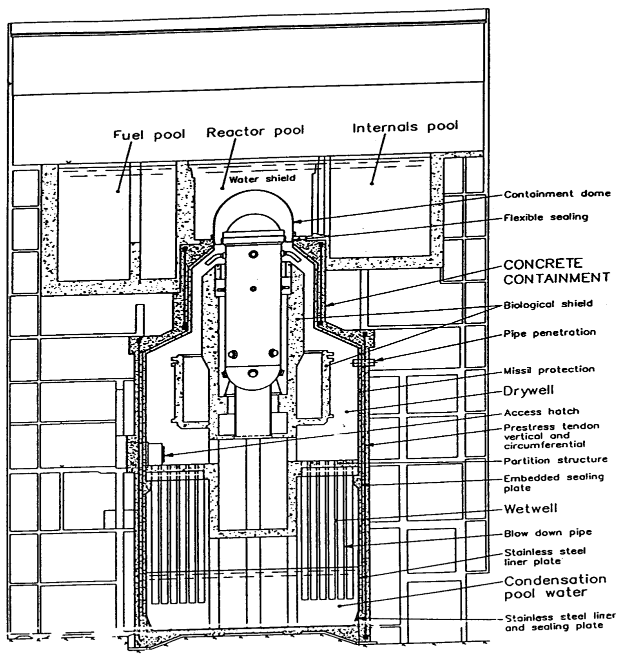


FIG. 2.6. ABB-Atom Type I design.

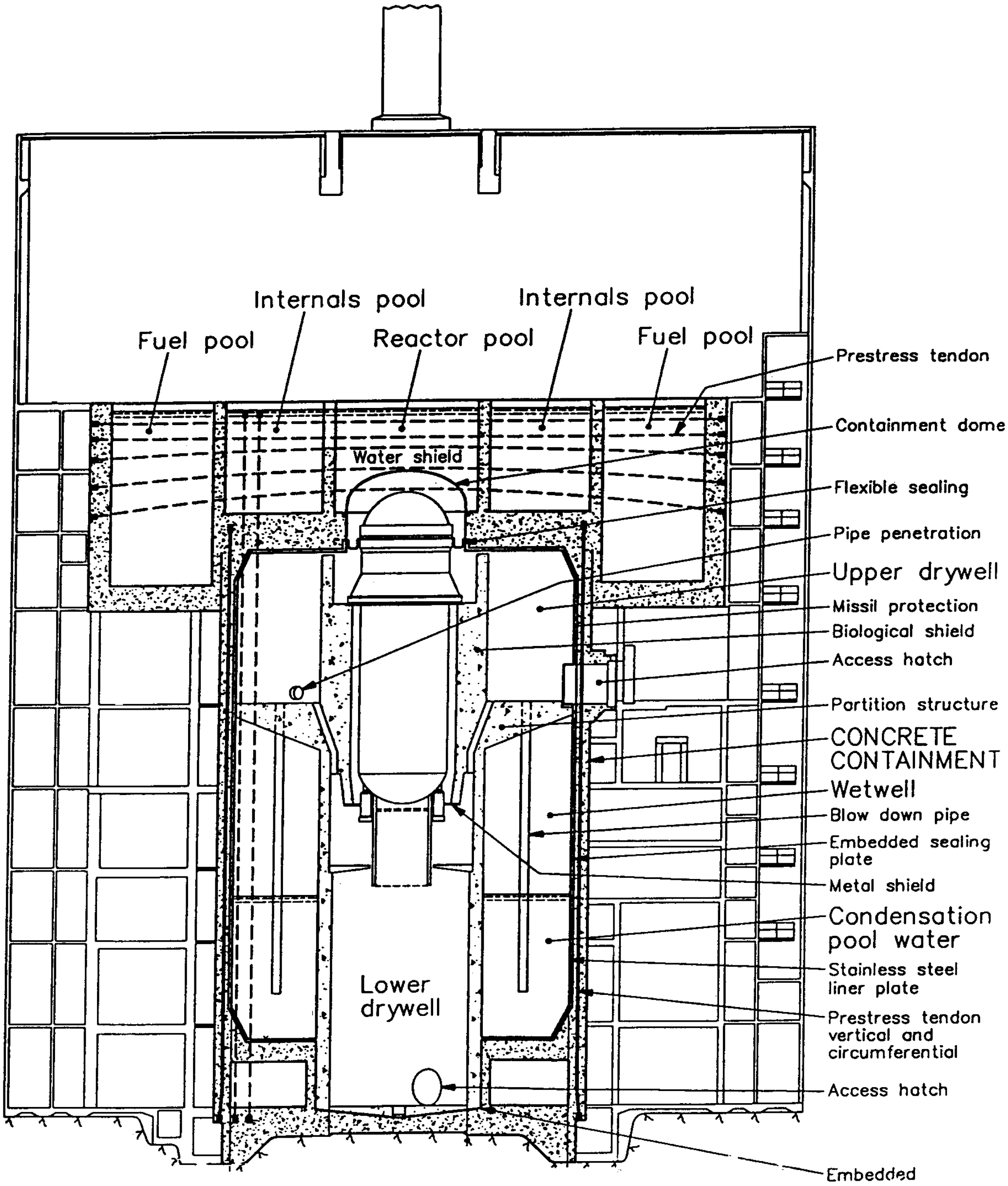


FIG. 2.7. ABB-Atom Type II design.

2.2.3.3. Materials

Metal components of the containment are fabricated and constructed primarily from low-alloy carbon steel type materials suitable for pressure-retaining parts. The embedded steel liner and embedded parts of penetrations are typically fabricated of SS141330 (i.e. ASTM A - 285) or SS141434 (ASTM A-106) carbon steel materials. Process piping components are fabricated from the same materials as the main process system (i.e. carbon steel or stainless steel piping). Low-alloy pressure vessel steel is used to fabricate the access openings. The older Type I containment domes are coated to provide protection against wetted outside and moist inside conditions. Containment domes for the newer Type II containments are fabricated from stainless steel SS142333 (ASTM A-304).

TABLE 2.4. DESIGN PARAMETERS FOR ABB ATOM PLANTS

	Type 1	Type 2
Design overpressure (bar) max	3.5–4.0	3.7–5.0
Design abs. pressure (bar) min	0.5	0.5
Design temperature dry well (°C)	150–170	172–175
Design temperature wet well (°C)	110–157	130–140
Dry well free volume (m^3) × 1000	3.68–4.98	4.55–5.86
Wetwell free volume (m^3) × 1000	1.86–2.96	2.85–3.03
Condensation pool water volume (m^3) × 1000	1.94–1.95	2.70–3.17

2.3. SUMMARY

The BWR containment is designed to act as the final barrier to the release of fission products to the environment that may occur as a result of a design-basis accident. A general description of the primary metal components of BWR containment systems is provided as well as the functions they provide. The design basis, description, and materials of construction for containment designs developed by three contractors (General Electric, Siemens-KWU, and ABB Atom) in Member States are described. With one exception, the containment designs are of the pressure-suppression type that consist of a drywell (reactor pressure vessel and recirculating and other associated piping) and a wetwell, or suppression chamber (contains a large volume of water). Materials used in the construction of the containment pressure boundary are primarily carbon or low-alloy steels. Stainless steels have been used in certain areas of the containment (e.g. wetwells and penetration bellows).

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Chapter 3

SERVICE CONDITIONS

This chapter describes the service (i.e. environmental and operating) conditions inside and outside the BWR containment. The service conditions indicate a possible presence of specific ageing mechanisms for the different components of BWR containments and are helpful in assessing the rate of degradation.

3.1. CONDITIONS INSIDE CONTAINMENT

Environmental conditions inside the BWR containments are controlled during normal plant operations (see previous chapter). Containment interior atmosphere is a controlled volume, with the pressure, temperature, and chemical makeup held at nearly constant values during operation. Heat generated by the nuclear reactor is removed by a ventilation and cooling system. Air chemistry inside the containment is also monitored and adjusted to maintain a nearly constant environment. Certain containments (e.g. BWR Mark I and Mark II) are inerted with nitrogen to prevent hydrogen-oxygen recombination. The oxygen content inside these containments is generally maintained at a level less than four percent by volume. Other containments (e.g. BWR Mark III) are filled with air.

Acceptable service temperature limits for the metallic materials used in containment construction are provided in applicable codes and standards (e.g. *ASME Code*). These limits have a significant influence on the allowable operating temperature range of the containment. One area of potential elevated temperature concern is in the containment shell near high temperature piping penetrations where temperatures can also fluctuate. To minimize the temperature effects most plants insulate the pipes within the penetrations.

Water is normally present inside the BWR containments in the pressure suppression pools and chambers. With the exception of one or two plants, chromated water (toxic) is no longer used in suppression pools, thus avoiding its circulation through the reactor core during emergency conditions and avoiding the problem of discharging toxic water when the torus is drained. The presence of non-chromated water however can increase the susceptibility of the suppression pool components to corrosion. Surface coatings and strict control of water quality are used to minimize the corrosion potential. Defective coatings, seals or gaskets, however, can lead to potentially corrosive situations if excess water results that can flow over or accumulate on pressure-retaining components. Although high radiation levels are generated inside the reactor pressure vessel, biological shielding effectively limits exposure of metal components of BWR containment systems to levels that are considered safe for occupational workers. Consequently, end-of-life fluence effects on the containment pressure boundary are anticipated to be quite small.

3.2. CONDITIONS OUTSIDE CONTAINMENT

Environmental conditions outside the containment can be significantly different from conditions on the inside. Exposed surfaces of containments that are exposed to the natural environment (e.g. Big Rock Point) are protected by an extensive coating system that can be periodically inspected and maintained. Typically, however, the steel containment vessel is surrounded by another structure, such as a reactor or shield building, that protects the containment from the wind, sun, rain, snow sleet, etc. Although the temperatures, humidity, and pressures within the reactor (or shield) building are controlled, certain local environments within these structures can be relatively harsh. The outer surfaces of metal containments that are in contact with compressible filler materials that separates the steel shell from the concrete reactor shield building, surfaces in contact with or embedded in concrete, and areas where water can accumulate probably have the greatest potential for degradation.

During construction of the reactor shield building in some plants (e.g. GE BWR Mark I) a compressible filler material was placed against the shell to form a permanent 51-to 76-mm gap between the steel shell and the concrete shield wall to accommodate thermal expansion and deformation under design-basis accident conditions. In some plants the filler material was removed while in others it was not. If water is introduced from flooding of the BWR drywells during refueling, or is present as a result of leaks from failed penetration seals, piping gaskets, or bellows expansion joints, the filler material can trap the moisture against the steel shell to cause corrosion. Contaminants (e.g. Na, K, Ca, Mg, SO_4, and Cl) that may be present in some of the fill materials may accelerate the corrosion process [3.1]. Areas of the containment where filler or other materials have been used to seal the space between structures in close proximity to the metal shell (e.g. floors) also provide locations where fluids can accumulate and potentially corrode the shell.

Carbon steel materials that are in contact with or embedded in Portland cement-based materials are normally exposed to a high pH environment that promotes the formation of a passive iron oxide film that tends to inhibit corrosion. However, even in this high pH environment chloride ions that penetrate the concrete can destroy the passive film on the steel and produce corrosion. Potential sources of chloride include seawater for plants affected by ocean environments, and groundwater contaminated by chlorides that permeates through the concrete to the level of the embedded carbon steel components. Fluid intrusion (e.g. water, cleaning fluids, and decontamination fluids) between the carbon steel shell of the metallic pressure boundary adjacent to the concrete due to a breakdown of the interface seal can also produce corrosion.

In some designs (e.g. GE BWR Mark I) a sand pocket is located at the concrete-steel interface to reduce stress concentrations. These sand pockets are connected to drains that keep excessive moisture from accumulating near the carbon steel shell. However, when the drains malfunction (e.g. improper construction or blockage) water can accumulate adjacent to the steel shell to cause corrosion.

REFERENCES TO CHAPTER 3

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Chapter 4

AGEING MECHANISMS

The ability of the containment pressure boundary to perform satisfactorily under the design basis (as well as under higher loading conditions, such as resulting from a severe accident and seismic margin earthquake) is influenced by the complex interactions between containment system components, service conditions and ageing mechanisms that are present. Analysis of the potential impact of ageing mechanisms must be done taking into account all the appropriate system information, including the design, materials, fabrication and installation data, and operating and maintenance histories.

Ageing mechanisms are specific processes that gradually change characteristics of a component with time and use. Ageing degradation are those cumulative changes that can impair the ability of a component to function within acceptance criteria. The rate of degradation is influenced by the sustained service conditions; service conditions outside prescribed limits (e.g. higher temperature and humidity, water leakage, and acid spills) which are caused by design, fabrication, installation, operation, and maintenance errors, can accelerate the rate of degradation.

This section describes the ageing mechanisms that can affect metal components of BWR containments and evaluates the potential significance of the effects of these mechanisms on the continued performance of BWR containments safety functions throughout the plant service life. Areas of the containment pressure boundary that are at risk to degradation are identified. Additional information to that presented in this section is available [4.1-4.4].

4.1. STEEL CONTAINMENT VESSELS

4.1.1. General corrosion

Corrosion is a chemical or electrochemical reaction between a material and the environment that produces a deterioration of the material and its properties. General corrosion of steel is degradation that produces uniform thinning and proceeds without appreciable localised attack. This type of corrosion is characterised by slow, nearly uniform loss of metal thickness over a wide area. General corrosion begins at an exposed metal surface and progressively alters the geometry of the affected component without changing the chemical composition of the material or its microstructure. Degradation initiates with the formation of a corrosion product layer and continues as long as at least one of the reactants is able to diffuse through the layer and sustain the reaction. The composition and characteristics of the corrosion product layer can have a significant influence on the corrosion rate.

Under the temperature conditions of a BWR containment, primary corrosion mechanisms are electrochemical processes that require an electrolyte — normally an aqueous solution of salts, acids, caustics, or oxygen. Leaks in systems, flanges and pools (e.g. suppression, fuel, or refuelling), or condensation, water losses during maintenance, and groundwater outside containment can provide sources of humidity. Historical data for corrosion of carbon steel exposed to an industrial environment indicate general corrosion rates in the range of 0.003 to 0.03 mm/yr [4.5]. Specific forms of general corrosion that could potentially affect the BWR containment metal components include atmospheric, aqueous, galvanic, and stray-current.

Atmospheric corrosion is the gradual degradation or alteration of a material by contact with substances such as oxygen, carbon dioxide, water vapour, and sulphur and chloride compounds that are present in the atmosphere. This is probably the most common form of general corrosion and is characterised by uniform thinning. Because atmospheric corrosion is an electrolytic process, only a very thin film of water is required to accelerate degradation. Although the rate of atmospheric corrosion is dependent on the humidity, temperature, and levels of sulphate, chloride, and other atmospheric pollutants, it is usually not constant with time and tends to decrease as the length of exposure increases. In nuclear power plants atmospheric corrosion can be suspected whenever uncoated carbon and low-alloy steel components are exposed to air with a relative humidity that exceeds about 70 percent.

Corrosion of metals in aqueous environments occurs when two or more electrochemical reactions take place on the surface causing the metal or alloy to change from a metallic state to a non-metallic state. The result may be either dissolved species or solid corrosion products. The driving force behind the process is the change in energy of the system as the metal converts to a lower energy form (e.g. rusting). *Aqueous corrosion* is similar to atmospheric corrosion except the metal surface is continually immersed. The corrosion rate of steel submerged in an aqueous solution depends on the temperature, flow rate, pH, and chemistry. Exposure to seawater or other types of severely contaminated water provide very corrosive environments for carbon, low-alloy, and stainless steel components. Stainless steel components are particularly susceptible to stress-corrosion cracking in chloride environments and in high-purity neutral-pH water that contains dissolved oxygen such as the recirculating coolant in a BWR. External surfaces of BWR Mark I metal containment shells and embedded portions of the liner of reinforced concrete containments are potentially susceptible to aqueous corrosion.

Galvanic corrosion is accelerated corrosion that occurs when a metal or alloy is electrically coupled to a more noble metal in the same electrolyte. The three requirements for galvanic corrosion are (1) materials possessing different surface potentials, (2) a common electrolyte, and (3) a common electrical path. The driving force behind the flow of electrons is the difference in potential between the two metals with the direction of flow depending on which metal is more active. The more active (less noble) metal becomes anodic and corrosion occurs while the less active metal becomes cathodic. Galvanic corrosion has been known to occur in dissimilar-metal, butt-welded piping systems that carry electrolytic solutions, with the most severe corrosion occurring adjacent to the weld on the anodic member. Destruction of weld metal in carbon steel piping systems also has occurred because the welds were anodic to the base metal. Physical damage to welds and base metal in nuclear plants can occur in locations where dissimilar metals are in contact and an electrolytic solution is present. Potential locations for galvanic corrosion in nuclear power plants include near vent line bellows (dissimilar-metal welds between stainless and carbon steels), flanged hatches in the wetwell, penetrations, and where equipment is attached to the containment through differential metal connections (e.g. supports, ducts, and grounding wires).

Stray-current corrosion is corrosion resulting from direct current (DC) flow through paths other than the intended circuit. After the electrical current leaves its intended path, it can pass through soil, water, or another electrolyte to find a low-resistance path, such as a buried metal pipe or some other metal structure, and flow to and from the structure, causing accelerated corrosion. Corrosion occurs at the point where the stray current leaves the metal structure and enters the surrounding electrolyte. Degradation due to alternating current (AC) is less than that caused by DC and decreases in severity as the frequency increases. Primary potential sources of stray current in nuclear power plants include cathodic protection systems, high-voltage DC systems, and DC welding operations.

4.1.2. Localised corrosion

Localised corrosion is similar to general corrosion except the rate of attack is usually much faster and the size of the affected area is significantly smaller. Damage caused by localised corrosion is often difficult to detect and quantify because visible surface flaws tend to be small and often do not provide a good indication of the extent of damage that has occurred under the surface. The most significant consequence of localised corrosion of steel used to construct metal components of BWR containment is loss of section caused by crevice or pit formation. This type of physical degradation is more likely to result in containment leakage than reduced load-carrying capacity. Pits and crevices caused by localised corrosion are thought to have no measurable effect on the mechanical properties of the containment steels. Specific forms of localised corrosion that could potentially affect the BWR containment metal components include crevice, pitting, and localised biological. Filiform corrosion is localised surface damage that occurs under organic coatings in the form of randomly distributed threadlike filaments that appear as worm-like blemishes. Filiform corrosion results from a nick, cut, pore, scratch, or disruption in a coating to produce a separation between the coating and the host steel component. This type of corrosion is not addressed below as it is considered to be a maintenance concern.

Crevice corrosion is localised attack of a metal surface adjacent to an area that is shielded from full exposure to the environment because of close proximity between the metal and the surface of another material. Narrow openings or spaces between metal-to-metal or nonmetal-to-metal components, cracks, seams, or other surface flaws can serve as sites for corrosion initiation. Moisture can enter into a crevice adjacent to an embedded steel surface at the steel-concrete interface, at an aged embrittled sealant, or through cracks in the concrete. Chances for crevice corrosion increase as the gap decreases (degree of tightness increases) and the depth or distance from the mouth increases. Stainless steels are more prone to crevice corrosion than carbon steels, particularly in the presence of chlorides. In nuclear power plants areas under hatch gaskets and bolts where coatings have deteriorated, locations where the metal pressure boundary is embedded in concrete, and the exterior drywells of the Mark I and II plants if the compressible fill has been left in place. Degradation of filler material in the gap between the concrete and steel can produce chemicals dissolved in water that can be very corrosive.

Pitting corrosion is localised degradation of a metal surface confined to a point or small area that takes the form of cavities. The cavities are generally irregularly shaped and may or may not become filled with corrosion products. Pitting usually affects metals that are covered with a very thin, often invisible, adherent protective surface film with the pits forming at weak spots in the surface film and at sites where the film is damaged mechanically under conditions where self-repair will not occur. Pitting corrosion is one of the most common types of localised corrosion encountered in aqueous environments and its significance depends on the thickness of the component and its penetration rate, which usually decreases with time. Pitting of all containment pressure boundary components made from all forms of steel is possible whenever they are exposed to aqueous environments. Potential locations include the torus wall where the coating has deteriorated, outer surface of Mark I containment near the sand pocket region and containment vessel wall near the concrete-metal interface where a sealant is missing or has deteriorated.

Localised *biological corrosion*, or *microbiologically-induced corrosion*, is deterioration of metal as a result of the metabolic activity of micro-organisms. The corrosion rate of metals in aqueous environments tends to be dependent on the rate at which dissolved oxygen can be delivered to the metal surface. Biological corrosion is usually localised because micro-organisms tend to settle in discrete colonies rather than uniformly over the surface of a material. Anaerobic corrosion due to attack by sulphate-bearing bacteria and aerobic corrosion under oxygenated conditions can cause degradation of iron and carbon steel, but localised biological corrosion of stainless steels can also occur. Biological corrosion occurs in stagnant or flowing water at moderate temperature. Potential areas where localised biological corrosion can occur are the Mark I sand pocket region, suppression pool region, containment sump region where concrete cracking permits water to come into contact with metal, areas where standing water accumulates, and sump line penetrations. Both aerobic and anaerobic microbes have been found in the sand pocket regions next to Mark I drywell exterior surfaces.

4.1.3. Mechanically-assisted degradation

Any degradation that is caused by mechanical action is considered mechanical-assisted degradation. Actions that involve both a corrosion mechanism and mechanical wear or fatigue also fall into this category. Under certain conditions, mechanical wear, maintenance and repair activities, and equipment failures can cause total loss of section thickness, wall thinning, discontinuities, stress concentrations, and dimensional changes in component geometry. Potential sources of mechanically-assisted degradation include erosion, fretting, cavitation, corrosion fatigue, surface flaws, arc strikes, and overload conditions. In general, the steels used to construct the containment pressure-retaining boundary components would not be expected to be routinely subjected to wear caused by erosion, fretting, or cavitation, and corrosion fatigue is not expected to be a generic concern [4.3]. Possible exceptions include the lubrite contact surfaces between the Mark I metal containment torus support column base plates and basemat due to relative motion during heatup, cooldown, and pressure testing; and steam impingement during safety-relief valve or other discharge may cause erosion of the passive surface coating (or even some of the metal surface) in the vent lines.

Steels that are used to construct the metal components of BWR containment systems are ductile materials that generally bulge, stretch, bend, or neck prior to fracture. Surface flaws such as notches, cracks, grooves, gouges, dents, and tool marks can be created during routine operations, inservice maintenance, repair actions, or equipment failures that generate missiles or pipe whips. The resulting stress concentrations, if located in a critical region, can contribute to premature structural failure at loads below those permitted in design or loss of leaktight integrity can occur. Flaws located in aqueous environments also can serve as initiation sites for filiform or crevice corrosion. Arc strikes formed during a welding process can cause loss of ductility in mild and low-alloy steels, hardening of higher carbon and alloy steels, or localised cracking in higher strength hardenable grades of steel. Equipment failure, excessive piping loads, and unanticipated thermal expansion or contraction are examples of overload conditions that can prematurely deform, bulge, stretch, bend, buckle, or neck pressure-retaining components. These effects can have a detrimental effect on the containment structural capacity and leaktight integrity.

4.1.4. Environmentally-induced cracking

Environmentally-induced cracking is a type of ageing degradation that occurs when cracks are produced in metals as a result of exposure to an environment. Stress-corrosion cracking, and hydrogen-induced cracking are two types of environmentally-induced cracking that potentially can impact the metal components of the BWR containment system. In general, as the yield strength and stress applied to a metal increase, its resistance to environmentally-induced cracking decreases. Hydrogen stress cracking should not be a serious problem for the metal components of BWR containment systems because neither hydrogen at high pressure nor hydrogen at high temperature are found inside the containment.

Stress-corrosion cracking is an ageing mechanism that requires the simultaneous action of a corrodent and sustained tensile stress to initiate and propagate cracks in metals and alloys. Relatively low tensile stresses, often below the yield strength of the material, can cause stress-corrosion cracking. These stresses may be produced by applied loads, residual stresses, or wedging action caused by the growth of corrosion products. Compressive stresses are beneficial in reducing or eliminating stress-corrosion cracking. Carbon, low alloy, and stainless steels exposed to various aqueous solutions (e.g. sulphates, hydroxides, chlorides, ammonia, fluorides, and carbonates) are susceptible to stress-corrosion cracking. Aqueous solutions present in nuclear power plants that could contribute to stress-corrosion cracking include such things as groundwater containing chlorides or sulphates and certain types of decontamination fluids. Certain alloys that may be resistant to general corrosion in a particular environment may be susceptible to corrosion in the presence of stress (e.g. austenitic stainless steel in the presence of a chloride environment). Stainless steel components and structures used at nuclear power plants that may be susceptible to stress-corrosion cracking include bellows, components of some electrical and piping penetrations, and some wetted portions of the ABB Atom designs. The ductile low-alloy steels used for construction of nuclear power plant containment vessels are not susceptible to stress corrosion, but to corrosion associated with mechanical wear or abrasion.

4.1.5. Fatigue

Fatigue is the progressive, localised, and permanent structural change that occurs in a material subjected to repeated or fluctuating strains at normal stresses that have maximum values less than the tensile strength of the material. Fatigue failure consists of three phases. During the crack initiation phase, initial fatigue damage leads to crack initiation. During the crack propagation phase, the crack grows to a critical size that depends on various factors including the material, environment, and stress level. When the crack reaches the critical size and the remaining uncracked section can no longer sustain the load, sudden failure of the remaining cross section occurs. Enhanced resistance to fatigue can be achieved by eliminating stress concentrations, avoiding the development of discontinuities, reducing residual stresses, and protecting the component from corrosion, erosion, and chemical attack.

The cyclic stresses to which the metal components of BWR containment systems may be subjected are the result of the following service-related loads:

- Startup/shutdown cycles (temperature transients);
- Pipe reactions (at penetrations);
- Leakage-rate and pressure testing; and
- Safety relief valve discharge testing (includes steam condensation loads).

The BWR containments are designed with sufficient margin to allow for these loads, and keep the overall fatigue usage small. However, heatup, cool down, and pressure testing operations introduce somewhat more severe cyclic stresses at several local sites in BWR containment metal components. These locations include sites with geometric discontinuities that act as stress raisers, and sites having adjacent materials with different thermal expansion coefficients. Examples of drywell sites with significant geometric discontinuities are the reinforcing plates near penetrations and the region connecting the cylindrical and spherical portions of the Mark I drywell or the Siemens-KWU "Baulinie 69" design. More critical are discontinuities, unforeseen in the design and not considered in the stress and fatigue analysis. Examples include misalignment resulting from weld shrinkage or offset between steel plates as a consequence of poor site construction which results in local wall thicknesses smaller than specified. If the calculated fatigue usage factor in such an area approaches the value of 1.0, the additional stresses resulting from the geometric discontinuity can be critical. Another area where fatigue can have an impact is the embedded portion of the drywell base. Under fluctuating temperature conditions separation at the concrete-metal interface (crevice) can occur due to different heat transfer coefficients. If moisture or water with dissolved corrosive chemicals enters into the crevice, corrosion assisted fatigue can result. Safety relief valve discharge, which occurs through the vent lines into the suppression pool, also can be a major source of fatigue in the suppression pool and the vent system components. Each discharge causes stresses in the suppression pool liner or the torus shell at the vent header/downcomer intersection. In GE Mark I containments the fatigue usage factor at the intersection between the downcomer and the vent header after 40 years of operation does not exceed 0.5 due to normal operation and routine testing.

Another potential fatigue source is the condensing of steam bubbles to produce pressure oscillations that in turn produce oscillating forces on the suppression pool liner. The suppression pool liner is normally anchored in the concrete with welded bolts, studs, or ribs. The oscillating forces can induce fatigue stresses in the liner plate and the spotwelds of the anchor-bolts. However, research in Japan has demonstrated that the suppression pool liner maintains its integrity under loadings of this type.

4.2. STAINLESS STEEL-RELATED COMPONENTS

Stainless steel components or structures are not normally used in nuclear power plant containments except for bellows, penetration assemblies (e.g. electrical and pipe penetrations involving stainless steel systems), seal structure between the pressure vessel flange and drywell of some designs, and the containment dome and wetwell cladding of the newer ABB Atom plant designs. Corrosion without stress, such as pitting and intergranular corrosion, requires combination with a halogen, which is normally avoided. However, contamination by corrosive chemicals could occur during maintenance as a result of decontamination of systems or tools. The major operating stresses in bellows result from pressure and from large relative deflections in axial and lateral directions. Pressure produces meridional membrane stresses, whereas deflections produce meridional bending stresses. The containment bellows are susceptible to low-cycle fatigue and stress corrosion cracking during normal operation.

Fatigue is a major design consideration for the bellows because of the number and magnitude of deflection cycles to which they are subjected. Fatigue analysis of bellows is generally performed in accordance with codes such as the ASME Code (Section III, Subsection NE). In practice, the design requirements for bellows generally specify a conservative number of anticipated cycles. Since each ply of the bellows is relatively thin, scratches or indentations incurred during fabrication or operation can create stress concentrations to reduce the fatigue life. Also, misalignment during installation can reduce the fatigue life by inducing additional stresses. One investigation determined that surface flaws have a greater impact on the fatigue life than misalignment [4.6]. In conducting a fatigue analysis for bellows, the as-installed geometry and fatigue strength reduction factors are considered [4.3].

Corrosion of stainless steel components or structures is more likely to occur if stresses such as residual stresses are present, or deformations or mechanical wear is present. The bellows of piping penetrations can have relatively high residual stresses from the cold forming fabrication process, superimposed with stresses due to deformations during transients or leakage-rate testing. Local damage due to impact or wear could also be sources of residual stresses. In both situations, the corrosion risk exists if humidity and chemicals are present. Transgranular stress-corrosion cracking of bellows has occurred resulting from exposure to chlorides, sulfides, or fluorides that might have accumulated during fabrication, installation, or operation in conjunction with the presence of high residual and tensile stresses [4.7]. Intergranular stress-corrosion cracking of bellows has not been reported.

4.3. COATINGS AND NON-METALLIC ELEMENTS

4.3.1. Coatings

Coatings are provided to protect structures and components from degradation (e.g. corrosion), and to facilitate decontamination. Most operating nuclear power plants have similar types of coatings (e.g. primer coating of inorganic zinc and intermediate and finish coatings of polyamide epoxy; and primer, intermediate, and finish coatings of phenolic epoxy enamel). These coatings have been applied according to manufacturer’s recommendations, with available industry standards used in the coating selection process [4.8]. Degradation of coatings can result from thermal effects, condensation and immersion, radiation, physical damage, and corrosion of base material. Areas where condensation and immersion are present provide the most likely sites of accelerated corrosion of the metal substrates. Areas of containment where protective coatings are subject to continual or periodic immersion include the suppression chamber, vent system, drywell components above drywell bellows (or drywell head), and drywell sump. Accessible surfaces with high corrosion risk such as these are inspected and recoated periodically. Inaccessible surfaces or areas embedded with concrete must be inspected in critical areas to the extent possible. Frequently this involves an evaluation of accessible areas in close proximity to determine if there are any signs (e.g. corrosion staining, presence of moisture, damaged moisture barriers) that might indicate the potential for corrosion activity.

Although ageing effects of qualified metal coatings have not been quantified, data on coating life is available for plants having ages in excess of 20 years. The coating life for wet or submerged service has been found to range from a minimum of 7 years to an upper limit of 15 years or more, depending on how the coating was applied and which product was utilised [4.3]. It is expected that in a relatively dry environment coatings on interior containment surfaces will last a minimum of 40 years, if sufficient maintenance is performed. Many of the degradation modes noted above depend strongly on the quality of the initial coating application and the possibility of maintenance.

4.3.2. Non-metallic elements

Several penetrations and airlock doors use mechanically tightened non-metallic seal material. Exposure of these non-metallic materials to modest temperatures and relativly low radiation doses for long periods of time can cause embrittlement and seal failure. Leak testing after each opening and periodic replacement of the seals is the best way to address potential ageing of these components.

Non-metallic seals (e.g. moisture barriers) between concrete and steel are used at several places in most containments to prevent ingress of moisture. High temperatures can cause embrittelment of this material to such an extent that they are not able to accommodate the differential expansions that occur during normal operation or pressure testing. Also, abrasion or impact can damage the seals. Visual examination of these seals can verify their condition and the necessity for repair or replacement.

4.4. SUMMARY

Table 4.1 presents a summary of potential degradation of pressure-retaining steel components [4.8]. Components of the GE BWR Mark series of containments that have been identified as being potentially at risk to ageing degradation are summarised in Table 4.2 [4.4].

TABLE 4.1. SUMMARY OF POTENTIAL STEEL CONTAINMENT DEGRADATION*

Factor	Description
Susceptible components	Metal containment shells, concrete containment liners, penetration sleeves and bellows, heads, nozzles, structural and non-structural attachments, embedment anchors, pipes, tubes, fittings, fasteners, and bolting items that are used to join other pressure-retaining components
Types of degradation	Degradation could involve loss of net section or wall thinning, coating degradation, cracks, pits, crevices, erosion, cavitation, surface flaws, arc strikes, plastic deformation, buckling, fracture, or bulging
Potential degradation areas	Areas of water accumulation; surfaces exposed to chemical, decontamination materials, or other fluid spills; flashed, caulked, or sealed joints; dissimilar metal connections; penetrations; condensation and leakage paths; sand pockets or cushions; heat trace areas; and locations with stray electrical currents
Visual degradation indicators	Rust, discoloration; staining, blistering and peeling of coatings; spalling of concrete; abrasion and wear; buckling or separation of liners; leakage from drains; and clogged drains

*Adaptation of material presented in Ref. [4.9].

TABLE 4.2. SUMMARY OF SELECTED GE BWR MARK SERIES CONTAINMENT COMPONENTS AND POTENTIAL AGEING MECHANISMS [4.4].

Containment components	Ageing mechanisms			
	Atmosphere Corrosion	Local Corrosion	Fatigue	Mechanical Wear
Common components				
Penetration sleeves			**X**	
Dissimilar metal welds		**X**		
Personnel airlocks				**X**
Equipment hatches				**X**
CRD hatch				**X**
Mark I steel containment				
Drywell exterior surface	**X**	**X**		
Drywell head				**X**
Embedded shell region		**X**		
Drywell support skirt		**X**		
Sand pocket region		**X**		
Torus interior surface at waterline	**X**	**X**		
Vent header			**X**	
Downcomers and bracing		**X**	**X**	**X**
Vent system supports				**X**
Torus seismic restraints				**X**
Torus support columns/Saddles				**X**
ECCS suction header	**X**			
Mark II steel containment				
Drywell exterior surface	**X**	**X**		
Drywell head				**X**
Suppression chamber exterior surface	**X**	**X**		
Suppression chamber interior surface at waterline	**X**	**X**		
Region shielded by diaphragm floor		**X**		
Embedded shell region		**X**		
Sand pocket region		**X**		
Support skirt		**X**		
Downcomer pipes and bracing		**X**	**X**	**X**
Mark III steel containment				
Embedded shell region		**X**		

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Chapter 5

INSPECTION AND ASSESSMENT METHODS

Ageing degradation can affect the ability of metal components of BWR containment systems to perform satisfactorily in the event of an accident by reducing its structural capacity or leaktight integrity. Of the ageing mechanisms reviewed in Chapter 4 for relevance to the metal components of containments, corrosion, fatigue, wear and erosion, and mechanical damage were identified as the primary potential mechanisms. The ageing problems of the containment differ from most other mechanical components because service loads primarily result from periodic pressure or leakage-rate testing. Mechanical loads due to the reaction forces of the penetrating systems also are normally low, so fatigue, with the possible exception of the bellows of high energy systems, is a limited (or isolated) problem. The most severe fatigue loads result from the blowdown of the safety or relief valves that produces vibrations of the pipes and pressure pulsations in the suppression pool. The most critical areas are related to inaccessible surfaces and in the suppression pool.

One of the significant problems associated with containment ageing is the inaccessible areas of the steel structures where the detection and assessment of possible degradation (primarily corrosion) is very difficult. Non-metallic seals and gaskets represent another ageing concern relative to containment leaktightness as these items can degrade under the influence of time and temperature. However, these items are periodically evaluated under well-defined inspection programmes, and can or are routinely replaced. Some penetration constructions can be susceptible due to high stresses that can be present in stainless steel bellows resulting in stress corrosion cracking. Bellows and penetration assemblies are evaluated through visual inspections and leakage-rate testing. Bellows are also subject to fatigue analysis. Coatings provided on many metal components to minimise corrosion occurrence also can degrade under the influence of the plant service environment (e.g. chemical attack, abrasion, and temperature). Coating systems need to be maintained so that they can continue to provide protection. Coating life is directly related to maintenance. The economic life of a coating is determined by comparing maintenance costs with recoating costs. Based on the high cost of recoating (e.g. surface preparation, application, and cleanup), the personnel dose, and outage-related costs, maintenance can be extensive before its costs exceed those of recoating. Inspection of coatings is generally done through visual inspections, but properties of in-place coatings can be determined (e.g. thickness and bond to substrate).

Information presented in this section is related to inspection requirements, methods of inspection, and assessment of degradation of the metal components of BWR containment systems.

5.1. INSPECTION REQUIREMENTS

National codes and standards require that the condition and leaktight integrity of nuclear power plant containments be periodically assessed (e.g. Ref. [5.1]). Although these requirements vary somewhat from country to country, they generally involve visual inspections and leakage-rate testing. Some Member States have in-service monitoring programmes in which instrumentation feedback is used to monitor and trend performance. Also, plants with prestressed concrete containments periodically are required to evaluate the performance of the post-tensioning system (e.g. tendon forces and presence of corrosion). In-service monitoring and post-tensioning system examinations will not be addressed.

A visual inspection is generally the basic method used as a first step in a typical inspection programme. A high quality visual inspection of exposed surfaces is able to detect and define areas of ageing-related distress that result in visible effects on the surface of the structure (e.g. corrosion products, flaking, discoloration, coating blistering, peeling, and abrasion). The containment surface inspection requirements apply to the metal containment pressure-retaining components and their integral attachment, and to metallic shell and penetration liners of concrete containments. Areas requiring inspection include the base metal and pressure-retaining weld surfaces that are accessible to either direct or remote visual examination (e.g. borescopes, periscopes and still or video cameras). To examine underwater portions of the suppression chamber, either the chamber must be drained or underwater examination techniques must be used. Drainage of the chamber relieves the hydrostatic pressure on the coating surfaces and may cause additional blistering or bursting of existing coating blisters. Underwater techniques have been developed that include desludging, ultrasonic mapping of critical areas, coating adhesion tests, dry-film thickness determinations, and repair of localised areas. Since crevice corrosion is possible at hatch locations; and under bolts, nuts, and gaskets; proper maintenance and use of grease and lubricants, and routine visual inspections are used to address and inspect for corrosion. Areas that do not generally require in-service inspection include inaccessible portions of the containment vessel and parts that are embedded in concrete. In the U.S. the visual inspections are performed prior to conduct of the integrated leakage-rate test.

Integrated leakage-rate tests are conducted by pressurising the containment with air to a pre-established level (e.g. peak pressure associated with a design-basis accident) and monitoring the leakage as a function of time. Both full- and partial-pressure testing have been utilised. Pressure, temperature and vapour pressure sensors are used during the test to sample the containment atmosphere. Changes in the contained air mass define the leakage rate. The method can be augmented by spraying a thin film of soap solution on cracked areas and visually monitoring the formation of bubbles during pressurisation. The presence and rate of bubble formation indicate the magnitude of the defect. Also the detection of leakage can be enhanced by incorporation of helium or halogen gas to the pressurising atmosphere and using a gas detector. The primary limitation of integrated leakage-rate testing is that it is performed while the plant is shut down. Also there have been concerns expressed that each pressurisation imposes a high magnitude, low frequency cyclic load on the containment that may affect its performance. Tests also are conducted to detect local leaks and to measure leakage rates across penetrations with flexible metal seals, bellows expansion joints, airlock door seals, accesses and penetrations with resilient seals or gaskets. Isolation valve leakage is also measured. Certain plants have leak-chase channels over the liner plate welds that can be used for leak testing.

5.2. INSPECTION METHODS

The primary goal of inspection is to identify the location, type and magnitude of structural imperfections or flaws. Non-destructive examinations, destructive tests, and in-service monitoring methods evaluate the presence and significance of indications of degradation of the containment pressure boundary detected by visual inspections and leakage-rate testing. These methods can also be applied to evaluate suspect areas. Evidence of degradation in the form of loss of section due to corrosion and the presence of cracking is of primary interest. Inaccessible regions of the containment pressure boundary require special consideration.

5.2.1. Non-destructive examinations

Non-destructive examination methods for metallic materials principally involve surface and volumetric inspections to detect the presence of degradation (i.e. loss of section due to corrosion or presence of cracking). The surface examination techniques primarily involve the visual, liquid penetrant, eddy current, and magnetic particle methods. Volumetric methods include ultrasonic and radiographic. Electrochemical corrosion monitoring techniques are also addressed.

Visual inspection is one of the most common and least expensive methods for evaluating the condition of a weld or component (e.g. presence of surface flaws, discontinuities, or corrosion). It is generally the first inspection that is performed as part of an evaluation process. It is beneficial for performing gross defect detection and in identifying areas for more detailed examination. It can identify where a failure is most likely to occur and when failure has commenced (e.g. rust staining or coating cracks). Once a suspect area is identified all surface debris and protective coatings are removed so that the area can be inspected in more detail. Visual examinations can be performed either with the unaided eye or optical magnifiers. Inspection mirrors, video cameras, and borescopes can be used for inspection of areas with limited accessibility. Three classifications of visual examinations are specified in the ASME Boiler and Pressure Vessel Code: (1) VT-1 (detect discontinuities and imperfections on the surfaces of components such as cracks and corrosion), (2) VT-2 (detect evidence of leakage from pressure-retaining components), and (3) VT-3 (determine general mechanical and structural condition of components and their supports). The effectiveness of a visual inspection is dependent on the experience and competence of the person performing the inspections. Also, without material or component removal, visual inspections are limited to accessible areas. Table 4.1, presented previously, provides an indication of some of the visual indications of steel containment-related degradation.

Liquid penetrant testing can be used to detect, define and verify surface flaws in solid or essentially nonporous components (e.g. cracks, porosity, laminations or other types of discontinuities that have a capillary opening to the surface). Indications of a wide spectrum of flaw sizes can be found with little capital expenditure regardless of the configuration of the test article or the flaw orientation. The procedure consists of cleaning the surface to be examined followed by application of a liquid penetrant. Surface defects or cracks absorb the penetrant through capillary action. After a dwell period, excess penetrant is removed from the surface and a developer is applied that acts as a blotter to draw penetrant from the defects to reveal their presence. Coloured or fluorescent penetrants may be utilised, with white light or black light, respectively, used for viewing. Effectiveness of the method is dependent on the properties of the penetrant and the developer. Limitations of the technique are that operator skill requirements are fairly high, only surface flaw defects can be detected, area inspected must be clean as scale or paint film may hide flaws, results are affected by surface roughness and porosity, and no permanent record of inspection is provided.

Eddy current inspection methods are based on electromagnetic induction and can be applied to electrically-conductive materials for detection of cracks, porosity, and inclusions, and to measure the thickness of non-conductive coatings on a conductive metal. In the flaw detection mode eddy current can detect surface connected or near surface anomalies. It is based on the principle that alternating current flow in a coil proximate to an electrical conductor will induce current flow in the conductor. The current flow (i.e. eddy current) creates a magnetic field that opposes the primary field created by the alternating current flow in the coil. The presence of a surface or near surface discontinuity in the conductor will alter the magnetic field (i.e. magnitude and phase) and can be sensed as a change in the flow of current in a secondary coil in the probe or change of inductance of the probe. The output signal from the detection circuit is fed to an output device, typically a meter, oscilloscope, or chart recorder. Flaw size is indicated by extent of response change as the probe is scanned along the test object. Eddy current techniques do not require direct contact with the test piece, and paint or coatings do not have to be removed prior to its application. For surface discontinuities of a given size, the sensitivity of eddy current decreases with distance below the surface. Best results are obtained when the magnetic field is in a direction that will intercept the principal plane of the discontinuity. Also, the technique requires calibration, is sensitive to geometry of the test piece, results may be affected by material variations, no permanent record is provided, and demagnification may be necessary following inspection.

Magnetic particle testing is used to detect surface and shallow subsurface discontinuities in ferromagnetic materials. A magnetic field is induced into the ferromagnetic material and the surface is dusted with iron particles that may be dry, suspended in a liquid, coloured, or fluorescent. The magnetic lines of force (flux) will be disrupted locally by the presence of the flaw with its presence indicated by the iron particles that are attracted by leakage of the magnetic field at the discontinuity. The resulting magnetically-held collection of particles forms a pattern that indicates the size, shape, and location of the flaw. Effectiveness of the method quickly diminishes depending on flaw depth and type, and scratches and surface irregularities can give misleading results. Special equipment, procedures, and process controls are required to induce the required magnetic fields (e.g. use of proper voltage, amperage, and mode of induction). Also, linear discontinuities that are oriented parallel to the direction of the magnetic flux will not be detected.

Ultrasonic testing uses sound waves of short wavelength and high frequency to detect surface and subsurface flaws, and measure material thickness. The most commonly used technique is pulse echo in which sound is introduced into the test object and travels through the material examined with some attendant loss of energy. Reflections (echoes) are returned to the receiver from internal imperfections or the component's surfaces. The returning pulse is displayed on a screen that gives the amplitude of the pulse and the time taken to return to the transducer. Inclusions or other imperfections are detected by partial reflection or scattering of the ultrasonic waves, time of transit of the wave through the test object, and features of the spectral response for either a transmitted or reflected signal. Operator interpretation is made by pattern recognition, signal magnitude, timing, and probe positioning. Flaw size, distance, and reflectivity can be interpreted. The technique has good penetration capability (i.e. up to 6 m for axial inspections), high sensitivity to permit detection of very small flaws, good accuracy relative to other non-destructive examination methods, only one surface has to be accessible, and rapid results are provided. For thickness measurements digital meters are commonly used. In the pulse-echo mode an ultrasonic transducer transmits waves toward the metal surfaces, signals are reflected from the front and back surfaces, and the difference in arrival times of the two signals is used to indicate the thickness. Metal loss is then calculated by taking the difference between the as-built thickness and the thickness measured. Two types of systems are available commercially – ultrasonic thickness gage (digital display) and digital gage (A-scan, echo signals are displayed on an oscilloscope). Ultrasonic testing is commonly used in nuclear plants to monitor wall thinning of the containment vessel caused by corrosion. Rough surface conditions such as could be present on the surfaces of the metal components of BWR containment systems present problems relative to signal scattering. Focused transducers provide the best results where rough surfaces are present. Because of its complexity, ultrasonic testing requires considerable technician training and skill. Also, good coupling between the transducer and component inspected is important, defects just below the surface may not be detected, and reference standards are required.

Radiographic techniques involve the use of penetrating gamma or X radiation and are based on differential absorption of the radiation. X radiographic inspection is applied to the detection of surface connected and internal anomalies as well as the internal configuration of a test object. The source is placed close to the material to be inspected and the radiation passes through the material and is captured on film placed on the opposite side of the test article from the source. A two-dimensional projection of the area being inspected is displayed on the film (permanent record). The thickness, density, and absorption characteristics of the material affect the intensity of radiation passing through an object. Possible imperfections are indicated on the film as density changes (i.e. series of grey shades between black and white). The choice of type of source is dependent on the thickness of material to be tested. Gamma rays have the advantage of portability. Gamma radiometry systems consist of a source that emits gamma rays through the specimen and a radiation detector and counter. Direct transmission or backscattering modes can be used to make measurements. The count or count rate is used to measure the specimen dimensions or physical characteristics (e.g. density and composition). Primary limitations of radiography are that radiation protection has to be observed while applying the method, personnel must be licensed or certified, results are not immediately available, the structure must be accessible from both sides, and detection of crack-like anomalies is highly dependent on the exposure geometry and orientation of the crack with respect to incident irradiation.

Acoustic emission inspection is based on monitoring and interpretation of stress waves generated by a structure under load. Acoustic emissions are small amplitude stress waves resulting from release of kinetic energy as a material is strained beyond its elastic limit (e.g. crack growth and plastic deformation). Material stress can come from mechanical or thermal loading, as well as from a variety of other means. The stress waves propagate throughout the specimen and may be detected as small displacements by piezo-electric transducers positioned on the surface of the material. A typical acoustic emission system consists of a number of sensors, preamplifiers, signal filters, amplifier, and a recording system. Signal measurement parameters most commonly used to interpret results include ringdown counts (threshold-crossing pulses), energy counts (area under rectified signal envelope), duration (elapsed time for ringdown counts), amplitude (highest peak voltage), and rise time (time from first threshold crossing to signal peak). Primary applications of acoustic emission inspection include continuous monitoring or proof testing of critical structures, monitoring of production processes, and experimental research related to material behaviour. Advantages of acoustic emission are that it is extremely sensitive, the entire structure can be monitored, it is relatively unobtrusive, onset of failure can be identified, and triangulation can be used to identify source location. Certain aspects of the corrosion process are detectable by acoustic emission (e.g. stress-corrosion cracking, hydrogen cracking, and gas evolution) [5.2]. Disadvantages are that it requires considerable technical experience to conduct the test and interpret results, background noise can interfere with signals, and a material may not emit until the stress level exceeds a prior applied level (i.e. Kaiser effect).

Thermographic inspection methods are applied to measure a variety of material characteristics and conditions. In the flaw detection mode they are used for detection of interfaces and/or variation of properties of interfaces within layered systems. The test object must be thermally conductive and reasonably uniform in colour and texture. The procedure involves inputting a pulse of thermal energy that is diffused within the test object according to thermal conductivity, thermal mass, inherent temperature differentials, and time of observation. The thermal state of the test object is monitored by a thermographic scanner camera that has infrared energy spectrum detection capability. Interpretation of results is done through visual monitoring of the relative surface temperature as a function of time and relating the time-dependent temperature differences to the internal condition of the test object. Results are recorded as a function of time and the process is relatively rapid. Specialised equipment is required and since the method is a volume inspection process, resolution is lost near the edges and at locations of nonuniform geometry change. Thermal inspection becomes less effective in the detection of subsurface flaws as the thickness of the object increases. Pulsed infrared techniques have been developed that can perform inspections through the thickness of test objects. The process basically entails providing heat through a thermal pulse or step heating, and dynamically collecting infrared images of the material surface. To be successful the heat applied at the top surface must penetrate to the bottom surface with a temperature differential of several degrees for good infrared contrast.

Electrochemical corrosion monitoring techniques are available to make measurements directly related to corrosion rate rather than indirectly in terms of the flaws produced by corrosion. Potential surveys, linear polarisation, and AC impedance are techniques that have been utilised. Electrochemical potential measurements using a standard half-cell (e.g. copper-copper sulphate) can be used to locate anodic portions of a structure (i.e. potential gradients indicate possibility of corrosion). The linear polarisation resistance method impresses DC current from a counter electrode onto the working electrode (e.g. steel structure). Current is passed through the counter electrode to change the measured potential difference by a known amount with the working electrode being polarised. An electronic meter measures the potential difference between the reference electrode and the working electrode. Measurements as a function of DC voltage applied across the cell provide an indirect measure of the corrosion current. The AC impedance-polarising technique utilises an alternating applied voltage with the data analysed as a function of frequency. The AC technique provides polarisation resistance as well as information on polarisation mechanisms at the anode and cathode which is important for interpretation of the AC impedance data. The technique requires rather sophisticated equipment (e.g. AC frequency generator and analyser system) and the Tafel slopes must be known to convert AC impedance data into corrosion rate information [5.3]. Each of these methods requires contact with the part of the structure monitored, and where corrosion rates are provided the rates are only since equipment installation and monitoring.

5.2.2. Destructive tests

Tests that alter the shape, form, size or structure of the material being tested are considered destructive. These tests may be performed to determine mechanical, physical, chemical, thermal, or other properties of the material, and to examine the material for microstructural imperfections, voids, or inclusions. Destructive tests commonly used to determine mechanical properties of metallic materials include tension, compression, ductility, shear, torsion, bend, creep, stress-relaxation, hardness, fatigue, or fracture testing. Metallography is another form of destructive testing that relates to determination of the constitution and structure, and their relation to the properties of metals or alloys. Testing generally requires removal of a test sample and examination in a laboratory set up. Destructive testing would not normally be conducted on metal components of BWR containment systems because the materials are not generally subjected to environments that would alter their constitution or structure.

5.2.3. Potential techniques for inaccessible areas

Inspection of inaccessible portions of metal components of BWR containment systems (e.g. fully embedded or inaccessible containment shell portions, the sand pocket region in Mark I and II drywells, and portions of the shell obscured by obstacles such as platforms or diaphragm floors) requires special considerations. Metal containment structures may be subjected to corrosion resulting from groundwater permeation through the concrete: a breakdown of the sealant at the concrete-containment shell interface that permits entry of corrosive fluids from spills, leakage, or condensation; or in areas adjacent to floors where the gap contains a filler material that can retain fluids resulting from condensation, and fluids used for cleaning or decontamination. Corrosion incidences that may occur at locations such as these can challenge the containment structural integrity and, if through-wall, can provide a leak path to the outside environment. Although no suitable technique for inspection of inaccessible portions of containment pressure boundaries has been demonstrated to date, several techniques have been proposed (i.e. ultrasonic inspection, electromagnetic acoustic transducers, half-cell potential measurements, high frequency acoustic imaging, magnetostrictive sensor technology, and guided plate waves), and are briefly discussed below.

Ultrasonic testing is commonly used to monitor wall thinning and can be used to detect and monitor corrosion if at least one side of the structure is accessible. In the pulse-echo mode an ultrasonic transducer transmits waves toward the metal surfaces, signals are reflected from the front and back surfaces, and the difference in arrival times of the two signals is used to indicate the thickness. Metal loss is then calculated by taking the difference between the as-built thickness and the thickness measured. In Germany, an extensive evaluation was conducted to evaluate the feasibility using ultrasonic methods to investigate inaccessible portions of the containment pressure boundary [5.4]. Non-destructive tests were performed on a containment and on calibration blocks containing corrosion damage. Results of this study indicate that it was possible to detect well developed corrosion pits with 45° angle beam 2 MHz search units within a distance of up to 130 mm from the interface between the concrete and steel pressure boundary. General corrosion was found to be difficult to detect.

Electromagnetic acoustic transducers (EMATs) consist of a transmitter and receiver, both of which contain a permanent magnet or electromagnet and a coil. The transmitter coil is excited by high radio-frequency current to induce an eddy current into the surface of the metal examined. The eddy current interacts with the magnetic field, generated by the transmitter coil to produce a Lorentz force in the metal which produces guided plate waves in the metal [5.5]. EMATs have advantage for detection of corrosion because a couplant is not needed, the ultrasound is generated directly in the metal rather than the transducer and is nondispersive, and the high-energy waves can travel relatively long distances parallel to the plate surface, the wave velocity is independent of plate thickness, and the ultrasound can be generated through a surface coating up to about 1.5-mm-thick. EMATs were used in the laboratory to detect simulated corrosion-like defects in a 2.1-m-wide by 4.9-m-long by 25.4-mm-thick plate [5.6]. Pulse-echo and through transmission-modes were evaluated. In the pulse-echo mode a flaw at least halfway through the plate thickness could be detected at distances to 4.6 m. In the through transmission-mode deep corrosion damage (i.e. >75% of the plate thickness) could be detected at a distance to 15.24 m, but its location could not be determined.

Half-cell potential measurements have been used with great success in the detection of corrosion of steel reinforcement in concrete structures. Potential measurements at a number of locations on the concrete surface using a reference electrode (e.g. copper-copper sulphate) connected to the steel reinforcement can be used to indicate the likelihood of corrosion occurrence (i.e. >90% probability of no corrosion activity, corrosion activity is uncertain, or >90% probability that corrosion is occurring). The surface of the concrete being investigated is usually divided into a grid system to define measurement locations. Results generally are plotted in the form of an equipotential diagram so that areas possibly exhibiting corrosion can be readily identified. Modified types of instrumentation, consisting of a number of half-cells mounted on a roller bar and automated data acquisition systems, have been developed to accelerate the process. Primary limitations of this method are that neither the magnitude nor the rate of corrosion are provided, concrete surface coatings or coatings on the steel present problems, measurements are affected by temperature and moisture, and concrete material constituents affect results (e.g. type of cement and chloride ingress). In order to obtain potential measurements on inaccessible portions of the metallic containment components, the electrodes would have to be placed near the component surfaces. For portions of these components that are embedded in concrete, this may entail drilling access holes so that the steel reinforcement in the concrete would not interfere with results provided. Although application of this technique to embedded portions of the containment pressure boundary appears feasible, no attempts at its application have been identified.

Preliminary analytical and experimental simulations were conducted to investigate the feasibility of *high frequency acoustic imaging* techniques for the detection and localisation of thickness reductions in the metallic pressure boundaries of nuclear power plant containments [5.7]. The analytical study used an elastic layered media code to perform a series of numerical simulations to determine the fundamental two-dimensional propagation physics. The experimental study utilised a commercial ultrasonic testing system to carry out several full-scale tests. The experimental studies were designed to also effectively restrict case scenarios to two dimensions. Measurements of 0.5 MHz shear wave levels propagated in 25-mm-thick steel plates embedded in concrete showed 1.6 dB of signal loss for each centimetre of two-way travel in untreated plates (compared with numerical predictions of 3-4 dB) and negligible loss in plates with a concrete debonding treatment applied. The return from a 4-mm rectangular slot cut across the width of a 25-mm-thick steel plate showed returns down 24 dB relative to the input and 4–6 dB higher than those obtained from both "V" shaped and rounded slots. The system displayed a dynamic range of 120 dB and measurement repeatability of 1–2 dB. Based on these results, measurable signals should be reflected from a 4-mm-deep round-faced degradation embedded in 30 cm of concrete. Returns would be expected to be down 75 dB relative to the input. The analytical simulation suggests that for the case of steel-lined concrete containments, the thin steel liner and additional concrete backing contribute to give unacceptable loss of signal to the concrete. Approximately 100 dB of signal loss is incurred for small degradations near the concrete interface. Due to this loss, it appears unlikely that acoustic imaging technology can be applied to this scenario. The study also concluded that currently available sensors cannot be used in array configurations to interrogate a large area due to their intrinsic narrow beam pattern, which does not allow steering. This limits these sensors to spot detection and mapping scenarios, where degradation is already suspected. For wide-area surveys, the use of scannable sensors appears to be applicable, but they will require development. The sensors would be manufactured by bonding many signal wires to a solid piezo-electric block on a substrate and then cutting it into individual sensors, leaving a line array of sensors in the substrate. The competing signals from unfocused source transducers and waveguide signal distortion remain as two significant barriers for localising and characterising degradations.

Magnetostrictive sensors are devices that launch guided waves and detect elastic waves in ferromagnetic materials electromagnetically to determine the location and severity of a defect based on timing and signal amplitude. Its primary application has been to piping systems [5.8], but preliminary numerical modelling results indicate that the technique is applicable to plate-type components. The technique is noncontact, couplant free, and requires minimum surface preparation. In addition, the technique has a sensing or inspection range from a single sensor location that can exceed several hundred feet on bare metals, the sensor can detect defects on the inside and outside diameters of pipe surfaces, and it can inspect structures whose surfaces are not directly accessible due to the presence of paint or insulation. Preliminary studies have been completed that demonstrate the feasibility of detecting and locating defects in plate-type components [5.9].

The *guided wave technique* (multi-mode guided plate waves) is more sensitive than techniques which utilise shear waves (e.g. electromagnetic acoustic transducers), provides a global inspection technique for characterising corrosion damage, follows the contour of the structure and can travel long distances (e.g. 100m depending on frequency and mode characteristics), and can interegate different regions or cross sections (i.e. depths) of the component inspected [5.10., 5.11]. The guided plate waves can be excited at one point on the structure, propagate over considerable distances, and be received at a remote point on the structure. This technique has been used with success to detect defects in piping materials, but no applications to plate-type materials have been identified.

5.3. ASSESSMENT METHODOLOGY

In-service condition assessments are an essential component for reliable continued service evaluations and informed ageing-management decisions. From an ageing management viewpoint, metal components of BWR containment systems that exhibit satisfactory long term performance and do no exhibit in-service degradation generally can be considered acceptable for continued service. However, components that are found during inspections and examinations to be deteriorated or damaged must be evaluated to determine whether continued service is appropriate, or whether a repair, replacement, or retrofit is needed. Damage is considered to be significant when it adversely impacts the structural capacity, leaktight integrity, or remaining service life of a component. Knowledge derived from condition assessments can serve as a baseline for evaluating the safety significance of any deterioration that may be present and defining in-service inspection programmes and maintenance strategies.

In-service monitoring involves periodic examination of the containment pressure boundary while it remains in service. This type of examination is different from preservice inspection that is conducted before a containment is placed into service and the in-service inspections that are performed on a regular basis over the operating life of the plant. In-service monitoring generally involves repeated examination of a flawed component or suspect area using one or more non-destructive examination techniques (e.g. periodic measurements of the torus shell in locations where corrosion has been observed). Results from periodic in-service monitoring can provide valuable information for assessing the condition of a degraded component, estimating its remaining useful life, and making informed ageing-management decisions. As an ageing-management tool, in-service monitoring results can be used to guide selection of the appropriate examination techniques, specify testing methods, and establish the frequency of subsequent inspections. Examples where in-service monitoring has been used include monitoring of the cumulative loss of metal due to corrosion or on-line electrochemical measurements to establish an average degradation rate.

Continued service evaluations are performed by qualified engineers and authorised personnel who determine the adequacy of components for their intended use [5.14]. The decision making process begins with an understanding of the in-service condition of each containment component. Condition assessments that provide essential information for continued service evaluations involve classifying the types of detected degradation, determining the root cause of the problem, and quantifying the extent of degradation.

Engineering evaluations are performed on a case-by-case basis by qualified engineers and authorised personnel who determine the adequacy of damaged or degraded components for their intended use. Acceptance criteria are generally established so that components with flaws, discontinuities, or areas of degradation that adversely affect the structural capacity, leaktight integrity, or remaining service life of the containment are not considered acceptable for continued service. Components that are found by engineering evaluation to have no effect on the structural capacity, leaktight integrity, or service life may be returned to service without removing the defect or repairing or replacing the defective component. For example, damaged components are considered acceptable if either the thickness of the base material is reduced by no more than 10 percent of the nominal thickness, or it can be demonstrated by analysis that the reduced thickness satisfies the requirements of the design specification.

Requirements for corrective actions to be taken when evidence of structural deterioration is discovered have been identified [5.12, 5.13]. Knowledge gained from condition assessments can serve as a baseline for evaluating the safety significance of any damage that may be present and defining in-service inspection programmes and maintenance strategies. Condition assessment results can also be used to estimate future performance and remaining service life. A diagram that illustrates the basic continued service evaluation process presented in Subsection IWE of the ASME Code is shown in Fig. 5.1.

One way to evaluate the significance of any degradation of the metal components of BWR containment systems on structural or leaktight integrity is by comparing its preservice condition to its present condition. Condition assessment accuracy depends on the availability of quantifiable evidence such as dimensions of a corroded surface area, section thickness, or changes in material properties. Section 5.2 provided background information for these determinations. Results from these investigations provide a measure of the extent of degradation at the time of examination. Changes with time can be determined through periodic measurements conducted under as similar conditions as possible (e.g. use of same measurement grid, equipment, and inspector). In-service monitoring provides another method of measurement for determining time-dependent changes in component geometry or material properties, and to detect undesirable changes in operating conditions that can affect the service life. Information required to characterise and quantify the condition of a degraded component, however, must be established on a case-by-case basis taking into account unique containment design features and plant operating conditions. General guidance on conduct of condition assessments is provided in documents such as Refs. [5.14, 5.15]. Information has been developed that specifically addresses assessment of age-related degradation mechanisms and programmes to address these mechanisms in BWR containments [5.16]. Also documentation has been prepared that addresses a BWR model containment inspection programme [5.17]. Specific guidance to assist in establishing an ageing management programme for metal components of BWR containments is provided in Chapter 8.

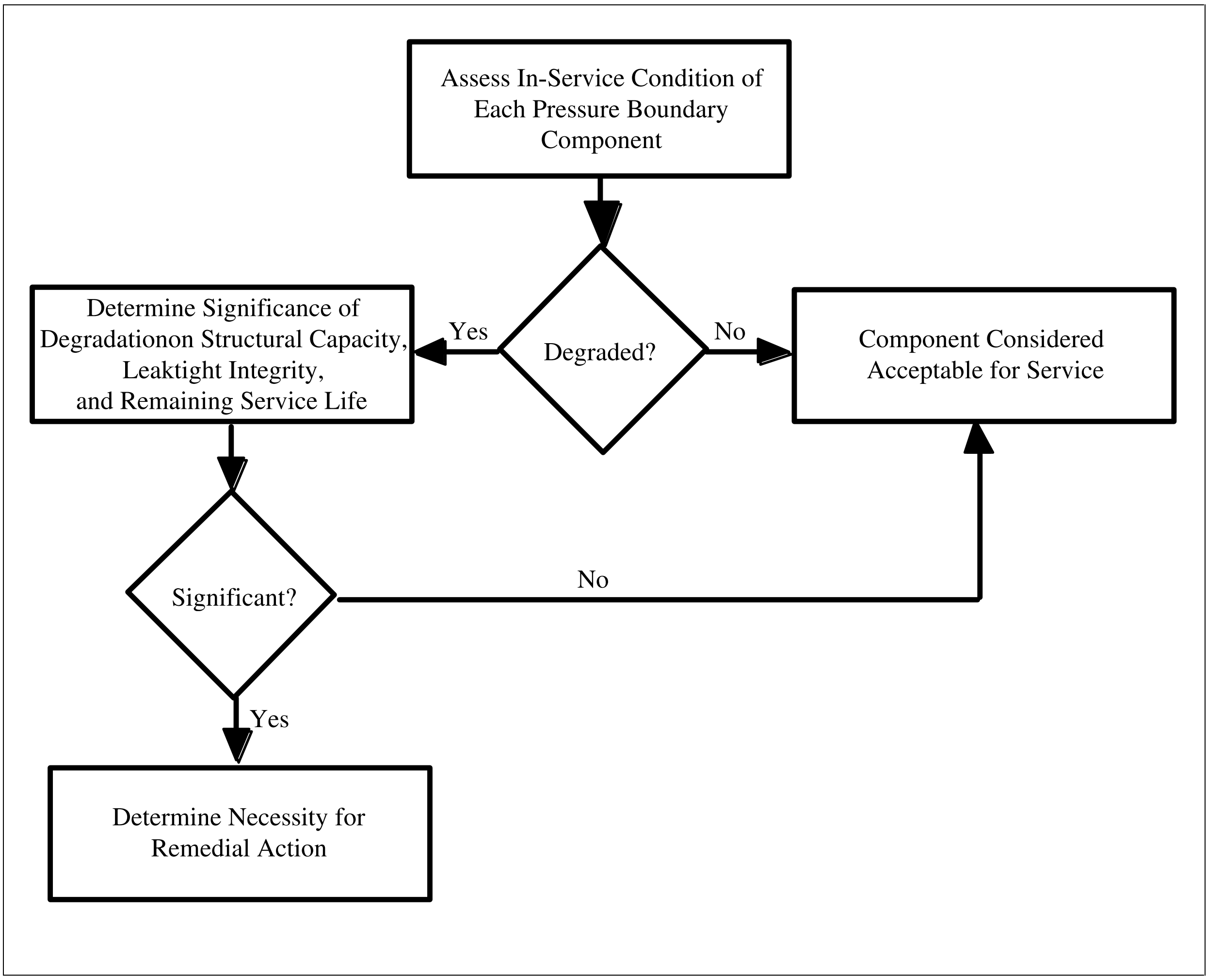

FIG. 5.1. Continued service evaluation process for containment pressure boundary components.

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Chapter 6

AGEING MITIGATION METHODS

Leaktightness of the BWR containment vessels is ensured by a continuous pressure boundary consisting of non-metallic seals and gaskets and metallic components that are either welded or bolted together. Non-metallic components are used to prevent leakage from pumps, pipes, valves, personnel airlocks, equipment hatches, manways, and mechanical and electrical penetration assemblies. The remaining pressure boundary consists primarily of steel components such as metal containment shells, concrete containment liners, heads, nozzles, structural and non-structural attachments, embedment anchors, pipes, tubes, fittings, fasteners, and bolting items that are used to join other pressure-retaining components. Each containment type includes numerous access and process penetrations that complete the pressure boundary. Although some of these components can be replaced, if necessary, most are intended to remain in service for the entire operating life of the plant. Prevention of corrosion of metal components of BWR containments is mainly achieved with a dry atmosphere, effective sealing methods, or use of protective coatings. Some surfaces are left uncoated, such as penetration sleeves, airlocks, vent systems, leak-chase channels, and areas embedded in concrete.

Since nuclear power plants have been in operation, the overall performance of the metal containments generally has been good. However, instances of wall thinning, coating degradation, moisture barrier deterioration, and component damage have been reported (see Chapter 7). Operating experience suggests that problems with the containment pressure boundary components can generally be related to general or pitting corrosion of steel components, cracking or loss of function of electrical penetration assemblies, and cracking and corrosion of expansion bellows. Past experience also suggests that degradation of metal containment shells can occur on the inside as well as the outside of the containment shell. Potential areas at greatest risk have been identified in Chapters 3 and 4.

Whenever containment degradation is detected, corrective actions are usually taken to identify and eliminate the source of the problem and thereby halt or slow down the rate of degradation (managing ageing mechanisms). One approach is to modify or minimise the effects of the environment that is the source of the problem. Examples of actions that could be implemented include reducing the oxygen level (requires a reduction to $<1\%$ by volume which may not be economically practical), and increasing insulation around hot piping penetrations. However, when significant wall thinning, cracking, surface defects, or leakage is detected and containment integrity is jeopardised, the degradation must be corrected by a repair or component replacement, or shown to be acceptable before the plant is returned to service (managing ageing effects). Under certain conditions an inspection programme may be implemented to periodically examine suspect areas or to monitor the long-term performance of a degraded component.

Contained in the balance of this section is an overview of potential methods for protection against ageing degradation (managing ageing mechanisms) and for the correction of unacceptable degradation (managing ageing effects). The primary ageing mechanism of concern is corrosion. Protection against corrosion primarily includes the application and maintenance of coatings to exposed steel that is at risk, and use of cathodic protection systems (See Sec. 6.1.2). Correction of ageing degradation is addressed in Sec. 6.2. Repair methods for the steel pressure boundary generally involve a welding procedure. For completeness, options for restoring bellows that are degraded due to mechanisms such as mechanical damage or stress-corrosion cracking are also addressed. Degradation of non-metallic seals and gaskets is not addressed as these items in most instances would be under a defined inspection programme and generally can be replaced.

6.1. CORROSION PROTECTION METHODS

Primary methods of protection against corrosion and its effects are the use of organic coating systems and cathodic protection. Removal of oxygen from the atmosphere can also prevent oxidation/corrosion of metal surfaces. However, the oxygen concentration needs to be reduced below 1% by volume to be effective, which is an expensive solution that has not been implemented.

6.1.1. Organic coatings

Organic coating systems are used in nuclear power plants to protect ferrous metal surfaces from corrosion and to facilitate decontamination of metal and concrete surfaces. Corrosion protection is needed for all exposed carbon steel components including surfaces of the containment shells, concrete containment liners, structural steel elements, uninsulated mechanical equipment, piping system components and related hardware, and mechanical machinery. Protective coatings also play an important role in achieving and maintaining radiological control by providing surfaces that can be more readily decontaminated. Although coatings are applied based primarily on economic considerations (e.g., potential consequences of corrosion occurrence), other factors that can influence their use include material compatibility, heat transfer characteristics, and consequences of failure during an accident. Proper application and maintenance of coating systems is important because failures can potentially result in blockage of containment sumps, blockage of flow passages to emergency systems, or chemical- or mechanical-induced damage to the reactor coolant system in the form of corrosion or abrasion.

Most of the BWR's utilise similar coating systems that have been applied according to manufacturer's recommendations (e.g., surface preparation and minimum and maximum dry-film thicknesses) with available industry standards utilised as guidelines for material selection. In large measure, qualification criteria for coatings applied to interior surfaces of early containments were not developed until after the fact. The criteria developed primarily focused on environmental qualification related to normal and accident temperatures and humidities, and were performed through artificial ageing tests and experiments representing simulated design-basis accident conditions. Results developed according to these criteria in conjunction with the satisfactory performance of the existing in-place coatings were such that the newer containments tend to use the same or similar materials.

Organic coatings can be damaged by exposure to service conditions such as elevated temperature, condensation, immersion and radiation, and also by physical damage and corrosion of base metal. Ways in which coatings can fail include checking (slight breaks in film that do not penetrate last applied coating), cracking, blistering, flaking, peeling, delamination, and scaling. Maintenance of the coating is important to reduce corrosion damage to the containment shell. Key parameters associated with coating maintenance are inspection to identify degraded areas, condition assessment, removal and surface preparation, selection of a compatible coating material, and coating application. Guidelines that address these parameters are available [6.1].

Maintenance of protective coatings in immersion areas of nuclear power plants requires special provisions. Coating degradation and pitting has been observed in suppression chambers. Materials located below the waterline are exposed to aggressive environmental conditions that may include ionising radiation and radiological contamination, high-pressure steam releases, decontamination operations, demineralised water immersion, and abrasive action from sludge [6.2]. Successful coating system performance depends on periodic inspections aimed at early detection of defects and failures, identification of repair alternatives, and timely execution of repairs. Options for coating inspection in immersion areas involve either draining to allow access for inspection personnel or underwater inspection by qualified divers. Problems associated with draining immersion areas include treatment of contaminated water, decontamination of internal surfaces, installation of rigging or scaffolding to provide access, and draining relieves the hydrostatic pressure that may lead to additional blistering or bursting of existing defects. The preferred alternative to draining is underwater inspection and repair. Techniques for desludging, coating inspection, and underwater coating repair have been developed [6.3]. When damaged coatings without significant metal loss are identified, the surface should be prepared and a material such as a 100 percent solids underwater-cured epoxy coating applied. Surface preparation would include cleaning using a power grinding tools, with the edges of the existing coating feathered, and adjacent coating slightly roughened to provide good adhesion for replacement coating [6.4]. The epoxy coating should be applied to the bare metal and overlapped with the existing coating. This process is suitable for spot repairs but is not recommended for major recoating [6.3].

6.1.2. Cathodic protection

Corrosion is an electrochemical process that causes metals to deteriorate due to a reaction with its environment. Electrochemical reactions occur whenever an anode and a cathode are electrically connected while immersed in an electrolyte. Cathodic protection is used to reduce or eliminate corrosion by making all anodic areas on the metal to be protected cathodic. Two types of cathodic protection have been widely used: sacrificial (galvanic) and impressed-current anode systems.

Sacrificial anode systems rely on a metal that is naturally more anodic to the structure being protected in the environment of interest (i.e., no external power source). Three metals – magnesium, aluminium, and zinc – are commonly used as sacrificial anodes. Magnesium is used routinely in buried soil applications, aluminium is most often used in offshore structures where lighter weight is important, and zinc is used in both fresh and marine water environments. The voltage difference between sacrificial anodes and cathodes is limited to about 1 volt or less depending on the anode material and specific environment. This reduces the current distribution pattern along the cathodes and makes the system best suited for application to smaller components (or areas of components).

Impressed current cathodic protection systems rely on an external electrical power source to provide the required direct current (DC). Rectifiers attached to alternating current systems are frequently used to provide the DC power. Since the impressed current systems utilise an array of electrical components, they are more complex than the sacrificial anode systems, potentially making them less reliable. To function properly, the positive terminal of the power source must be connected to the anodes and the negative terminal must be connected to the structure (cathode). Connection reversal can accelerate corrosion. The availability of larger voltage differences with the impressed current systems permits its use in a low-conductivity environment and permits remote anode placement. Overprotection, however, with a large voltage difference or too much external current can damage the component being protected, cause blistering or disbonding of the existing surface coating, and result in hydrogen embrittlement of high-strength steel. To ensure a long service life, anodes for impressed current systems are usually made from non-consumable materials that are naturally cathodic to steel such as high-silicon cast iron, titanium, platinum, or graphite. Impressed current systems are generally used for larger components (or areas of components).

6.2. CORRECTION OF AGEING DEGRADATION

6.2.1. Repair methods for steel containments

Codes and standards generally stipulate that any evidence of structural deterioration that could affect the structural capacity or leaktight integrity of metal containments must be corrected before the containment can be returned to service (e.g., Ref. [6.5]). More specifically, containment pressure boundary components with flaws, discontinuities, or areas of degradation that do not meet acceptance standards (e.g., Ref. [6.6]) may not be returned to service unless: (1) the unacceptable flaws, discontinuities, or areas of degradation are removed to the extent necessary to meet the acceptance standards, (2) a repair involving welding is performed such that existing design requirements are met, or (3) the component or portion of the component containing the unacceptable flaws or areas of degradation is replaced. Also, generally there is the option of performing an engineering evaluation to demonstrate that a BWR containment system with a specific defect present will maintain its required structural capacity and leaktightness until at least the next inspection (See Sec. 5.3).

6.2.1.1. Grinding

Steel containment components that contain defects may be returned to service provided the unacceptable flaw or discontinuity is removed or reduced to an acceptable size and the resultant section thickness created by the removal process is equal to or greater than the minimum design thickness. If the affected component has been reduced below the minimum design thickness, before being returned to service the component must be either repaired, replaced or demonstrated by an engineering evaluation to be fit for service.

Mechanical methods such as grinding are used for defect removal. Grinding is a process whereby metal fragments are removed from the surface of the component as it comes into contact with an abrasive substance such as a rotating aluminium oxide grinding wheel. In most applications the grinding wheel is manipulated manually. In areas where repair welding is not required, the affected area is faired into the surrounding area so that all sharp notches and severe discontinuities are eliminated. When repair welding is required, the cavity produced by the defect removal is finished smooth with bevelled sides and rounded edges so that suitable access for welding is provided. Methods such as magnetic particle or liquid penetrant are utilised to ensure that the indications have been reduced to an acceptable size by the defect removal procedure.

6.2.1.2. Welding

Steel containment components that have been reduced below the minimum design thickness either by degradation or defect removal may be repaired by welding and returned to service. Requirements for repair welding are generally provided in the original construction code. Welding repairs to the metal containment shell base material and welds can be categorised as those involving welding of similar materials, dissimilar materials, and austenitic stainless steel and nickel-base cladding. Welding repairs are performed using either the shielded metal-arc or the gas tungsten-arc welding process. With certain exceptions, welds are post-weld heat treated. Where preheat and post-weld heat treatment are impractical, butter bead – temper bead welding can be used. Underwater welding can also be performed. Reference [6.4] provides more detailed information on the three weld repair methods discussed below as well as information related to the weld repair categories noted above.

Shielded metal-arc welding is a manual welding process that uses heat generated by an arc between a covered metal electrode and the component to produce a coalescence of metals. Prior to welding, items to be joined are placed beside or in contact with each other. Welding begins when the welder momentarily touches the electrode on the base material to initiate an arc. The arc melts both the base material and the tip of the welding electrode creating a pool of metal that is continuously transferred to the base material until the electrode metal is consumed or the arc is extinguished. The quality of the weld depends on the design of the joint, selection of the electrode, technique and accessibility, and skill level of the welder. This is the most widely used welding process because it can be used in all orientations, with base-metal thicknesses of 1.6 mm and greater, areas with limited accessibility, and it can be used to apply cladding and hard surface layers [6.4].

Gas tungsten-arc welding is a high-temperature metal-joining process that uses heat generated by an arc between a non-consumable tungsten alloy electrode and the component. An inert gas is used to sustain the arc and to protect the molten metal from atmospheric contamination. Weld pool temperatures can approach 2500°C. The welds can be made with or without filler metal depending on the thickness of materials to be joined, and can join almost all types of metals ranging in thickness from a few thousandths of a millimetre to several millimetres. The process can be used for carbon and low-alloy steels, but it is primarily used for joining dissimilar metals, stainless steels, aluminium, magnesium, and reactive materials, and for root-pass welding of carbon and low-alloy steels. Welds produced by this process are high-quality, low-distortion, and free of splatter. The process, however, requires a relatively high skill level, and slightly more dexterity and co-ordination than shielded metal-arc welding.

Butter bead – temper bead welding is an alternative welding technique that is intended for use in the repair of metal containment pressure boundary components where preheat and post-weld heat treatment are impractical. The technique is suitable for use when the size or configuration of the repair leads to highly restrained weld joints, or the repair area is backed by water. Prior to welding the surface areas must be clean and free of scale, rust, moisture, or other surface contaminants. The procedure involves application of a butter layer of surfacing weld metal followed by the application of temper beads or a temper bead layer. As the welding progresses, the welder must apply a butter bead followed by temper beads or a temper bead layer. Maximum interpass temperature between applications may not exceed 260°C. Improper application of the temper bead or butter bead must be repaired by application of a new butter bead and temper bead. This welding sequence eliminates the need for post-weld heat treatment.

Underwater welding has been used for many years for special salvage operations and making temporary structural repairs. Practical difficulties encountered with underwater welding repairs include rapid quenching of the weldment by the surrounding water and susceptibility of the weldment to hydrogen embrittlement. Underwater welding can be done in either a dry or wet environment. Dry underwater welding is performed in a dry habitat and often requires construction of a customised high-pressure chamber around the welding zone. Shielded-metal arc, gas-metal arc, and gas-tungsten arc welding processes can be used for dry underwater applications, with shielded metal arc being the least desirable because of the smoke and fumes produced. Because of the requirement to provide a chamber, dry underwater welding is used primarily in special or unique situations. Wet underwater welding is performed at ambient pressure with the welder/diver in the water without any mechanical barrier between the water and the welding arc. This procedure has been demonstrated at depths greater than those associated with a metal containment, but relatively poor quality welds can result because of problems associated with heat transfer (more rapid cooling rate) and the presence of hydrogen in the arc atmosphere during the welding operation. Reference 5.7 provides additional information related to underwater welding.

6.2.1.3. Replacement

As an alternative to defect removal or repair, items or portions of containment pressure boundary components that contain flaws, discontinuities, or areas of degradation may be replaced with items that meet the acceptance standards. Items used as replacement are constructed, installed, and documented in accordance with a defined repair/replacement plan.

6.2.2. Repair methods for bellows

Stainless steel bellows expansion joints are used in nuclear plants as flexible seals between process piping and the containment vessel wall. The piping, which may carry steam or other liquids at high pressures and temperatures, moves under the influence of temperature changes and applied forces. Expansion and contraction of bellows accommodates the differential movement between the containment wall and piping while maintaining the leaktight integrity of the containment. Properly installed bellows are intended to serve as a pressure boundary between the inside of the containment vessel and the surrounding atmosphere. Despite efforts to protect the bellows during service, inadvertent mechanical damage can occur as a result of arc strikes, tool gouges, and scratches that may necessitate repair. Options associated with restoring the leaktightness and structural integrity of bellows that have been damaged include replacement of penetration assembly, replacement of damaged bellows, installation of new enveloping bellows, and in-place welding repairs. In some cases, removal or blending of small defects may be possible.

Replacement of an entire penetration assembly that contains a damaged bellows is an option, but is generally not considered. Removal of walls and equipment to provide the required access for replacement would be expensive and time consuming. Replacement of a damaged bellows having one or two plies and replacement with a new one is also a feasible option provided there is sufficient access for a crew of skilled craftsmen and their equipment.

The new bellows would be cut in half using longitudinal saw cuts. After removal of the damaged bellows, the new bellows would be reassembled around the penetration, longitudinally seam welded, and circumferential attachment welds completed to join the bellows ends to the pipe and containment shell. When the outer ply of a two-ply bellows is damaged and its removal could damage an otherwise sound inner ply, a larger enveloping bellows can be installed around the damaged outer ply. Installation would be similar to that for bellows replacement, except thick plate rings that extend outward from the existing bellows support pipe to the new bellows diameter are used to connect the new bellows to the pipe. Prior to installation, the new bellows assembly should be analysed and evaluated to demonstrate that the change in spring constant after its installation can be accommodated. If defects are small it may be possible to remove them. When access permits, a small contoured anvil can be pushed into position inside a dented or mashed convolution to force the damaged surface to return essentially to its original shape. External cosmetic work such as blending is usually required while the anvil is in place. When bellows are found damaged with dents or gouges that are not considered severe, the stress intensifying characteristics of the abrupt change in contour can be lessened by surface blending. If some surface metal was removed when the damage occurred, an appraisal of the loss must be made with respect to pressure requirements. Obviously, it is desirable that little or no additional metal be removed at the deepest point during blending.

REFERENCES TO CHAPTER 6

[6.1] ELECTRIC POWER RESEARCH INSTITUTE, "Guidelines on Nuclear Safety-Related Coatings," TR-109937, Palo Alto, California (April 1998).

[6.2] STUART, C.O., "Underwater Coating Inspections Cut BWR Maintenance Costs," *Power Engineering*, Vol. 91, No. 8, pp. 20-23 (August 1987).

[6.3] STUART, C.O., "Underwater Coating Repair Cuts Nuclear Maintenance Costs," *Power Engineering*, Vol. 97, No. 7, pp. 31-34 (July 1993).

[6.4] OLAND, C.B., and NAUS, D.J., "A Survey of Repair Practices for Nuclear Power Plant Containment Metallic Pressure Boundaries," NUREG/CR-6615, U. S. Nuclear Regulatory Commission, Washington, D.C. (May 1998).

[6.5] OFFICE OF FEDERAL REGISTER, "Code of Federal Regulations, Title 10 – "Domestic Licensing of Production and Utilization Facilities," *Code of Federal Regulations*," Title 10, Part 50, National Archives and Records Administration, Washington, D.C. (January 1, 1997).

[6.6] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, "Rules for Inservice Inspection of Nuclear Power Plant Components," *ASME Boiler and Pressure Vessel Code*, Section XI, Division 1, Subsection IWA, Feneral Requirements, New York, New York (July 1, 1995).

[6.7] AMERICAN WELDING SOCIETY, "Specifications for Underwater Welding," ANSI/AWS D3.6-93, Miami, Florida (1993).

Chapter 7

OPERATING EXPERIENCE

A key method of determining and assessing potential degradation mechanisms for containments is to review their operating and maintenance history. Since many containments have been in operation for twenty years or more, data for age-related degradation currently exists. Table 7.1 presents a summary of reported instances of degradation of containment pressure boundary components for commercial BWR plants in the United States. As the BWR Mark I containments are some of the oldest, most of the age-related experience reported is associated with these containments. Contained in the balance of this section is a summary of incidences related to degradation of containment components (e.g. drywell corrosion, torus corrosion, liner corrosion, and bellows cracking).

In reviewing the material in this chapter it should be noted that overall the operating experience for these components has been very good. The intent of information contained in this chapter is to provide a listing of degradation instances to assist operators having similar plants or operating conditions in setting up (or modifying) their in-service inspection/maintenance programmes.

7.1. BWR MARK I DRYWELL CORROSION

In the early 1980s, water was discovered leaking out the top of the sand bed through an annulus around the torus vent line of the GPU Nuclear Corporation's Oyster Creek nuclear power plant. Water on the torus room floor originating from drain lines also was observed following construction in 1969 [7.1]. The probable sources of water were the equipment storage pool and the refuelling cavity, or fuel pool. It was further concluded that leakage occurred only during refuelling when the refuelling cavity, the equipment storage pool, and fuel pool were flooded.

Water was again found leaking from the sand bed drains during the refuelling outage in 1983. A radiological analysis of water samples indicated that the water had the same radioactivity as water within the reactor, and the leak path was believed to have been from the refuelling cavity immediately above the drywell (Fig. 7.1). Initial investigations revealed that the leak was at the bellows drain line gasket. Later, leaks were also found through several through-wall fatigue cracks in the stainless steel liner of the refuelling cavity [7.2]. The cracks were along the perimeter of the liner plates where these plates were welded to the embedded channels [7.3]. The leaking coolant could have passed along the concrete side of the liner to the 76-mm annular space between the drywell shell and the concrete shield wall, and eventually into the sand pocket region. Table 7.2 identifies potential water sources that were investigated at Oyster Creek and summarises remedial actions that were taken to eliminate water from the sand bed [7.4].

During subsequent investigations, it was discovered that the five 102-mm diameter drains that had been installed in the sandbed during the original construction of the plant to remove water from the sand bed were clogged. This permitted water to saturate the sand and corrode the outside surface of the exposed carbon steel drywell shell. After compacted sand was removed from the drain lines during the twelfth refuelling outage in 1988, hundreds of litres (several hundred gallons) of water drained from the sandbed.

TABLE 7.1. INSTANCES OF CONTAINMENT PRESSURE BOUNDARY COMPONENT DEGRADATION AT COMMERCIAL NUCLEAR POWER PLANTS IN THE UNITED STATES

Plant Designation (Occurrence Date) Plant Type (Source)*	Containment Description (No. of Similar Plants)	Degradation Description	Detection Method
Vermont Yankee (1978) BWR/4 (Ref. 7.17)	Mark I Steel drywell and wetwell (22)	Surface cracks in the overlay weld-to-torus base metal heat-affected zone	Visual examination (As part of modifications to restore the originally intended design safety margins)
Hatch 2 (1984) BWR/4 (Refs. 7.18-7.20)	Mark I Steel drywell and wetwell (22)	Through-wall cracks around containment vent headers within the containment torus (Brittle fracture caused by injection of cold nitrogen into torus during inerting)	Visual examination of torus interior
Hatch 1 (1985) BWR/4 (Ref. 7.20)	Mark I Steel drywell and wetwell (22)	Through-wall crack in nitrogen inerting and purge line (Brittle fracture caused by injection of cold nitrogen during inerting)	In-service inspection testing using magnetic particle method
Monticello (1986) BWR/3 (Ref. 7.21)	Mark I Steel drywell and wetwell (22)	Polysulfide seal at the concrete-to-shell interface became brittle allowing moisture to reach the steel shell	Visual examination (A small portion of the drywell shell was excavated as a part of a life extension study)
Oyster Creek (1986) BWR/2 (Refs. 7.22-7.24)	Mark I Steel drywell and wetwell (22)	Defective gasket at the refuelling pool allowed water to eventually reach the sand cushion region causing drywell shell corrosion	Visual examination of uncoated areas and ultrasonic inspection
Fitzpatrick (1987) BWR/4 (Refs. 7.21 and 7.25)	Mark I Steel drywell and wetwell (22)	Degradation of torus coating with associated pitting	In association with general visual examination of uncoated areas and ultrasonic inspection (Technical specification surveillance performed during outage)
Millstone 1 (1987) BWR/3 (Ref. 7.25)	Mark I Steel drywell and wetwell (22)	Degradation of torus coating	In association with general visual examination of uncoated areas and ultrasonic inspection (The torus had been drained for modifications)
Oyster Creek (1987) BWR/2 (Ref. 7.25)	Mark I Steel drywell and wetwell (22)	Degradation of torus coating with associated pitting	In association with general visual examination of uncoated areas and ultrasonic inspection

TABLE 7.1. (cont.)

Plant Designation (Occurrence Date) Plant Type (Source)*	Containment Description (No. of Similar Plants)	Degradation Description	Detection Method
Brunswick 1 (1987) BWR/4 (Ref. 7.26)	Reinforced concrete with steel liner (9)	Corrosion of steel liner	General visual examination of coated areas
Nine Mile Point 1 (1988) BWR/5 (Ref. 7.27)	Steel drywell and wetwell (22)	Corrosion of uncoated torus surfaces	Visual examination of uncoated areas and ultrasonic inspection
Pilgrim (1988) BWR/3 (Ref. 7.25)	Steel drywell and wetwell (22)	Degradation of torus coating	In association with general visual examination of uncoated areas and ultrasonic inspection (Licensee inspection as a result of occurrences at similar plants)
Brunswick 2 (1988) BWR/4 (Ref. 7.26)	Reinforced concrete with steel liner (9)	Corrosion of steel liner	General visual examination of coated areas
Dresden 2 (1988) BWR/3 (Ref. 7.28)	Steel drywell And wetwell	Coating, electrical cable, and valve operator component degradation due to excessive operating temperatures	Visual examination of uncoated areas and ultrasonic inspection (Ventilation hatches in the drywell refuelling bulkhead inadvertently left closed)
Quad Cities 1 (1991) BWR/3 (Refs. 7.15, 7.29, and 7.30)	Steel drywell and wetwell (22)	Two-ply containment penetration bellows leaked due to transgranular stress-corrosion cracking	General visual examination (Excessive leakage detected)
Quad Cities 2 (1991) BWR/3 (Refs. 7.15 and 7.29)	Steel drywell and wetwell (22)	Two-ply containment penetration bellows leaked due to transgranular stress-corrosion cracking	General visual examination and leakage-rate testing (Excessive leakage detected)
Dresden 3 (1991) BWR/3 (Ref. 7.15)	Steel drywell and wetwell (22)	Two-ply containment penetration bellows leaked due to transgranular stress-corrosion cracking	General visual examination and leakage-rate testing (Excessive leakage detected)
Cooper (1992) BWR/4 (Ref. 7.31)	Steel drywell and wetwell (22)	Corrosion of interior torus surfaces and corrosion stains on exterior torus surface in one area	General visual examination

TABLE 7.1. (cont.)

Plant Designation (Occurrence Date) Plant Type (Source)*	Containment Description (No. of Similar Plants)	Degradation Description	Detection Method
Brunswick 2 (1993) BWR/4 (Refs. 7.26 and 7.32)	Reinforced concrete drywell and wetwell with steel liner (9)	Corrosion of steel liner	General visual examination and visual examination of coated areas (Follow-up inspection based on conditions noted in 1988)
Brunswick 1 (1993) BWR/4 (Ref. 7.32)	Reinforced concrete drywell and wetwell with steel liner (9)	Corrosion of steel liner	General visual examination and visual examination of coated areas (Inspection initiated as a result of corrosion detected at Brunswick 2)
Brunswick 2 (1999) BWR/4 Ref. 7.33)	Reinforced concrete drywell and wetwell with steel liner (9)	Corrosion of liner ranging from clusters of surface pitting corrosion to a 2-mm-diameter hol	General visual examination and visual examination of coated areas (Inspection initiated as a result of corrosion detected at Surry)

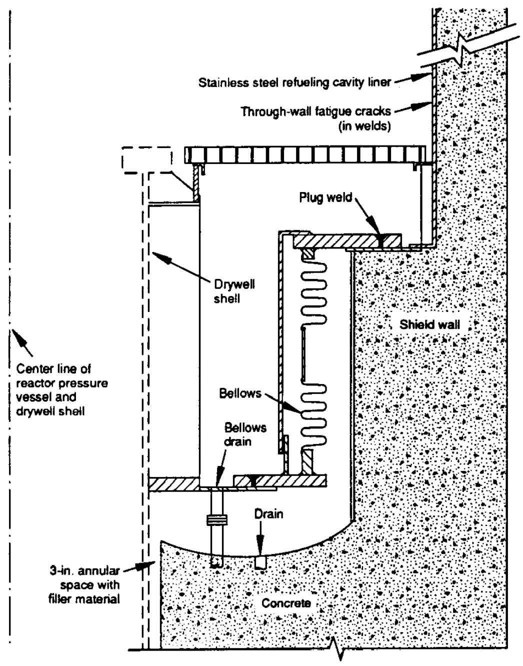


FIG. 7.1. Drywell to cavity seal in BWR Mark I metal containment .

TABLE 7.2. POTENTIAL WATER SOURCES AND REMEDIAL ACTIONS TAKEN TO ELIMINATE WATER FROM THE SANDBED AT THE OYSTER CREEK NUCLEAR POWER PLANT [7.4].

Potential water source	Detection method	Remedial action
Spent Fuel Pool	Vacuum Box Testing	Repaired by underwater welding.
Reactor Cavity Seal Bellows	Pressure Testing	No leaks detected.
Reactor Cavity Seal Drain Line	Pressure Testing	Gasket repaired.
Reactor Cavity Seal Under Drain	Video Surveillance	Modified concrete trough contour.
Reactor Cavity Liner	Visual Inspection, Dye Penetrant Testing, and Vacuum Box Testing	Extensive leaks identified. Temporary coated during refuelling outage, stainless steel tape and elastomeric coating applied.
Skimmer System Piping	Helium Leakage Testing	System was isolated.
Equipment Storage Pool	Visual Inspection, Dye Penetrant Testing, and Vacuum Box Testing	Repaired by welding.
Sandbed Drains	Examination of Drain Lines	Drains were unclogged allowing water to drain from the sandbed.

Because corrosion of the outside surface of the drywell shell was suspected, extensive ultrasonic testing (UT) was performed from inside the containment to determine the extent and severity of the degradation. These measurements revealed that thinning was most severe in the sand bed region where the original plate thickness was 29.3 mm and that shell thicknesses in some local areas were as low as 20.3 mm. These findings were particularly alarming because the minimum acceptable drywell shell thickness in the sand bed region was 18.8 mm. Verification of the UT measurements was achieved by removing 51-mm diameter cores from the drywell shell and physically measuring their thickness. Holes produced by the core drilling operation were replaced with machined plugs that were seal welded to the drywell shell from inside the containment. Once completed, inspected, tested, and accepted, the leaktight integrity of the Oyster Creek containment was restored to its original condition. One conclusion resulting from analyses of the water, corrosion products, and the core samples was that the red-lead coating applied to the outside surface of the drywell does not provide adequate corrosion protection to carbon steel subject to dilute acidic water conditions (i.e., the red-lead coating is cathodic to the carbon steel so the steel is sacrificial relative to the lead) [7.5, 7.6]. Figure 7.2 shows the corroded area on the outside surface near the sand pocket region [7.2].

Initial efforts to stop the corrosion process involved fixing leaks in the drywell-to-refuelling-cavity seal and installing a cathodic protection system. In 1988, anodes were inserted into the sand bed through small-diameter holes through the concrete biological shield wall. This scheme for arresting the corrosion process by controlling the flow of current between anodic and cathodic surfaces was only effective for a short period of time. As the sand around the anodes dried out, the electrical circuit between the cathode (drywell shell) and the anodes was broken thereby rendering the system ineffective. The ineffectiveness of the cathodic protection system was verified by UT measurements. Analysis of time-dependent UT

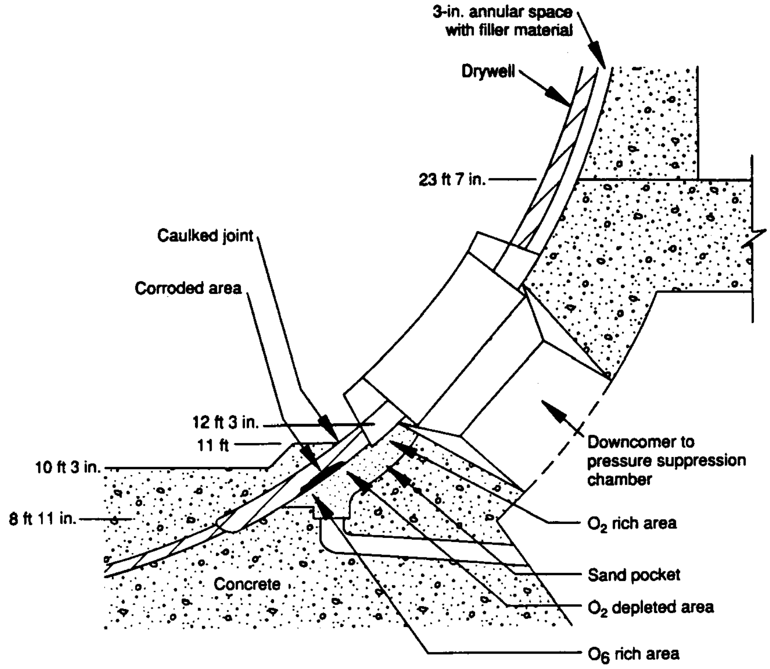


FIG. 7.2. BWR drywell base showing corroded area on outside surface near sand pocket region.

data revealed that the rate of corrosion before and after installation of the cathodic protection system was the same.

After attempts to stop the corrosion process by application of cathodic protection failed, aggressive efforts were undertaken to remove the sand and apply a protective coating of epoxy paint to accessible areas of the drywell shell in the sand bed region. Access to the sand bed was provided by drilling 508-mm diameter holes through the concrete biological shield wall about 305 mm away from the ten vent lines. These holes, which were completed in November 1992, were large enough to allow workers to crawl from the torus region into the sand bed. About one week into the fourteenth refuelling outage that started on November 28, 1992, workers entered the sand bed and began vacuuming out the sand. As the sand was being removed, workers discovered other problems.

- The floor of the sand bed was rough and irregular (large voids were found in some parts of the sand bed floor).
- Segments of reinforcing bars were not embedded in concrete.
- The drain pipes were protruding about 76 to 102 mm above the rough concrete floor surface.

According to the original design documents, a smooth concrete floor with troughs leading to the five drains was to be constructed to serve as the floor for the sand bed. Because this work was never performed, some standing water always remained at the bottom of the sand bed to sustain the corrosion process even when the drains were functioning properly. In order to solve this problem, a new sand bed floor was installed using an epoxy-based system to fill the voids, cover the exposed reinforcing bars, and raise the floor to the level of the top of the five drain pipes. Even though the original design called for sand to be installed in the sand bed to provide transitional radial support for the drywell shell, sand was not reinstalled after the floor was repaired and the walls were painted. This consensus decision between GPU Nuclear Corporation, General Electric, and NRC personnel was based on results of detailed analytical studies performed to resolve this issue. Results of the entire Oyster Creek investigation also provided the basis for reducing the containment peak pressure from 427 kPa to 303 kPa and for establishing a new minimum drywell shell thickness in the sand bed region of 13.7 mm.

During cleaning of the drywell shell surface prior to painting, the workers discovered that the corrosion was relatively uniform and that it could be easily removed with scrapers and hand-held equipment. By the end of January 1993, the drywell shell was cleaned and painted with a two-part, self-curing epoxy coating allowing the plant to return to service at the end of the refuelling outage in early February. Application of the protective coating on the outside of the drywell shell was not required by the NRC because coatings provide no specific safety-related function to mitigate the consequences of postulated accidents.

GPU Nuclear Corporation continues to monitor the long-term performance of the drywell shell as part of its overall ageing management strategy. Monitoring activities include:

- periodic visual examinations of the epoxy paint,
- UT measurements of the drywell shell above the sand bed, and
- inspections for leakage from the reactor cavity.

So far, no additional thinning of the drywell shell has been detected, the epoxy paint appears to be in excellent condition, and efforts to eliminate water from the sand bed region have been effective.

In the event that remedial actions are required in the future, GPU Nuclear Corporation has prepared contingency plans for repairing the drywell shell to restore its structural integrity. Figure 7.3 provides a schematic of corrosion damage in an inaccessible area such as could occur in the drywell shell. The four repair welding techniques that are proposed in the plan include:

- replacement plate repair welding,
- doubler plate repair welding,
- stiffener plate repair welding, and
- surface overlay repair welding.

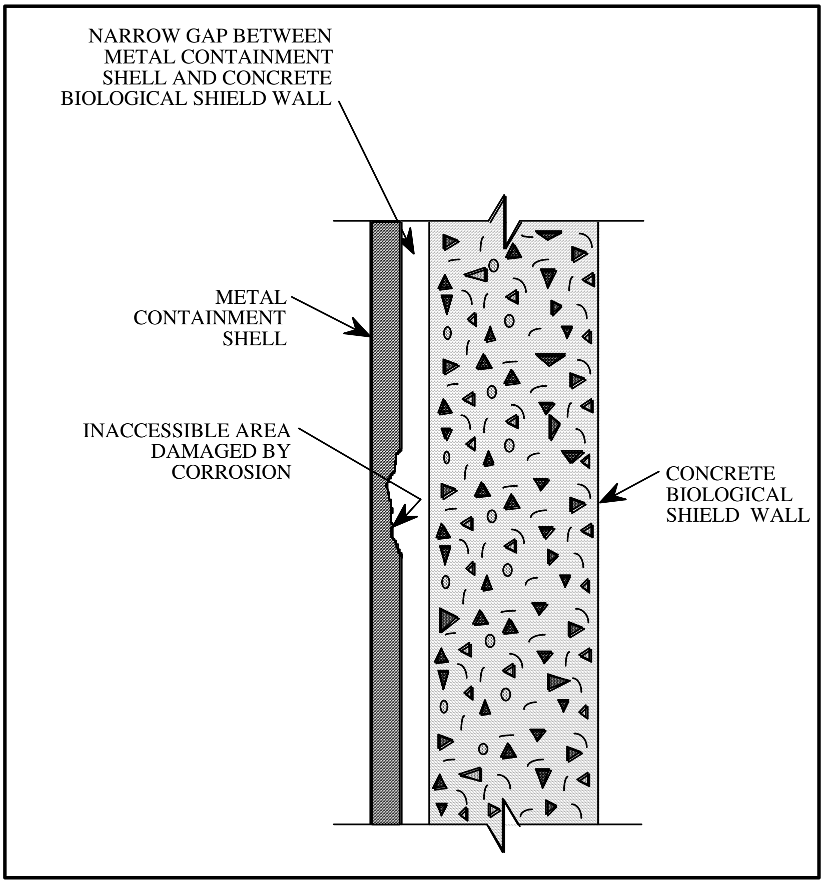
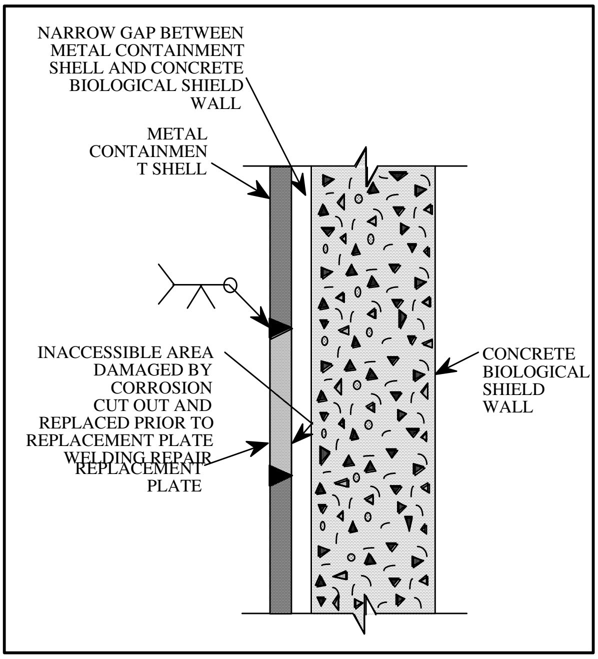
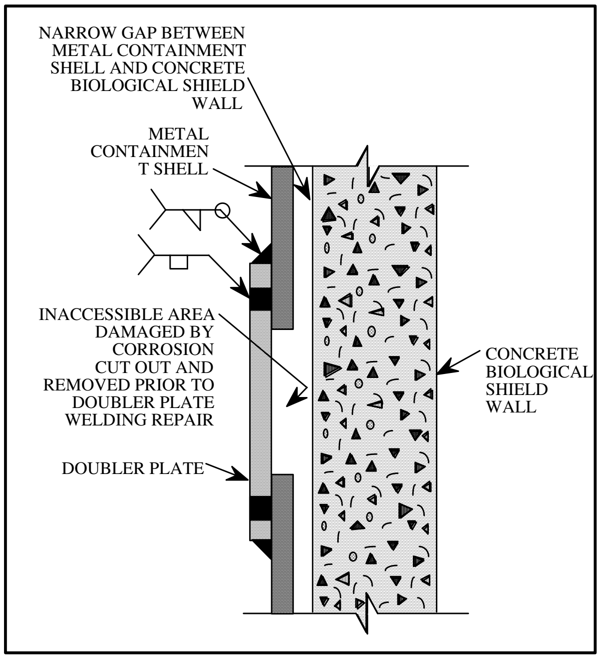


FIG. 7.3. Illustration of inaccessible area of metal containment exhibiting significant corrosion.

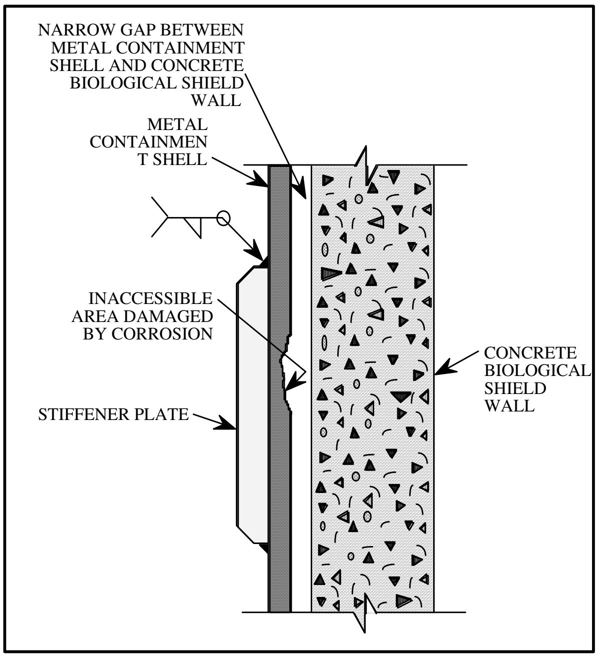
In the replacement plate welding repair, the structural integrity of the containment is restored to its preservice condition by removing the defective area, replacing it with new plate material, performing the necessary repair welding and post-weld heat treatment, and conducting the required non-destructive evaluations and leakage-rate tests. The doubler plate welding repair technique restores the structural capacity and leaktight integrity by removing the damaged portion of the metal shell, fitting a larger plate over the hole, performing the necessary repair welding and post-weld heat treatment, and conducting the required non-destructive evaluations and leakage-rate tests. Stiffener plates can be used to strengthen the remaining shell without affecting the leaktightness of the containment. Use of stiffener plates eliminates the need for repair of the corroded surface by providing additional structural elements to restore structural integrity. Because this repair technique does not involve repair of the corroded surface, it has no effect on the leaktight integrity of the containment. Surface overlay welding eliminates the need for repair of the corroded surface by providing sufficient replacement metal to restore structural integrity. This repair technique is also desirable because it has no effect on the leaktight integrity of the containment. Schematics illustrating each of the four proposed repair welding techniques are provided in Figure 7.4.



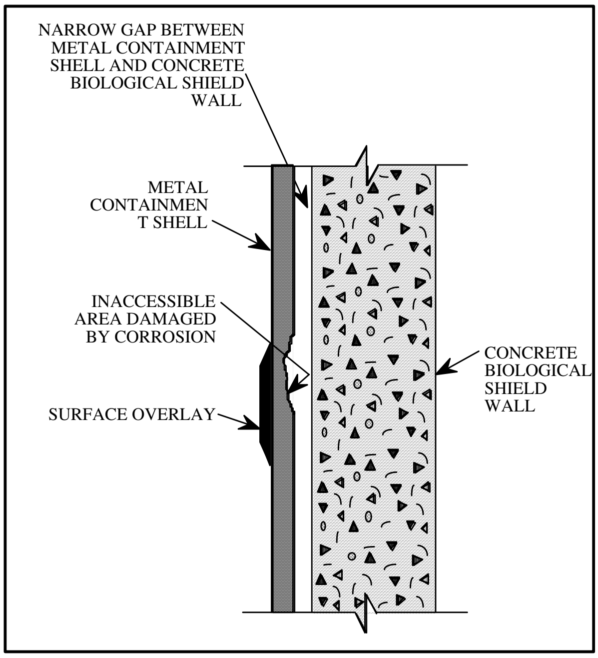
Replacement plate technique.



Doubler plate technique.



Stiffener plate technique.



Surface overlay technique.

FIG. 7.4. Examples of weld repair techniques to address corrosion occurrence on inaccessible portions of metal containment shell

7.2. BWR MARK I TORUS CORROSION

Most areas on the outside surface of the torus of Niagara Mohawk Power Corporation's Nine Mile Point Unit 1 nuclear power plant are accessible for visual inspection, and the surface is coated to prevent corrosion. The inside of the torus is partially filled with water, and all surfaces above and below the water line are not coated. Consequently, progressive thinning of the torus shell due to corrosion is considered likely.

Niagara Mohawk has monitored the thickness of the torus shell since 1975 because of its degradation potential and the resulting significance to containment integrity. After inspections in March 1988 revealed that the torus wall was very near its minimum allowable thickness, additional calculations were performed establishing a worst case minimum wall thickness of 11.4 mm. These calculations reflected a reduction in condensation oscillation loads and indicated that the most critical location is at the bottom of the torus shell. Periodic UT was performed to quantify the amount of wall thinning that had occurred and to estimate the rate of corrosion. Based on visual inspections performed inside the torus, non-destructive examination results, and laboratory analyses of water samples, Niagara Mohawk concluded that the observed wall thinning was being caused by general corrosion and that local attack (pitting, crevice, and biological corrosion) was not occurring.

Following this engineering evaluation, a new corrosion monitoring programme was initiated in August 1989 to measure the thickness of all 40 mid-bay plates on the bottom surface of the torus. Part of the programme included suspending metal samples fabricated from ASTM A 516, Gr. 70 material in the torus water so that the thickness of these samples could be periodically measured. ASTM A 516 material was used because the chemistry of this material was similar to the ASTM A 201, Gr. B steel used in the original construction, but was no longer available. Prior to installation, the samples were preconditioned in the same way that laboratory corrosion test specimens are preconditioned prior to exposure testing [7.7].

Since the monitoring programme began, UT measurements have been performed at six-month intervals. Every effort is made to use the same personnel and equipment to examine the same locations during each inspection. Results of these UT measurements are used to update the thickness of the plates and to estimate remaining service life of the torus. At the current rate of corrosion, it is estimated that the torus shell will be at its minimum acceptable thickness about the year 2007, which is near the end of its 40-year initial operating license.

In addition to the monitoring programme, a number of options to mitigate the effects of corrosion have also been considered by Niagara Mohawk. These options, which have not been implemented as yet, include:

- using a corrosion inhibitor in the torus water;
- inerting the torus with nitrogen during outages;
- coating the inside surface of the torus;
- installing a cathodic protection system; and
- modifying the torus to improve its structural capacity.

Addition of corrosion inhibitors to the torus water was rejected for the following reasons. Chemicals to scavenge oxygen, such as hydrazine, would require removal prior to startup; could produce undesirable gases, such as ammonia; could cause pH problems; and could be a safety risk (i.e., carcinogenic). In addition, the possible gains in terms of reducing corrosion were estimated to be minimal. Maintaining a nitrogen purge on the torus during outages was rejected because of safety concerns. Nitrogen could escape or leak from the torus into the drywell resulting in pockets of low oxygen concentrations. Atmospheric conditions like this could create a potential suffocation hazard for workers. Despite the potential benefits of an effective coating system, application of either an organic or a metal spray protective coating on the inside surface of the torus was rejected. Application of either coating system would require removing the water, sludge, and debris from the torus and thoroughly cleaning all exposed surfaces in preparation for the coating. Additional reasons for rejection include outage critical path impacts, the relatively short service life of a coating in this environment, the need for extensive long-term maintenance, and as low as reasonably achievable considerations. Application of a metal spray, such as zinc, zinc-aluminium, or aluminium, would provide at least one distinct advantage over an organic coating. The metal spray coating could be classified as non-safety related. Unlike organic paints and epoxies that fail by producing loose flakes or sheets that could potentially clog the emergency core cooling system (ECCS), consequences of metal spray coating failure would likely not affect plant safety. A catastrophic metal spray failure would result in sheets of metal falling from the surface and sinking to the bottom of the torus without affecting the performance of the ECCS. Compared to organic coatings, metal spray coatings take somewhat longer to apply. However, the time difference is not considered significant. Installation of sacrificial anodes and an impressed current cathodic protection system also were considered to stop the corrosion process. In the sacrificial anode concept, either zinc anodes would be placed in the water and electrically connected to the torus shell, or zinc screens would be welded to the torus shell surface. After installation, the zinc would create a passive protection system requiring no periodic maintenance. However, use of sacrificial anodes was not considered feasible due to low conductivity of the water. An alternative approach based on an impressed current cathodic protection system for this application would be more complex than the sacrificial anodes system just described. This active system would require installation of an electrical conductor 360 degrees around the torus with direct current applied between the cable and the torus shell. Direct current would be supplied by a rectifier powered by an alternating current source. For the system to function properly and provide the required corrosion protection, the conductor would need to be installed under water and supported by structures attached to the torus shell. Due to concerns about loads imposed on the conductor and its support structure during a loss of coolant accident, and the impracticality of installing a suitable system, installation of an impressed current cathodic protection system was also not considered feasible.

The most viable option involves structural modifications to the torus shell to enhance its ability to resist applied loads. In this concept, eight stiffener rings would be fillet welded to the outside surface of the torus shell in each of the 20 bays. The stiffener rings would be fabricated from 457-mm wide by 12.7-mm thick carbon steel plates rolled through the thickness to conform to the outside surface of the torus shell. Each ring would be prefabricated in sections that would be approximately 4-m long. This dimension was selected to facilitate movement into the torus room. Once inside, the ring sections would be assembled on the floor, welded together, and then turned and lifted into position. All eight stiffener rings for each bay would be spaced longitudinally at 305-mm intervals. The four centre stiffener rings would extend 210 degrees around the shell and be centred about the bottom of the torus. Inherent flexibility of the thin stiffener rings would allow spreading to fit a curvature greater than 180 degrees. In order to minimise the impact on plant operations, it is proposed that these stiffener rings would be installed while the plant is in operation. Under these conditions, the torus would contain water making it necessary to weld some parts of each stiffener ring with water backing. Figure 7.5 shows a conceptual view of how these four stiffener rings would be installed on each of the 20 bays [7.2]. Two adjacent stiffener rings would extend 15 degrees above the horizontal centreline on the outer part of the torus and terminate at the inner column wing plates. The final two stiffener rings would be coped between the outer column wing plates and the mitre joint between bays on the inner part of the torus. To accommodate obstructions such as penetrations and reinforcing pads, the stiffener rings would be coped around or bridged to adjacent rings to provide an acceptable load path. The configuration of the stiffener rings to accommodate these obstructions would be developed to allow sufficient access for inspection of the stiffener plate to torus shell fillet welds. If these structural modifications were performed, installation of the stiffener rings would re-establish an adequate corrosion allowance for the projected remaining plant life plus a 20-year extension.

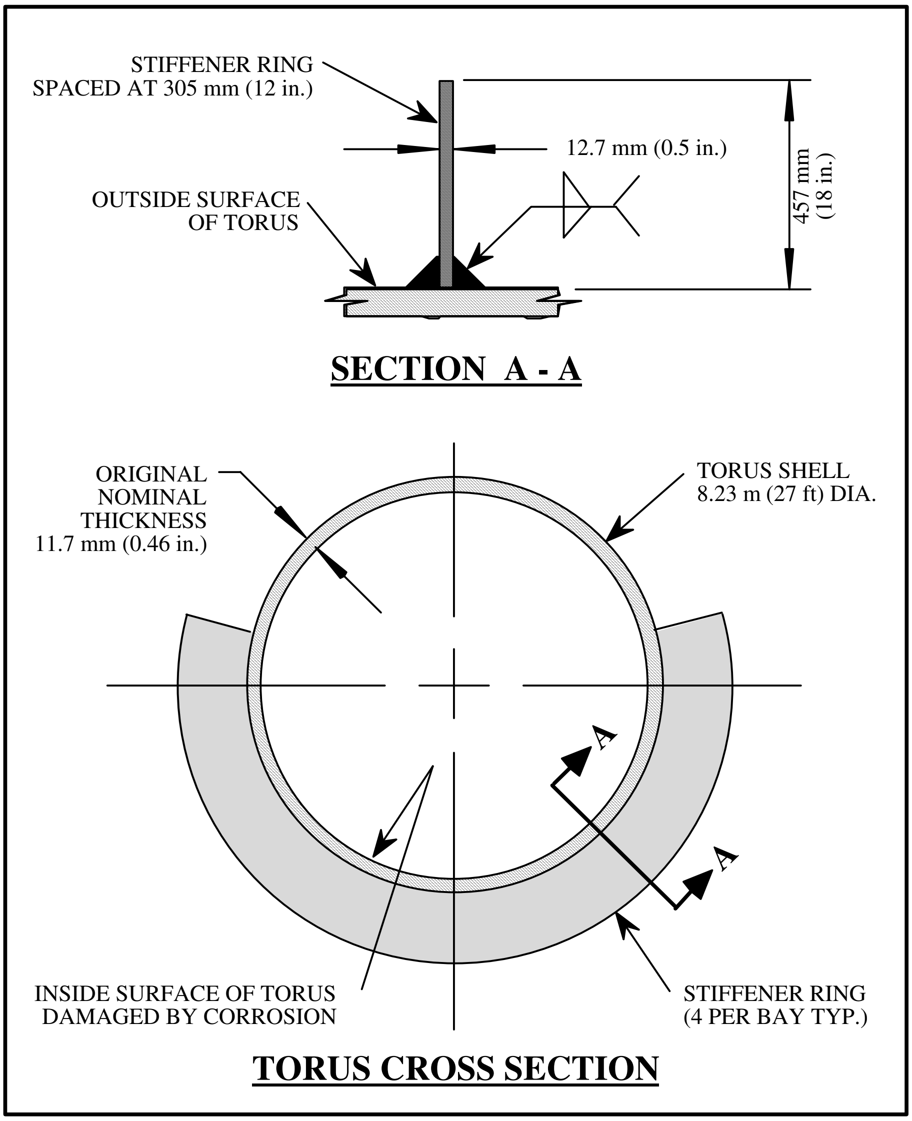


FIG. 7.5. Conceptual view of how four of eight stiffener rings would be installed on the outside surface of each bay of the Nine Mile Point Unit 1 torus.

7.3. BWR LINER PLATE CORROSION

Liners of reinforced and post-tensioned concrete containments are typically constructed using relatively thin (about 6.4-mm thick) carbon steel plates that are welded together to create a leaktight barrier against the uncontrolled release of radioactivity to the surrounding environment. Although liner plates are not designed to carry loads, corrosion could have a detrimental effect on containment reliability and availability under design basis accidents and beyond design basis events. Any liner plate thinning can create geometrical transitions that influence strain concentration. This influence could change the failure threshold under challenging environmental or accident conditions and may reduce the design margin of safety. Corrosion that results in thinning, pitting, or cracking is of particular concern when the entire thickness of the liner plate is affected. Holes, pits, and cracks that penetrate completely through the liner plate disrupt the pressure boundary and may create pathways to the surrounding environment. Most of the instances of liner plate corrosion have primarily occurred in PWR plants in the United States, France, and Germany. The only identified incidence of liner plate corrosion involving a BWR plant that was considered significant occurred at the Brunswick 1 and 2 in the United States [7.8 – 7.13], and Barsebäck Unit 2 and Forsmark Unit 1 in Sweden [7.14].

7.3.1. Brunswick Units 1 and 2

General and pitting corrosion affecting as much as 50 percent of the nominal 8-mm-thick liner was detected at several locations along a narrow band around the inside circumference of both drywells at Brunswick Units 1 and 2. The corrosion was caused by an accumulation of water at the junction of the drywell liner and the concrete floor surface (Elev. 4'-6") at the bottom of the containment. Degradation of sealing materials applied around the inside circumference of the containments at this junction allowed the water to enter and accumulate in this region. Procedures used by the licensee to quantify the extent and depth of the corrosion damage involved:

- removing concrete adjacent to the liner to provide access for inspection (Unit 1 only),
- cleaning (sandblasting and wire brushing) the liner plates,
- selecting designated inspection zones,
- measuring the base metal plate thickness using ultrasonic testing methods, and
- determining the depth of pitting and general corrosion using dental moulding compound.

Metal loss and pitting depth measurements revealed that there were locations of the liners that were below the minimum acceptable thickness of 5 mm. Five such sections were identified in Unit 2, but the damage in Unit 1 was more severe. Corrosion was observed around virtually the entire circumference of the Unit 1 drywell. The damage even extended below the level of the concrete floor surface making removal of concrete adjacent to the liner necessary. Although corrosion of the drywell liner for Unit 1 was more severe than the corrosion for Unit 2, the leaktight integrity of both Brunswick containments was never jeopardised because the thinning and pitting did not penetrate completely through the liner plates.

In order to restore damaged liner plates to the required minimum thickness and thereby allow the units to be returned to service, the licensee performed a series of construction activities. Details of these activities are described below.

- Areas with significant metal loss or pitting were repaired by overlay welding in which weld metal was deposited on the damaged liner plates to supplement the existing thickness. During welding, efforts were taken to limit the interpass temperature of the liner plates to 79°C.

- Following welding, each area was examined using the liquid penetrant test method. Results of this test were used to determine the acceptability of the welding repairs.

- All damaged and repaired areas were recoated.

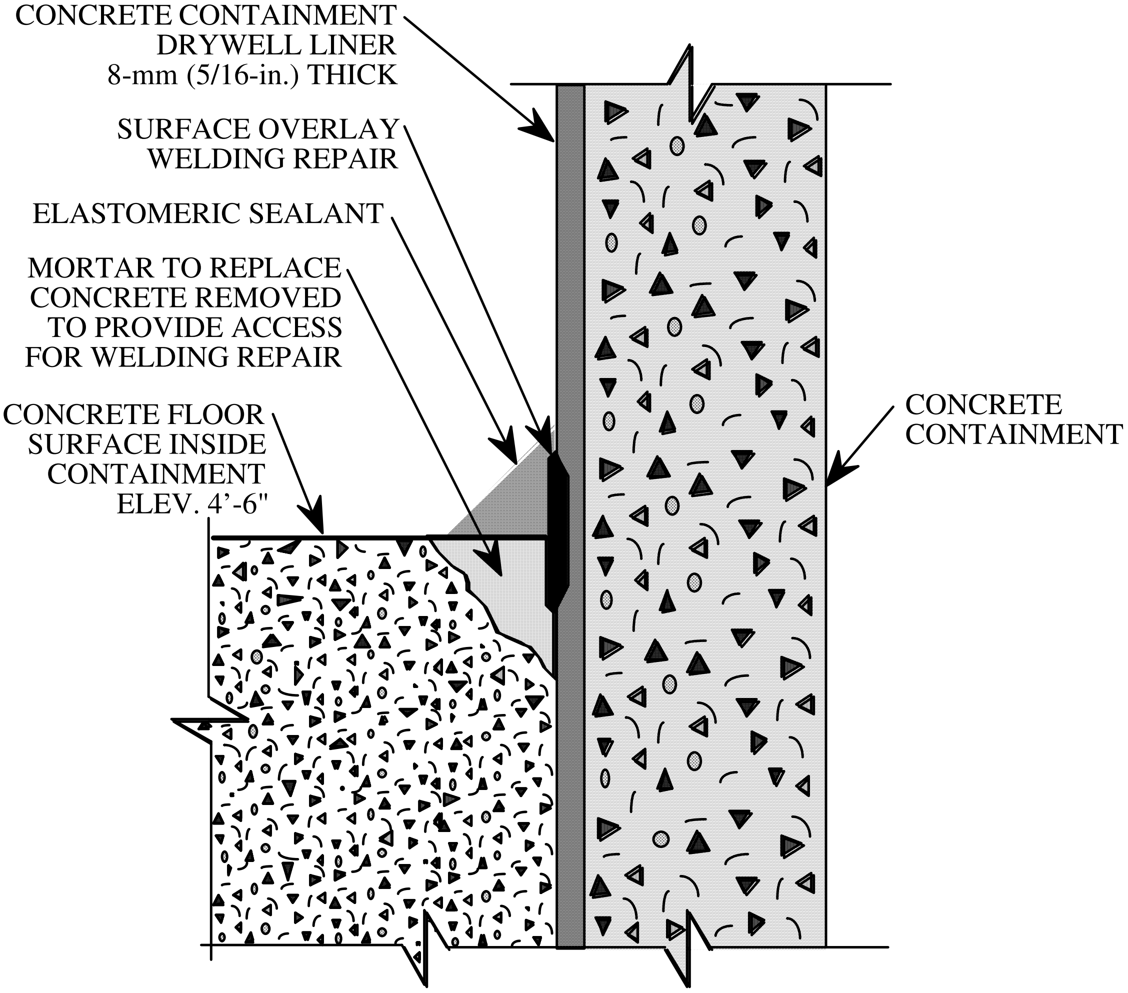
- Mortar was placed in Unit 1 to return the concrete floor to its original elevation and configuration.

- Intersections of the concrete floors and drywell liners were sealed with an elastomeric sealant.

Figure 7.6 shows a cross section of the drywell liner repair for the Unit 1 containment.

7.3.2. Barsebäck Unit 2

At Barsebäck Unit 2 the containment liner is a carbon steel plate, 7 mm thick, that is located between an outer concrete wall, thickness approximately 900 mm, and an inner concrete wall, thickness approximately 200 mm. By embedding the liner in concrete, it is hoped that liner corrosion would not occur. During a leakage-rate test, however, it was found that the containment did not meet the maximum value of less than 0.35% pressure drop in 24 hours. A freon sniffing test and application of soap film to the outside of the containment identified the source of excessive leakage as being close to an electrical cable penetration through the containment wall. Examination of the liner by means of holes drilled through the inner concrete layer revealed a number of holes in the liner, with the largest having a diameter of 20 mm. The source of the holes was corrosion associated with an area of poor concrete filling (i.e. voids) in conjunction with the presence of excess water from the concrete mix. Some areas of corrosion attack were also found at other penetration locations, but the corrosion had not penetrated the liner thickness at any of these locations. Subsequent examination of about 25% of the penetrations at Barsebäck Unit 1 by drilling holes and using ultrasonics to determine the liner thickness found no thinning of the liner. The explanation for the difference between the two plants was that a slightly different manufacturing process had been used at Unit 1.



NOTES:

1. CONCRETE AT THE FLOOR-LINER INTERSECTION WAS REMOVED TO PROVIDE ACCESS FOR INSPECTION AND WELDING REPAIR ACTIVITIES.

2. OVERLAY WELDING WAS PERFORMED TO RESTORE PORTIONS OF THE LINER DAMAGED BY CORROSION.

3. AFTER WELDING, AFFECTED AREAS WERE COATED, MORTAR WAS PLACED TO RESTORE THE FLOOR SURFACE, AND AN ELASTOMERIC SEALANT WAS APPLIED AT THE FLOOR-LINER INTERSECTION.

FIG. 7.6. Cross-sectional view of the drywell liner repair performed inside the Brunswick Unit 1 BWR/4 Mark I concrete containment.

7.3.3. Forsmark Unit 1

During the 1997 integrated leakage-rate test at Forsmark Unit 1, leakage above acceptable values was recorded. The excess leakage occurred in the toroid of the reactor containment, which had been degraded by corrosion. The toroid, consisting of an outer and inner steel plate, connects the steel liner in the concrete structure of the containment building to the flange of the drywell head (Fig. 7.7). The toroid is a relatively thin, flexible member that allows for thermal expansion of the drywell head relative to the adjacent concrete structure. A cover of mineral wool was placed on the outer surfaces of the toroid and flange to form space adjacent to the concrete to accommodate thermal expansion (Fig. 7.8). In addition, a plastic film was placed between the mineral wool and concrete (probably to protect the mineral wool while concrete was being placed). Corrosion protection of the toroid was provided by epoxy paint, but due to design errors relative to the reactor vessel insulation, temperatures reached greater than 100°C. Since the expected maximum temperature at this location was 65°C, the higher temperature probably degraded the corrosion-protection coating.

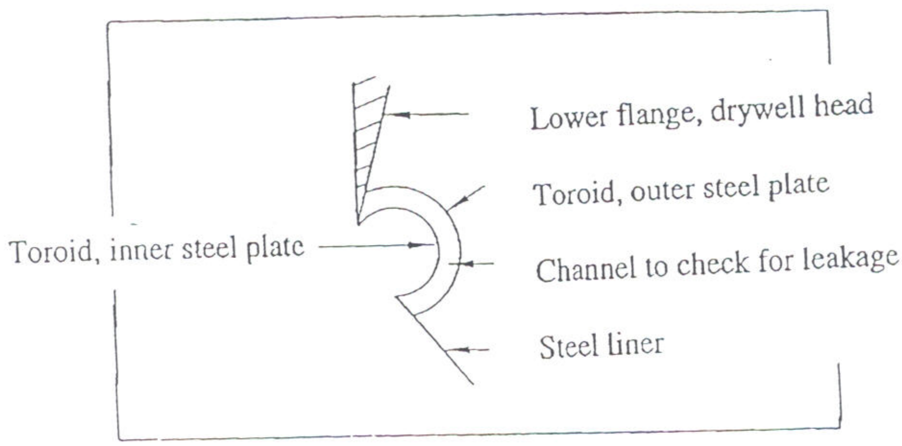


FIG. 7.7. Toroid Forsmark Unit 1.

The cause of toroid corrosion was cracking in the welds of the steel liner at the bottom of the reactor pool that was situated just above the toroid. The cracks permitted moisture from the pool to leak into the mineral wool where it was trapped adjacent to the toroid by the plastic film. Corrosion started in areas where the epoxy paint had been damaged.

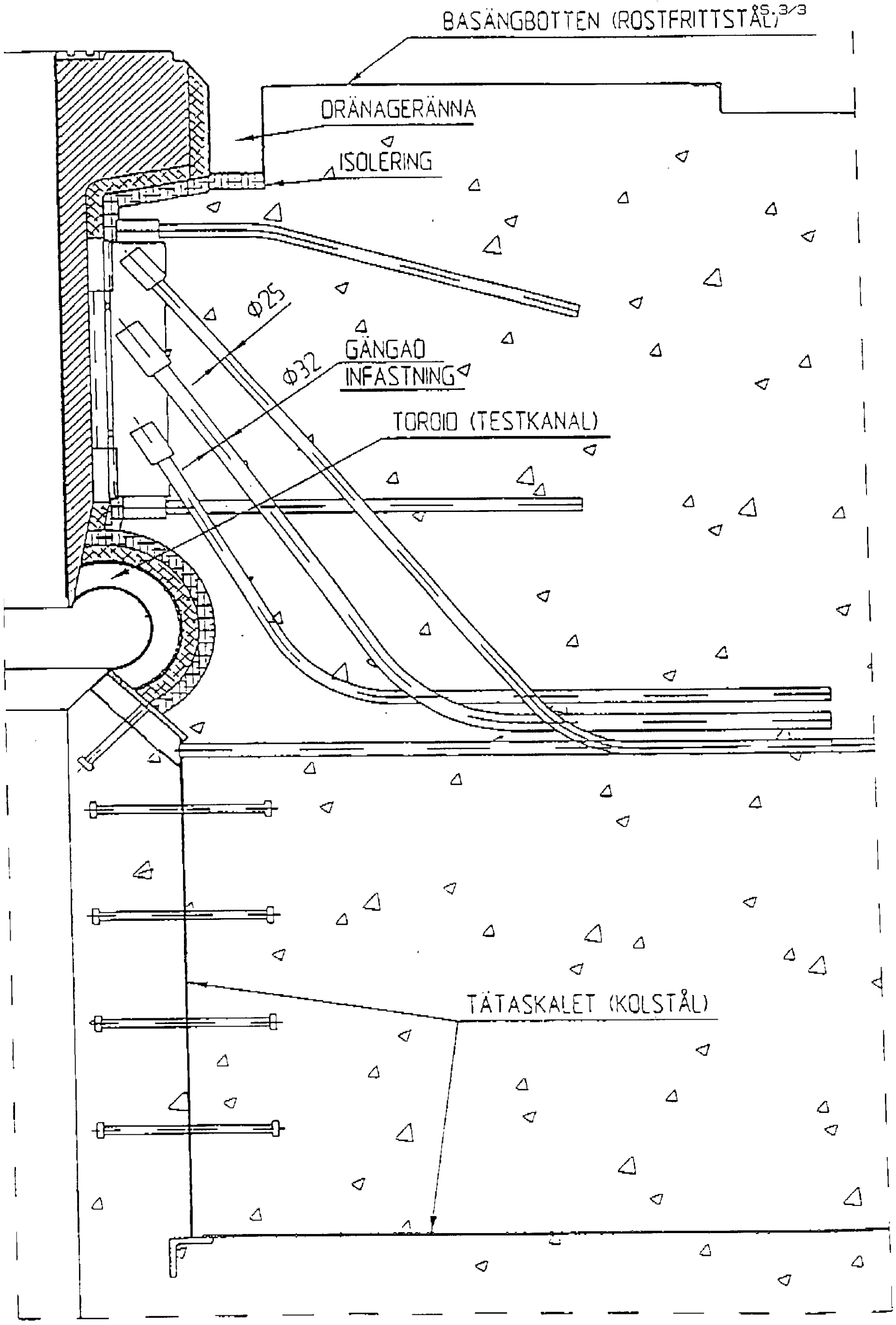


FIG. 7.8. Mineral wool located on outer surfaces of toroid and flange.

Initial inspection identified significant damage that resulted in the complete toroid being removed for more detailed examination. The pool liner welds were inspected by penetrant testing and then repaired by welding. The plastic film and mineral wool were removed and the inner plate repaired by welding prior to reinstallation. Also, a permanent ventilation system was installed to provide dry air in the space behind the toroid. The relative humidity of the air emerging from this area is monitored to check for any increases in relative humidity.

7.4. BWR/3 MARK I BELLOWS CRACKING

While conducting a containment integrated leakage-rate test at Quad Cities Unit 1, excessive leakage was detected from the drywell ventilation penetration bellows [7.15]. Application of a soap film to the bellows surface while the containment was pressurised showed one large indication and over 100 smaller indications. A metallurgical investigation identified the cause of the indications as transgranular stress corrosion cracking with the primary indication initiating on the bellows inner surface. No evidence was found that the crack growth was due to fatigue. Chlorides, fluorides, and sulphides were identified as the responsible corrosive initiators. They may have accumulated during fabrication, construction, or operation. A review of the bellows size, configuration, and design displacements at all penetrations of Quad Cities Unit 1 containment revealed that the ventilation penetration bellows was among the most highly stressed bellows in the plant. The greater than expected leakage rate from the ventilation bellows likely resulted in part from some of the maintenance activities that took place prior to the leakage-rate test, opening the transgranular stress corrosion cracks. Transgranular stress corrosion cracking was also identified as a failure mechanism that resulted in leakage from bellows at Quad Cities Unit 2 and Dresden Unit 3 [7.16].

7.5. OTHER EXPERIENCE

In Japan, according to their Incident Reporting System, there have not been any problems in the BWR containment vessels with respect to torus corrosion, liner plate corrosion, or bellows cracking.

In Germany, inspections of steel containments of BWR plants have not revealed any significant degradation. This is attributed to details addressed in the design phase with special attention paid to exclusion (to the extent possible) of the possibility for corrosion attack, and surveillance requirements that have been imposed. Documented cases of corrosion attack include sealing surfaces, condensation chamber where coating degradation has occurred, below insulation where moisture can accumulate, and at the interface where the steel shell becomes embedded in concrete. Only a small number of such events have occurred over the last 25 years; however, only limited information is available on the condition of the steel shell in areas where it is not accessible for inspection. Corrosion potential for these areas is assessed through companion corrosion test specimens that are maintained while being subjected to similar environments. Some limited studies have been conducted to evaluate the applicability of non-destructive testing methods (e.g. ultrasonics) with respect to optimal life extension of the plants. Corrosion in the plants is expected to be a slow process, however, because they are inerted by nitrogen.

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Chapter 8

AGEING MANAGEMENT PROGRAMME FOR METAL COMPONENTS OF BWR CONTAINMENTS

The information presented in this report indicates that BWR service conditions can cause ageing degradation of metal components of BWR containments and thus impair their safety functions. The primary ageing mechanisms that may potentially impact the structural capacity, leaktight integrity, or service life of BWR containments are corrosion of metal components and stress corrosion cracking of bellows. Other potential degradation mechanisms include fatigue and mechanical wear. Areas of concern are where surfaces are inaccessible for inspection (e.g. areas adjacent to floors, where the containment vessel is embedded in concrete, and locations adjacent to equipment or other structures). Therefore, a systematic ageing management programme (AMP) for the metal components of BWR steel containment vessels, including the bellows, is needed to preserve the overall safety of the plant.

The preceding chapters of this report dealt with important elements of an ageing management programme whose objective is to maintain the fitness-for-service of the metal components of a BWR containment throughout plant service life. Chapters 2, 3 and 4 contain information on important aspects of understanding these components and their ageing. Chapter 5 provides information on various techniques for detecting ageing and assessing its effects. Chapter 6 contains information related to methods for prevention and mitigation of unacceptable ageing effects. Chapter 7 summarises operating experience in terms of degradation, inspection, and repair of metal components of BWR containments; although in all cases the containment function was preserved, the degradations experienced have been of concern to the safety authorities in various countries.

This chapter describes how the above elements are integrated within a plant specific AMP for metal components of a BWR containment utilising a systematic ageing management process, which is an adaptation of the Deming's "plan-do-check-act" cycle for ageing management, Fig.8.1. Such an ageing management programme should be implemented in accordance with guidance prepared by an interdisciplinary ageing management team organised at the corporate or owners' group level. For guidance on the organisational aspects of a plant ageing management programme and interdisciplinary ageing management team refer to IAEA Safety Report Series No.15 [8.1].

A comprehensive understanding of metal components of BWR containment systems, their ageing degradation, and the effects of degradation on the containment's ability to perform its design functions is a fundamental basis for the AMP. Knowledge of the plant, and of the impact of any degradation, is fundamental in making decisions about the inspection requirements, evaluating inspection results, and choosing any remedial strategies. Plant specific knowledge is enhanced by drawing on external experience related to behaviour of metal containment structures such as is available in technologies related to pressure vessels and piping, or the offshore oil industry.

In order to maintain the fitness-for-service of the metal components of a BWR containment throughout plant service life, it is necessary to control within defined limits their potential aged-related degradation. Effective degradation control is achieved through a systematic ageing management process consisting of the following ageing management tasks, based on understanding of ageing:

- operation of the plant within specified operating conditions aimed at minimising the rate of degradation, in particular, error-induced accelerated degradation (managing ageing mechanisms);

- inspection, monitoring and condition assessment consistent with requirements aimed at timely detection and characterisation of any degradation to determine fitness for service (detecting ageing effects);

- maintenance, i.e. repair, or replacement to correct or eliminate unacceptable degradation (managing ageing effects).

Existing ageing management programmes for containment structures have generally focused on managing ageing effects (i.e. the AMP is based on periodic inspection or monitoring of the structure, with remedial measures being implemented to deal with any observed degradation before serviceability is lost). An alternative (proactive) approach to AMP involves an additional step of monitoring and controlling the operational environment aimed at minimising ageing degradation. Such an approach may be appropriate, in particular for inaccessible parts of the structure, where detection of degradation would be difficult, or where repair of any degradation would be costly.

A systematic ageing management programme for metal components of BWR containment co-ordinates programmes and activities contributing to the above ageing management tasks in order to detect and mitigate ageing degradation before containment safety margins are compromised. This programme reflects the level of understanding of the BWR metal containment ageing, the available technology, the regulatory licensing requirements, and the plant life management consideration/objectives. Timely feedback of experience is essential in order to provide ongoing improvements in the understanding of the ageing degradation and in the effectiveness of the ageing management programme. The main features of the ageing management programme, including the role and interfaces of relevant programmes and activities in the ageing management process, are shown in Fig.8.1 and discussed in the following sections.

8.1. UNDERSTANDING AGEING

Understanding BWR steel containment ageing is the key to effective management of its ageing. In addition it is vital with respect to: integrating ageing management activities within a systematic ageing management programme; managing ageing mechanisms through prudent operating procedures and practices (in accordance with technical specifications); detecting and assessing ageing effects through effective inspection, monitoring, and assessment methods; and managing effects using proven maintenance methods. This understanding consists of: knowledge of BWR steel containment materials and material properties, stresses and operating conditions, likely degradation sites and ageing mechanisms, condition indicators and data needed for assessment and management of ageing and the effects of ageing on safety margins.

The understanding of BWR steel containment ageing is derived from baseline data, operating and maintenance histories, and external experiences. This understanding should be updated on an ongoing basis to provide a sound basis for the improvement of the ageing management programme consistent with operating, inspection, monitoring, assessment, and

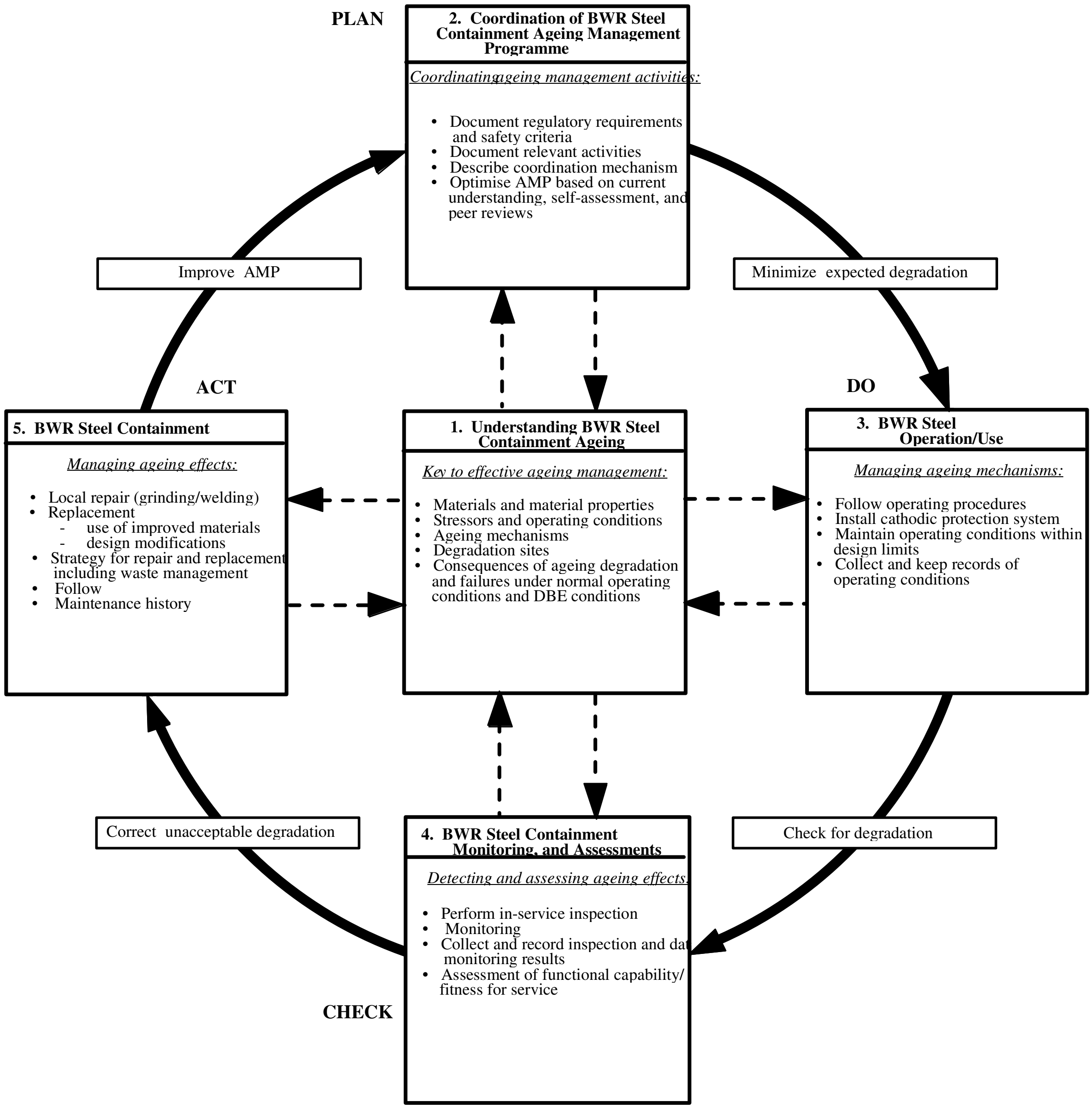

FIG. 8.1. Key elements of a metal components of BWR containment systems ageing management programme (AMP) and their interfaces.

maintenance methods and practices. Table 8.1 provides a listing of several potential sources of information important to understanding ageing of the BWR steel containment.

The baseline data consists of the performance requirements, the design basis (including codes, standards, regulatory requirements), design documents, the manufacturers data (including material data), and the commissioning data (including pre-service inspection data). The operating history includes such things as service loadings, environmental conditions, and various procedures. The maintenance history includes design modifications, replacement parts/components, inspection records, and assessment and timing of maintenance

TABLE 8.1. POTENTIAL SOURCES OF PLANT DATA

Type of data	Sources	Information
Baseline	Design documents	Design life Design philosophy Design codes/standards Material design properties Design stresses/strains Static design loadings Dynamic design loading Hazard design loading
Construction & Commissioning	Construction and Record Drawings	Substructure (Foundations) Superstructure Fabric and finishes Construction details Construction sequence
	Specifications	Construction standards Material sources Material properties Level of QA/inspection/testing Construction sequence Construction methods
	Designers/contractors	Design variations Specification variations Temporary works Temporary loads Construction history Levels of supervision
	Quality control records	Certified material test records Liner acceptance test results Penetration leakage-rate results
	Preoperational test records	Structural integrity test records Leakage test records
Operational History	Plant operating procedures	Service loadings Environmental conditions Fault loadings Safety procedures Maintenance procedures
Inspection & Surveillance	Inspection records	Visual inspection data Leakage-rate tests Ultrasonic thickness tests for liner
	Plant management/operatives	Plant history Maintenance history

performed. Retrievable up to date records of this information are needed for making comparison with applicable codes, standards, regulatory rules, and other external experience.

External experience consists of the operating and maintenance experience of (a) BWR steel containments of similar design, materials of construction, and fabrication; (b) BWR steel containments with similar operating histories, even if the BWR steel containment designs are different; and (c) relevant research results. It should be noted that effective comparisons or correlation with external experience requires a detailed knowledge of the BWR steel containment design and operation. The present report is a source of such information. However, this information has to be kept up to date using feedback mechanisms provided, for example by owner groups. External experience can also be used when considering the most appropriate inspection method, maintenance procedure, and technology.

8.2. CO-ORDINATION OF THE AGEING MANAGEMENT PROGRAMME

Existing programmes relating to the management of BWR steel containment ageing include operations, surveillance and maintenance programmes as well as operating feedback, research and development and technical support programmes. Experience shows that ageing management effectiveness can be improved by co-ordinating relevant programmes and activities within an ageing management programme utilising the systematic ageing management process. Safety authorities increasingly require licensees to define and implement such ageing management programmes for selected systems, structures, and components important to safety. The co-ordination of a BWR steel containment ageing management programme includes the documentation of applicable regulatory requirements and safety criteria, and of relevant programmes and activities and their respective roles in the ageing management process as well as description of mechanisms used for programme co-ordination and continuous improvement. The continuous ageing management programme improvement or optimisation is based on current understanding of BWR steel containment ageing and on results of periodic self assessment and peer reviews.

8.3. OPERATION/USE OF BWR STEEL CONTAINMENT

Plant operation has a significant influence on the rate of degradation of NPP systems, structures, and components. Exposure to operating conditions (e.g. temperature, pressure, humidity, radiation, and aggressive chemicals) outside design limits could lead to accelerated and premature degradation. Since operating practices influence the containment operating conditions, NPP operations staff has an important role within the ageing management programme to minimise age-related degradation by maintaining operating conditions within prescribed (design) limits.

Operation of plant systems, and inspection and testing of the metal components of the containment system according to procedures, and record keeping of relevant operational data (e.g. environmental conditions, test conditions, and results) also are essential for an effective ageing management programme. In particular, it is prudent to attempt to control and monitor the operating environment of inaccessible parts of the containment pressure boundary (e.g. interface where the metal pressure boundary becomes embedded in concrete) where detection and repair of degradation would be difficult and costly).

8.4. INSPECTION, MONITORING, AND ASSESSMENT

The inspection and monitoring activities for metal components of BWR containments systems are designed to detect and characterise significant component degradation before safety margins are compromised. Together with an understanding of ageing degradation, the results of the inspections provide a basis for decisions regarding the type and timing of maintenance actions to correct detected ageing effects. Also, these results can impact decisions regarding changes in operating conditions and practices to control significant ageing mechanisms.

Current inspection and monitoring requirements and techniques for metallic components are described in Chapter 5. In general, the rigor and extent of the inspection increases as the containment or component develops problems. Normally, a visual inspection of accessible surfaces is conducted. Visual inspections may be supplemented by non-destructive and destructive tests in areas exhibiting distress.

It is extremely important to know the accuracy, sensitivity, reliability, and adequacy of the non-destructive methods used to identify and evaluate the particular type of suspected degradation. The performance of the inspection method(s) must be evaluated in order to rely on the results, particularly in cases where they are used as part of a fitness-for-service assessment. Inspection methods capable of detecting and quantifying expected degradation are therefore selected from those proven by relevant operating experience. Information on capabilities of non-destructive methods for inspection and monitoring of metal containment system component's ageing are presented in Chapter 5.

Systematic and effective record keeping is an important part of the inspection process. It is this data that underpins evaluation of the current condition as well as estimates of future performance. For visual inspections, permanent records are generally made of the condition of component at the time of survey, and may be used subsequently for trending behaviour (e.g. identifying active/inactive cracks, and monitoring crack growth or wall thinning). Items most often identified include cracking and the presence of staining or corrosion products for metal components, and blistering, flaking, peeling, cracking, and delaminations of coating materials. Records may consist of detailed drawings, photographs/videos, or a combination of these techniques. To avoid subjectivity, photographs recording the extent of degradation should, where possible, be backed up by quantitative measurements.

Quantitative data provided by other testing and monitoring techniques also should be recorded appropriately. Practical guidance on the implementation of an effective system for data collection and record keeping for the purpose of ageing management is given in Ref. [8.2].

8.5. MAINTENANCE, REPAIR, AND REPLACEMENT

Most maintenance and remedial work is implemented in response to an identified defect in the structure. Table 7.1 gives some examples of containment pressure boundary component degradation for BWR plants. Depending on the degree of degradation and the residual integrity (i.e., structural and leaktight) of the structure, the objective of a remedial measures programme might be any one, or a combination, of structural, protective, or cosmetic. Decisions on the type and timing of the maintenance actions are based on an assessment of the observed ageing effects, available decision criteria, an understanding of the applicable ageing mechanism(s), and the effectiveness of available maintenance technologies. Typical options that would be considered in response to unacceptable plant degradation are:

- Enhanced surveillance to trend progress of deterioration. This is the initial approach adopted as part of the evaluation process during the early stages of degradation.

- Maintenance/operational changes to prevent deterioration from getting worse (if safety margins are acceptable). This might include modified operating conditions (e.g. reducing reactor power, particularly in the shorter term while repairs are planned).

- Local repairs to restore parts of a structure to a satisfactory condition.

- Replace component.

Chapter 6 provides methods for prevention and mitigation of ageing effects for steel components of BWR containment systems and bellows.

REFERENCES TO CHAPTER 8

[8.1] INTERNATIONAL ATOMIC ENERGY AGENCY, "Implementation and Review of Nuclear Power Plant Ageing Management Programme," Safety Series No. 15, IAEA, Vienna, Austria (1999).

[8.2] INTERNATIONAL ATOMIC ENERGY AGENCY, "Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing," Safety Series No. 50-P-3, IAEA, Vienna, Austria (1991).

Chapter 9

SUMMARY AND CONCLUSIONS

9.1. SUMMARY

Experience shows that the exposure of metal components of BWR containments to BWR service conditions can cause their ageing degradation and impair their safety functions. Therefore, proper management of ageing of the metal components of BWR steel containment systems, including the bellows, is considered important to preserve the overall safety of the plant. The primary ageing mechanisms that may potentially impact the structural capacity, leaktight integrity, or service life are corrosion of metal components and stress corrosion cracking of bellows. Other potential degradation mechanisms include fatigue and mechanical wear. Areas of concern are where surfaces are inaccessible for inspection (e.g. areas adjacent to floors, where the containment vessel is embedded in concrete, and locations adjacent to equipment or other structures).

Visual inspection and maintenance of the accessible components, and pressure and leakage-rate testing can be realised periodically. Air locks, bolted covers or penetrations, and isolation valves should be inspected and leak tested. For penetration components, integrated leakage-rate tests, pressure tests, and visual examinations conducted at longer inspection intervals is reasonable. In-service inspection guidelines are furnished in documents such as provided in References [9.1-9.4].

Special care must be established to maintain the integrity of the wall of the suppression pool and other components that are subjected to environments that may be conducive to corrosion. The most effective method to manage ageing of inaccessible portions of the containment pressure boundary is proper maintenance. It is important to avoid leakage of fluids that may accumulate in the inaccessible areas. Most effective are suitable measures realised during the design and construction phases. An evaluation of the as-built documentation can help to analyse the corrosion risk. After start-up care must be taken to avoid, detect, and repair all fluid leakage in and around the containment. Also, waterstops and seals between concrete and steel, and coatings on floors and walls should be maintained as they provide effective measures to help prevent corrosion.

An ageing management programme (AMP) should be developed having the objective of timely detection and mitigation of any degradation that could impact NPP safety functions. Its main characteristic is a systematic, comprehensive, and integrated approach aimed at ensuring the most effective and efficient management of ageing. A comprehensive understanding of metallic components of BWR containment systems, their ageing degradation, and the effects of degradation on the containment's ability to perform its design functions is a fundamental element in the AMP. This understanding is derived from knowledge of the design basis, including applicable codes and regulatory requirements; the design and fabrication, including material properties and specified service conditions; the operation and maintenance history, including commissioning and surveillance; the inspection results; and generic operating experience and research results. Figure 8.1 provides the key elements of an ageing management programme and Chapter 8 provides guidance on development of such a programme.

9.2. CONCLUSIONS

Based on information contained throughout this report, and experience of contributors to the report, several conclusions can be derived.

- The performance of metal components of BWR containment systems in nuclear power plants has been good. However, as these structures age, incidents of ageing degradation are likely to increase the potential threat to their functionality and durability. The most commonly observed form of degradation has been corrosion. Degradation factors of primary concern would be corrosion of metal components and stress corrosion cracking of bellows.

- Techniques for detecting the effects of metal component ageing (i.e. inspection and performance monitoring) are sufficiently developed to provide vital input for evaluating the structural condition of BWR steel containment components. Periodic application of these techniques provides data that can be used to trend performance and form the basis of other ageing management actions. One area of concern where these techniques require additional development is related to locations of the containment pressure boundary that are inaccessible, such as portions that are embedded in concrete.

- Methods for conduct of condition assessments of metal components such as steel containment vessels are fairly well established and generally start with a visual examination of the structure's surfaces. Application of supplemental examinations and testing have primarily been associated with assessments of degradation occurrence or suspected occurrence.

- Maintenance and repair techniques for metal components are well established and when properly selected and applied are effective. Effective implementation of a repair strategy requires knowledge of the degradation mechanisms, the environment, proper preconditioning of the structure to be repaired, correct choice of repair technique and material, and quality workmanship.

- Many utilities worldwide have responded to the potential for age-related degradation through implementation of ageing management programmes. These programmes in large measure have been in response to requirements contained in codes and standards and have, generally, adopted an approach in which the effects of ageing are managed. A characteristic of the most effective AMPs is the clear definition and documentation of a systematic programme of activities aimed at understanding, effectively monitoring, and mitigating ageing effects. A particular feature is the routine trending of surveillance and test data to estimate future performance. This has value in ensuring continued containment reliability, and hence plant availability.

- A framework for ageing management of metal components of BWR containment systems has been defined (Fig. 8.1). The proposed approach is consistent with existing IAEA guidelines. An understanding of the issues involved is the basis for an effective AMP. The AMP is broken down into a sequential series of tasks: (1) *Co-ordination of ageing management* to integrate ageing management activities, in particular the inspection and monitoring requirements and appropriate acceptance criteria (drawing on and integrating existing plant practice); (2) *Operation/use* of plant within design limits to minimise age-related degradation, in particular that which is error-induced; (3) *Inspection, Monitoring and Assessment*s to detect and characterise significant component degradation before fitness-for-purpose is compromised; and (4) *Maintenance* to correct any unacceptable degradation (i.e. manage ageing effects).

REFERENCES TO CHAPTER 9

[9.1] INTERNATIONAL ATOMIC ENERGY AGENCY, “In-Service Inspection for Nuclear Power Plants, Safety Guide No. 50-SG-02, Vienna, Austria, 1980.

[9.2] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, "Requirements for Class MC and Metallic Liners of Class CC Components of Light Water Cooled Plants," Section XI, Subsection IWE in *American Society of Mechanical Engineers Boiler and Pressure Vessel Code*, 1998 edition with Addenda, New York, New York, 1998.

[9.3] RSK, “RSK-Leitlinie für Druckwasserreaktoren,” LL-DWR 10.81, 1981.

[9.4] KTA, “Reaktor Sicherheitsbehälter aus Stahl, Teil: Wiederkehrende Prüfungen”, KTA 3401.4, 6.91.

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