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Understanding and Managing Ageing of Material in Spent Fuel Storage Facilities

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UNDERSTANDING AND MANAGING AGEING OF MATERIAL IN SPENT FUEL STORAGE FACILITIES

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FOREWORD

At the end of 2004 there were over 400 operational nuclear power plants in IAEA Member States. There were also 274 research or test reactors in operation, as well as ten under construction and six planned. In addition, there are 97 independent power reactor storage facilities, wet and dry, that are not directly attached to a reactor building and 57 away from reactor spent fuel storage facilities at research reactors. The need to understand and manage ageing of systems, structures and components has emerged as a priority as the ages of these storage facilities increase, in some cases well beyond their originally expected lifetimes.

Several organizations, notably the IAEA and, at the national level, the United States Nuclear Regulatory Commission, have developed guidance regarding implementation of ageing management in nuclear power plants. While the ageing management approaches would be similar for research and test reactors, their systems are less complex and operating conditions are generally less aggressive.

The IAEA Coordinated Research Project (CRP) on Ageing of Materials in Fuel Storage Facilities drew from strategies developed for ageing management in nuclear power plants and recommended adaptation of these methods to smaller fuel storage facilities at research and test reactors. The participants in the CRP provided many valuable insights to age related phenomena that have occurred at storage facilities in their countries. They also suggested ageing management strategies that have been successfully applied in their respective facilities. The recommendations of the CRP provide valuable guidance to operators of smaller fuel storage facilities.

The members of the CRP who contributed to the drafting and review of this report are identified at the end of this report. In particular, the contributions of K. Simpson from the United Kingdom and A.B. Johnson, Jr., from the United States of America are acknowledged. The IAEA officer responsible for this report was I.G. Ritchie of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

This report stems from the Coordinated Research Project (CRP) on Ageing of Materials in Spent Fuel Storage Facilities. It includes a section on the highlights of the contributions of the participants in the CRP and sections on the status of the understanding of ageing of selected types of material and on managing ageing. Managing ageing is of key importance in many countries for the owners and operators of many facilities, including power reactors. There is a large measure of agreement on the general approach, which is summarized in this report. This report also includes a brief section on specific approaches for fuel storage facilities.

The value, in principle, of collating, summarizing and disseminating information on how individual fuel storage facility owners or operators specify operating conditions, monitor actual conditions and judge the need for maintenance and remediation activities in their facilities was considered by the participants in the CRP. A questionnaire was drawn up, given a limited trial and subsequently slightly modified; the latest version of the questionnaire is included in this report.

Thus while this report is in part a reflection of the CRP's activities and includes some specific recommendations, the content has been broadened to try to appeal to those who may be in the early stages of setting up ageing management programmes (AMPs), either for new or older fuel storage facilities.

The CRP participants were aware of closely related IAEA activities, particularly in relation to the storage of civil reactor fuel. Accordingly, there is discrimination here on the ageing of fuel material from research reactors rather than civil reactors. However, there is no such selectivity with regard to the material in fuel storage facilities.

2. UNDERSTANDING AGEING CONCEPTS AND TERMINOLOGIES

The IAEA Spent Fuel Storage Glossary [1] defines extended storage as "storage of fuel units beyond the period of time originally planned". In numerous countries, "beyond the period of time originally planned" evolved to be the only interim option. It is important that the effects of extended operation be anticipated and monitored to ensure that the excellent safety record of fuel storage technology is maintained. At several fuel storage facilities, expensive recovery operations have been the consequence of failure to effectively manage degradation of fuel and/or facility components.

The generally accepted parlance in nuclear technology for the effects of extended operation is 'ageing', defined as a "general process where characteristics of a system, structure or component (SSC) gradually change with time or use" [2]. The effects of ageing may be beneficial, as in the curing of concrete, or they may be deleterious, as in the corrosion of metal or degradation of insulating material due to the effects of radiation. The same concepts and terminologies that apply to nuclear reactor ageing are relevant to ageing in fuel storage facilities.

2.1. AGEING TERMINOLOGIES

The sources of ageing terminologies are provided in Refs [2, 3]. Selected terms are defined below.

Ageing. General process in which the characteristics of a system, structure or component (SSC) gradually change with time or use.
Natural ageing. Ageing of an SSC that occurs under pre-service and service conditions, including error induced conditions.
Error induced conditions. Adverse pre-service or service conditions produced by design, fabrication, installation, operation or maintenance errors.
Ageing degradation. Ageing effects that could impair the ability of an SSC to function within acceptance criteria. Examples are reduction in diameter from the wear of a rotating shaft, loss in material strength from fatigue or thermal ageing, swelling of potting compounds and loss of dielectric strength or cracking of insulation.
Ageing effects. Net changes in the characteristics of an SSC that occur with time or use and are due to ageing mechanisms. For examples of negative effects, see ageing degradation; examples of positive effects are increase in concrete strength from curing and reduction of vibration from wear-in of rotating machinery.
Ageing mechanism. A specific process that gradually changes the characteristics of an SSC with time or use. Examples are curing, wear, fatigue, creep, erosion, microbial fouling, corrosion, embrittlement and chemical or biological reactions.
Artificial ageing. Simulation of natural ageing effects on SSCs by application of stressors representing plant pre-service and service conditions, but perhaps different in intensity, duration and manner of application.

Figure 1 shows the ageing sequence, the key elements that lead to age related degradation or failure. The sequence illustrated is for the case of through-wall cracks that developed in a fuel storage pool (FSP) stainless steel pipe during service [4]:

(a) Pre-service conditions: sensitization of welds due to lack of a proper weld procedure and possible overstressing when the ends of matching pipe ends are forced together.
(b) Operating conditions: borated water at about 30°C; possible contaminants.
(c) Ageing stressors: stress in weld area and abnormal metallurgy (sensitization) in heat affected zones.
(d) Ageing mechanism: intergranular stress corrosion cracking (IGSCC).
(e) Ageing effect: through-wall crack development (nine locations).
(f) Ageing degradation: degradation of pipe wall in heat affected zones.
(g) Degraded condition or failure: degraded pipe that had to be replaced.

Figure 1 provides spent nuclear fuel storage facility (SNFSF) staff with a concise insight into the elements that contribute to degradation and/or failure of components in their facilities. The sequence emphasizes that ageing processes may be influenced by histories that precede the onset of service of an SCC.

Appendix II provides a concise compilation of terms on the following topics [2, 3]:

(i) Degradation:
— Causes of degradation;
— Degradation/ageing.

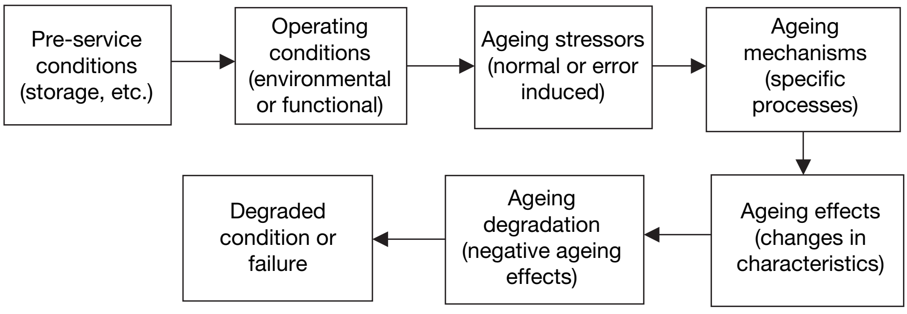

FIG. 1. Ageing sequence [2].

(ii) Life cycle:
  — Life;
  — Failure.
(iii) Ageing management:
  — Maintenance;
  — Condition assessment.

The groupings of related terms lead to an understanding of the various elements of ageing. Definitions of each term are found in Refs [2, 3]. Figure 2 provides a concise view of the elements of maintenance, refurbishment and repair, which are key factors in effective ageing management.

While the relatively benign conditions in most SNFSFs place maintenance on a less stringent basis than that of safety related SSCs in high temperature service, the elements of maintenance deserve attention at the level demanded by the potential vulnerabilities in SNFSF SSCs. A facility that can show evidence of a systematic approach to ageing management will have a sound basis on which to address extension of its operating period, if that becomes necessary.

As previously indicated, a major key to the effective control of ageing degradation in any facility is to have a staff that is attuned to the important concepts that comprise the understanding and managing of ageing. Operators of SNFSFs can use Appendix II and Figs 1 and 2 to facilitate examination of their programmes to understand and effectively manage ageing degradation in their facilities.

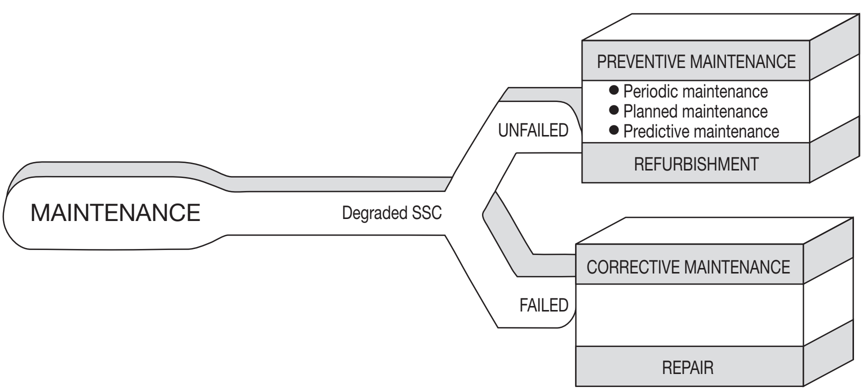

FIG. 2. Terms related to maintenance [2].

When it is necessary for SNFSF staff to design an AMP specific to their facility, guidance to develop the programme is provided in the GALL (Generic Aging Lessons Learned) report [5], which is based on ten programme attributes that have been identified by staff of the United States Nuclear Regulatory Commission (NRC). The ten AMP attributes should be applied with judgements regarding relevance to how the AMP is to be implemented in the specific SNFSF; for example, if there is no relevant operating experience for the SSC in the SNFSF, operating experience from similar facilities may be useful to assess potential vulnerabilities.

3. STATUS OF UNDERSTANDING AGEING OF MATERIAL IN FUEL STORAGE FACILITIES

There is now more than 50 years of experience in operating wet storage facilities and more than 30 years of experience in operating dry storage facilities. There have been systematic assessments of the durability of fuel and material in the environments often encountered in wet or dry SNFSFs. Generally there is a good understanding of the susceptibility of an individual material to degrade in different environments. Major sources of data that apply to this understanding will be referenced in this section, which can be regarded as providing an overview on a material by material basis.

The behaviour of different types of material in specific storage facilities naturally depends on the conditions encountered there. This brings monitoring and inspection into the process of ageing management, which are covered in Section 4.

3.1. ALUMINIUM

Aluminium is prominently used in power, research and test reactor fuel storage technology. Aluminium clad fuels are stored predominantly in FSPs, but some aluminium clad fuels are in dry storage [6–8]. Also, there are major initiatives to move inventories of aluminium clad fuel to dry storage [9, 10].

Aluminium alloys are used in nuclear fuel and fuel storage facilities in:

(a) Fuel cladding;
(b) Fuel matrices;
(c) Fuel structural components;

(d) Fuel storage racks;
(e) Fuel storage canisters;
(f) Fuel handling devices.

Aluminium components thus have varied histories or pre-treatments before entering the fuel storage environment, and this pre-storage history can affect behaviour.

3.1.1. Reactions with water

Aluminium alloys react with water to form hydrated oxides. The oxide form depends on temperature. Below the range of 60–70°C, bayerite (beta $Al_2O_3{\cdot}3H_2O$) forms over an initial amorphous film; above this range, boehmite (alpha $Al_2O_3{\cdot}H_2O$) forms over the initial amorphous film.

Aluminium based material has had a wide range of durabilities in aqueous environments, related principally to water quality control [11]. In aggressive waters, aluminium alloys are subject to corrosion attack, principally by pitting [11, 12].

3.1.1.1. Significance of reactor formed oxide films

Oxide films that form during exposure to a water coolant in reactor service seem to protect aluminium cladding, even in aggressive FSP waters [11, 13]. However, if the oxides are damaged during fuel handling operations, pitting of the exposed aluminium can be expected, particularly in aggressive waters [11, 13].

3.1.1.2. Significance of low temperature oxide films

Cases have been described involving the installation of unfilmed aluminium alloy racks in FSPs at ambient temperatures, resulting in copious hydrogen evolution as an oxide film formed. After the film reached a stable thickness, hydrogen evolution ceased. In FSPs with high purity water, storage racks, canisters, tools, etc., have remained in service for periods beyond 40 years with minimal evidence of degradation. In contrast, aluminium based material has undergone severe pitting in aggressive water conditions over periods of a few months [12] to three years [11]. In aggressive water chemistries the film formed in low temperature water has not proved to be substantially protective [13].

3.1.1.3. *Summary of aluminium alloy corrosion factors in fuel storage pools*

A large database on the corrosion of aluminium clad material has been generated from the CRP on Aluminium Alloy Corrosion [14] and by application of corrosion surveillance programmes [11, 12–14]. An evaluation of these data indicates that the most important factors contributing to aggressive corrosion of aluminium are:

(a) High water conductivity (e.g. 100–200 μS/cm).
(b) High concentrations of aggressive impurity ion concentrations (e.g. Cl^-).
(c) Deposition of cathodic particles on aluminium (Fe, etc.). Sediments on the top surfaces of aluminium alloy coupons caused pitting. No pits have been observed on the bottom surfaces of these coupons.
(d) Sludge containing Fe, Cl^- and other ions in concentrations more than ten times the concentrations in the water.
(e) Galvanic couples between dissimilar metals (stainless steel–aluminium, aluminium–uranium, etc.).
(f) Crevices: in non-ideal water chemistry, chloride ion concentrations can become elevated in crevices [15] and support a local pH that is less than that in the bulk water.
(g) Damage to protective oxides.
(h) Stagnant water.

Also note that radiation may influence aluminium alloy corrosion in some circumstances. Aluminium alloy components that have resided adjacent to irradiated fuel have been inspected by metallography; however, the evidence suggests that corrosion is not substantially accelerated [16].

The above factors, while operating both independently and synergistically, influence corrosion of aluminium based material.

Aggressive water chemistries may be long lasting or they can occur inadvertently in FSPs and be corrected after a short period. Inadvertent creation of adverse water chemistry can result from the addition of certain biocides to the FSP water to mitigate biological growth. To cite an example, addition of a biocide caused elevated corrosion rates on aluminium alloys, carbon steels and copper alloys that were exposed to FSP water [11, 13]. Coupons of the FSP material were used to evaluate the effects of the chemical excursion. Corrosion effects peaked after about nine months and then sharply diminished, indicated by coupons added periodically to the FSP waters. By two years after the onset, aluminium alloy corrosion rates had diminished to pre-excursion values and continued at low rates for the next 13 years. However, aluminium alloy canisters that were in the pool during the entire period of the excursion developed copious pits that appeared to continue to propagate, even after good water chemistry was restored, until the 2.3 mm walls were fully penetrated. Candidate methods for the control of biological species in FSPs are summarized in Ref. [11].

The single most important key to preventing corrosion is establishing and maintaining satisfactory water chemistry control. Water conductivities below about 5 μS/cm generally ensure that aggressive impurity ions such as chloride are in the parts per billion range. When chemistry is maintained in this regime, corrosion of aluminium alloys is minimized [11, 14]. An FSP storing aluminium clad fuel for 40 years had aluminium alloy storage racks that were examined after 34 years in service. Significant corrosion was not observed at aluminium–stainless steel contacts [17]; however, pits with depths of up to 2 mm were observed in the rack structural tubes.

Uniform corrosion rates for aluminium alloys in FSP waters are generally low, even in more aggressive waters. In low conductivity waters, rates of 2.5×10^{-4} mm/a have been measured [11, 13]. Pitting rates are reported to be low in low conductivity waters (<0.001 mm/a in the absence of particulates such as iron). In high conductivity waters (e.g. 170–630 μS/cm), pitting rates were not precisely defined, but appeared to be in the range of 0.4–8 mm/a.

Good water chemistry alone does not always guarantee that corrosion will be prevented, as seen by the extensive testing conducted in Argentine storage pools, where iron oxide particles deposited from the water caused pitting even in high purity water [14]. This has been seen in other FSPs. Pitting mechanisms can involve both galvanic and/or oxygen depletion cells.

3.1.2. Drying of wet stored aluminium clad fuel

Investigations have been conducted to identify effective drying procedures for wet stored aluminium clad fuel before transfer to dry storage [9, 10]. In particular this relates to the inventory of residual bound and free water that can pass into dry storage. The United States of America has sought to develop acceptance criteria for aluminium clad fuel in dry storage [9, 18].

3.1.3. Reaction with air

Aluminium is resistant to oxidation in dry or moist air; for example, aluminium clad fuel was used in the two air cooled reactors at Windscale in the United Kingdom [19]. However, although the temperatures encountered in the dry storage of aluminium clad fuel will be modest, there is a need to consider whether radiolysis of stagnant moist air atmospheres will generate acidic species.

3.2. URANIUM METAL

Uranium metal, often natural or with only slight enrichment, was used in many early large reactors. The choice of cladding material varied, examples being zirconium, aluminium and magnesium. Much of the irradiated uranium fuel was, or is, reprocessed to extract plutonium for defence purposes or for supply to fast breeder reactors. However, residual amounts of unreprocessed fuel remain in wet and dry storage.

The move from the use of high enriched uranium (HEU) to the use of low enriched uranium (LEU) in research and test reactors has stimulated interest in alloys of uranium containing a few per cent of stabilizing components such as molybdenum (see Section 3.2.7). The ageing of uranium–aluminium alloy fuels is covered in Section 3.3.

Unirradiated uranium metal is held in dry storage. Fissile ^{233}U, derived from thorium, can be categorized as unirradiated uranium.

The effect of irradiation on the chemical degradation of uranium is not large, and comment is made as needed in the following sections. These sections address the reactions of uranium metal in a range of environments relevant to wet and dry storage.

3.2.1. Reaction with water

The reaction of uranium metal with water has been the subject of many investigations. The main reaction relevant to ageing that degrades metal fuel is:

$$U + 2H_2O \text{ (liquid)} = UO_2 + H_2 \tag{1}$$

The corrosion product is generally not adherent, and the reaction rate settles to a constant value at a fixed temperature. The pH of the water does not have much effect on the corrosion rate in the pH range pH4–12 [20].

The degree of oxygenation of the water is considered significant. The presence of dissolved oxygen retards the uranium corrosion rate in water (see also reaction with moist air, Section 3.2.5) [21]. It is thought that the oxygen is reduced in preference to water to balance the electrochemical (anodic) oxidation of uranium. If dissolved oxygen is not replenished, it may become exhausted, so that the corrosion rate of uranium in water may move through a transition [22, 23].

Radiolysis of water by gamma radiation has been reported [20] not to affect the reaction rate of uranium with water. This is supported by (a) small samples seeming to represent the corrosion of fuel elements and (b) unirradiated uranium corroding very much like irradiated uranium when surface area effects are accounted for (see below).

There is still much debate over the mechanism and quantity of uranium hydride formed when uranium reacts under water. Hydride is considered by some [22] to form adjacent to the uranium metal, lying underneath the thin layer of adherent oxide. It may be that this hydride is formed in the early stages of the reaction and then remains in situ without further augmentation. Uranium hydride is not stable in water, although the reaction may be slow or may become slow. The slow reaction may be due to what some call passivation of the hydride surface by oxide formation before all the hydride has reacted. There is a consensus that operational problems do not result from the amount of hydride formed when uranium corrodes entirely under water and water has continued access to the corrosion product surface.

The purity of the uranium has been observed to affect the corrosion rate of unirradiated uranium in water; for example, aluminium additions in the 600 ppm region gave corrosion rates a factor of five higher at 100°C than when around 1100 ppm aluminium was added [22]. Low levels of iron are inferred not to have an effect, based on the agreement of results of different studies.

The effect of irradiation on the corrosion rate of uranium has probably received most attention for fuel from the UK's Magnox reactors. Depending principally on the irradiation temperature and local alloy composition, regions of porosity can develop in the uranium bar due to the accumulation of gaseous fission products [24]. If water contacts such swollen uranium, the corrosion rate is more rapid than with unswollen uranium if rates are still expressed as a rate per unit geometric surface area [22]. This increase in rate has been attributed to an increase in the surface area accessible to water. An increase of a factor of two in rate is generally included to allow for the effect of irradiation (i.e. for fuel classed as unswollen or with low swelling).

The release of fission products follows the progress of uranium oxidation: there is no leaching of fission products from unreacted uranium metal in water. The release is close to 100% for the fission product caesium in uranium that has corroded [22], and caesium activity in the water can be a convenient means of tracking uranium corrosion. The amount of strontium dissolved in the water is only a fraction [22, 23] of that originally present in the uranium that has corroded. Strontium is known to be absorbed strongly in solids, and thus strontium release to water may be affected, for example, by the presence of cladding corrosion products as well as by the uranium corrosion products. The actinides and rare earths remain with the corrosion product particulate.

The oxide produced when uranium corrodes in water is considered to be UO_{2+x}. The extent of hyperstoichiometry may depend on the dissolved oxygen level. The particulate is fine [22]. As already remarked, the corrosion product is not adherent and may oxidize further in a conducive environment.

3.2.2. Reaction with dry air

The reaction of uranium with dry air has been well studied. The ageing of material in long term storage concerns temperatures well away from those under which rapid uranium oxidation occurs, and so ignition is not considered here. This also means that the reaction of uranium with nitrogen can be excluded, since this becomes significant only at elevated temperatures (c. 400°C and above). The very early stages of oxidation of clean metal in oxygen and dry air (no water vapour) have been studied spectroscopically, and some references are given in Ref. [25]: the behaviour is not relevant to the times involved in ageing of material and is not considered further here.

The oxide that is thermodynamically stable in contact with uranium metal is not the oxide that is thermodynamically stable in contact with air. Thus the overall oxidation path may have several stages, as the first formed oxide may be oxidized further. The oxidation rate is generally measured via weight gain as a function of time per unit surface area of metal surface. The metal surfaces of cut samples are often prepared initially, for example, by polishing, so that the surface area is known. To convert the weight gain to metal loss, knowledge of the stoichiometry of the oxide formed is needed.

The kinetics of oxidation of uranium metal in dry air initially shows parabolic behaviour; that is, the rate of reaction decreases with time and the oxide, having fluorite structure and composition UO_{2+x}, can be considered protective. With protective oxidation, the effect of oxygen partial pressure on the rate is very small, and for practical (fuel storage) purposes is within the scatter of the published data [25, 26]. When a certain thickness of oxide is reached (of the order of 1 μm), spallation occurs. This leads to an oxidation rate that appears to be effectively constant with time, although it may be described as a sequence of parabolic episodes. Kinetic equations combining both stages of reaction have been fitted to the database for oxidation of uranium (for an oxide with a fluorite structure). The early stages of reaction are detected at lower temperatures, for example 250°C and below.

At higher temperatures, the duration of the protective phase may be so short and oxidation of UO_2 to U_3O_8 so rapid that the rate of uranium oxidation in dry air can be considered constant and the product close to U_3O_8.

The effects of impurities and minor alloying additions on the oxidation rate in dry air are fairly small, and often within the scatter to be expected in data from different laboratories; thus reviews [26, 27] have assessed the broad range of data and have produced representative or best estimate results.

The effect of irradiation on the oxidation rate of uranium metal in dry air does not seem to have received much attention [28], since moist air atmospheres are more frequently encountered in practice or are likely to be present in fault conditions. The oxidation of uranium in moist air is covered in Section 3.2.5.

3.2.3. Reaction with hydrogen

Uranium reacts with hydrogen to give uranium hydride. The kinetics has been well studied for process applications, and a good summary of the kinetics and thermodynamics is given in Ref. [29]. The reaction rate with hydrogen passes through a maximum as the temperature increases, because the back reaction, dissociation of uranium hydride, becomes important. The optimum temperature for hydrogenation is often quoted as being in the range 200–250°C [29]. Uranium hydride is in equilibrium with 0.1 MPa of hydrogen at about 400°C.

The product of reaction of uranium metal with pure dry hydrogen is very reactive towards oxygen or air. With a plentiful supply of oxygen the heat generated may elevate the temperature so as to drive the hydride oxidation to completion (and passivation will not be observed).

A point to note in connection with long term storage is that reaction of uranium with hydrogen occurs faster than with oxygen or water vapour at temperatures below very roughly 250°C.

3.2.4. Reaction with water vapour in the absence of oxygen

There is a large body of literature on the reaction of water vapour with uranium metal, which dates back more than 50 years and is still being added to. The basic oxidation reaction consumes water vapour and produces hydrogen. Hydrogen can also react with uranium metal to produce uranium hydride (see Section 3.2.3). With uranium immersed in liquid water, the water can always be considered to be in excess; however, with water vapour the composition of the atmosphere may change if hydrogen is allowed to accumulate.

Hydrogen not only reacts with uranium metal but can readily diffuse into it. The way in which water vapour interacts with a clean uranium metal surface has been studied by a variety of techniques [25, 30–32]. One point of interest is the behaviour of hydrogen atoms in adsorbed water vapour. The overall reaction sequence is:

$$U + 2H_2O\ (gas) = UO_2 + H_2 \qquad (2)$$

It can be postulated that there is no direct connection of hydrogen with uranium, the bonding being entirely with oxygen. It can also be postulated that a water vapour molecule adsorbs to give a bound hydroxyl group and an adsorbed hydrogen atom. The latter may be desorbed or may dissolve in the uranium metal. Fairly recent studies using thermal desorption mass spectrometry [32] are seen as supporting three modes of hydrogen liberation: from binding with oxygen in a hydroxyl group, from direct absorption of hydrogen and from within the metal. The temperature at which the last of these was observed depended on the thickness of the oxide.

Many laboratory studies that follow continuously the reaction kinetics of uranium metal with water vapour have used flowing systems in which hydrogen gas cannot accumulate (see, for example, Ref. [33] and references in Refs [25, 26]). Earlier studies, for example Ref. [20], in which the kinetics was followed by intermittent measurements in sealed systems, recognized a hydrogen deficit compared with the stoichiometry of Eq. (2). Evidence for some hydride formation below the main oxide layer (nearer the metal than the main oxide layer) is cited in Ref. [34].

The overall rate of uranium metal corrosion is determined by the reaction with water vapour. The reaction rate is found to be proportional to the square root of the water vapour pressure. Reviews of literature data, for example Refs [25, 26, 35], have provided representative or best estimate expressions for the rate of reaction per unit surface area of uranium.

The form of the oxidation product from reaction of water vapour with uranium metal is a fine powder below about 373 K [25]. However, around 423 K exfoliation occurs in sheets about 1 μm thick; the sheets are made up of many laminar flakes.

Under notionally dry storage conditions, where the supply of water vapour is constrained or becomes exhausted and reaction product hydrogen can accumulate in an oxygen free environment, there is incontrovertible evidence that reaction of uranium with hydrogen can occur even at ambient temperatures. The reaction of uranium with hydrogen is known to initiate at physical discontinuities in the surface microstructure of the uranium [36]. This leads to pitting of the uranium metal surface and the appearance of blisters or pimples on the surfaces of loosely clad uranium metal. There appears to be an induction period before the secondary reaction with hydrogen begins. If controlled oxidation of highly reactive uranium hydride can occur, for example via moisture or a low rate of oxygen supply, there is evidence that the uranium hydride becomes surface oxidized or passivated. This surface oxidized form does not react quickly and visibly with air at room temperature, as shown by measured oxidation rates [37]. Attention may need to be given to the possibility of slowly developing thermal transients, which may, depending on the circumstances, lead to eventual ignition. Irradiated, unswollen uranium reacts with oxygen free water vapour much as unirradiated uranium [28].

3.2.5. Reaction with moist air

Moist air presents uranium metal with alternative reactants, and a product of one of them that can itself react with uranium. This system has been extensively studied and is relevant to uranium metal fuel storage under intended or fault conditions.

The reaction with water vapour producing hydrogen is inhibited by the presence of oxygen down to the sub-100 ppm level [25]. In air with water vapour pressures of a few kilopascals, such as are encountered outdoors or in ambient conditions, the kinetics tends to resemble that in dry air [26, 38]. Equations to represent the range of kinetic behaviour as a function of temperature and water vapour pressure are derived in, for example, Refs [26, 27].

The effect of irradiation on the kinetics of uranium oxidation in moist air has been studied [39]. Swelling of the uranium, which can locally increase the surface area to above the geometric value, is considered to be the main reason for an increase in reaction rate compared with unirradiated uranium.

The effect of impurities or minor alloying additions on the reaction in moist air or moist argon–oxygen mixtures is not marked within the scatter of the published data. Fairly recent oxidation rate data collected on Hanford N reactor fuel [28] in moist air generally relate well with older work.

The form of the oxidation product in moist air has been observed as interlocking sheets [25] or fragile flakes [40] that detach from the parent metal.

3.2.6. Other reactions relevant to ageing

Most of the work referred to above was carried out in hot cells or in the laboratory on prepared uranium samples in controlled atmospheres or known water chemistries. In the context of ageing during storage, the uranium may be in contact with other metals, may be precorroded or preoxidized before being passed to storage, or may encounter radiolysed atmospheres or various water chemistries.

As mentioned in Sections 3.2.3 and 3.2.4, uranium can react with hydrogen to form uranium hydride. In a storage environment the hydrogen can come from reaction of uranium with water or from reaction of a fuel cladding such as Magnox with water (i.e. there may be a supplementary source of hydrogen).

Although the formation of uranium hydride during storage is undesirable, it can be argued that uranium hydride is not hazardous in practice if the hydride is undisturbed. Incidents have occurred involving rapid oxidation when fuel had been moved, corrosion products were dislodged or containers of uranium were opened up to air. New storage facilities normally seek to prevent or avoid conditions conducive to uranium hydride formation. With older storage facilities it is useful to consider whether hydride might be present and to assess the potential significance for fuel retrieval and repackaging operations.

3.2.7. Effect of alloying constituents

There has long been an interest in improving the mechanical properties of uranium by addition of other metals at the several per cent level. From a perspective of neutron economy, such large additions are not desirable with natural uranium. However, the international drive to run research reactors with LEU rather than HEU has renewed interest in, for example, alloys with a composition around U + 10 wt% Mo.

Uranium–silicon intermetallic compounds, usually U_3Si (or U_3Si_2 in material testing reactors (MTRs)) as a solid dispersion in aluminium, have allowed efficient fuelling with LEU in research reactors such as the NRU in Canada.

Additions such as zirconium, niobium, molybdenum and silicon to uranium increase the corrosion resistance of the unirradiated material [41]. Irradiation effects are complicated by the very high burnup that LEU alloys may achieve (in comparison with natural uranium).

Information on the oxidation behaviour in air suggests that the oxidation rate (measured as the rate of weight gain) is slower for the alloys than for unalloyed uranium [42]. Uranium oxidation products have been identified for a U + 10 wt% Mo alloy [42], but the behaviour of the molybdenum was not reported.

Where there is a dispersion in aluminium, for example of U_3Si, the properties of the aluminium often determine the overall fuel behaviour. However, the rate of oxidation of U_3Si in air is reported to be lower than that of uranium metal [43].

3.3. URANIUM–ALUMINIUM ALLOY FUELS

Uranium–aluminium alloy fuels have been commonly used in research reactors in a variety of fuel element designs. The U:Al ratio can be such (~40 wt%) as to allow enriched uranium to be used (note that low enriched is <20 wt% ^{235}U; high enriched is >20 wt% ^{235}U). However, such compositions are reported [44] to lead to brittle alloys that are difficult to fabricate. The use of HEU in a lower U:Al ratio, for example 10–20 wt% U, is more common. Thermodynamically, the overall composition would lead to two phases, UAl_4 and Al, but with an increasing amount of aluminium as the U:Al ratio goes down. However, the method of fuel manufacture allows different UAl intermetallic compounds, for example UAl_2 and UAl_3, as well as UAl_4, to be incorporated into an aluminium matrix. Irradiation temperatures are often low enough (around 200°C) for the resulting composites not to reach thermodynamic equilibrium.

When HEU is used, the fuel elements can be taken to a very high burnup, for example 50 at.%, and thus the loss of uranium and the concentration of fission products at the end of their service life, when the fuel is put into storage, are not negligible in terms of the overall chemical composition.

3.3.1. Reactions with water and air

As indicated above, the irradiated fuel has an aluminium matrix, which dominates its corrosion and oxidation. The fuel is very much less reactive than uranium metal. The corrosion and oxidation of aluminium are discussed in Section 3.1. The studies on fuel that have been carried out tend to report relative behaviour. Caskey [9] has reviewed the corrosion of UAl_x–Al composites in water and air.

It is reported [45] that the corrosion rate of UAl_x–Al in water at 563 K (reactor conditions) is about twice that of 8001 Al. For pool conditions, a bounding corrosion rate for UAl fuel material of 5.1 μm/a has been used for assessment purposes, based on a testing programme [46]. Cermet type fuel made by powder blending and compaction may behave less favourably in contact with water. Spent dispersion fuel from the BR2 Belgian reactor was readily attacked and released through a small pinhole [47]. In contrast, spent alloy fuel from the Australian HIFAR reactor remained intact, even though the breached area was much larger [48]. Dispersion fuel particles are not fully bonded to the matrix during fabrication, unlike alloy fuel, which is metallurgically bonded to the matrix.

3.4. URANIUM OXIDE–ALUMINIUM DISPERSION FUELS

Dispersion fuels consisting of an aluminium matrix containing U_3O_8 particles or UO_2 particles have been made and used in certain research and test reactors; for example, the High Flux Isotope Reactor in the USA uses a U_3O_8–Al fuel with aluminium cladding. Neither oxide is stable in the presence of aluminium [44]. Reaction with aluminium is slow at typical irradiation temperatures.

3.4.1. Reactions with water and air

As with UAl alloy fuels, the corrosion and oxidation of uranium oxide–aluminium dispersion fuels in water and air are generally dominated by the behaviour of the aluminium [47], but data seem very limited.

3.5. URANIUM SILICIDE–ALUMINIUM DISPERSION FUELS

Uranium silicide, U_3Si, in an aluminium matrix is the current fuel for the Canadian NRU reactor. Other reactors (e.g. Osiris in France and the JMTR in Japan) use U_3Si_2. The uranium density in this fuel composite can be sufficiently high that reactors can operate with LEU. The recovery of residual enrichment, if desired, is difficult, and long term storage is normal. However, uranium silicide generally does not react with water and air under storage conditions. This can be seen as augmenting the protection afforded by the aluminium matrix, as discussed above.

3.6. STAINLESS STEEL

Stainless steels have important roles in the wet storage of spent nuclear fuel, prominently as pool liners, fuel storage racks, canisters, piping, heat exchangers and fuel handling equipment. Some stainless steel components have remained in service in FSPs for more than 40 years without evidence of significant age related degradation, even in aggressive environments [11]. Predominantly, stainless steels have an excellent history of satisfactory service in FSP environments. Occasionally the metallurgical condition is affected by component assembly, for example welding or brazing, with adverse effects on corrosion resistance [18].

A small fraction of light water reactor (LWR) power reactor fuel has austenitic stainless steel cladding. Advanced gas cooled reactor (AGR) fuel is clad with stainless steel. Several test/research reactor fuels have had stainless steel cladding [18], but the dominant cladding type is aluminium. Relatively little experience with dry storage of stainless steel clad fuels has been published, although work has been carried out in support of proposals for the dry storage of AGR fuel. A preliminary assessment of expectations for LWR stainless steel clad fuel durability in dry storage has been documented [49]. Irradiation damage is now recognized as a means of changing the metallurgical structure of stainless steels so that corrosion resistance in subsequent storage may be impaired. The details (temperature and dose) of the irradiation conditions affect how far the microstructural changes develop and whether they persist (or are annealed out). This can lead to differences along a fuel rod or along a fuel assembly. The microstructural changes can lead to very narrow chromium depleted zones along grain boundaries, but precipitation of chromium carbides, typically associated with classical (thermal) sensitization, does not occur.

3.6.1. Behaviour of stainless steels in fuel storage pool waters

Three main factors may affect the corrosion behaviour of stainless steel fuel cladding or FSP components:

(a) The metallurgical condition, particularly the known or estimated degree of sensitization, either from thermal or radiation sources, as mentioned above.
(b) The storage environment, including the purity level of the FSP water.
(c) The stress level, a condition that is generally difficult to quantify but may become apparent upon analysis as a factor in IGSCC or other cracking phenomena.

The three factors generally interact to determine whether a given stainless steel cladding or component will have impaired durability. Sensitized stainless steels have degraded in FSP waters that were deemed to have low impurity levels [11, 49]. Stainless steels considered to have low levels of sensitization have survived without substantial degradation in FSP waters that were highly contaminated [11].

Uniform corrosion rates of stainless steels in FSP waters are negligible in the context of several decades or even a century of service life. In water with 7.5 ppm chloride and incident radiation of 3 × 105 R/h, the aqueous corrosion rate is 0.3 μm/a [11, 50]. In a caustic environment (e.g. Magnox FSP water), the stainless steel aqueous corrosion rate is indicated to be <0.1 μm/a [50].

3.6.2. Behaviour of stainless steels in dry storage

Under dry conditions the oxidation rates of stainless steel components, whether sensitized or not, are very low. Corrosion may occur if the humidity is such as to allow liquid film formation on the metal surface, and the corrosion is often facilitated by the presence of salts. In moist air environments, radiation can produce acidic oxides of nitrogen, which can promote local corrosion. Storage in an inert gas atmosphere is the typical practice for the dry storage of stainless steel power reactor fuel.

Fuel assemblies irradiated in a US liquid metal test reactor (EBR II) were placed in dry wells at the Idaho National Engineering and Environmental Laboratory (INEEL). Serious degradation of the cladding occurred, abetted by sensitization of the cladding during service and by the moist conditions in the air cover gas that existed in the dry wells [11].

UK studies on irradiated AGR stainless steel fuel cladding indicated that IGSCC occurred in time frames of a month at 35–50°C when the sensitized cladding material was suspended in air in a hot cell. At relative humidity above about 15%, degradation of the stressed material was promoted. The investigators hypothesized that radiation induced formation of nitric acid in moist air was the key factor in the degradation of the corrosion-prone material [51]. Unsensitized stainless steels would be expected to resist degradation by nitric acid under the test conditions. In fact, nitric acid treatments are applied to passivate stainless steels.

The durability of stainless steels in dry storage conditions has been satisfactory in cases in which observations have been possible after several years of dry storage service [52, 53]; for example, the stainless steel basket and lid used with a CASTOR V/21 cask were inspected after 14 years in a pressurized water reactor (PWR) dry fuel storage demonstration [53]. There was no evidence of significant degradation of the stainless steel.

3.7. ZIRCONIUM ALLOYS

Zirconium alloys are the dominant type of fuel cladding material used in power reactors. A few test/research reactors have been powered with fuel with zirconium alloy cladding. Defence reactors in the USA have sometimes used zirconium alloys as fuel cladding and pressure tube material. In the Western world, the cladding alloys have traditionally been the zircaloys, although there have been recent initiatives to introduce advanced alloys. In reactors of Soviet design, the cladding material has been principally Zr–1Nb.

Occasionally, zircaloy clad fuel has suffered mechanical damage in the discharge route to wet storage. In the case of the Hanford N reactor fuel, which was routinely dropped on to a hard surface, the damage to elements allowed extensive aqueous corrosion of the uranium metal to occur [13]. This uranium corrosion further mechanically disrupted the zircaloy cladding.

3.7.1. Zirconium alloy corrosion in water

Over more than 40 years of experience with several million LWR rods, power reactor fuel with zirconium alloy cladding has had excellent durability in wet storage [11, 50]. The evidence includes examination of CANDU fuel rods after up to 27 years of wet storage in Canada, which showed no changes in the cladding [54]. US examinations [55] after 20 years in wet storage indicated no detectable changes in zircaloy PWR cladding characteristics compared with properties measured on rods from the same assembly soon after the fuel rods were first put into wet storage. Zircaloy cladding on unbreached Hanford N reactor elements has typically exhibited good durability in wet storage now approaching 30 years.

Destructive and non-destructive examinations of fuel rods, visual evidence and coupon studies [11, 13, 54–58] all support resistance to aqueous corrosion. There have been no reports of fission gas evolution, indicative of cladding failure in wet storage. Rod consolidation campaigns have been conducted without any indication of storage induced degradation. There is a sufficient database to indicate that wet storage of fuel with zirconium alloy cladding can be extended for at least several decades.

Zircaloy corrosion rates in wet storage are too low to be detected by standard weighing techniques. However, zircaloy forms very thin coloured interference oxides that permit visual indications of oxide formation. Zircaloy coupons were exposed in two Hanford FSPs for a period of three years [13]. In this period, the coupons retained their silver appearance, suggesting that oxide growth had not progressed into the first interference colour range (20 nm thickness) [11]. The rate is suggested to be <0.007 μm/a. Weight measurements on the same coupons indicated weight changes of <0.01 μm/a. Extrapolations of kinetic data from higher temperatures suggest a rate of <0.07 μm in 50 years [11]. While generally supportive of extended zirconium alloy durability, the extrapolation is not regarded to have quantitative credibility. However, an even lower estimate of 1×10^{-6} μm/a has been reported [50].

Measurements on Russian fuel rods with Zr–1Nb cladding are reported to indicate measurable corrosion rates in wet storage [50].

3.7.2. Reactions of zirconium clad fuel in dry storage

The resistance of zircaloy cladding to corrosion in water and in water vapour, the absence of a hydrated zirconium oxide and the low oxidation rate of zircaloy cladding in air mean that there are few chemical problems in drying zircaloy clad fuel, or indeed in storing unfailed fuel incompletely dried. For power reactor fuel the decay heat promotes drying when the fuel is removed from water. However, the high decay heat of power reactor fuel means that cladding temperatures in dry storage need to be limited by design. Acceptable cladding temperature limits tend to be based on mechanical properties such as creep, but cladding strength can be degraded by internal fission product attack. There are initiatives under way in the USA to analyse and possibly relax conservatism regarding acceptable cladding temperature limits. There is also a strong interest in extending the technical basis to license storage of power reactor fuel assemblies with burnups above 45 000 MW·d/MTU [59].

Power reactor fuel with zirconium alloy cladding has been placed into dry storage in approximately a dozen countries [50]. The technical basis for satisfactory dry storage of fuel clad with zirconium alloys includes hot cell tests on single rods, whole assembly tests, demonstrations using casks loaded with irradiated fuel assemblies and theoretical analysis [50, 52, 58, 60]. The principal method of monitoring cladding behaviour in the assembly and cask storage tests and demonstrations has been to periodically sample cover gases for evidence of ^{85}Kr, the presence of which offers an indication that cladding has been breached. The fact that only a few rod failures have been detected provides evidence that cladding integrity is satisfactory in dry storage [60].

The CASTOR V/21 dry storage demonstration referred to above involved visual, non-destructive and destructive examination of PWR zircaloy clad fuel rods after 14 years in dry storage [53]. There was no indication of degradation of the cladding or fuel and little or no fission gas release from fuel pellets during the period of dry storage. The cladding retained significant creep ductility after dry storage.

3.8. CARBON STEEL

Carbon steel is an iron based alloy with 0.05–1% carbon and small amounts of other elements, principally manganese (~1%). Carbon steel has minimal representation in power reactor FSPs but is used in piping, filter housings, heat exchangers and valve bodies in some FSPs. Some test reactors have structural steels and piping in FSPs. Painted carbon steel containers (skips) are used for fuel discharges at some power reactors (e.g. Magnox). Storage facilities at US federal FSPs utilize carbon steels in storage racks, piping, buckets and fuel handling equipment [11, 13]. Magnox fuel was stored in carbon steel canisters at the EUREX plant at Saluggia, Italy, for 19 years [61].

Carbon steel/cast iron material is used in dry storage technology, including storage cask bodies, components and overpacks.

3.8.1. Carbon steel corrosion in water

Waters in FSPs are generally saturated with oxygen from contact with the atmosphere. Oxygenated water promotes corrosion of carbon steels. A typical corrosion rate is 250 μm/a [11]. Iron corrosion is insensitive to pH in the range pH4–10. In Hanford FSPs, corrosion of carbon steel pipes and fuel storage racks over a period of approximately 30 years produced voluminous reddish brown corrosion products that contributed to a large sludge inventory.

Mild steel is often protected by galvanized (zinc based) or organic coatings. A coating has mitigated corrosion of a carbon steel liner in the Halden reactor FSP, which has been in service since 1959. However, in a Hanford FSP organic coatings on fuel storage racks degraded and exposed the carbon steel to corrosion, as mentioned above. The coatings involved three layers of epoxy paint, with a total thickness of 0.12 mm. However, despite producing copious reddish brown corrosion products on the rack surfaces exposed at areas of degraded paint, the rack structure mechanical integrity was confirmed to be satisfactory after 25 years of service. The coating degraded principally through insertions and withdrawals of storage canisters, possibly combined with radiation induced degradation.

Carbon steels exposed in FSPs have been subject to the following corrosion mechanisms:

(a) Uniform corrosion: a carbon steel bar on a fuel hanger corroded 0.25 cm after exposure to an aggressive water (up to 760 ppm chloride) for 24 years [11]. Zinc coated carbon steel corroded at the rate of 100 μm/a in 40 years [11]. Dosing of two Hanford FSPs with biocide caused corrosion excursions on carbon steel coupons and, by inference, contributed to corrosion of carbon steel FSP components [13].
(b) Pitting corrosion: large, shallow pits developed on zinc coated carbon steel over 40 years in an aggressive FSP water [11].
(c) Galvanic corrosion: carbon steel welded to stainless steel was subject to accelerated corrosion in the aggressive chemistry cited above [11].

Carbon steel corrosion products: the relatively complex array of iron corrosion products and their interactions are summarized in Ref. [62]. The initial corrosion product on steel in oxygenated water at 22°C was γ-FeOOH, lepidocrocite, which, after four weeks, reverted to Fe_3O_4, magnetite. Sludge resulting from corrosion of steel components in the Hanford K East FSP involved the species FeOOH, lepidocrocite form, Fe_2O_3, hematite, and Fe_3O_4, magnetite.

3.8.2. Carbon steel behaviour in dry storage

Carbon steel, nodular cast iron and forged steel applications in the dry storage of spent nuclear fuel include the following:

(a) Cask bodies (nodular cast iron, carbon steel, low alloy steel, forged steel);
(b) Cooling fins (SA 283 or SA 285 grade A);
(c) Basket assemblies (carbon steel with metallic coatings).

Degradation of exterior carbon steel surfaces exposed to the atmosphere is minimized by coatings that are subject to periodic inspection and repair as required. Carbon steel surfaces inside cask seals are often protected with a metal coating, generally zinc or aluminium. Otherwise, deterioration of the steel surfaces can be minimized by effective removal of reactive species, for example water and oxygen, in the drying process. The integrity of cask seals and/or welds is essential to preclude ingress of moisture or air.

3.9. CONCRETE

The term ‘concrete’ covers types of material with a very wide range of compositions. Just as steels are specified for particular conditions in service, so are concretes. However, steels are usually made in a fabrication facility and, after inspection or quality control procedures, are sent to the site. Concrete is often mixed on the site or fairly locally and then has to be poured. In this sense, concrete manufacture is somewhat analogous to on-site welding. The curing conditions for poured concrete are important if the set concrete mass is to have the desired product quality. A further point is that the curing of concrete extends into the period of service. Thus concretes with notionally the same composition specification can lead to components with different qualities after manufacture.

In the context of ageing of material in spent fuel storage facilities, general understanding of the ageing of concrete is probably less important than it is for metals or alloys, the behaviour of which can be better characterized, for example, in the laboratory (a brief survey of methods to assess concrete properties is given below). References given here are mainly to standards, design guides, assessments and compilations, rather than to the original literature, in view of the vast international attention to concrete specification, manufacture, delivery and performance in service.

3.9.1. General understanding of the ageing and degradation of concrete

In the most general terms, concrete is made by mixing water with powdered cement and inert or semi-inert material. Dry cement may consist of several compounds, including calcium silicates, calcium aluminates and calcium oxide. The reactions of the cement with water lead to a monolithic hard structure. The hydration reactions are exothermic, and the different components of the cement differ in the total amount of heat evolved and in the rate of heat generation [63]. The curing temperature affects the rate of heat evolution from the cement and also the extent to which the semi-inert components of the mixture undergo reaction.

Concrete may be put into service while it contains some free water [64]. This water may continue to participate in hydration reactions for many years [63], often gradually augmenting the properties, such as hardness, of the concrete. This may be considered as natural ageing or maturing.

Thermal environments in service can promote degradation. The freezing of free water can cause cracking of concrete. In-service exposure of concrete to higher temperatures some way above 100°C can cause evaporation of free water and possibly some loss of bound water. Conditions in service are often set to avoid such extremes, or in the case of low temperatures the concrete may be specified to accommodate sub-zero temperatures [65, 66]. Temperature gradients can cause movement of water within the concrete monolith.

Chemical environments can also result in changes to concrete [64]. Among these, and important for reinforced concrete, is carbonation. This is the incorporation of carbon dioxide from the air into the calcium hydroxide that is usually present in concrete. The associated reduction in alkalinity reduces the protection of mild steel against aqueous corrosion. The iron corrosion products can stress the concrete sufficiently to cause cracking. The depth of carbonation increases slowly with time. In practice, the local pH around reinforcing bars is normally preserved by ensuring that the bars are buried or given a concrete cap of adequate depth.

The components of concrete are not strictly insoluble. However, the volume of liquid with which concrete must be contacted needs to be very large to produce a significant effect. Thus in waste repositories that are backfilled with grout and sited below the water table, the solution of the grout affects the period for which alkaline conditions can be claimed [67]. In a concrete walled storage pool, the concrete might become saturated with water, but the lack of refreshment of the water limits any dissolution of concrete constituents. Strong acids and alkalis can also attack concrete [64], but are unlikely to be encountered in quantity in spent fuel storage facilities.

In principle, neutron irradiation can cause swelling through atom displacement and gamma irradiation can lead to gas evolution from radiolysis of water. Some dose–damage data are available [68].

Concrete is hard, strong in compression but weak in tension. Thus conditions in service that produce tensile stresses are undesirable. Corrosion of reinforcing bars [66] is a source of stress that has already been mentioned. Temperature gradients can also generate stresses, such that acceptable limits on temperature gradients in concrete monoliths are often specified.

3.9.2. Concrete specification and design of durable concrete structures

A concrete is normally specified against a known set of requirements; for example, it may be important to develop hardness early on, or it may be important to withstand freezing conditions early in service [65, 66]. Many national and international organizations issue guides on how to specify concretes for particular applications (e.g. Ref. [69]). Design guides often focus [64] on the linkage between the specification of 'the mix' and the structure into which the mix will be made. Thus good durability (resistance to degradation) can be seen as stemming from attention to structural detailing, material, execution and curing.

3.9.3. Monitoring and inspection

Perhaps because of the difficulty in predicting the ageing behaviour of a concrete structure or component, much attention has been given to techniques for monitoring and inspection; for example, an IAEA report on the assessment and management of ageing of concrete containment buildings [68] describes some 20 test methods for exploring more than 30 properties or characteristics of concrete. They are summarized in Table 1. Other techniques may be applicable to concrete that is saturated with water or forms a container for water, for example FSPs [50].

A general checklist is provided in Ref. [64] for the investigation and appraisal of deteriorated concrete. It is noted in Ref. [64] that the investigation may include 'recalculation', since advances in computation may be able to demonstrate a current fitness for purpose for material that had to be treated more conservatively at the time of design and construction.

The interpretation of monitoring and inspection results (i.e. deriving an inspection frequency or deciding on the need for or timing of remedial action) depends very much on the duty of the concrete component. The expectations placed on reactor containment buildings may be much greater than those placed on some spent fuel storage facilities in certain respects, for example to

TABLE 1. METHODS TO ASSESS CONCRETE PROPERTIES OR CHARACTERISTICS

Concrete property or characteristic	Evaluation method																			
	Air permeability (S)	Audio methods (N)	Break-off methods (S)	Carbonation depth (D)	Chloride testing (S)	Core testing (D)	Infrared thermography (N)	Instrumentation (N)	Magnetic methods (N)	Modal analysis (N)	Petrographic methods (D)	Probe penetration (S)	Pullout testing (S)	Radar (N)	Radiation/nuclear (N)	Rebound hammer (N)	Stress wave transmission (N)	Tomography (N)	Ultrasonic pulse velocity (N)	Visual inspection (N)
Alkali–carbonate reaction											X									
Air content	X										X									
Acidity				X	X															
Alkali–silica reaction											X									
Bleeding channels											X									X
Cement content											X									
Chemical composition											X									X
Chloride content					X	X														
Compressive strength			X			X						X	X			X			X	
Concrete cover						X			X					X						
Aggregate content											X									
Mixing water content											X									
Corrosive environment	X			X	X															X
Cracking		X				X		X			X				X		X	X	X	X
Creep						X		X												
Delamination		X				X	X				X				X		X	X	X	X
Density						X									X					
Elongation						X		X												
Embedded parts														X	X			X		
Frost damage											X									
Honeycomb						X					X				X			X	X	X
Modulus of elasticity						X													X	
Modulus of rupture						X														

TABLE 1. METHODS TO ASSESS CONCRETE PROPERTIES OR CHARACTERISTICS (cont.)

	Evaluation method																			
Concrete property or characteristic	Air permeability (S)	Audio methods (N)	Break-off methods (S)	Carbonation depth (D)	Chloride testing (S)	Core testing (D)	Infrared thermography (N)	Instrumentation (N)	Magnetic methods (N)	Modal analysis (N)	Petrographic methods (D)	Probe penetration (S)	Pullout testing (S)	Radar (N)	Radiation/nuclear (N)	Rebound hammer (N)	Stress wave transmission (N)	Tomography (N)	Ultrasonic pulse velocity (N)	Visual inspection (N)
Moisture content						X					X									
Structural performance		X						X		X										X
Permeability	X										X									
Pullout strength													X							
Aggregate quality											X									X
Freeze–thaw resistance											X									
Soundness						X									X			X		
Splitting — tensile strength						X														
Sulphate resistance											X									
Tensile strength			X			X														
Concrete uniformity											X					X				X
Voids						X								X	X		X	X	X	X
Water/cement ratio											X									

N: non-destructive method; S: semi-destructive method; D: destructive method.

withstand loadings and provide physical barriers. Collection of relevant data may be incidental to the operation of a power plant, whereas with a storage facility distanced from reactor operation, the need for and value of monitoring and inspection may incur greater scrutiny.

3.10. NEUTRON ABSORBERS

Neutron absorber materials in FSPs include liquid and solid forms. The liquid forms are boron-containing (e.g. boric acid), generally in pressurized water reactor FSPs. The borated chemistry is maintained because the FSP water mixes with primary system water. In general, credit is taken for borated water in the criticality design. However, although typical boron concentrations are higher than 2000 ppm boron in PWR pool waters, criticality designs limit boron credits to a fraction of the total boron in the pool water. Solid neutron absorbers include [50]:

(a) Boraflex: boron carbide in a silicone rubber binder.
(b) Boral: boron carbide particles in an aluminium matrix with aluminium alloy cladding.
(c) Boronated aluminium alloy: generally used in baskets for storage/transport casks.
(d) Boronated stainless steel alloy: Type 304 stainless steel with up to 1.9% natural boron.
(e) Cadminox: cadmium metal in a leak proof cladding.

The absorbers are incorporated into fuel storage racks for criticality control. Since the neutron absorbing capacity needs to be assured, age related degradation of the neutron absorbers needs to be factored into AMPs.

Ageing of Boraflex occurs due to radiation induced degradation of the silicone rubber matrix. Results from some accelerated test coupons in the USA have been found to be unreliable predictors of Boraflex panel degradation. The GALL report [5] includes guidance for an AMP that relies on periodic inspection, testing, monitoring and analysis of the criticality design to ensure that the required 5% subcriticality margin is maintained. The frequency of inspection and testing depends on the Boraflex condition but has a maximum of five years. Testing includes: (1) neutron attenuation testing ('blackness testing') to determine gap formation in the silicone matrix; (2) sampling for silica in the FSP water, along with boron loss; and (3) analysis of criticality to ensure that the 5% criticality margin is maintained.

Boral compacts may be subject to copious hydrogen gas evolution when first exposed to FSP water if the high surface area internal surfaces have not been subject to adequate prefilming prior to insertion into the FSP. However, hydrogen evolution disappears after a short period, after stable aluminium oxide films are established. The Boral may be subject to galvanic corrosion in contact with stainless steel in some water chemistries (i.e. high conductivity), so the rack design must address mitigation of contact with FSP water, while also accommodating a gas relief feature to avoid accumulation of radiolytic gases, which have caused rack panels to expand against fuel assemblies, affecting assembly removal until holes were drilled to relieve the gas pressure.

The durability of boronated aluminium depends on the water purity in wet storage or on a non-aggressive cover gas in dry storage.

Boronated stainless steel is expected to have low corrosion rates in the normal range of FSP purity levels. However, welding procedures need to address mitigation of sensitization to avoid IGSCC. The maximum natural boron loading is 1.9%, in order to accommodate material workability. The amount of boron needed can be mitigated by use of ^{10}B, although this increases the cost.

Cadminox does not have wide usage. Cladding cadmium metal sheets with a corrosion resistant material facilitates its durability. However, there have been cases that involved direct exposure to water of cadmium sheets for periods of more than a decade.

Since neutron fluxes from spent nuclear fuel are low (e.g. fluence of 10^{14} n/cm^2 in 20 years), boron consumption is not an issue in ageing of neutron absorbers in the expected time frame of interim storage.

The CASTOR V/21 cask dry storage demonstration referred to above included a spent fuel basket fitted with borated stainless steel plate for criticality control. The boron content was approximately 1%. Polyethylene neutron absorbers were mounted in the cask wall. Neutron measurements at the cask surface indicated that the neutron absorber performance was satisfactory [53].

3.11. METAL SEALS

Dry storage casks and canisters with bolted closures have configurations that have metal or elastomer seals. The metal seals include:

(a) O rings: uncoated, stainless steel, Inconel, Nimonic steel.
(b) O rings: coated, including aluminium, silver, copper.
(c) Double O rings, including pressure monitoring in the space between the two seals.

The type of material of which the seals are made is likely to affect their degradation in either normal service or off-normal conditions (see Section 6.3).

A problem with a cask seal is illustrated in the following case history. A leak developed in a TN-32 secondary cask seal [70]. The seal comprised a

Nimonic 90 spring enclosed in an Inconel 600 sleeve. The spring and sleeve assembly were enclosed inside a soft grade 1050 or 1100 aluminium cover, comprising the malleable sealing surface. This O ring resided just inside the bolt circle. The lid assembly, associated cask and seal pressure monitoring equipment were enclosed inside a watertight weather cover, consisting of a steel dome and rubber O ring seal. The cover was intended to keep water from the lid bolts and the instrumentation installed in the lid. When a leak was discovered, the fuel cask was opened, resulting in observation of a leak involving the secondary seal. There was noticeable corrosion in the gap between the lid and the cask flange, observed as rusting of the carbon steel lid and flange involving an area of about seven lid bolts (of a total of 48 bolts). The metallic O ring aluminium jacket was pitted at several locations within the affected area. The root cause analysis suggested that the cause was galvanic corrosion. Rainwater had been observed leaking through an instrument conduit penetration on the top of the weather cover. The fitting was loose, providing the leak path.

Several cask types were involved in dry storage demonstrations at the INEEL. A CASTOR V/21 cask was loaded with PWR zircaloy clad fuel in 1985. In 1999 and 2000 the cask and cask contents were examined. The primary cask lid seal comprised two concentric O rings, one metal and one elastomer. The metal O ring was in excellent condition, having no sign of degradation. The sealing surfaces on the cask and lid were shiny [53].

3.12. NON-METAL SEALS, LINERS, COATINGS, ETC.

Non-metals are involved in both wet storage and dry storage facilities. In wet storage, the applications include:

(a) FSP liners;
(b) Rubber boots on FSP gates;
(c) Liners for ion exchange vessels;
(d) Gaskets;
(e) Filters;
(f) Coatings.

In dry storage, the principal applications are as gasket material and filters.

3.12.1. Non-metal fuel storage liners/coatings

The main elements of the FSP structure are thick concrete walls and floors that are designed to maintain integrity, even under seismic conditions. Metals, mainly stainless steel, are the common liner material. However, numerous FSPs have used epoxy liners or coatings to cover the concrete walls and floors. To minimize epoxy degradation, radiation doses need to be limited to a 1 MGy lifetime radiation dose. At a Canadian FSP, where water temperatures have regularly exceeded 30°C, minor damage to the wall coating has been noted. To mitigate the issue of epoxy degradation, alternative types of paint material have been developed and applied in the UK [50]. The concrete walls of the Savannah River Site's Receiving Basin for Offsite Fuel (RBOF) away from reactor (AFR) storage facility were coated with Amercoat. The facility operated from 1964 to 2003. Some bubbles were observed to form in the coating, but repair of the coating was not required.

3.12.2. Liners for ion exchange vessels, valve chambers, etc.

Elastomers are used as liners in ion exchanger vessels, valve chambers, etc. Ageing is addressed in the GALL report [5]. The environment is chemically treated oxygenated water of up to 50°C, with a range of levels of radiation. Ageing effects include hardening and cracking. Ageing management involves determining the qualified life of the linings under service conditions. Linings are replaced as dictated by the qualified life determination.

3.12.3. Gaskets

Several cask types have been involved in dry storage demonstrations at the INEEL. A CASTOR V/21 cask loaded with PWR zircaloy clad fuel in 1985 was examined in 1999 and 2000. The seal had two concentric O rings, one metal and one elastomer, that provided the primary lid seal. Upon inspection in 2000, the elastomer was observed to be in excellent condition. It was flexible, having no evidence of embrittlement. There was no evidence of cracking, breaks or delamination [53].

Ethylene propylene copolymer is one of the more radiation resistant types of material. Fluoride-containing material should not be used in applications that will involve high radiation doses, particularly if moisture is available to form hydrofluoric acid.

3.12.4. Coatings: dry storage

The epoxy coating on the exterior of the CASTOR V/21 cask at the INEEL was in good condition, with only minor areas that had peeled after exposure to the local environment for approximately 14 years. The environment is dry, with annual temperature extremes of 37.8°C to –28.9°C [53].

4. MANAGING AGEING IN SPENT NUCLEAR FUEL STORAGE FACILITIES

Effective ageing management involves taking informed actions to mitigate degradation of SSCs in wet or dry storage facilities. The actions are based on an understanding of the types of material and environments at the facility. The key elements of ageing management, as addressed in standard ageing terminology, involve maintenance and condition assessment, with their multiple functions as outlined in Appendix II. Important to effective ageing management is development of an ageing management plan that identifies the SSCs that need specific actions to mitigate ageing and the AMPs that are to be applied to each SSC. Criteria for selection of SSCs requiring specific ageing management are given in Section 4.4.

Staff at storage facilities that are in the design and early construction phases have the opportunity to apply principles of ageing management that anticipate and mitigate problems that might otherwise emerge as troublesome during service. While facilities that are in operation were generally designed to minimize material degradation, many are now or will be in operation longer than was anticipated in the original design. Furthermore, licensing authorities are requiring systematic evidence that ageing is effectively managed during the extended operation of nuclear power plants. Those facilities that can demonstrate that they already have and are applying an effective ageing management plan will be in a strong position to meet regulatory requirements for extended operation. Nuclear power plants are generally subject to comprehensive regulatory requirements for ageing management. Operators of smaller storage facilities will generally have an ageing management plan that is less complex than is the case for larger facilities. They may have more leeway in the application of systematic ageing management. In fact, some respondents to the questionnaire on storage facility ageing (see Section 1) indicated that they have not yet implemented an ageing management plan. Ageing management has emerged as a prominent consideration in the operation of nuclear facilities. It is a recommendation of this CRP that all spent fuel storage facilities have effective ageing management plans (Section 8).

More than half a century of experience with wet storage facility operation demonstrates that proper attention to material selection and environmental control can result in several decades of operation without substantial impacts of SSC failures. However, storage facility operators that have neglected the essentials of ageing management have had to deal with degraded components, elevated radiation inventories in the pool water, expensive, premature component replacements, and costly and complex waste management. The contrasting experience demonstrates that ample investment in effective ageing management is cost effective.

Approaches to effective ageing management have been developed by several organizations, including the IAEA and the NRC, and in countries participating in this CRP. Alternative approaches are summarized in this section to assist facility operators to develop ageing management plans that are relevant to their facilities.

4.1. ALTERNATIVE METHODS TO APPLY EFFECTIVE AGEING MANAGEMENT

Operators of power reactors and large independent spent fuel storage installations (ISFSIs) are guided in AMPs by regulatory requirements and industry standards. The operators generally have staff and budgets that are consistent with the scope of the ageing management needs. Operators of test/research reactors have access to smaller staff and budgets, but generally deal with more benign conditions (e.g. temperatures and radiation levels) than are characteristic of larger facilities. This section summarizes alternative methods that have been devised to provide a systematic approach to the design and implementation of effective ageing management in nuclear facilities. Storage facility operators that seek to implement or upgrade the basis for ageing management in their facilities can choose the alternative that most effectively addresses their specific needs.

4.1.1. IAEA method to implement ageing management in nuclear facilities

The IAEA has developed a general format for analysing ageing management needs and has applied the method to ageing management recommendations for several major nuclear power plant components, including reactor pressure vessels, PWR steam generators and concrete structures. Concrete structures are relevant to wet and dry storage facilities, so the IAEA method for ageing analysis of concrete structures is illustrated here (Fig. 3). Details of the ageing management needs for concrete structures are provided in Ref. [68]. The same method can be adapted to other SSCs in storage facilities.

A merit of the IAEA ageing analysis method is that it identifies and integrates the essentials of understanding and managing ageing. A drawback of the existing IAEA method is that it does not address ageing management details of SSCs that are specific to types of fuel storage material other than concrete. Although several IAEA reports address corrosion of SNFSF material

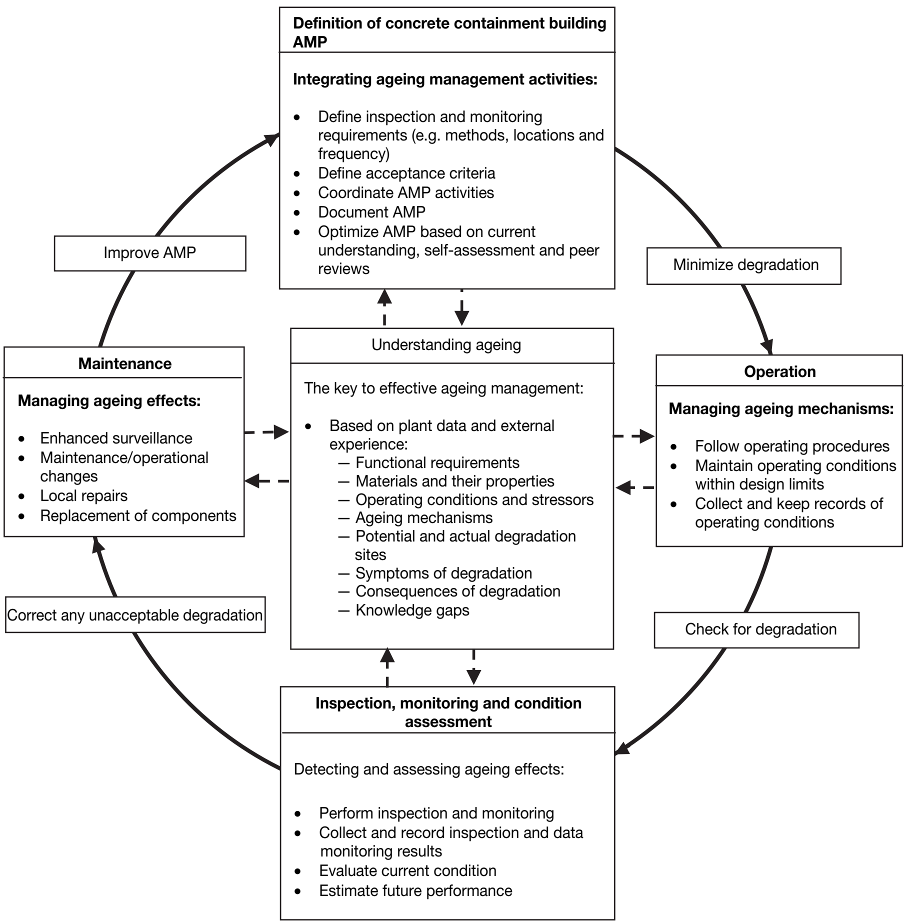

FIG. 3. IAEA ageing management assessment method for concrete structures [1].

[11, 50, 71], the application to ageing management of SSCs in storage facilities is mainly described in general terms.

Fuel storage facility operators that choose to apply the IAEA method can draw guidance from the IAEA template for each SSC that requires ageing management, using the template that is illustrated for concrete. The general template is published in Ref. [72]. Each SSC will also need an AMP (see Section 4.4 for an illustration of AMP development focused on aluminium alloys).

4.1.2. Nuclear Regulatory Commission method for ageing management in nuclear fuel storage facilities

Nuclear power plant operators in the USA have begun to apply for operation beyond the current licensing period of 40 years. Central to regulatory approval for licence renewal is the requirement that the utility operators demonstrate that the needs of ageing management are understood and will be effectively implemented in the 20 years of additional operation. Over a period of nearly two decades, the NRC and the US utility industry have invested in developing the technical basis to implement ageing management strategies for nuclear power plants in the period of extended operation.

Major guidance specific to understanding and managing ageing in nuclear power plant SSCs important to safety is provided in the GALL report [5]. The report includes information directed to SSCs in LWR FSPs. The following are the merits of the US approach:

(a) The method provides a template that shows the technical requirements for understanding and managing ageing on one page, including references to specific AMPs for each SSC (illustrated in Appendix I).
(b) The template provides information on structures, types of material, environments and ageing management effects/mechanisms that are specific to SSCs in wet storage facilities. The illustrations in Appendix I apply principally to facilities that operate with deionized water chemistry, with one illustration that relates to a borated water system.

Facility operators that choose to use the template will need to adapt it to the specific types of material and environments in their facility if they differ from those addressed in the illustrations. While the SSCs addressed in the Appendix I templates include those that are regarded as important to safety in boiling water reactor (BWR) FSPs, in an FSP at a test/research reactor the SSCs that are addressed may be substantially fewer and involve a different set of types of material, environments and mechanisms. Some of the guidance provided in the Appendix I templates may be immediately applicable to non-LWR storage facilities. However, the principal merit is illustration of a template that provides guidance to efficiently assess the ageing management considerations in SNFSFs, wet or dry.

The AMPs that are identified in the templates are often drawn from AMPs that also apply to other SSCs in nuclear power plants. Specific AMP information may not be readily available to operators of test/research reactors. Also, some types of material that are common in test/research reactors are not addressed in the power reactor templates. Prominent by their absence are aluminium alloys. Operators with aluminium alloys in their facility will need to adapt the template. This would involve identifying the information needed to complete the template as it applies to aluminium alloy systems or components that exist in the specific storage facility. Section 3.1 provides information on aluminium alloy corrosion. Provided in Section 4.4 is an illustration of the considerations involved in the development of an AMP, in this case applied to aluminium alloys.

To illustrate considerations that narrow the number of mechanisms that need to be considered in an ageing management review, stainless steels have very low uniform corrosion rates in benign water chemistries (0.03 μm/a), so there is no practical need to monitor stainless steel systems or components for uniform corrosion. However, stainless steel piping has undergone IGSCC in a few PWR FSPs [4], suggesting that surveillance of weld areas of stainless steel components is justified.

4.1.3. Ageing management method: Argentina

Summarized in Section 5.1 is a formal process applied in Argentina to systematically identify and evaluate the critical SSCs in spent fuel storage facilities. The elements of the process are shown in Fig. 4. The approach is consistent with the methods developed by the NRC and the IAEA. The method has been applied to both new (ISFSI) and older (at-reactor) storage facilities. It will also be applied to new storage facilities, including dry storage if implemented.

4.1.4. Ageing management: Germany

The German contribution to the CRP involved ageing management of metal casks. The ageing management considerations for metal casks are summarized in Section 5.3. The German approach involves two main principles:

(a) In the design and licensing phases, any possibility of ageing effects must be considered, including service conditions (radiation, temperature, mechanical stresses and environmental conditions) as they relate to the material of construction, with emphasis on metallic seals.
(b) The operational phase needs to involve appropriate surveillance in order to ensure proper operation of all safety relevant components over the entire storage period.

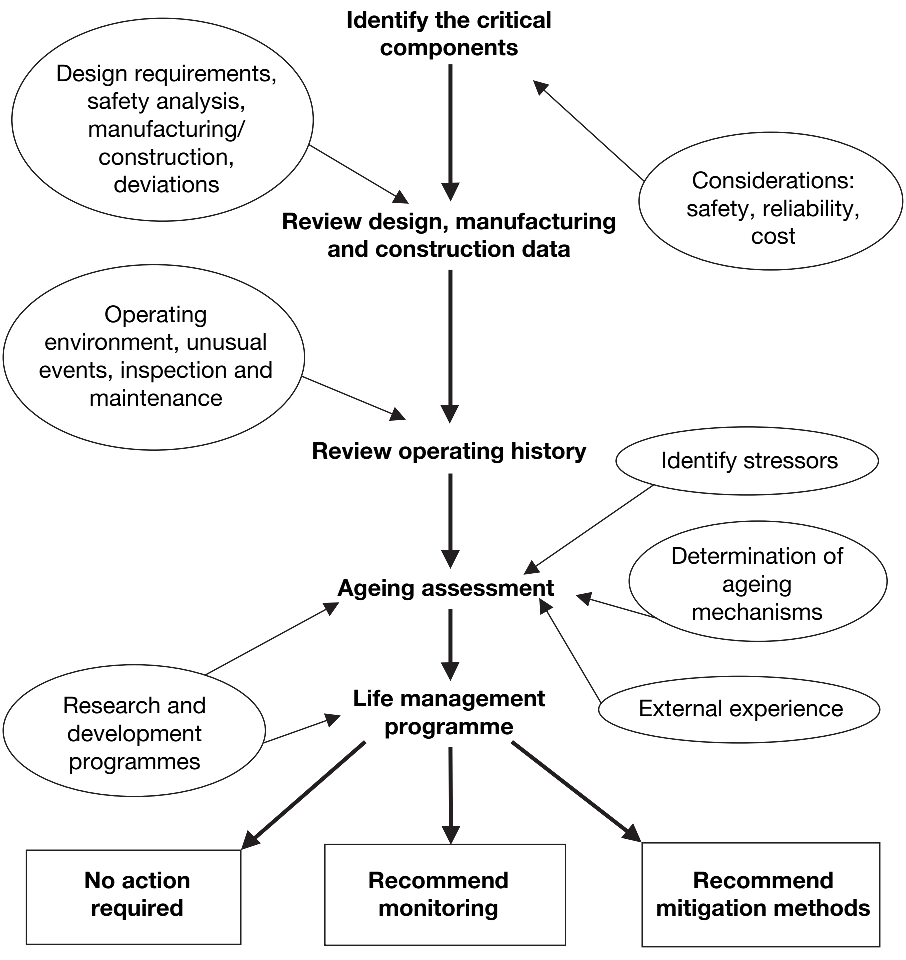

FIG. 4. Ageing management method applied in Argentina.

4.1.5. Ageing management: United Kingdom

Summarized in Section 5.4 are the elements of ageing management as they are applied by British Nuclear Fuels (BNFL) to nuclear fuel storage facilities. An important perspective is that the process of managing ageing involves all phases of a facility's life, including design, operation and decommissioning. The design phase is strongly based on UK and international standards and on internal design guides, codes of practice and standards. Specific examples are cited in Section 5.4.

Plant ageing is managed by a policy of understanding the degradation mechanisms to the extent possible and using that understanding to target SSCs considered as vulnerable or for which there are uncertainties. However, when the behaviour of a material, such as 304L stainless steel, is well defined and degradation effects are minimal, no monitoring or maintenance is prescribed.

Targeting of research and development and inspection is driven by the perceived consequences of failure and by uncertainty in understanding degradation mechanisms. Economic consequences are also considered in prioritization. Inspection rigor and frequency depend on the severity and level of understanding of the operating conditions. After the facility is in operation, inspections may be modified — more frequent if operating conditions are more aggressive than anticipated, less frequent if the conditions are more benign than expected. If inspection methods are deemed to be inadequate, development programmes may be instituted. Results of routine plant inspections and asset surveys are used to support the plant safety case by demonstrating that the plant can still satisfy its safety functions.

Asset care is the latest initiative in the overall management of ageing, involving inspection, examination, testing, maintenance, and refurbishment and replacement to optimize facility service life and reliability.

4.2. CATEGORIES OF SPENT NUCLEAR FUEL STORAGE FACILITY STRUCTURES, SYSTEMS AND COMPONENTS THAT REQUIRE SPECIFIC AGEING MANAGEMENT

Guidance regarding the SSCs in FSPs that need specific ageing management is summarized in this section [73].

(a) SSCs important to safety:
   - (i) SSCs that maintain the conditions required to store spent fuel safely;

(ii) SSCs that prevent damage to the spent fuel during handling and storage;
(iii) SSCs that provide reasonable assurance that spent fuel can be received, handled, packaged, stored and retrieved without undue risk to the health and safety of the public.

These SSCs address the following important functions: criticality control, shielding, confinement, heat transfer and structural integrity.

(b) SSCs not important to safety but whose failure could prevent an important safety function from being fulfilled, for example coatings and liners on FSP walls.

4.3. AGEING MANAGEMENT OF NUCLEAR FUEL

The prime consideration in an SNFSF is that the integrity of the fuel cladding should be preserved under storage and handling conditions. The stored fuel must be protected against degradation that leads to gross rupture, or the fuel must be confined such that degradation of the fuel will not pose operational problems with respect to its removal from storage [73]. The facility's design and operation are required to:

(a) Ensure containment of radioactive species consistent with the release regulations;
(b) Maintain subcriticality under all credible conditions;
(c) Ensure adequate confinement and containment of the spent fuel under all credible conditions of storage;
(d) Allow ready retrieval of the spent fuel from the storage facility.

Durabilities of the leading cladding types are addressed in Section 3 and the cited references. While zirconium alloys have been durable in a range of water chemistries, water chemistry control is the principal key to maintaining satisfactory durability for other cladding material. It is also important that mechanical damage of the fuel units (assemblies, elements, etc.) be minimized in fuel handling operations.

Maintaining the fuel in subcritical configurations requires that age related degradation of storage racks, canisters, etc., be mitigated. It is important to consider that the requirement extends to maximum credible conditions, including seismic conditions. Thus if age related degradation has compromised the integrity of critical storage systems or components, the capability of the systems or components to perform their intended function must be judged on the basis of their viability in a design basis seismic event.

In cases where fuel cladding has been breached, vulnerability of the exposed fuel material depends on the fuel type. Uranium oxide is largely inert when exposed to water at FSP temperatures [11]. LWR fuel with cladding defects has been frequently stored without isolation in canisters. Composite fuels, for example uranium oxide encased in aluminium metal, also generally release radioactive species slowly. By contrast, cladding breaches that expose uranium metal fuel facilitate radiation releases by aqueous corrosion of the uranium at FSP temperatures [13] (Section 3.2).

4.4. ILLUSTRATION OF AGEING MANAGEMENT PROGRAMME DEVELOPMENT APPLIED TO ALUMINIUM ALLOYS IN WET STORAGE

Regardless of the ageing management method selected by an operator of a smaller storage facility, it seems likely that AMPs specific to at least some of the SSCs in the facility will need to be devised. The GALL report [5] outlines ten attributes of an effective AMP. Application of the AMP guidance in the GALL report as it might apply to aluminium alloys in wet storage is illustrated in Table 2.

The guidance provided in this section can be similarly applied to the development of AMPs for other SSCs.

4.5. AGEING MANAGEMENT IN DRY STORAGE FACILITIES

The elements of ageing management that have been addressed for wet storage facilities are equally applicable to dry storage facilities, although the configurations and operating conditions (and, in some cases, types of material) differ from wet storage facilities. Bases for the development of ageing management plans and AMPs are addressed in this section.

Participants in the CRP provided useful input to dry storage experience, including dry cask storage in Germany (Section 5.3), experience at a dry storage facility in the Russian Federation used to store fuel with stainless steel cladding (Section 6.6) and experience with dry storage of aluminium clad fuel in Australia (Sections 5.2 and 6.2). The design of a dry storage facility for pressurized heavy water reactor (PHWR) fuel in Romania is summarized in Section 6.5.

TABLE 2. ATTRIBUTES OF EFFECTIVE AGEING MANAGEMENT PROGRAMMES

Element	Description
1. Scope of the programme	The scope of the programme should include the specific structures and components subject to an ageing management review. Effective ageing management of aluminium alloy fuel cladding and facility components in wet storage facilities.
2. Preventive actions	Preventive actions should mitigate or prevent the applicable ageing effects. (a) Control water chemistry parameters to prevent or mitigate pitting, crevice and galvanic corrosion. At the onset of pitting corrosion, determine whether it is chemically or biologically driven, because countermeasures depend on which mechanism is involved. (b) Minimize damage to reactor formed oxide films during discharge from the reactor and fuel handling operations. (c) Eliminate or minimize cathodic particle deposition (e.g. iron oxides) on aluminium alloy surfaces. (d) Eliminate or minimize aluminium alloy contact with sludge layers on the facility floor. (e) Recommended: maintain water flow to avoid stagnant conditions.
3. Parameters monitored or inspected	Parameters monitored or inspected should be linked to the effects of ageing on the intended functions of the particular structure or component. As a minimum: water conductivity (<5 µS/cm), chloride (<1 ppm), pH (pH4–8.5). Other: sulphate (<1 ppm), heavy metals, radiochemical species. **Note:** The parameter limits indicated above are higher than the limits that generally apply in wet storage facilities at nuclear power plants [50]. While the more restrictive specifications are preferable, justification for the higher values is based on operation at numerous facilities at higher values for extended periods with minimal impacts on aluminium alloys.
4. Detection of ageing effects	Detection of ageing effects should occur before there is a loss of intended function of any structure or component. This includes aspects such as method or technique (i.e. visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one time inspections to ensure timely detection of ageing effects. (a) Visual inspection of aluminium alloy surfaces when accessible, including use of binoculars or underwater optical devices. (b) Inspection/analysis of components (e.g. racks, canisters, tools) if removed from the pool.

TABLE 2. ATTRIBUTES OF EFFECTIVE AGEING MANAGEMENT PROGRAMMES (cont.)

Element	Description
	(c) Use of corrosion specimens that are inspected/analysed periodically. The specimens need to represent alloys and conditions present in the systems or components. Coupons should be positioned at representative and extreme locations relevant to aluminium alloy locations. **Note:** Maintaining cladding integrity is the priority consideration, particularly if the fuel involves uranium metal.
5. Monitoring and trending	Monitoring and trending should provide for prediction of the extent of the effects of ageing and timely corrective or mitigative actions. Systematic monitoring and trending of aluminium alloy system or component degradation is not generally conducted in wet storage facilities. If pitting is observed on coupons or easily accessible components such as fuel handling tools, the pit frequency and depth could be measured, but to be statistically significant many measurements would be necessary. However, the general severity of the pitting attack, the systems or components involved, and when in facility operation it is first observed, will determine whether corrective/ mitigative actions are justified.
6. Acceptance criteria	Acceptance criteria against which the need for corrective actions will be evaluated should ensure that the particular intended structure or component functions are maintained under all current licensing design basis conditions during the period of operation. Evidence of degradation will be judged in terms of the ability of the specific system or component to perform its intended function in the expected service period. Pitting on heavy section rack members would be of less concern than significant pitting observed on fuel cladding.
7. Corrective actions	Corrective actions, including root cause determination and prevention of recurrence, should be timely. If significant degradation or failure of one or more aluminium alloy system or component is detected, root cause analysis or another assessment will be applied to provide an informed basis for corrective actions. Also, possible effects of the corrective action on other material in the storage facility will be considered in the planning phase.
8. Confirmation process	The confirmation process should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective. If corrective actions are implemented, for example mitigation of pitting attack, subsequent monitoring is needed to assess whether the actions have effectively dealt with the cause of the pitting.

TABLE 2. ATTRIBUTES OF EFFECTIVE AGEING MANAGEMENT PROGRAMMES (cont.)

Element	Description
9. Administrative controls	Administrative controls should provide a formal review and approval process. For smaller facilities, the need for administrative controls will generally be decided on a facility specific basis.
10. Operating experience	Operating experience involving the AMP, including past corrective actions resulting in programme enhancements or additional programmes, should provide objective evidence to support a determination that the effects of ageing will be adequately managed so that the intended structure and component functions will be maintained during the period of operation. If aluminium alloy system or component degradation is observed in a specific facility, the history of aluminium behaviour in the facility should be accessed to determine whether prior problems have been observed, and, if so, the corrective actions that were taken and whether they are judged to have been appropriate. Experience with aluminium alloy behaviour in other relevant facilities may be useful.

Note: The entries for aluminium alloys may be modified by staff of specific facilities if justified by special considerations.

4.5.1. Implementation of ageing management in dry storage facilities

Section 4.1 outlines alternatives for devising effective ageing management in storage facilities. Operators of dry storage facilities can review the options and select the method that is most effective for analysing the ageing management needs in their facility.

4.5.2. Development of ageing management plans

Ageing management plans involve identification of the SSCs that need specific actions to mitigate age related degradation and development of appropriate AMPs for each SSC.

4.5.3. Development of ageing management programmes

An AMP identifies the considerations to be taken into account to effectively manage age related degradation in SSCs. The characteristics of an AMP are provided in Section 4.4, illustrated for an aluminium alloy AMP.

Guidance regarding issues of fuel storage in metal casks is provided in Sections 5.3 and 6.3. AMPs in other dry storage facility designs, silos, vaults and dry wells can be devised by applying the elements of ageing management summarized in this report.

4.5.4. Ageing management of nuclear fuel in dry storage

Mitigation of age related degradation of nuclear fuel in dry storage involves the same requirements that apply to fuel in wet storage (Section 4.4). However, the ageing mechanisms and approaches to mitigate age related degradation of the fuel cladding are markedly different in dry storage. Thermal creep is the leading failure mechanism for LWR fuel. Hydrogen redistribution and hydride reorientation also need to be minimized. It is therefore necessary to design the storage facility such as to avoid exceeding the cladding temperature limit. The NRC has recently accepted 400°C as the limit for fuel with zirconium alloy cladding at burnups at or below 45 GW·d/MTU [74]. For aluminium clad fuel, the interim limit is 200°C, but this is subject to specific fuel characteristics and regulatory requirements [9]. Magnox fuel is discharged at about 350°C to a vault with a carbon dioxide cover gas. When the cladding temperature decreases to 150°C, the fuel is transferred to a vault with air as the cover gas.

4.6. MONITORING, INSPECTION AND ANALYSIS NEEDS

In wet and dry storage facilities, ongoing verification is required to ensure that the performance of critical SSCs is adequate to meet the requirements of effective ageing management. Elements of condition monitoring are summarized in Appendix II. Planning is needed to ensure that resources are focused on the priority ageing management issues.

4.6.1. Condition monitoring in wet storage facilities

Spent fuel, fuel storage racks and other components in high radiation fields are not readily accessible for close inspection. Methods that may be applied to define the status of a material’s condition are summarized in this section.

4.6.1.1. Destructive examinations

Detailed examinations of irradiated fuel assembly material are conducted in hot cells, but have limited application since they are expensive. Detailed metallographic examinations of sections from unirradiated components obtained from several FSPs after periods of service in water, involving stainless steels and aluminium alloys, have been summarized [16].

4.6.1.2. Coupons

Coupon assemblies representing metals involved in FSP components and fuel cladding have been inserted into FSPs [13, 14]. Periodic examination of the coupons provides valuable insights regarding material behaviour. The coupons need to represent, to a practical extent, the material and conditions represented in the facility components, for example typical welds, crevices, galvanic factors, oxide films (damaged and undamaged) and the presence or absence of radiation. Coupons proved valuable to assess the corrosion effects of biocide dosing, including the period that was required for corrosion rates to return to normal [13]. The results of an extensive IAEA coupon study of aluminium alloy corrosion in FSPs have been reported [14]. American Society for Testing and Materials standards for designing and conducting corrosion coupon programmes are given in Ref. [75].

4.6.1.3. Underwater photography and video cameras

Underwater photography and video cameras have been valuable to assess the status of fuel assemblies and FSP components.

4.6.1.4. Binoculars

Inspection of fuel assembly and component surfaces using binoculars has been useful in order to judge the status of corrosion. In some cases, fuel assemblies are moved as close to the water surface as permitted by radiation dose considerations.

4.6.1.5. Inspection of components removed from interim service, during maintenance or during decommissioning

Fuel storage racks have been subject to visual inspection after periods of service in FSPs after being removed from the FSPs [16]. As components are subject to maintenance, there may be opportunities to assess the condition of subcomponents or nearby components. As components are removed from service, it is recommended that a protocol be applied to assess the condition, at least to the extent of a systematic assessment of the surface condition. This provides cost effective information that may be valuable if licence extension is applied for or as a basis for judgements regarding the need for refurbishment. Documenting key historical, material composition and condition, and environmental information facilitates interpretation of cases involving significant age related degradation. FSP decommissioning also presents valuable opportunities to assess material condition. Decommissioning of a Hanford FSP provided an opportunity to evaluate the condition of concrete walls and floors and metal rebars [76]. Queries to the staff revealed that no evidence of concrete degradation or significant rebar corrosion was observed after 17 years of reactor service and 49 years prior to decommissioning (although in a dry environment). It is important that information obtained during interim inspections or decommissioning be documented, so that it becomes accessible to the facility or to other facilities that might otherwise have to make substantial investments to provide the information essential to make cases for licence extension or for ageing management decisions.

4.6.1.6. Non-destructive evaluations

Poolside evaluations of fuel rods have been conducted after significant periods of wet storage [57]. The evaluations provide evidence regarding possible through-wall or incipient defects that may be significant if the fuel is transitioned to dry storage. Data on oxide thickness and cladding mechanical properties can also be derived.

4.6.1.7. Sipping

Sipping involves either wet or gas phase measurement to detect whether radioisotopes are escaping from fuel rods. Wet methods have been applied to test/research reactor fuel assemblies (elements) that are candidates for US take-back programmes [14, 47]. Sipping has also been conducted in gaseous environments in closed chambers or hoods. Fuel with substantial decay heat causes temperature increases in the fuel rods, which facilitates effusion of mobile radioisotopes from rods that have through-wall defects.

4.6.1.8. Water chemistry monitoring

Water chemistry monitoring has been addressed elsewhere [50, 77]. In some cases selective monitoring has been conducted to determine the degree of non-uniform parameters, for example pH and conductivity in various sections of a specific facility [15]. Other special FSP water monitoring has involved electrochemical measurements to assess corrosion potentials of specific types of material in the chemistry conditions in a specific facility [15]. Interpretation of radioactivity levels and radioisotopes is valuable to assess whether there are leaking fuel rods in the FSP.

4.6.1.9. Laboratory data

Corrosion data measured under laboratory conditions that are relevant to the environments that apply to a given facility can provide estimates of component lifetimes. As an example, laboratory studies on irradiated AGR cladding in moist air provide the basis to judge the lengths of time that stainless steels can reside in moist air conditions if they are suspected to be sensitized [51]. More broadly, corrosion data for the array of SNFSF material under relevant conditions provide a basis to analyse the durability of SNFSF material if properly applied. Creep rate measurements have been valuable in modelling fuel rod behaviour under dry storage conditions at elevated temperatures [78], as an example of relevant laboratory measurements.

4.6.1.10. Theoretical assessments

Some properties of SNFSF material are not amenable to direct measurement, for example the modes of failure of fuel cladding during dry storage. Models have been developed to evaluate zirconium alloy cladding in dry storage [59].

4.6.1.11. Inspection, surveillance and condition monitoring

Determination of the status of age related degradation in an SNFSF needs to involve systematic programmes of inspection, surveillance and condition monitoring, and related actions. Cracking in coolant system piping of a PWR FSP [4] was discovered in a periodic walk-down of the piping system.

4.6.2. Condition monitoring in dry storage facilities: inspection and analyses

The principal designs of dry storage facilities include dry wells, metal and concrete casks, silos and vaults. Condition monitoring requirements vary with design. However, general approaches will have similarities. Some SCCs are accessible to direct inspection during service; others are inaccessible. The following general approaches are applied. Descriptions of specific monitoring methods for all dry storage facility designs are beyond the scope of this report.

4.6.2.1. Direct inspections and monitoring

For SSCs that are accessible, inspections and monitoring can be used to verify performance and current condition. Examples include determining the condition of concrete structures and pads, external coatings and housings, and instrumentation and cables.

4.6.2.2. Analyses when direct inspections are not practicable

Radiation surveys, when compared with baseline values, provide evidence of the condition of polymeric neutron shield material. Corrosion of internal components can be estimated from the temperature and composition of the internal environment. Water permeation resistance of concrete might be employed to make judgements regarding the condition of embedded steel reinforcements.

4.6.3. Ageing management programmes

AMPs need to focus on ensuring that essential functions are being maintained. The same methodology that was illustrated for aluminium alloys in wet storage (Section 4.4) can also be applied to devise AMPs for dry storage. Operators need to consider normal, off-normal and accident conditions in ensuring that the effects of age related degradation will not compromise any performance requirements. Ageing assessments of SNFSFs need to ensure that no essential function of the facility would be compromised by a seismic event, in particular accounting for the potential consequences of degraded SSCs.

In addition to the ageing management guidance provided by the GALL report [5], numerous databases have been established. Also, standards are in preparation to provide SNFSF operators with a sound basis for ageing management that will be recognized by licensing authorities, for example Refs [79, 80].

4.7. SUMMARY OF ACTIONS TO IMPLEMENT EFFECTIVE AGEING MANAGEMENT IN FUEL STORAGE FACILITIES

The following recommendations offer operators of fuel storage facilities a basis for the effective management of age related degradation in their facilities:

(a) Develop an ageing management plan to be used as a basis to investigate whether SSCs in the facility need specific actions to mitigate age related degradation, either in current operation or in future operation. Devise and document AMPs as needed to ensure that each identified SSC will continue to perform its intended function in all foreseeable situations, normal and off-normal. A prioritization process may be needed that focuses resources on critical SSCs that have credible ageing mechanisms in the environments in a given facility.
(b) Select from the alternative ageing management methods one that facilitates development of ageing management strategies that are optimum for the specific facility.
(c) Involve the facility's management in planning the financial and staffing needs that will be required to effectively understand and manage age related degradation in the facility.
(d) Maintain a systematic and sustained approach to conduct and document the required ageing management actions. Evidence of effective ageing management will facilitate regulatory approvals if operation of the facility is needed beyond the original licensed period.

5. PERSPECTIVES ON MANAGING AGEING FROM COORDINATED RESEARCH PROJECT PARTICIPANTS

This section includes contributions from Argentina, Australia, Germany, the UK and the USA containing specific information regarding the application of the principles of ageing management applied in their facilities, wet and/or dry.

5.1. APPROACH TO MANAGING AGEING IN ARGENTINA

The Comisión Nacional de Energía Atómica (CNEA) in Argentina employs formal processes to systematically identify and evaluate the critical SSCs in Argentine spent fuel storage facilities. A plan is produced for facility surveillance, operation, monitoring and maintenance activities to ensure that any component degradation is within the original design specifications. This is essential for attainment of the planned lifetimes of the facilities.

A Technology Watch Programme is being established to ensure that observations and funding problems that could impact on the facility's lifetime are systematically investigated so that mitigating programmes can be designed.

The methodology Argentina uses is shown in Fig. 4. The approach is consistent with those developed by the NRC and the IAEA. This methodology has been applied to both the new (separate from reactor) and the older (at-reactor) pool buildings that hold fuel from Atucha I. The fuel assembly used in Atucha I is a 1 m long cluster of 36 rods of zircaloy 4 clad UO_2 fuel. Use of the methodology has contributed to extending the lifetime of both storage facilities for Atucha I fuel. The methodology is also applied to storage of the CANDU fuel from Embalse.

Consideration is being given to the possible future movement of Atucha I fuel from wet to dry storage. The same approach to ageing management will be applied to any new Argentine storage facilities.

5.2. THE AUSTRALIAN NUCLEAR SCIENCE AND TECHNOLOGY ORGANISATION'S APPROACH TO MANAGING AGEING

The Australian Nuclear Science and Technology Organisation (ANSTO) utilizes wet and dry storage facilities to manage its inventory of aluminium clad HEU aluminide research reactor fuel. It has implemented a spent fuel management strategy that has a strong focus on condition monitoring. Decades of experience gained in Australia and other countries clearly illustrate that the fuel and the components of the storage facility can be maintained in sound condition when the storage environment is consistent with the design requirements.

At ANSTO the liners of the wet and dry storage facilities are fabricated from stainless steel. All other components are fabricated from aluminium or aluminium alloys, including the fuel cladding. The following parameters are identified to have a strong influence on the integrity of the components and are routinely monitored:

(a) In wet storage:
  (i) Water conductivity;
  (ii) pH;
  (iii) Chloride concentration;
  (iv) Activity levels.
(b) In dry storage:
  (i) Cover gas pressure;
  (ii) Moisture;

(iii) Oxygen level;
(iv) Fission gases such as ^{85}Kr.

Acceptable operating levels have been determined for each parameter measured. If the levels are observed to exceed the determined criteria, the cause of the elevated levels is identified, and the appropriate remedy is implemented.

5.3. APPROACH OF THE GERMAN FEDERAL INSTITUTE FOR MATERIALS RESEARCH AND TESTING TO MANAGING AGEING

The German Federal Institute for Materials Research and Testing (BAM) is involved in dry cask storage site operation surveillance performed by the competent State authorities. The institute also investigates the long term behaviour of storage casks under service conditions. Based on this experience, the following approaches to managing ageing have been developed.

Material ageing effects, if they occur, do not lead to a safety relevant reduction of the properties of the cask, owing to their thick walls. The most sensitive components of casks for dry storage of spent fuel are the metallic seals used with the lid system responsible for the safe enclosure of the nuclear inventory. These metallic seals consist of an inner helical spring made of a nickel based alloy, an inner metallic jacket made of stainless steel and an outer jacket made of aluminium or silver. Seal installation requires a wide range of quality assurance measures, starting with fabrication standards and ending with the installation procedure itself.

Within the German licensing procedures the long term suitability of such seals has been investigated and demonstrated taking into consideration the required service conditions. The service conditions are characterized by the absence of water, which has to be removed carefully by suitable drying procedures if the casks are loaded under water. The cask body and the cavity between the lids are filled with inert gas in order to minimize possible corrosion reactions. Finally, protection measures against environmental influences are taken by using a sealed protection plate above the barrier lid system with its metallic seals.

The worldwide operation experience of several hundred dry storage casks in service for up to about 20 years (in 2003) shows few problems with the long term stability of this concept. Specifically, no safety relevant material ageing mechanisms have occurred. Problems were minor and included unsuitable operation conditions such as water getting into sealing systems or the reaction of condensed atmospheric humidity with outer cask metal components because of an inadequate quality of epoxy coatings.

In connection with the German concept of realizing on-site storage facilities for the interim storage of spent fuel from power reactors, the German Reactor Safety Commission (RSK), which advises the Federal Environment Ministry, published Safety Guidelines for Dry Interim Storage of Irradiated Fuel Assemblies in Storage Casks in 2001 [81]. These recommendations contain a chapter entitled Long-term and Ageing Effects, Long-term Monitoring, which defines the following requirements:

(a) An observation programme on long term and ageing effects during the storage period must be submitted.
(b) Special attention must be paid to components developed for the entire period of use, for example the casks, including safety relevant components such as the sealing systems and neutron absorbers, and the storage facility.
(c) The safety relevant properties of system and component parts must be guaranteed for the entire service period, and cask handling inside the facility must be possible at any time.
(d) The observation programme has to consider the following demands:
    (i) Inspection of the storage building and all other storage components: status report every 10 years.
    (ii) Spot check inspection of storage casks.
    (iii) Assessment of findings of recurrent inspections.

Two main conclusions about the German approach to managing ageing in dry spent fuel storage facilities can be made:

— Firstly, the design and licensing phase must consider the possibility of ageing effects by referring to the service conditions (radiation, temperature, mechanical stresses and environmental conditions) in combination with the types of material to be used. Metallic sealing systems for the long term safe enclosure of the radioactive inventory should be used under guaranteed dry and inert conditions.
— Secondly, the operation phase should be accompanied by an appropriate observation programme in order to ensure the required conditions for all safety relevant components over the whole storage period.

5.4. APPROACH OF BRITISH NUCLEAR FUELS TO MANAGING AGEING

The approach taken by BNFL to managing ageing is developed at the design stage and builds upon experience gained through designing, building, operating and decommissioning nuclear facilities since 1947. Much of the current philosophy was developed during the design and construction of the Thermal Oxide Reprocessing Plant (THORP).

The process of managing ageing encompasses all the phases of the facility's life; that is, design, operation and decommissioning. Underlying all these phases is the input from the results of research and development and plant inspection activities.

5.4.1. Design

Early in any design process there is a need to define the types of material to be used in the construction of plant and equipment. This material selection is based on many issues: the numerous technical requirements of the material as well as its cost, availability, proposed service life of the facility, decommissioning, etc. These issues are formally dealt with via the use of British Standards and internal design guides, codes of practice and standards that have evolved over the years to represent BNFL's 'know-how'. Design guides, etc., are incorporated into a set of BNF standards. Examples include:

(a) Codes of practice: the selection of material for vessels and piping or concrete shielding structures.
(b) Standards: heavy shielding grout, density 2900–6000 kg/m^3.
(c) Material evaluation: for contact with stainless steel.

These standards give the designer guidance on the types of material to use in specific environments. This guidance is based on BNFL's previous experience, supported where necessary by internal/external reports and/or laboratory testing. For difficult areas where corrosion is an important issue, a corrosion audit is undertaken. More information on this and corrosion prevention for BNFL Sellafield reprocessing plants is given in Ref. [82].

As part of the process of design for new plants, the routine maintenance and monitoring requirements are prescribed and incorporated into the plant's manual. This information is replicated in the group's engineering maintenance database, which supports plant management through providing prompts for undertaking statutory maintenance, safety checks, etc., and also acts as a mechanism for recording plant reliability and inspection reports.

5.4.2. Research and development and plant inspection

Plant ageing is managed by a policy of understanding the degradation mechanisms as far as possible and, from that knowledge, targeting for particular attention plants considered as vulnerable or for which we are uncertain of our understanding of the degradation processes; for example, THORP receipt and storage LWR storage racks have been fabricated from 304L stainless steel. In their storage environment, deionized water, the worst case measured corrosion rate for such steels is 0.03 μm/a. Given that the minimum material thickness used is 20 mm, corrosion over the design life is negligible and therefore there are no prescribed monitoring or maintenance requirements.

Where the processes are uncertain or unknown, further research into the degradation processes, or programmed inspection of the plant, or both, will be undertaken. Targeting of research and development and inspection is driven by the perception of the consequences of failure as well as the uncertainty in the understanding of degradation mechanisms; if ageing or failure of certain SSCs have a high safety consequence, those SSCs will receive more attention than those with no safety consequences of failure. Similarly, the economic consequences of ageing are considered, hence SSCs whose breakdown would severely curtail plant operations are also prioritized.

The policy was first developed for THORP, where targeted programmes of inspection are established. Inspection may be limited to visual inspection or may involve quantitative non-destructive testing such as ultrasonics, and in certain areas, such as dissolvers, on-line thickness monitors have been installed to facilitate the process. The programmes are updated as the actual plant operating conditions are better known and understood; therefore, for example, an SSC found to be operating at higher temperatures than originally planned or carrying a higher burden of corrodents may become targeted for inspection, whereas SSCs perceived to be benign may have inspection frequencies reduced. Similar policies are being developed for other plants.

Plant items are planned for replacement or repair on the basis of the above strategy as inspection and predictive laboratory data dictate. In some instances inspection methods are deemed to be inadequate and development programmes are introduced as a consequence, for example for blockage detection or long range inspection of pipework.

In some cases in which ageing is not perceived as an issue in the short term but is considered to become an issue in the long term, for example in the lifetime prediction of concrete structures, strategies on the above lines are being developed, but these may allow a longer term view for development. In such instances alliances with universities to develop the technology required over an extended period are under consideration.

In most instances there are not sufficient data available to allow the use of statistical lifetime prediction methods, but the potential for the application of such methods is being explored.

The results of routine plant inspections and periodic asset surveys are used to support the plant safety case through demonstrating that the plant in question can still satisfy its safety functions.

5.4.3. Long term ageing management (asset care)

The latest initiative in the overall management of ageing is asset care, which replaces individual plant refurbishment programmes. The purpose of the asset care programme is to ensure that there is a consistent approach taken across the portfolio of plants on the Sellafield site. Asset care is the process to ensure that the asset remains fit for purpose throughout its intended design life and decommissioning stages. This embraces inspection, examination, testing and maintenance to optimize its life and reliability, as well as asset care projects such as refurbishment and replacement.

The prerequisites of an effective asset care programme include life cycle design, asset register, an understanding of asset condition, maintenance and spares strategies and processes, and resources to implement and monitor the effectiveness of the strategies to meet defined standards.

5.5. APPROACH TO MANAGING AGEING IN NUCLEAR FUEL STORAGE FACILITIES IN THE UNITED STATES OF AMERICA

The ageing management methods developed by the NRC and the US nuclear power industry involve the following actions:

(a) Identification of the SSCs that are candidates for comprehensive ageing management.
(b) Assessment of design, material and construction information.
(c) Assessment of the facility's operating history.
(d) Assessment of age related degradation of selected SSCs:
    (i) Effects of stressors (chemical, electrical, mechanical, radiation);
    (ii) Inspection, surveillance, maintenance, repair, etc.

5.5.1. Technical basis for the approach to ageing management in nuclear power plants in the United States of America

The NRC invested in the development of a comprehensive and systematic approach to understand and manage ageing in nuclear power plants in order to facilitate implementation of regulatory oversight of ageing management in plants that are licensed for extended operation [5, 83]. The comprehensive compilation of ageing management guidance is known as the GALL report [5], which is structured to simplify and integrate the concepts of understanding and managing ageing by combining the two concepts on a one page template that is applied to relevant SSCs important to safety. The template also identifies the AMPs that apply to each SSC. Most important to this IAEA report, the GALL report includes specific templates for SSCs in BWR and PWR wet storage facilities. The information for FSPs is illustrated in Appendix I, focusing principally on BWR SSCs, which are characterized by deionized water chemistries, which are widely used in research and test reactors. While specific to LWR wet storage facilities, the template is applicable to other reactor types, to AFR storage facilities and to dry storage facilities.

The template addresses the specific structure (e.g. the heat exchanger), the material, the environments and the ageing effects/mechanisms that apply. The crux of the ageing analysis involves applying corrosion rates or other kinetic data that are appropriate to the relevant material, environments and ageing mechanisms. Ageing data and assessments are illustrated and referenced in Section 3 of this report. Operators of many FSPs will not find all the types of material in their facilities included in the templates in Appendix I; for example, many FSPs store aluminium alloy fuel cladding or components. The facility staff can use the template to analyse the key ageing considerations for aluminium alloy systems or components in the environments relevant to the facility. Aluminium alloy ageing information is given in Section 3.1 and in Ref. [82].

Operators of smaller SNFSFs that apply the template provided in the GALL report (Appendix I) need only adopt SSCs, environments, mechanisms and AMPs that have identified priorities in their facilities. In a given facility only a small number of SSCs and related material may need to be addressed in the facility ageing management plan.

The AMPs referenced in the LWR FSP templates often relate to programmes that apply to other parts of the nuclear power plant. Simpler or more relevant AMPs may be more applicable to smaller FSPs. Alternative AMPs can be devised by facility staff using the guidance provided in Section 4.4. The guidance includes an illustration based on an actual case involving ageing considerations for aluminium alloys in SNFSFs.

The templates are also applicable to systematic analysis for understanding and managing ageing in dry storage facilities. Typical types of material and components in dry storage casks are addressed in Section 5.3. Characteristics of other dry storage facilities are provided in Ref. [84]. The durability of concrete is an ageing issue in most dry storage facilities. Section 3.9 addresses ageing issues relevant to concrete. In addition, Section 4.1.1 includes an IAEA method for analysing ageing characteristics of concrete and references an IAEA report that addresses the details of concrete ageing.

Understanding ageing in dry storage facilities has the same basis as in wet storage facilities and involves systematic assessment of material, environments and potential ageing effects. Age related degradation has occurred in canisters, casks, dry wells and vaults when moisture ingress was not effectively precluded, involving, in some cases, vulnerable fuel cladding and/or fuel material and, in other cases, facility components.

5.5.2. Basis for managing ageing in fuel storage facilities

Effective ageing management requires commitment on the part of management to allocate sufficient financial and staff resources to facilitate the required actions. The generally benign appearance of SNFSF environments may result in an undercommitment to effective ageing management. Ageing management also requires a staff that is committed to understanding the potential vulnerabilities of facility SSCs, to being alert for unusual phenomena and to systematically applying the AMPs that are necessary to mitigate age related degradation of the facility SSCs.

6. HIGHLIGHTS OF THE COORDINATED RESEARCH PROJECT

The contributions of the participants in the CRP ranged widely, from specific technical investigations through facility assessments to reports of national procedures. The following sections reflect what the participants regard as the highlights of the work they contributed.

6.1. ARGENTINA

Argentina has two nuclear power plants in operation: Atucha I, a 370 MW(e) pressure vessel type reactor, and Embalse, a 600 MW(e) CANDU type reactor, both operated by Nucleoeléctrica Argentina SA. Atucha I has six pools in two separate buildings for spent fuel storage, all of which are lined with stainless steel. Fuel that fails in the reactor is stored in special cans. Good water chemistry control is maintained. In Embalse the spent fuel stays in the pool for seven years before transition to dry storage; the dry systems are above ground concrete silos, lined with carbon steel. The zircaloy clad fuel is stored in stainless steel baskets. The maximum storage time in the silos is about 20 years (in 2003), and the silos have worked well.

The main objective of the CRP study in Argentina was to develop a life management programme for the spent fuel and to analyse the different candidate technologies for dry storage of the commercial spent fuel from Atucha I. A formal process to systematically identify and evaluate the critical SSCs was employed for analysis.

The facilities are in good condition, but monitoring of concrete ageing will be necessary if the pools of Atucha I are to be operated after the nuclear power plant is closed. The decision will be made within the CNEA either to continue with wet storage in the reactor pool or to utilize dry storage. If it is necessary to use dry storage for the spent fuel of Atucha I, one of the economically more viable options may be a concrete cask system.

6.2. AUSTRALIA

ANSTO has stored research reactor fuel from operation of the HIFAR reactor since 1958. HIFAR is a DIDO class tank type reactor that utilizes a heavy water moderator and coolant. HIFAR fuel assemblies are fabricated from aluminium clad HEU aluminide alloy curved plates. ANSTO has operated both wet and dry storage facilities to store over 1600 spent fuel assemblies. Thus ANSTO has one of the longest histories of dry storage of research reactor fuel.

From its operating records, routine monitoring of its spent fuel storage facilities, previous detailed examinations of spent fuel and inspections performed prior to recent shipments, ANSTO is readily able to certify the soundness of the vast majority of the fuel it has in storage. By the end of 2003 more than 1500 spent fuel assemblies had been inspected and transported to France, the UK or the USA. There are a small number of assemblies that have been classified as unsound (those which were sectioned as part of a post-irradiation examination campaign) and a small number that have been designated as questionable and that require further investigation.

The assemblies that were sectioned for post-irradiation examination have facilitated a valuable assessment of the performance of aluminium clad aluminide fuel in dry storage. Aluminium has been shown to be acceptable for continued storage for many decades when stored under the correct conditions. In this particular case the sectioned elements are representative of the worst possible cases for dry storage, since (a) a corrosive agent was already present before commencement of storage, due to the presence of brazing flux from the fabrication process, and (b) the uranium–aluminium alloy fuel was directly exposed to the storage environment, due to sectioning to remove samples for metallographic examination in 1967.

Re-examination in 1983 and 1996 showed that all fuel plates maintained their physical integrity and showed little change over a 30 year period, including 25 years of dry storage. Close examination of sectioned plates did reveal areas of additional corrosion by 1996 compared with 1967, but only in the immediate vicinity of the cut edges, where samples removed exposed the fuel alloy to the storage environment, and adjacent to known occlusions of brazing alloy at the plate edges. It was also known that the assemblies had been exposed to a small quantity of water during their period of storage.

These observations are consistent with quantitative assessment performed by sip testing of spent fuel assemblies with small breaches of the cladding. ANSTO has been able to demonstrate that those fuel assemblies with known cladding breaches yield very low rates of activity release and are suitable for either further storage or shipment. After irradiation, solid and gaseous fission products are evenly distributed throughout the fuel. Since there is no interconnection, a minor cladding breach permits release of fission products only from the area immediately beneath the breach. Hence, with a minor cladding breach, the great bulk of an MTR fuel plate continues to be effectively contained by the remaining sound cladding. Furthermore, because the fuel alloy and cladding were fabricated from pure aluminium (grade 1050), there appears to be no preferential galvanic attack on the exposed fuel, even where cladding has been removed.

This work has demonstrated that aluminium clad fuel plates provide considerable resistance to degradation, even where defects are present. Any activity release is restricted to the area immediately beneath the breach and is therefore generally small. These observations provide considerable support for the viability of dry storage technologies for extended interim storage of aluminium clad research reactor fuel.

6.3. GERMANY

Metallic sealing systems for spent fuel and high active waste casks are qualified for storage periods of up to 40 years. Commonly used is the Helicoflex type of seal illustrated in Fig. 5. The sealing systems of storage casks have to guarantee the safe enclosure of the nuclear inventory over long periods. Depending on the type of nuclear waste (e.g. spent fuel or non-heat generating waste) and the different requirements of the storage facilities, different sealing systems can be used.

The BAM has been investigating details of the long term behaviour and tightness of such sealing systems under real storage conditions (moisture,

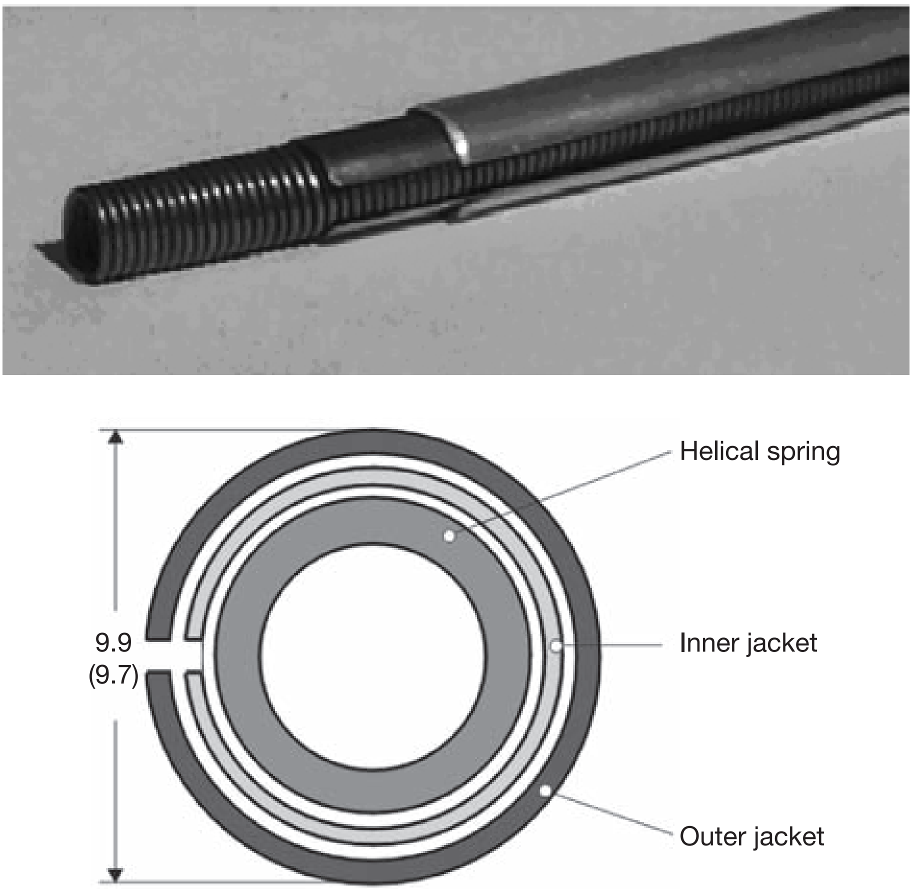

FIG. 5. Helicoflex type of seal for spent fuel and high active waste casks.

temperature, radiation) for many years. International experience with operating metallic sealing systems has shown that only very few problems have occurred. In such cases the presence of water in connection with atmospheric conditions (oxygen) has always been responsible. Under inert conditions no relevant material ageing effects and no decrease in the quality of the sealing systems are expected.

As part of this IAEA CRP the consequences of pool water remaining between the inner and outer jacket have been examined theoretically and experimentally. The tests used outer aluminium jackets, which were considered to be more vulnerable to corrosion. Eight test seals were prepared for use with two different water qualities (borated pool water and sodium chloride solution as an aggressive medium). The test seals were assembled in special flange systems giving the opportunity of periodic leakage rate measurement and a visual inspection of the surface of the gaskets. Four flange systems have each been stored at room temperature and at 80°C. Leakage rate measurements have been made every six months over a period of two years and are planned to continue to be made for several more years. So far with both water qualities and at both test temperatures there has been no increase in the leakage rate, and visual inspection has shown no traces of corrosion on the visible surface.

The theoretical analysis that BAM has done in the past predicted no significant corrosion of the inner or outer jacket of a Helicoflex type seal when pool water remains in the jacket gap of metal seals, as long as oxygen is excluded. So far, the long term tests, in which oxygen is extremely limited, are giving results that agree with the theoretical assessment.

Altogether, the operational experience and the experimental investigations have confirmed that metallic sealing systems do not show relevant degradation effects under normal cask storage conditions when operated under inert conditions. This includes limited pool water enclosures between the jackets of metallic seals, where no air or oxygen supply exists and relevant corrosion reactions are inhibited. Among other things, the results and conclusions may be interesting for development of storage facilities as well as for extension of storage periods, which may become important for existing facilities in the future.

6.4. KAZAKHSTAN

The objective of the work of the Institute of Nuclear Physics of the National Nuclear Centre in Kazakhstan was to characterize the condition of BN-350 fuel assemblies with irradiation temperatures in the range 280–400°C and irradiation damage up to 80 dpa that had been stored for 5–20 years in the reactor spent FSP.

Visual inspection, scanning electron microscopy, transmission electron microscopy and remote optical metallography were used to determine the depth of corrosion during wet storage and the extent of in-reactor sensitization of the irradiated stainless steel (08Cr16Ni11Mo3, 12Cr13Mo2Nb, 12Cr18Ni10Ti) shrouds and wire wraps (08Cr16Ni15Mo3) from representative fuel assemblies. Mechanical tests were also used to establish strength and ductility.

Metallographical investigations have revealed that the steel products corroded to a maximum depth of ~30 μm after storage in water. It was determined that the corrosion mechanism is mainly intergranular corrosion. The γ (austenite) can transform to α (martensite) by irradiation in the reactor and by deformation through handling of fuel out of the reactor. Corrosion of martensite is observed to be greater than that of austenite.

Studies of samples from spent fuel assemblies have shown that the material of the hexagonal shroud outer wall surface differs from that of the inner wall surface in colour, microhardness, smoothness, thickness of the corrosion layer and elemental composition of the corrosion layers and metal surfaces. Taken in conjunction with the results of mechanical property testing and microscopic examination, these findings suggest that when the BN-350 assemblies are dried and sealed into canisters, both standard core and blanket assemblies may be safely handled, transported and kept in dry storage. All the BN-350 fuel assemblies have now been placed in dry storage in stainless steel canisters with argon cover gas.

A lesson to be learnt from these studies is that to determine the physical and mechanical properties and especially the microstructure of large irradiated structures of stainless steel, such as the shroud of the BN-350 fuel, it is necessary to examine samples from different levels and both inside and outside surfaces. Variations in surface microstructure, which profoundly influence corrosion, may be associated with the initial conditions of deformation, surface preparation, gradients of temperature and neutron fluence or even interactions with coolant.

6.5. ROMANIA

Romania is planning to transfer CANDU type fuel from the Cernavoda nuclear power plant, after a suitable cooling period in wet storage, to a CANSTOR type intermediate dry storage facility. This is made up of about 30 modules, each module being essentially a concrete dome with 29 galvanized carbon steel fuel cylinders loaded from the top and cooled by natural convection through ten input and 12 output holes. Computer codes (e.g. ANSYS) have been used during the CRP to investigate the effects of covering the input/output holes of the CANSTOR on the volume of concrete subjected to freeze–thaw cycles. Calculations also carried out at the Centre of Technology and Engineering for Nuclear Projects (CITON) as part of the CRP show that the procedure of covering the input/output holes during the second part of the module lifetime will prolong the lifetime of the galvanized carbon steel fuel cylinders (due to reduced relative humidity obtained by trapping the winter air inside the module and the increased mean temperature of the cooling air), but will decrease the lifetime of defective fuel (due to accelerated oxidation of UO_2 exposed at cladding defects because of the mean annual temperature increase). Fortunately, however, this shortened lifetime of defective fuel is still much longer than the design life (50 years) of the storage module. The maximum temperature for the storage of defective fuel in air was considered in a Canadian assessment to be 160°C (but more recent studies have shown that it can be raised to 180°C), while the end of the life of the defective fuel was taken as a 0.6% increase in weight gain by oxidation, thus avoiding major disruption of fuel pin cladding.

In addition, optimizing the geometry and coating (black paint) of the fuel baskets and the configuration of covered/uncovered input/output holes can significantly reduce the maximum temperature of the fuel.

6.6. RUSSIAN FEDERATION

All Russian nuclear power plant and research reactors currently use wet storage facilities for the interim storage of spent nuclear fuel at the final stage of the fuel cycle. Former assessments of the wet storage of spent nuclear fuel indicated that a storage time of 30–40 years was permissible. Today the residence of some fuel types in wet storage has approached or even exceeded this period. An assessment of structural material and spent fuel has been conducted with the aim of extending the spent fuel storage time and service life of storage facilities. Wet storage of spent nuclear fuel will remain the predominant technology for the next 20 years.

The Physical Energy Institute (FEI) has carried out an inspection of a dry storage facility that has operated since 1960. Visual inspection of about 1000 spent fuel assemblies from a prototype transport nuclear installation stored for 20–30 years in the dry facility of the FEI, where the fuel is held in air, showed no sign of anomalous corrosion damage or mechanical deformation of the fuel assemblies.

The cladding material of fuel elements from the AM reactor (stainless steel) irradiated to a mean burnup of 17.6 MW·d/kg U after a 38 year storage in a dry facility showed high mechanical strength and satisfactory plasticity. At room temperature, at which the tests were carried out, the ultimate strength of the steel was 820–1095 MPa, the yield point was within 610–930 MPa and the common elongation was 4–37.5%.

Heavy and conventional concretes had no mechanical damage in the form of crumbs or cracks after 37 years of facility operation. It was found that gamma irradiation to 10 kGy at 300 K did not cause any change to the samples studied. After a 120 kGy dose the strength of cement stones (the major binder in concrete) increased by 30% compared with non-irradiated samples.

In a radiation induced thermal field, concrete evolves gaseous components — water vapour, hydrogen, CO_2, CO. This fact should be considered in justifications for reliable and safe storage of spent fuel in sealed ferroconcrete containers.

On the basis of data covering a 10–30 year storage period of ARBUS type spent fuel assemblies, an experimental dependence was identified for the release of ^{137}Cs from a spent fuel assembly during storage in water. By extrapolation, the critical period for wet storage of fuel assemblies with aluminium alloy cladding was determined as 41 ± 4 years. Experimental data are also given on the condition of the inner and outer surfaces of the aluminium tank used to store the spent fuel assemblies. The data suggest the development of pitting corrosion. The experimental data on the long term wet storage of spent fuel assemblies from the EWA, MARJA and ARBUS reactors suggest development of isolated corrosion in fuel rods under wet storage and allow one to forecast that the extrapolated time period for wet storage of aluminium clad fuel in water with good chemistry control is limited to 41 ± 4 years. The case in point is the degradation of fuel rods caused by (pitting or nodular) corrosion of the fuel rod cladding. The storage period for spent fuel assemblies in each case will be governed by the thickness of the fuel rod cladding specified for manufacture and by control of the water chemistry.

In addition, studies were carried out at an AFR storage facility of the Leningrad plant with RBMK-1000 (high power channel type reactor) spent fuel (Zr–1Nb cladding) stored in stainless steel cans to protect it from mechanical damage. The fuel in the cans is in a water environment. The water is not changed or cleaned. The Cl^- and F^- ions and various oxides are accumulated and create an acid environment that is dangerous from the corrosion viewpoint. According to the data obtained by RIAR, the main mechanism of RBMK fuel rod damage is the interaction between the rod cladding and debris.

New dry storage technologies for spent fuel are currently under development. These include the construction of dual purpose metal and concrete casks and pads for their installation at the nuclear power plant, the reprocessing plant (PO Mayak) and research institutes. The design and construction of the federal dry storage facility for spent nuclear fuel from different types of reactor is carried out at the Mining Chemical Combine (Krasnoyarsk). The capacity of the dry storage facility will be 33 000 tU; it will commence operation in 2006.

Knowledge of the ageing process of structural material is used to design and construct new spent fuel storage facilities. It will also be used to modify and improve existing facilities.

Minatom is actively involved in numerous programmes to investigate the maximum permissible storage periods for different types of spent fuel in wet and dry storage facilities and possibilities to extend the service life of existing facilities. In 2002 Minatom approved a further special research programme that was developed by RIAR (Dimitrovgrad) and ICC Nuclide (St. Petersburg). The scope of this programme includes investigation of the behaviour of the structural material of spent fuel from the MIR reactor, can, basket and cask during long term dry cask storage and ageing processes of the material.

The TUK-108/1 cask was developed for the storage and transport of spent fuel from submarines. A prototype cask was fabricated at the Izora plant for use in hot and cold testing. In 2002 the technical design of the test facility was developed and approved by Minatom. The prototype cask is being used in a new large scale test facility.

The objective of the test facility is to:

(a) Test the behaviour of the spent fuel inside the first unit of a metal–concrete cask (TUK-108/1);
(b) Test the behaviour of the structural material of the cask;
(c) Assess the safety of the cask dry storage technology;
(d) Increase the dry storage knowledge base.

The results of the programme will be used to support the safety analysis and licence application for a cask storage facility. A new large storage facility will be constructed on the RIAR site to provide a storage solution for spent fuel from all research reactors.

6.7. UNITED KINGDOM

Sellafield operations and management services activities in the area of ageing of material in spent fuel storage facilities over the duration of the CRP have been:

(a) Work in support of continued operations safety cases (COSCs) for existing storage facilities;
(b) Specific projects in support of the long term storage of AGR fuel.

6.7.1. Continued operations safety cases

The introduction of COSCs in 1998 presented the opportunity to report on the status of plant/material of varying ages that had experienced a variety of environmental conditions. This major exercise has substantiated all class 1 and 2 safety functions for individual plants through a review of build, modifications, design calculations and walk-down surveys.

An example of material substantiation is the assessment of the service condition of painted mild steel storage racks used in the site's first generation LWR storage pond. The approach taken has been to review the current condition established through in-pond closed-circuit television examinations with an assessment of the anticipated corrosion rate given the service to date and knowledge of the pond chemistry over the period. In this case occasional pitting corrosion due to coating degradation through operations or the quality of the original finish was observed. The degree of corrosion observed was, however, found to be significantly less than would have been anticipated for unpainted mild steel. This information has been used by engineers to establish whether the mechanical requirements are met at present and to assess performance for the required lifetime.

The main conclusions from the walk-down surveys and substantiation of the SSCs of the four operational spent fuel storage facilities subject to the COSC process were:

(a) The only common long term life limiting components were those of polymeric material used in the expansion joints of the storage ponds;
(b) Not unexpectedly, the older facilities show the greatest signs of ageing, mainly because the plant and the pond civil structure are exposed to the full environmental effects of the coastal location;
(c) Most of the ageing effects reported are a result of localized pitting where coatings have degraded.

6.7.2. Initial service life assessment for the Fuel Handling Plant (ponds)

In support of the long term storage of AGR fuel, an initial service life assessment of one of the site's spent fuel storage facilities, the Fuel Handling Plant, has been undertaken. After consideration of the major plant degradation processes influenced by the facility's local environs and build, apart from evidence of early age shrinkage cracking, the main causes of long term civil structure degradation were concluded most likely to arise from carbonation of the concrete, leading to reinforcement bar corrosion, and radiation hardening of non-concrete elements (polymers).

Experimental data and models for the Fuel Handling Plant's concrete specification have been used to predict the service life of the facility. These models predict that, away from cracks, carbonation induced corrosion is unlikely to lead to cracking of the cover concrete in sound concrete within 100 years.

Degradation data for the non-concrete components have also been analysed in combination with the expansion joint design and the predicted lifetime integrated dose. Although this predicts that the primary waterbar in the expansion joint will no longer perform its design duty, the secondary waterbar will be unaffected through radiation hardening for hundreds of years, given that it has 1.5 m of shielding.

6.8. UNITED STATES OF AMERICA

The US perspective includes inspections of more than 50 wet storage facilities in 20 countries and 11 dry storage facilities in five countries. Predominantly, ageing of material has been effectively addressed in these facilities. However, degradation of some types of material reached advanced stages in several facilities. Aluminium alloy fuel cladding and FSP components have degraded in several FSPs when exposed to waters with high conductivities and chloride contents. Stainless steel pipe welds in FSP coolant system pipes developed IGSCC through-wall cracks at nine locations in one facility. Carbon steel FSP components have been subject to formation of copious corrosion products, which were released to the water. Epoxy coatings on FSP walls have been observed to degrade when exposed to high radiation exposures. Uranium metal fuel corroded and released fission products and transuranic species to FSP waters when cladding failed, either by corrosion or by mechanical damage. Magnox fuel cladding degrades in open FSPs, even in well controlled water chemistry. Storage at high pH (pH13) extends durability in water, but the eventual option is reprocessing, although extended durability in dry storage has been demonstrated. Even in dry storage, leakage of water into the dry store caused corrosion of a few containers of Magnox fuel. The experience with the durability of spent fuel with zirconium alloy cladding has been without evidence of significant degradation for more than four decades, except in cases of mechanical damage.

In contrast to the examples of degraded material in FSPs cited above, durability has predominantly been the dominant observation for the leading types of material in FSPs. Aluminium alloy fuel cladding and facility components, stainless steel clad LWR fuel and facility components, and zircaloy clad fuel have all had satisfactory storage durability in FSPs for times approaching or exceeding four decades. However, aluminium durability depends on sustained water chemistry control. The zircaloys and non-sensitized stainless steels have been durable even in degraded water chemistry.

In dry storage facilities, the experience has also been predominantly positive. However, sensitized stainless steel cladding degraded in moist air environments at multiple facilities.

Major initiatives have been developed to ensure that age related degradation of material is systematically controlled in nuclear power plants. The NRC developed a basis to grant 20 year licence extensions to nuclear power plants; central to assurance of safe operation is the requirement to demonstrate that age related degradation will be effectively controlled over the lifetime of the nuclear power plant. The NRC has issued specific guidelines for understanding and managing ageing in the plant SSCs, including those in FSPs.

The nuclear power plants have mandates to implement programmes to control age related degradation and have allotted staff and resources to address the actions that are needed. The methods developed by the NRC to address age related degradation are based on two fundamental concepts: understanding ageing and effectively managing ageing. The associated actions are closely integrated. The guidance documents address each material in each relevant SSC in nuclear power plant fuel storage facilities.

7. EXCHANGE OF INFORMATION AND KNOW-HOW

7.1. QUESTIONNAIRE ADDRESSING AGEING OF STRUCTURES, SYSTEMS AND COMPONENTS IN FUEL STORAGE FACILITIES

One element of the CRP was the development and testing of a questionnaire addressing ageing in nuclear fuel storage facilities. The questionnaire was tested by CRP participants and was then revised to more effectively elicit input from future participants in a broader survey. The revised questionnaire is shown in Appendix III.

7.1.1. Specific results of the questionnaire test conducted under the auspices of the coordinated research project

The results of the test questionnaire are briefly summarized here to illustrate the range of information and to share the data and information obtained. The CRP participants that responded to the questionnaire were from seven countries, representing 15 facilities:

(a) Three power plants.
(b) Eight ISFSIs: five wet, two dry.
(c) Three test reactor sites (four reactors).
(d) One reprocessing plant.

The first facility started operation in 1947; other early start dates were 1957, 1959 and 1962. The latest start dates were 1992 (dry) and 1995 (wet). The first facility to be started was shut down in 1992; all other facilities remain in operation.

7.1.1.1. Wet storage

Water chemistries:

— Water conductivities (μS/cm): range, 0.5–3. Conclusion: good water chemistry control in all facilities included in the survey. Other facilities not included in the survey have operated with degraded chemistries (>100 μS/cm) for periods of service, resulting in accelerated corrosion of components [11].
— pH range: pH4.3–7.5. Conclusion: most facilities operate above pH5.5. Pressurized water reactors have boric acid pH control, resulting in low pHs at ambient temperatures. No ageing effects of low pH operation were noted.
— Chloride (ppm): <0.006–1.1.

Conclusion: Good chemistry control.

Fuel types:

— PHWR (zry-4 cladding);
— VVER-1000/440 (Zr–1Nb cladding);
— RBMK-1000 (Zr–1Nb cladding);
— Research reactor (aluminium alloy cladding);
— BWR (zry-2 cladding);

— PWR (zry-4 cladding);
— Magnox (magnesium alloy cladding);
— AGR (stainless steel cladding).

Conclusion: The predominant cladding types in storage were represented in the survey.

Principal material in wet storage facilities:

— Stainless steels (liners, heat exchangers, baskets/racks, fuel handling).
— Carbon steels (baskets/racks, liners, heat exchangers, fuel handling). A deficiency of the questionnaire is that it did not elicit whether the steels were coated.
— Aluminium alloys (baskets/racks, fuel handling tools).
— Epoxy (wall coatings).
— Concrete (structures).
— Polymers (ion exchange resins, seals).
— Boral (neutron absorber).
— Copper alloys (heat exchangers).

Conclusion: The predominant types of SSC material in wet storage facilities were represented in the responses to the survey.

Operating experience. Components that were replaced/upgraded and reasons:

— Storage tray lifting tool: after three years, due to material failure.
— Man bridge: after one year, due to electrical failure.
— Liner: upgraded after 15 years in one facility, 25 years in another facility (stainless steel material).
— Baskets/racks: replaced or upgraded (recoating of carbon steel components) after 15 years.
— Facility roof: scheduled replacement after 21–30 years.
— Crane hold down bolts: failed due to overstressing after 21 years; ductile failure of low strength bolt; replaced with bolts of higher strength but same ductility.
— Aluminium alloy racks: replaced due to corrosion of vulnerable welds.

Operating experience. Stability of operating conditions:

All facilities reported that operating conditions had not been changed during operation except to address specific material problems.

Conclusion: The questionnaires reflected minimal impacts of ageing mechanisms, presumably due to consistently good control of operating conditions.

Key ageing processes:

Fourteen ageing processes that are prominent in storage facilities are summarized in section 6 of the questionnaire (Appendix III). Responses to queries regarding the ageing processes are summarized below.

— One facility operator indicated past observation of the following ageing processes: cement phase corrosion, cracking of concrete and corrosion of carbon steel.
— Two facility operators indicated that the only past ageing process involved biological attack.
— One facility operator indicated that degradation of epoxy and radiolytic generation of aggressive radiolytic species were concerns.
— All other operators did not identify any past problematic ageing processes.
— Most facility operators acknowledged future concerns for several of the processes.
— Most facility operators indicated that some processes are subject to monitoring, including, collectively, IGSCC, gate seals, epoxy degradation, stainless steel liner failure, cement degradation, corrosion of rebars, corrosion of carbon steel, radiolytic generation of aggressive species and biological attack.
— Most facility operators identified ageing processes that they considered life limiting. However, this is an aspect of the questionnaire that requires further definition; for example, IGSCC was indicated as life limiting in several responses. However, other facilities not included in the survey have experienced IGSCC failures resulting in replacement of piping but not resulting in termination of facility operation. Even in facilities with serious ageing problems, none to our knowledge have resulted in shutdown of a facility, although in some facilities age related degradation has resulted in complex and expensive operations. It is recommended that the query relating to life limiting be eliminated, because it is subject to multiple interpretations.

Conclusion: The survey reflected minimal indications of past impacts of age related degradation resulting from suitable operating conditions. Operators indicated concerns about possible future impacts of several ageing processes, prompting initiatives to monitor and thereby anticipate and mitigate development of the processes.

Ageing management plan:

- Nine wet storage facility operators indicated that they have an ageing management plan. Operators of three facilities indicated that they do not have an ageing management plan.
- Those with ageing management plans indicated that the plans include planned maintenance and systematic monitoring, including, collectively: (a) annual examination of buildings and concrete structures; (b) external inspection of the pool liner, baskets and piping; and (c) monitoring of corrosion of material by use of samples. Water quality control was indicated.

Conclusion: The survey indicated that, predominantly, wet storage facility operators are taking a systematic approach to understanding and managing ageing.

7.1.1.2. Dry storage

Facility type:

- Underground concrete facility, commissioned in 1962;
- Ductile cast iron casks (CASTOR), first operated in 1992.

Cover gas:

- Underground facility: air.
- Metal casks: helium, inside casks.

Fuel type:

- Research reactor: aluminium alloy?;
- BWR, PWR, research reactor, high temperature gas cooled reactor (HTGR).

Maximum storage temperature:

- Fuel cladding: normally 20°C.
- Cask surface: 110°C.

Material:

- Underground facility: heat exchanger — stainless steel; baskets — stainless steel; racks — carbon steel; fuel handling — stainless steel; joints — stainless steel; monitoring system — stainless steel; cleanup system — stainless steel; structures — concrete.
- CASTOR cask: baskets/racks — stainless steel, aluminium alloy, zirconium alloy; neutron absorber, seals — stainless steel, aluminium alloy, IN/Nimonic; joints — stainless steel; leak monitoring — stainless steel; coating — epoxy; cleanup system — stainless steel.

Operating experience:

- Neither facility had component replacements or upgrades;
- Neither facility changed operating conditions.

Key ageing processes:

- Neither facility indicated impacts of ageing processes in past operations.
- The operator of the metal cask facility indicated future concerns about IGSCC of stainless steel, failure of seals, stainless steel liner failure, carbon steel corrosion and degradation of concrete.
- The operator of the metal cask facility indicated that the following are monitored: IGSCC, seals, stainless steel liners, concrete degradation and corrosion of carbon steel.
- As with wet storage operators, the metal cask facility operator identified life limiting ageing processes. However, to be explored is whether any of the concerns would result in shutdown of the facility.

Ageing management plan:

- One facility operator indicated that an ageing management plan exists. One operator indicated that an ageing management plan does not exist.
- The facility with a plan indicated that the plan includes planned maintenance and systematic monitoring. Planned maintenance includes annual examination of building and concrete structures, tubing (monitoring lines?) and research on concrete samples.
- Systematic monitoring includes radiation measurements and monitoring of aerosols.

Overall conclusion: The two facilities reflect the general range of types of material in similar facilities. These facilities have not experienced significant age related degradation.

7.2. INFORMATION ON REACTORS/STORAGE FACILITIES IN DECOMMISSIONING

When facilities are decommissioned, there are opportunities to observe, examine and even test material as it is removed from locations that are otherwise not accessible. Contact was made with several facilities that are in various stages of decommissioning in order to assess the extent and type of information that may have been accessed during decommissioning operations.

7.2.1. Receiving Basin for Offsite Fuel

The RBOF is located on the Savannah River Site in South Carolina, USA. The facility stored irradiated fuel from test reactors and demonstration power reactors, starting in 1963. The last fuel was shipped from the facility in October 2003. The facility is now empty of fuel, but decommissioning has not commenced. The sequence of the fuel removal operation is outlined in Ref. [85]. Over a time frame of 40 years, the facility received and stored many fuel types from foreign and domestic reactor operators.

7.2.1.1. Material and material durability

The observations on material behaviour in the 40 years of operation of the RBOF facility are based principally on visual observations. However, stainless steel and aluminium alloy specimens were removed from the RBOF and sent to the Pacific Northwest National Laboratory for metallographic examination after 15 years of service [16]. Aside from shallow pits on an aluminium alloy fuel assembly spacer, the material had not been subject to significant degradation. Aluminium clad fuel assemblies were occasionally brought near the surface of the pool for inspection using binoculars. No significant degradation was observed after the fuel had been stored for about a decade. Some fuel assembly material was inspected using underwater video. Corrosion coupons were inserted into the RBOF pool and were examined periodically. No significant corrosion was observed. Water chemistry control in the facility has been consistently good (see below).

The main FSP structure was concrete coated with Amercoat. The coating developed small bubbles but was functional for the full service period without repair. The fuel storage rack and bucket material was aluminium alloy. The racks and buckets were functional for 40 years without evidence of significant degradation.

The facility had areas that were lined with stainless steel; also, fuel elements were stored in stainless steel buckets, and there were stainless steel work tables. All stainless steel components were functional over the period of RBOF operation, without evidence of significant degradation.

Fuel assembly/element cladding material included aluminium alloys, zirconium alloys and stainless steels. There were no cases of serious degradation of the cladding material on fuel units exposed to RBOF water, with one exception, which involved zircaloy clad uranium metal fuel that had developed a through-wall cladding defect. Aqueous corrosion of the exposed uranium resulted in advanced degradation of the fuel assembly.

7.2.1.2. Water chemistry and temperature

Water chemistry parameters were maintained at a benign level over the history of facility operation. Nominal values were:

— Water conductivity: 0.7–1.5 μS/cm.
— pH: pH5.6–6.5.
— Chlorides: <100 ppb.
— Cu: 0.015 ppm.
— Hg: <2 ppb.
— Fe: 0.001–0.015 ppm.
— Water temperature range: 18–25°C.

7.2.1.3. Summary

Only a small fraction of the fuel and material exposed to the RBOF water was examined in detail, but the examinations and general observations suggested that material exposed to the RBOF water was subject to minimal degradation. The SSCs remained functional for the 40 year period of service. The key factor for successful operation was the consistent attention to water chemistry control (Section 7.2.1.2).

7.2.2. Decommissioning of test reactors

Operators of three US test reactors that were in various stages of decommissioning were contacted to assess whether useful insights on age related phenomena had been gained. In general, observation of material condition was not a planned element of decommissioning operations. Some useful insights were accessed in discussions with observers of the decommissioning operations. However, particularly in cases of known or suspected material degradation, a systematic approach to material assessment offers the prospect of gathering useful information on the ageing of material.

7.2.2.1. Facility A

The test reactor began operation in 1957. Final shutdown occurred in July 2003. During operation only two fuel assemblies (aluminium clad MTR type) developed pinhole leaks during operation. The fuel storage area was lined with tile. The fuel was stored on aluminium alloy racks. Dismantling of the reactor had not begun. There is no evidence of substantial material degradation, based on general observations.

7.2.2.2. Facility B

The reactor operated from 1960 to June 1998. The pool walls were concrete painted with epoxy. The coating had good durability. The principal operational problem was recurring leakage through the concrete wall. The cause was a major deficiency in the initial constitution of the concrete. Also, pipe penetrations were in direct contact with the concrete, which also resulted in leak locations at the pipe–concrete interfaces. The MTR aluminium clad fuel held up well during reactor service. Only three assemblies developed leaks. The underground steel pipe was in good condition. Water chemistry was controlled to <5 μS/cm. Removing surface contamination to releasable limits was troublesome on an aluminium alloy gate exposed to pool water. The contamination was removed in a basic solution but returned when re-exposed to pool water.

7.2.2.3. Facility C

The reactor began operation in 1972 and continued to operate as of late 2003. The aluminium alloy containment tank has been observed to develop intermittent small leaks, indicated by slow changes in the water level beyond the evaporation rate. The water chemistry is well controlled (<1 μS/cm, pH6–7). The leakage disappears about one month after it appears. The leakage rate is small. The location of the leak has not been identified.

7.2.3. Hanford C reactor decommissioning

The Hanford C reactor was decommissioned to an interim safe storage status that included removal of the fuel storage concrete structure [76]. Assessment of material condition was not an item on the decommissioning agenda. However, observers indicated that no evidence of degraded concrete or corroded metal rebars was observed while the fuel storage structure was dismantled. The reactor operated for 17 years. The dismantlement occurred 49 years after the reactor was commissioned. Data regarding the condition of the concrete and rebars could have been obtained at minimal cost if there had been an intent to assess the aged condition of the structure.

8. RECOMMENDED ACTIONS

The CRP participants offer the following recommendations to operators of fuel storage facilities, based on systematic consideration of storage experience and ageing issues.

(a) All SNFSFs should have effective AMPs that address the durabilities of their SSCs.
(b) The strategies for ageing management already adopted in some countries should be more widely utilized.
(c) It should be noted that in some countries strategies for ageing management evolve in cooperation with the regulators. Where this is not the case, regulators may find that AMPs drawn up by the SNFSF operators help with regulatory assessment of proposals for continued operation or life extension.
(d) Ageing management should, where possible, be addressed at the design stage of the SNFSF. Designers need quantitative guidance on rates of degradation of material.
(e) Awareness of the behaviour of the types of material in SNFSFs should be maintained during facility operation. Good behaviour, which often predominates, should be recorded as well as observations of degradation.
(f) Use should be made of opportunities afforded by maintenance, replacement and decommissioning activities to investigate material in support of programmes to understand the ageing of material.

(g) The prepared questionnaire should be circulated internationally to identify the range of experience, good and bad, with ageing of material in SNFSFs.
(h) Exchange of experience on ageing of material in SNFSFs, good and bad, should be encouraged for the benefit of the whole nuclear community.
(i) The results of the survey should be analysed to identify the spread of observations on material ageing problems, how common they are and how they are resolved. Compilation of a database may be helpful.
(j) The results of such a survey should be used to focus attention and resources where they are most needed.

Appendix I

BASIS IN THE UNITED STATES OF AMERICA FOR UNDERSTANDING AND MANAGING AGEING IN NUCLEAR FUEL STORAGE FACILITIES: FORMAT AND CONTENT

The format and content of ageing management guidance provided for passive components in the GALL report [5] are illustrated in Tables 3–7. Table 3 shows components in borated water PWR FSPs. Tables 4–7 address components in deionized water BWR FSPs. Storage pools in test/research reactors and defence reactors predominantly contain deionized water. However, some FSP waters have high pHs maintained in order to mitigate corrosion of certain fuel types. Other FSPs use treated river water. Still other facilities have maintained unusual chemistries, for example using chlorination to mitigate algae, resulting in high chloride contents in the pool waters. Facilities with chemistries not addressed by the GALL report need to modify the recommended programmes to ensure that ageing management is effective, even in unusual chemistries, or that changes to the chemistries are implemented if the environments are too aggressive to accommodate reasonable ageing management measures.

If the ageing management approach that is embodied in the GALL report is deemed to offer useful input to ageing management at a specific reactor, then it may be useful for the reactor staff to access details in the report [5], including the tabular information illustrated in the tables in this appendix and the content of AMPs that are indicated in the sixth column. However, if the facility staff intends to derive AMPs that are specific to their facility, there is sufficient guidance in this report, in the template provided in the tables in this appendix and in the illustration regarding development of an AMP for aluminium material in Section 4.4.

The designations in the tables refer to sections and table numbers in the GALL report, to facilitate access to the report for additional information.

TABLE 3. AUXILIARY SYSTEMS (SPENT FUEL POOL COOLING AND CLEANUP (PRESSURIZED WATER REACTOR))

Item	Structure	Material	Environment	Ageing effect/mechanism	AMP	Further evaluation
A3.3-b	Valves (check and hand valves)	Carbon steel with stainless steel cladding	Chemically treated borated water	Crack initiation and growth/stress corrosion cracking	Chapter XI.M2, Water Chemistry, for PWR primary water in EPRI TR-105714	No
A3.3.1	Body and bonnet					
A3.3-c	Valves (check and hand valves)	Body: carbon steel; bolting: carbon steel or low alloy steel	Air, leaking chemically treated borated water	Loss of material/boric acid corrosion	Chapter XI.M10, Boric Acid Corrosion	No
A3.3.1	Body and bonnet (external surface)					
A3.3.2	Closure bolting					
A3.3-d	Valves (hand valve only)	Elastomers	Chemically treated borated water	Hardening cracking/ elastomer degradation	A plant specific AMP that determines and assesses the qualified life of the linings in the environment is to be evaluated	Yes, plant specific
A3.3.3	Elastomer lining					
A3.4-a	Heat exchanger (serviced by closed cycle cooling water system)	Carbon steel	Shell side: closed cycle cooling water (treated water)	Loss of material/ general, pitting and crevice corrosion	Chapter XI.M21, Closed-Cycle Cooling Water System	No

TABLE 3. AUXILIARY SYSTEMS (SPENT FUEL POOL COOLING AND CLEANUP (PRESSURIZED WATER REACTOR)) (cont.)

Item	Structure	Material	Environment	Ageing effect/mechanism	AMP	Further evaluation
A3.4.1	Shell and access cover					
A3.4.2	Channel head and access cover					
A3.4-b	Heat exchanger (serviced by closed cycle cooling water system)	Carbon steel; low alloy steel	Air, leaking chemically treated borated water	Loss of material and boric acid corrosion	Chapter XI.M10, Boric Acid Corrosion	No
A3.4.1	Shell and access cover					
A3.4.2	Channel head and access cover (external surface)					
A3.4.3	Closure bolting					

TABLE 4. AUXILIARY SYSTEMS (SPENT FUEL POOL COOLING AND CLEANUP (BOILING WATER REACTOR))

Item	Structure	Material	Environment	Ageing effect/mechanism	AMP	Further evaluation
A4.1-a	Piping	Stainless steel	Chemically treated oxygenated water up to 50°C	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, Water Chemistry, for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control, see Chapter XI.M32, One-Time Inspection, for an acceptable verification programme	Yes, detection of ageing effects is to be evaluated
A4.1.1	Piping, fittings and flanges					
A4.2-a	Filter	Stainless steel; carbon steel with elastomer lining or stainless steel cladding	Chemically treated oxygenated water up to 50°C	Loss of material/ pitting and crevice corrosion (only for carbon steel after lining/cladding degradation)	Chapter XI.M2, Water Chemistry, for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control, see Chapter XI.M32, One-Time Inspection, for an acceptable verification programme	Yes, detection of ageing effects is to be evaluated
A4.2.1	Housing					

TABLE 4. AUXILIARY SYSTEMS (SPENT FUEL POOL COOLING AND CLEANUP (BOILING WATER REACTOR)) (cont.)

Item	Structure	Material	Environment	Ageing effect/mechanism	AMP	Further evaluation
A4.2-b	Filter	Elastomers	Chemically treated oxygenated water up to 50°C	Hardening, cracking/ elastomer degradation	A plant specific AMP that determines and assesses the qualified life of the linings in the environment is to be evaluated	Yes, plant specific
A4.2.2	Elastomer lining					
A4.3-a	Valves (check hand valves) Body and bonnet	Stainless steel; carbon steel with elastomer lining or stainless steel cladding	Chemically treated oxygenated water up to 50°C	Loss of material/ pitting and crevice corrosion (only for carbon steel after lining/cladding degradation)	Chapter XI.M2, Water Chemistry, for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control, see Chapter XI.M32, One-Time Inspection, for an acceptable verification programme	Yes, detection of ageing effects is to be evaluated

TABLE 5. AUXILIARY SYSTEMS (SPENT FUEL POOL COOLING AND CLEANUP (BOILING WATER REACTOR))

Item	Structure	Material	Environment	Ageing effect/mechanism	AMP	Further evaluation
A4.3-b	Valves (hand valve only)	Elastomer	Chemically treated oxygenated water up to 50°C	Hardening, cracking/ elastomer degradation	A plant specific AMP that determines and assesses the qualified life of the linings in the environment is to be evaluated	Yes, plant specific
A4.3.2	Elastomer lining					
A4.4-a	Heat exchange (service by closed cycle cooling water system)	Carbon steel	Shell side: closed cycle cooling water	Loss of material/ general pitting and crevice corrosion	Chapter XI.M2, Closed-Cycle Cooling Water System	No
A4.4.1	Shell and access cover					
A4.4.2	Channel head and access cover					
A4.4.3	Tubes					

TABLE 5. AUXILIARY SYSTEMS (SPENT FUEL POOL COOLING AND CLEANUP (BOILING WATER REACTOR)) (cont.)

Item	Structure	Material	Environment	Ageing effect/mechanism	AMP	Further evaluation
A4.4-b	Heat exchange (service by closed cycle cooling water system)	Channel head and access cover: stainless steel; carbon steel with stainless steel cladding; tubes and tube sheet: stainless steel	Dematerialized oxygenated water	Loss of material/ general pitting and crevice corrosion	Chapter XI.M2, Water Chemistry, for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control, see Chapter XI.M32, One-Time Inspection, for an acceptable verification programme	Yes, detection of ageing effects is to be evaluated
A4.4.2	Channel head and access cover					
A4.4.3	Tubes					
A4.4.4	Tube sheets					
A4.5-a	Ion exchanger (demineralizer)	Stainless steel; carbon steel with elastomer lining	Demineralized oxygenated water	Loss of material/ pitting and crevice corrosion (only for carbon steel after lining degradation)	Chapter XI.M2, Water Chemistry, for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control, see Chapter XI.M32, One-Time Inspection, for an acceptable verification programme	Yes, detection of ageing effects is to be evaluated

TABLE 5. AUXILIARY SYSTEMS (SPENT FUEL POOL COOLING AND CLEANUP (BOILING WATER REACTOR)) (cont.)

Item	Structure	Material	Environment	Ageing effect/mechanism	AMP	Further evaluation
A4.5.1	Shell					
A4.5.2	Nozzles					

TABLE 6. AUXILIARY SYSTEMS (SPENT FUEL POOL COOLING AND CLEANUP (BOILING WATER REACTOR))

Item	Structure	Material	Environment	Ageing effect/mechanism	AMP	Further evaluation
A4.5-b	Ion exchanger (demineralizer)	Elastomers	Chemically treated oxygenated water up to 50°C	Hardening, cracking/ elastomer degradation	A plant specific AMP that determines and assesses the qualified life of the linings in the environment is to be evaluated	Yes, plant specific
A4.5.3	Elastomer lining					
A4.6-a	Pump	Stainless steel; carbon steel (with stainless steel cladding)	Demineralized oxygenated water	Loss of material and crevice corrosion	Chapter XI.M2, Water Chemistry, for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control, see Chapter XI.M32, One-Time Inspection, for an acceptable verification programme	Yes, detection of ageing effects is to be evaluated
A4.6.1	Casing					

TABLE 7. AUXILIARY SYSTEMS (A2 SPENT FUEL STORAGE)

Item	Structure	Material	Environment	Ageing effect/mechanism	AMP	Further evaluation
A2.1-a	Spent fuel storage racks	Boraflex	Chemically treated oxygenated (BWR) or borated (PWR) water	Reduction of neutron absorbing capacity/Boraflex degradation	Chapter XI.M22, Boraflex Monitoring	No
A2.1.1	Neutron absorbing sheets					
A.2-b	Spent fuel storage racks	Boral boron steel	Chemically treated oxygenated (BWR) or borated (PWR) water	Reduction of neutron absorbing capacity/general corrosion	A plant specific AMP is to be evaluated	Yes, plant specific
A.2.1	Neutron absorbing sheets					
A2.1-c	Spent fuel storage racks	Stainless steel	Chemically treated oxygenated (BWR) or borated (PWR) water	Crack initiation and growth/stress corrosion cracking	Chapter XI.M2, Water Chemistry, for BWR water in BWRVIP-29 (EPRI TR-103515) or PWR primary water in EPRI TR-105714	No
A2.1.2	Storage racks					

Appendix II

TERMS FOR DEGRADATION, LIFE CYCLE AND AGEING MANAGEMENT

The terms for degradation, life cycle and ageing management given in Table 8 are taken from the Glossary of Nuclear Power Plant Ageing [3].

TABLE 8. PRINCIPAL COMMON AGEING TERMS LISTED BY CATEGORY (SYNONYMS NOT INCLUDED)

Degradation	
Causes of degradation	Degradation/ageing
Condition:	Characteristic
• Service conditions	Condition:
• Pre-service conditions	• Degraded condition
• Environmental conditions	Ageing:
• Functional conditions	• Natural ageing
• Operating conditions	• Premature ageing
• Normal conditions	• Normal ageing
• Error induced conditions	• Artificial ageing
• Design basis event	• Accelerated ageing
• Design basis event conditions	• Age conditioning
• Design conditions	Ageing mechanism
Stressor:	Ageing effects:
• Normal stressor	• Combined effects
• Error induced stressor	• Simultaneous effects
• Design basis event stressor	• Synergistic effects
	Degradation:
	• Ageing degradation
	• Normal ageing degradation
	• Error-induced ageing degradation
	Ageing assessment

TABLE 8. PRINCIPAL COMMON AGEING TERMS LISTED BY CATEGORY (SYNONYMS NOT INCLUDED) (cont.)

Life cycle	
Life	**Failure**
Age Time in service Life: • Installed life • Service life • Remaining life • Design life • Remaining design life • Qualified life Retirement	Failure: • Degraded failure • Complete failure • Random failure • Common cause failure • Common mode failure • Wear-out Failure cause: • Root cause • Failure mechanism • Failure mode Failure analysis: • Failure evaluation • Failure modes and effects analysis • Failure trending Mean time between failures

Ageing management	
Maintenance	**Condition assessment**
Ageing management Life management Maintenance: • Preventive maintenance • Periodic maintenance • Planned maintenance • Corrective maintenance Repair Refurbishment Overhaul Replacement Servicing Post-maintenance testing Rework	Predictive maintenance In-service inspection In-service test Surveillance Surveillance requirements Condition monitoring Condition indicator Functional indicator Testing Diagnosis Acceptance criterion

Appendix III

QUESTIONNAIRE ADDRESSING AGEING IN NUCLEAR FUEL STORAGE FACILITIES

AGEING OF MATERIAL AT SPENT FUEL STORAGE FACILITIES

Please submit one questionnaire for each storage facility.

1. STORAGE FACILITY

Facility name:	
Country:	
Owner:	
Operator:	
Date of commencement of operation:	
Planned shutdown date:	
Design life of facility:	
Wet storage	❏
Dry storage	❏

2. CONTACT

Who is the correct person to contact for your site? Please give name, address, telephone number, fax number and email address.

Contact person:			
Title:			
Institute:			
Address:			
City and state:			
Country:		Postal code:	
Telephone number:		Fax number:	
Email address:			

3. STORAGE FACILITY DETAILS

Please indicate the type of facilities that you operate.

Wet storage	❑

Please supply details of pool water chemistry.

pH		Boron (ppm)		Chloride (ppm)	
Conductivity (μS/cm)		Lithium (ppm)		Sulphate (ppm)	
Activity ^{137}Cs (Bq/L)					

Other? ____

Dry storage	❑		

Please indicate the specific type of dry storage that your facility uses.

Type	
Cover gas	
Maximum licensed fuel clad temperature (°C)	

Please indicate the types of fuel stored in your facility:

Fuel types	BWR	❑	PWR	❑	AGR	❑
	MAGNOX	❑	PHWR	❑	VVER	❑
	RBMK	❑	RESEARCH	❑	HTGR	❑

Other fuel types? ____

What is the maximum design basis burnup of the fuel in your facility (GW·d/t HM)?

____

4. MATERIAL

Please indicate which types of material are currently in use in your facility.

Material	Alloy/type	Components									
		Structures	Liners	Tubing/ heat exchanger	Baskets/ racks	Fuel handling	Seals[a]	Joints[b]	Leakage system	Monitoring system	Cleanup system
Stainless steel											
Carbon steel											
Aluminium alloy											
Zirconium alloy											
Concrete											
Polymers											
Neutron absorber											
Copper alloys											
Epoxy											
Others											

[a] In dry storage facilities.
[b] In wet storage facilities.

5. OPERATING EXPERIENCE

Have you repaired, refurbished or replaced components or structures in your facility? Please specify.

(a) Was it a scheduled maintenance[1] (SM) or an upgrade (U)?
(b) Was the repair refurbishment or replacement due to an unexpected failure or degradation?
(c) How long after commissioning was the replacement or upgrade made?
(d) Did you perform any examinations including root cause analysis?

Component	a	b	c	d
	SM ❑ U ❑	Yes ❑ No ❑	_______ Years	Yes ❑ No ❑
	SM ❑ U ❑	Yes ❑ No ❑	_______ Years	Yes ❑ No ❑
	SM ❑ U ❑	Yes ❑ No ❑	_______ Years	Yes ❑ No ❑
	SM ❑ U ❑	Yes ❑ No ❑	_______ Years	Yes ❑ No ❑
	SM ❑ U ❑	Yes ❑ No ❑	_______ Years	Yes ❑ No ❑
	SM ❑ U ❑	Yes ❑ No ❑	_______ Years	Yes ❑ No ❑
	SM ❑ U ❑	Yes ❑ No ❑	_______ Years	Yes ❑ No ❑
	SM ❑ U ❑	Yes ❑ No ❑	_______ Years	Yes ❑ No ❑
	SM ❑ U ❑	Yes ❑ No ❑	_______ Years	Yes ❑ No ❑
	SM ❑ U ❑	Yes ❑ No ❑	_______ Years	Yes ❑ No ❑

If you answered yes in column b please provide details about each failure below or on a separate sheet:

Have you changed operating conditions? Yes ❑ No ❑

If yes, then please provide further details below:

[1] Maintenance includes preventative maintenance, periodic maintenance, planned maintenance and corrective maintenance.

6. KEY AGEING PROCESSES

The following are examples of ageing phenomena that have occurred in spent fuel pools and dry storage facilities. Please indicate which of the following have been encountered at your facility, which you are concerned about for the future, which you are monitoring or tracking and which you think might be life limiting. Please add additional ageing processes that apply to your facility.

	Encountered in the past?	Future concern?	Monitored?	Corrective action identified?
Corrosion of piping				
Failure of gate seals				
Degradation of neutron absorbers				
Corrosion of aluminium racks				
Degradation of organic coatings				
Stainless steel liner breach				
Evidence of concrete degradation				
Corrosion of rebar				
Corrosion of carbon steel				
Radiolytic generation of aggressive chemical species				
Galvanic attack				
Biological attack				
Dry storage cask seal degradation				
Copper corrosion				
Pitting corrosion				

Stressors are agents or stimuli that stem from service or pre-service conditions that either singly or in combination can cause ageing degradation. Examples: heat, radiation, humidity, pressure, vibration, aggressive species.

If you have experienced any signs of ageing in your facility, expected or unexpected, would you please fill in the attached Ageing Sequence sheet. If you have had several such experiences, please use one sheet for each process identified.

Please list in the space provided below the corrective actions identified in the table above.

7. AGEING MANAGEMENT PLAN

Do you have a documented ageing management plan? Yes ❑ No ❑

Do you conduct activities that contribute to ageing management in your facility? Yes ❑ No ❑

What specific activities are carried out?

Have you gained any information from your monitoring which would be of value to material ageing studies?

8. CONTINGENCY PLAN

What is your plan to relocate your fuel if degradation of your fuel or storage facility warrants it?

9. AGEING SEQUENCE

This sequence facilitates understanding of a given ageing process. For each ageing process that you have identified in 6 above, please provide information on the attached sheet. An example is provided below.

Name of ageing process as recorded in 6: Cracking of spent fuel pool pipe.

1. Pre-service conditions	Chloride contamination during pre-service storage
2. Operating conditions	Standby spent fuel pool storage system
3. Ageing stressors	Stressed pipe Thermally sensitized Chemical contamination (Cl^{-})
4. Ageing mechanisms	Intragranular stress corrosion cracking
5. Ageing effects	Pipe wall cracking
6. Ageing degradation	Through-wall cracking in heat affected zones
7. Degraded condition or failure	Leakage of radioactive water requiring pipe replacement

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ABBREVIATIONS

AFR	away from reactor
AGR	advanced gas reactor
AMP	ageing management programme
ANSTO	Australian Nuclear Science and Technology Organisation
BWR	boiling water reactor
CANDU	Canada deuterium–uranium reactor
COSC	continued operations safety case
CRP	coordinated research project
FSP	fuel storage pool
GALL	Generic Aging Lessons Learned
HEU	high enriched uranium
IGSCC	intergranular stress corrosion cracking
ISFSI	independent spent fuel storage installation
INEEL	Idaho National Engineering and Environmental Laboratory
LEU	low enriched uranium
LWR	light water reactor
MTR	material testing reactor
NRC	Nuclear Regulatory Commission
PHWR	pressurized heavy water reactor
PWR	pressurized water reactor
RBMK	high power channel type reactor
RBOF	Receiving Basin for Offsite Fuel
SNFSF	spent nuclear fuel storage facility
SSC	structure, system or component
THORP	Thermal Oxide Reprocessing Plant

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The management of ageing is of key importance in many countries for the owners and operators of many facilities, including power reactors. There is a large measure of agreement on the general approach, which is summarized in this report. It includes sections on the ageing of selected types of material and on the management of ageing. This report also includes a brief section on specific approaches in the context of fuel storage facilities and some specific recommendations. Moreover, the content has been broadened to try to appeal to those who may be in the early stages of setting up ageing management programmes either for new or older fuel storage facilities.

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