IAEA-TECDOC-1871

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IAEA-TECDOC-1871

Material Properties Database for Irradiated Core Structural Components for Lifetime Management for Long Term Operation of Research Reactors

Report of a Coordinated Research Project

IAEA
International Atomic Energy Agency

MATERIAL PROPERTIES DATABASE FOR IRRADIATED CORE STRUCTURAL COMPONENTS FOR LIFETIME MANAGEMENT FOR LONG TERM OPERATION OF RESEARCH REACTORS

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REPORT OF A COORDINATED RESEARCH PROJECT

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2019

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Printed by the IAEA in Austria
May 2019

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Material properties database for irradiated core structural components for lifetime management for long term operation of research reactors : report of a Coordinated Research Project (2014-2018) / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2019. | Series: IAEA TECDOC series, ISSN 1011–4289 ; no. 1871 | Includes bibliographical references.
Identifiers: IAEAL 19-01238 | ISBN 978–92–0–102519–7 (paperback : alk. paper)
Subjects: LCSH: Nuclear engineering — Safety measures. | Materials — Databases. | Nuclear reactors — Cores.

FOREWORD

Research reactors play a key role in the development of the peaceful uses of nuclear energy and technology. However, more than 50% of operating research reactors are over 45 years old, and continued operation has to be carefully assessed, especially from the point of view of structural materials. In many instances, data on radiation induced changes of research reactor core materials resulting from exposure to very high neutron fluence — needed to evaluate the reliability of research reactor core components and establish the technical basis for continuous operation — are not available.

Neutron based irradiation programmes at research reactors cover a wide range of experiments, including irradiation of sample materials to determine the effects of neutrons on their microstructural and mechanical properties under various fluences. Most of these irradiations are carried out under simulated conditions to represent the environment of conventional nuclear power plants. Hence the information on research reactor structural materials is more limited. To support extended operation of research reactors, a predictive and/or preventive maintenance programme based on reliable data from inspection and irradiation programmes is critical.

To address this need, the IAEA has developed a Research Reactor Material Properties Database containing selected relevant data on irradiated core and core support structures provided by Member States. The data have been validated and organized according to the components in which the materials are used. The present publication describes the processes that led to the design, structure and collation of information for the database, and discusses uncertainties regarding the risk and reliability of the core structural materials of research reactors. This database can also be used to support the safe operation of existing research reactors and design of new research reactors.

The IAEA is grateful to the experts who contributed to the analysis, compilation and validation of the Research Reactor Material Properties Database, and to those experts who supported the drafting and review of this publication. The IAEA thanks B. van der Schaaf (Netherlands) for his valuable support, guidance and chairing the research coordination meetings. The IAEA officers responsible for this publication were D.V.H. Rao of the Division of Nuclear Installation Safety and R.C. Sharma of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

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1. INTRODUCTION

1.1. BACKGROUND

As per the IAEA Research Reactor Database, more than 50% of operating research reactors (RRs) are over 45 years old and 30% are over 50 years old. Continued safe and efficient operation depends on, among others, the predictability of the behaviour of structural materials of major components such as reactor vessels and core support structures, many of which are difficult to replace or refurbish. Management of the ageing process requires predictions of the behaviour of materials subjected to irradiation.

Ageing management of RRs includes a comprehensive effort of engineering, operation, inspection and maintenance strategy to ensure the reliability and availability of structures, systems and components (SSCs) that are important to safety. Ageing related degradation mechanisms can result in unplanned outages, as well as lengthy shutdowns, and potentially the need for additional regulatory approvals. These burdens can be mitigated by utilizing available data and implementing appropriate maintenance and surveillance programmes. In many instances data for radiation-induced damages of RR core structural component materials resulted from exposure to a high neutron fluence (in the case of aluminium, for example, 10^{23} $n{\cdot}cm^{-2}$ thermal neutron fluence resulting in the transmutation of about 1% aluminium to silicon) are not generally available because the materials and operating conditions are diverse and specific to each reactor type. The operating temperature of RRs is generally below 85°C, and hence a large amount of data generated for nuclear power plant (NPP) applications cannot be directly applied to RRs.

A structured database will support the aging assessment of core materials used in the RRs for their continued safe operation and life extension. The database can be used by RR designers, safety analysts, operators and regulators to help in prediction of ageing related degradation. This would be useful in minimizing unpredicted failures of core components and in mitigating the undesirable consequences of lengthy and costly unplanned shutdowns.

Therefore, effective sharing of experimental results related to the materials used in core structural components is needed to evaluate the safety, risk and reliability of reactor core components. Some RR may be required to extend their operating life to provide unique irradiation capabilities. Consequently, the behaviour of core materials need to be better understood so that timely action can be taken for refurbishment/replacement of affected components. In addition, predicting the life-limiting components (non-replaceable) will contribute considerably to the decision-making process on operation and replacement schedules.

Recognizing the need for a reliable and validated database for irradiated core structural components of RRs, IAEA launched a Coordinated Research Project (CRP) entitled "Establishment of Material Properties Database for Irradiated Core Structural Components for Continued Safe Operation and Lifetime Extension of Ageing Research Reactors". The CRP was joined by 11 organizations from 11 Member States (MSs) who signed research contracts or agreements. The CRP directly provided a forum for the establishment of the Research Reactor Material Properties Database (RRMPDB) for irradiated core structural components. The database is a compilation of data on changes in material properties from RR, operators' input, comprehensive literature reviews and experimental data from RRs.

The CRP has identified activities to address the gaps of the database. The database is managed by the IAEA with controlled access to participated MSs.

1.2. OBJECTIVES

This publication is an output of a CRP organized (during 2012–2017) by the IAEA. The objective of this publication is to provide a comprehensive background and relevant information on the properties of irradiated core structural materials for RRs. This publication is to help the RR community to predict ageing related degradation to support safe and reliable long-term operation and lifetime extension of operating RRs, as well as to the design of new RRs. This publication describes results of the CRP, the process of data collection, validation and creation of the database and its maintenance and utilization.

Specific objectives of the CRP were:

- To obtain the data (from dedicated experiments on irradiated specimens as well as tests on components taken out-of-service) on relevant material properties from RRs;
- To collect and collate irradiation data through comprehensive literature reviews;
- To evaluate and identify the data gaps;
- To compile an evaluated, reviewed and assessed database that can be used for sharing relevant information among interested MSs;
- To establish structure and content of the database with provision for incorporation of new data in future as it becomes available;
- To specify further activities needed to address the identified data gaps of the database for potential follow-up activities by the IAEA.

The users of the RRMPDB are identified mainly as:

- All RR operators;
- RR design organizations;
- Regulators;
- Research organizations/universities.

1.3. SCOPE

The publication analyzed the collected data on materials typically used in RRs, as well as materials being considered for future applications. Such materials include aluminium alloys, zirconium alloys, beryllium, stainless steels, graphite, concrete and plastic/organic materials. The properties considered (from recognized international refereed sources) are as follows:

- Physical and chemical properties: density, microstructure and chemical composition as a function of fluence (n,γ).
- Mechanical/engineering properties: tensile strength, ductility, fracture toughness, corrosion resistance, fatigue strength, crack propagation rates as a function of fluence (n,γ).

Good practices in material applications are discussed. These include suitability for the application and compatibility with other materials.

1.4. STRUCTURE

The publication consists of 5 sections, 6 appendices and 2 annexes:

- Section 2 of this publication provides a description of features of the materials used in core structural components.

- Section 3 includes a review of the data sources for these materials.

- Section 4 describes the methodology of data validation and identification of the data gaps.

- Section 5 provides an overview of the RRMPDB structure and guidance for users.

Appendices 1–6 provide a list of the references for each material group that existed in the database at the time of publication of the TECDOC. Annex 1 is a list of typical components of RRs that are subjected to the effects of radiation. Annex 2 is a list of the ideal set of materials properties that are desirable.

2. MATERIALS FOR RR CORE STRUCTURAL COMPONENTS

For the design of a RR and its life management, data on the physical properties of the core materials used, including changes due to irradiation, are required. The environment in a RR varies depending on the operating temperature, power level and neutron flux, etc. Material selection takes into account not only the environment but also the type of the core component. Core structural materials commonly used include aluminium, zirconium, beryllium and its alloys, steels, graphite and concrete. The materials selected for the RRMPDB have quite different properties. This is not surprising, considering the large differences between the functions of different components and their respective environments. The component determines the selection of the material with the most appropriate behaviour for a specific set of design requirements.

Not only do the inherent properties of these materials exhibit a wide spectrum, but also the effects of neutron and gamma irradiation differ to a high degree. The literature and datasets for each material describe the materials property changes brought about by radiation exposure. Interpolation or extrapolation is often necessary to estimate the condition of a particular component in a specific RR. Understanding radiation damage and ageing mechanisms that act on the materials in RR SSCs is important. This section introduces the main effects of radiation for all materials included in the RRMPDB. The user will thus become aware that not only the mechanical properties test data are important, but, for example, the fractographic character of mechanically tested specimens could often provide valuable information on the status of reactor structural materials. For instance, optical and electron microscopies can also reveal the physical and chemical properties of a material after a long-term of services. Thus, the microscopic information contributed by the Member States (MSs) has been included in the database.

2.1. ALUMINIUM AND ITS ALLOYS

Aluminium and its alloys such as the 1-, 2-, 5- and 6- thousand series, are widely used as materials for construction of core components and core support structures, as they possess good radiation damage resistance and show very little swelling up to high thermal and fast neutron fluence. Their water corrosion resistance is also good when pH and water chemistry are maintained within desirable limits. The mechanical properties of the 5XXX and 6XXX series are highly adequate in the temperature range of RR operation. Their formability and relatively easy of welding, with limiting prior- and post-welding heat treatments required, give the designers a relatively large-degree-of-freedom in designing complex or irregularly shaped components. Aluminium and its alloys have been frequently selected for vessels and piping related components near high neutron fluxes in RRs such as pressure vessels, low pressure tanks, connecting pipes, beam port nozzles, core support structures and poolside irradiation facilities. Aluminium core supporting components such as grid supporting cores, core boxes and other internal support structures can be found in many RRs. Components with a more or less basic tube-like shape such as control guide devices, experimental facilities and isotope production holders and tools are also often constructed from aluminium and its alloys. Related aluminium components that experience high neutron radiation levels, such as cold neutron sources, thermal columns and neutron guide tubes, have operated as intended in RRs. In several reactors heavy water tanks and their accessories have been constructed of aluminium and its 5XXX and 6XXX series alloys. The list in Section 5.1.2 indicates well the wide application of 5XXX and 6XXX series aluminium alloys as RR core structural materials.

Aluminium alloys have a high tolerance to radiation effects when irradiated at ambient temperatures due to their low melting point (T_m). This is because the homologous temperature of aluminium alloys at room temperature is around 0.32. In metals it is known that noticeable thermal diffusion of vacancies occurs at homologous temperatures above 0.3. This can be compared with ~0.175 for austenitic steel, ~0.17for ferritic steel and ~0.14 for α–Zr. This thermally induced movement of vacancies at room temperature promotes mutual recombination of vacancies and interstitials resulting in a lower density of point defect clusters that initiate damage to microstructure.

In particular, 5XXX and 6XXX series alloys exhibit a good combination of mechanical, thermal, corrosion resistance and irradiation swelling resistance properties in a RR environment, which makes these alloys a suitable choice for in-core structures and reactor vessel components of RRs. The components of these reactors can experience high neutron fluence, up to several 10^{23} $n{\cdot}cm^{-2}$, during their operational life. Substantial damage to the material's microstructure and mechanical properties can occur under these high fluence conditions.

2.1.1. Literature on Aluminium alloys

A substantial amount of literature has been published on the irradiation behaviour of aluminium alloys [1–12]. The available dataset on 6XXX series alloys is considerably larger due to their widespread use in several RRs and cold neutron sources [1, 3–9]. On the other hand, only limited data was found on 5XXX series alloys [2, 8, 10, 11]. The published data from the surveillance programme of the HFR vessel in the Netherlands is also included in this review [12]. Although 5XXX and 6XXX series alloys are fundamentally different in their as-fabricated microstructure and properties, the data on irradiated 6XXX series aluminium alloys has great relevance to 5XXX series data. The 5XXX series alloys slowly transmute chemically into 6XXX series alloys with neutron irradiation due to the Si content produced by transmutation [8].

2.1.2. Microstructure of unirradiated 5XXX and 6XXX series aluminium alloys

The microstructure of 6XXX series alloys is carefully engineered by a suitable age hardening treatment to form coherent precipitates within the matrix to obtain the required mechanical properties [13, 14]. On the other hand, the manufacture of the 5XXX series yields very limited precipitates within the matrix. The Mg atoms present in the solid solution provide the required strength properties.

2.1.3. Irradiation induced damage mechanisms in aluminium alloys

The damage caused by neutron irradiation is the major degradation mechanism leading to irradiation hardening and embrittlement of aluminium alloys. Both thermal and fast neutrons cause damage in aluminium alloys. Displacement damage by fast neutrons and transmutation damage by both thermal and fast neutrons are the two major damage mechanisms in irradiated aluminium alloys [2, 8, 9]. The relative contribution of these different damage mechanisms and the resulting impact on mechanical properties depend mainly on: the alloy compositions and micro-structures, thermal to fast fluence ratio (TFR) and the irradiation temperature. Figure 1 shows schematically the damage mechanisms and resulting effects of neutron irradiation on aluminium alloys.

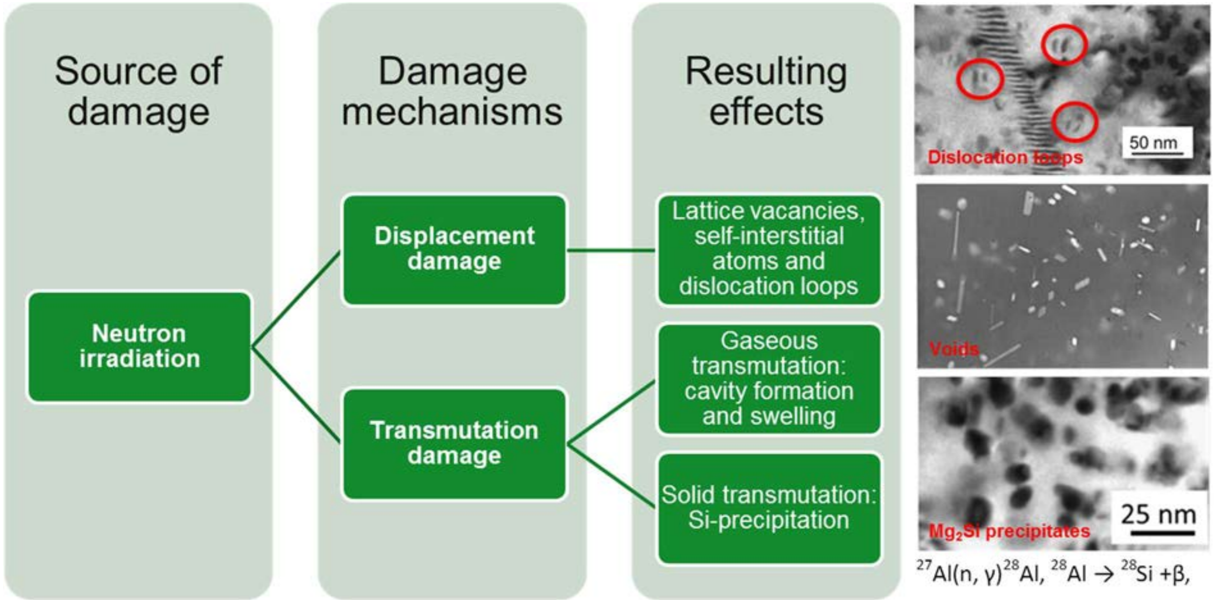


FIG. 1. Damage mechanisms and resulting effects of neutron irradiation on aluminium alloys (Courtesy of M. Kolluri).

2.1.4. Displacement damage

As in other metals, displacement damage is initiated by the production of primary knock-on atoms (PKAs) through elastic collision of fast (high energy) neutrons with the aluminium matrix. The resulting PKAs trigger displacement cascades leading to the formation of lattice vacancies, self-interstitial atoms and dislocation loops. With increasing fluence, dislocation loops grow and encounter other loops or dislocation networks. When the loops interact with each other, they coalesce and form dislocation networks that further evolve under irradiation.

The irradiation induced dislocation density determines the extent of irradiation hardening and embrittlement resulting from displacement damage. It is known from the literature that the dislocation density in irradiated metals evolves towards a saturation value with increasing fluence [15]. This occurs when the dislocation annihilation rate reaches the value of the production rate. The resulting contribution of displacement damage to irradiation hardening and embrittlement remains nearly constant above the irradiation dose levels at which dislocation density reaches a saturation value. Therefore, transmutation produced Si plays a dominant role in contributing to irradiation hardening of aluminium alloys as discussed further in the next section. A detailed discussion on the evolution of displacement damage in aluminium alloys can be found in references [8, 15].

2.1.5. Transmutation damage

Transmutation damage in aluminium can be caused by both fast and thermal neutrons. Thermal neutrons cause transmutation of aluminium into silicon through the following sequential reactions (n-capture and beta-decay), leading to an increase in Si content with increasing thermal neutron fluence.

$$^{27}Al(n,\gamma)^{28}Al,\ ^{28}Al \rightarrow {}^{28}Si + \beta \qquad (1)$$

Fast neutrons produce gaseous products like helium and hydrogen through (n,α) and (n,p) transmutation reactions [9]. In most metals, the gaseous transmutation products play a larger role in the development of radiation damage microstructure than non-gaseous products. However, aluminium alloys used in RRs are different in this respect. Depending upon the thermalization of the neutron spectrum, the solid transmutation product Si can have a stronger effect on radiation damage structure than gaseous transmutation products. The transmutation produced Si precipitates either in elemental form as in pure aluminium (1100 and 6061 alloys) or in Mg_2Si precipitates as in 5XXX series alloys until all Mg in solid solution is consumed. The structure, size and distribution of these precipitates (Si and Mg_2Si) in the microstructure; determines the resulting mechanical properties of irradiated alloys.

2.2. ZIRCONIUM AND ITS ALLOYS

The major advantages of the application of zirconium and its alloys such as Zircaloy-2, Zircaloy-4 and Zr-2.5Nb in RR core environments are their low neutron absorption cross-section, excellent mechanical properties and high corrosion resistance. The formability of zirconium and its alloys has some limitations related to its crystallographic texture, leading to anisotropy. The welding of zirconium and its alloys can be accomplished for many physical shapes, however, the procedures require added precautions such as gas-shielding and post-weld heat treatment (PWHT). Many RR core components have been constructed from zirconium and its alloys.

Generally, the irradiation growth of a zirconium alloy is determined by the texture. The degree of deformation of the material also affects the irradiation growth; the recrystallized material shows very little growth, of less than 0.1%, for a fluence of up to 10^{21} n·cm^{-2} (>1 MeV), which leads to saturation [23–25].

The irradiation growth rate of a zirconium alloy decreases as irradiation temperature decreases; and a higher neutron irradiation dose is required to increase the irradiation growth rate [26–28]. At temperature below ~130°C the migration of vacancies is minimal [29–30]. In a zirconium alloy, neutron irradiation forms point defects, extended defects (clusters, pores and dislocation loops), whereby irradiation embrittlement may occur. A study on the reduction of fracture toughness of Zircaloy-4 showed that the reduction of plane–strain fracture toughness (K_{IC}) did not occur at 130–310°C at a fluence up to 2×10^{21} n·cm^{-2} (>1 MeV) [31]. Therefore, the fracture characteristics of Zircaloy-4 due to irradiation effects under a relatively low operating temperature are not significantly changed. This makes zirconium and its alloys suitable for RR application, however, it must be remembered that there is very limited data available for high fluences at RR operating temperatures.

2.2.1. Zirconium alloys

A comprehensive analysis has been undertaken for the zirconium alloys intended for use in RRs. A series of tests have been undertaken, aiming to provide confidence in the prediction of material behaviour over time. Its nuclear characteristics and high resistance to corrosion, mechanical properties and ductility make zirconium alloys a suitable material for the construction of several components located close to the core. Zircaloy-4 (Nuclear materials: grade R60804), and Zircadyne (Non-nuclear materials: grades R60702 and R60705) alloys were included in the analysis. Chemical requirements follow standards ASTM B493, B550, B551, B653 and B658 and ASTM B351, B352 and B353 for nuclear application (i.e., low Hf content). Zircaloy-4 and Zircadyne grade R60702 have a predominant α (hexagonal closed packed (hcp)) phase rich in Sn with Fe. Cr and Ni present as inter-metallic precipitates. Zr-2.5 Nb alloys (e.g., the Russian nuclear grade TU 95 166–98) are two phased (α/β), where the α phase is saturated in Nb (1 wt. %) and the remaining Nb together with Fe or Cr impurities is present in the metastable β phase.

2.2.2. Irradiation effects

Zirconium alloy properties are affected through the effects of radiation [42]. This aspect is of particular significance in those structural components that are expected to be placed close to the reactor core for the life of the RR. For example, a core chimney wall may receive neutron fluence of the order of 4.9×10^{22} $n\cdot cm^{-2}$ over 40 years and operate at temperatures around 60ºC. As a result of the integrated neutron flux, point defects are produced, as well as clusters and linear defects, all of which increase tensile strength and reduce the ductility of the material. This increase in strength tends to saturate at low fluence. In the case of Zircaloys, material hardening has been observed at high fluence, a phenomenon related to the dissolving of precipitates during irradiation [32]. Amorphization of intermetallic particles has also been observed.

The irradiation of zirconium and its alloys causes the component to grow. Irradiation growth is a directionally oriented increase in dimensions that is determined by the preferred orientation of the grain structure or texture. The magnitude of the growth depends on the metallurgical state, i.e. texture and thermal treatment. For cold worked materials, the initial growth with fluence is linear, and its magnitude depends on the strain level within the material. For re-crystallised materials at low fluence (approximately 10^{21} $n\cdot cm^{-2}$), a smaller initial growth (<0.1%) soon saturates [34–36]. The growth rate is also influenced by temperature and creep.

From a comparison between the irradiation-induced growth and microstructural relationships for binary and multi-component zirconium alloys, it follows that the pre-transition growth is related to defects. The accelerated growth stage at high fluence is related to the formation of defects. At high temperatures (typical of NPPs) and high neutron fluence the phenomenon of a change in the growth rate, also called breakaway [37], has been observed. Temperature has a large effect on the level of neutron fluence at which breakaway occurs. As temperature drops, the initiation of the rise in growth rate moves towards higher neutron fluence [38]. At low temperatures such as the ones found in a RR, the available information shows that the effects of temperature, fluence and alloying elements on microstructure is such that the vacancy mobility (below 130°C) is significantly reduced. This, together with the influence of defects on zirconium alloy microstructure, leads to the conclusion that the accelerated growth phenomena occurring under the operating conditions expected in such reactors is unlikely [39, 40].

The most important area in which zirconium alloys are used in RRs is the reflector vessel, including the chimney (e.g., Australia's OPAL reactor) or inner shell (e.g., the Republic of Korea's HANARO reactor) surrounding the core. This environment has demineralized water on the outside of the reflector vessel with a low conductivity (<5 µS/cm). Neutral pH and heavy water with the same conductivity are also controlled on the inside. Various corrosion processes have been analysed, including pitting corrosion, stress corrosion, erosion corrosion, hydrogen effects and thinning effects. Under the strict chemical control regime proposed for the light and heavy water systems of RRs, none of these effects are significant. The chemistry control for both the light and heavy water is the key factor to ensure that the effects of these processes are minimized.

In a radiation environment wherein zirconium and zirconium alloys may be used, only fast neutrons that have sufficient energies will affect the microstructure and those mechanical properties that are structural sensitive.

Other radiation sources such as gamma rays and ion bombardment are capable of altering the atomic structure (e.g. ionization) and microstructure but generally are incapable of any significant penetration of metallic materials and are therefore of little practical importance in terms of altering mechanical behaviour. Review is limited to the effects of fast flux neutron irradiation. It has been well established through electron microscopy and thermal activation studies that fast neutron irradiation strengthening in pure face centered cubic (fcc) metals is at least qualitatively attributable to two types of defect clusters produced by a momentum exchange between high energy neutrons and lattice atoms. The two types of defects that are believed to be the result of the momentum exchange are:

- Large interstitial loops whose density tends to saturate at a low fluence.
- Small, variably sized planar vacancy clusters whose density continues to increase approximately linearly with fluence.

It is generally supposed that the irradiation strengthening mechanism in body centered cubic (bcc) and hcp metals is similar to that in fcc metals. The mechanical behaviour of hcp Zr and dilute Zr alloys is complex due to inherent crystallographic anisotropy and dynamic strain ageing characteristics. Veevers, et al. [41] show that for Zr-2 strain ageing occurs at temperatures ranging from about 200 to about 450°C. The 300°C strain ageing peak has been attributed to interstitial oxygen and has a significant effect on creep strength [41]. After irradiation to a fluence of about 5×10^{19} $n \cdot cm^{-2}$ or higher, oxygen strain ageing appears to be effectively suppressed, presumably because the oxygen is trapped at irradiation defect clusters. In PIE annealing tests, an increase in yield strength is observed when annealed at temperatures above the irradiation temperatures.

Non-nuclear grade zirconium alloys R60705 and R60702 may be used in those places where there are no low neutron absorption cross-section requirements, such as flanges, frames, piping connections, external piping (e.g. heavy water outlet and inlet, venting pipe), skirts, braces and brace anchoring, etc. Zirconium alloys R60702 and R60705 are subjected to integrated flux below 10^{20} $n \cdot cm^{-2}$ during 40 years of reactor operation. This integrated flux value is well below the thresholds in which noticeable growths or modifications may be realized to affect the mechanical properties [33].

2.2.3. Welding

The welding process and procedures mandate the provision of the necessary protection for the weld and heat affected zone by inert gases to prevent weld contamination from oxygen and nitrogen [43]. The tungsten inert gas welding process is commonly used with argon acting as a cover gas. A local shielding system is used to protect the weld and the heat affected zone until the material cools down sufficiently to ensure weld stability and to avoid contamination. Post-welding thermal treatments are used to improve the toughness and reduce the local residual stresses induced by the fusion of Zircaloy-4 and Zr-2.5Nb during welding. [44]. Welding of Zr-2.5Nb alloys is usually followed by stress-relieving, particularly to reduce the likelihood of delayed hydride cracking (DHC). The necessary requirements for DHC is dissolved hydrogen, a sustained stress (approximately >30 MPa) and time for hydride formation and consequent crack growth.

2.3. BERYLLIUM

The major strength of beryllium for application in RRs is the combination of a large scattering neutron cross-section combined with low thermal neutron absorption cross-section. The formability, weldability, and neutron radiation effects are such that it is not attractive as a structural material. Therefore, its application in RR core components is limited to solid reflectors in the form of beryllium metal, Al-Be alloy or beryllium oxide, as also indicated in the ANNEX 1. Neutron irradiation greatly affects the fracture resistance of beryllium. Studies have shown that tensile strength is reduced for beryllium blocks exposed to neutron irradiation at 66°C and 5×10^{21} $n\cdot cm^{-2}$ (> 0.1 MeV) [16-18]. Neutron irradiation causes a small amount of dimensional change for beryllium. According to the data on beryllium irradiation test, only about 0.3% of the length change was observed at 70°C and 5×10^{21} $n\cdot cm^{-2}$ (>0.1 MeV). For higher fluence around 6.6×10^{21} $n\cdot cm^{-2}$ (> 1 MeV), the material becomes almost completely brittle [18].

However, it is considered that asymmetrical deformation occurs in the beryllium reflector due to the difference in flux density of each block. Because of this, beryllium reflectors are recommended for periodical examination.

Helium bubble formation after Be transmutation, which essentially depends on irradiation temperature, should not be an issue for mechanical properties of RR materials due to the fact that He atoms are present as subatomic complexes. These cannot even be observed by transmission electron microscopy because of their size [18].

2.4. STAINLESS STEELS

Austenitic stainless steels are versatile materials that have been used in RR core components. The major difference between austenitic stainless steel and aluminium is the higher neutron capture cross section of steel, which limits it to a few applications in the core. Another difference is the activation of austenitic stainless steel, which is much longer lasting than that of aluminium, limiting some applications of steel in irradiation devices.

Stainless steels exhibit excellent mechanical properties, good radiation damage resistance and very little swelling up to high thermal and fast neutron fluence in the temperature range encountered in RR cores. The water corrosion resistance is good, but application of 304 versus 316 or 347 (Nb stabilized) should be justified. The steel formability and relatively easy welding give the designer a large freedom in designing complex or complex-shaped components. These major properties result in the wide application of austenitic stainless steels, i.e. AISI 304L, AISI 316L and AISI 347; see Annex 1. Vessels and piping related components located near a high neutron flux in RRs such as pressure vessels, low pressure tanks, connecting pipes, beam port nozzles and poolside irradiation facilities made from steel are common. Core supporting components such as grids supporting core, core boxes and other internal support structures can be found in many RRs. Components with a more or less tube-like shape such as control guide devices, experimental facilities and isotope production holders and tools are also often constructed from austenitic stainless steels. Beam related structures are limited to neutron guide tubes.

The neutron irradiation behaviour of stainless steel, such as hardening, loss of ductility, swelling, irradiation induced creep and fracture toughness, is determined by irradiation conditions such as temperature, neutron flux, fluence and neutron energy spectrum [20–22]. In its alloys irradiation damage originates mostly from fast neutrons. Stainless steels have strong resistance to corrosion over a wide range of environments and temperatures. The swelling phenomenon depends on the operating temperature and neutron fluence. Under thermal neutrons, He is mainly formed by transmutation of ^{58}Ni. For a RR, components operate at temperatures below 70°C and are expected to see a lifetime fluence of approximately 1×10^{23} $n\cdot cm^{-2}$. These conditions are well below the conditions at which swelling becomes significant [21].

Austenitic stainless steels are welded with fillers of similar materials, e.g. low carbon filler metals. Shielded metal arc welding, gas metal arc welding or gas tungsten arc welding are used with appropriate selection of cover gases so that carburization or an increase in the nitrogen content of the weld is avoided. This ensures problems such as hot cracking or intergranular corrosion are avoided. Degradation of corrosion resistance in the heat affected zones of the weld metal is prevented by using low carbon steels and appropriate welding practices [22].

2.5. GRAPHITE

The value of graphite for application in RRs lies in its nuclear properties that allow for efficient use for reflection and moderation of neutrons; see Annex 1. In some special irradiation devices, the very high temperature properties of graphite are used but always in combination with metals and alloys. In the case of nuclear grade graphite, the amount of ash should be below 300 ppm, and the amount of equivalent boron content should be kept below 2 ppm. Neutron irradiation of graphite causes volumetric expansion and shrinkage by releasing carbon atoms into graphite, and this volume change is highly dependent on the irradiation temperature. Generally, at a temperature of 300°C or lower, irradiation growth occurs rapidly without initial shrinkage. The IG-110 graphite reflector used in Japan's JRR-4 reactor exhibited a maximum irradiation growth rate of $7.13\times10^{-25}\%\cdot m^2/n$ at a dose of 2.5×10^{20} $n\cdot cm^{-2}$ (>0.18 MeV) [19]. However, as the irradiation amount increases, the irradiation growth rate decreases, and the volume change due to irradiation is minimal. Irradiation of graphite at low temperatures (<300°C), typical when used as the reflector material in RRs, may lead to accumulation of a significant amount of Wigner energy which can be a potential fire hazard under conditions of loss of cooling or power transients. Reduction in thermal conductivity can compound such problems.

2.6. CONCRETE

The application of concrete in RRs is mainly for structural support and shielding. Because concrete is generally reinforced by steel bars, its properties are dependent on those of both the matrix and reinforcement materials. As reinforced concrete ages, changes in its properties occur as a result of continuing microstructural changes, e.g. slow hydration, crystallization of amorphous constituents and reactions between cement paste and aggregates, as well as environmental influences. These changes are not necessarily detrimental to the point that concrete is unable to meet its functional and performance requirements. However, reinforced concrete can suffer undesirable changes over time because of improper or a violation of specifications, adverse performance of the cement paste matrix or aggregate constituents or either physical or chemical attack upon the reinforcing steel.

The main issue described from the literature is state-of-the-art non-destructive testing methods and technologies for the inspection of reinforced concrete structures. Inspection has mainly been devoted to estimate the durability and service life of structures involved in not only experimental reactors but in any reinforced concrete structure used in nuclear applications. In this context, some approaches must ensure concrete structures meet their functional requirements throughout their service life, including life extensions.

Several degradation mechanisms have been identified as potentially harmful to the integrity, stability and durability of the structures: drying shrinkage and creep, steel corrosion, freeze–thaw, alkali-silica reaction, sulphate attack, nuclear radiation effect, thermal degradation, leaching and efflorescence. Most of these mechanisms require a specific monitoring technique.

2.7. OTHER MATERIALS

Contributions on other materials include the experimental investigations on RR cable insulation materials by Algeria. Cables with both polyvinyl chloride (PVC) and polyethylene (PE) were exposed to gamma and neutron irradiation. Above gamma radiation levels of 10^6 Gy, PVC insulation properties were affected significantly. Polyethylene insulation properties up to 1.7×10^{11} $n\cdot cm^{-2}$ thermal neutron fluence are hardly affected [45, 46].

The information on carbon steel ageing was contributed by Argentina.

3. DATA SOURCES

There are three major sources for experimental results to be included in the database. These are:

- open literature,
- CRP experimental results,
- open institutional experimental results,
- data from in-service components, and
- data from decommissioned components.

The data from open literature are derived from journal articles, conference proceedings and open reports from reputed institutes on the irradiation behaviour of materials relevant for RR cores and related structures. These data underwent a review process using criteria established early in the CRP project. The criteria concentrate on the details of the origin of the materials, data accuracy bands and calibrations. More details are given in Chapter 4.

Several participants conducted experiments in the frame of this CRP that filled gaps identified early in the project. The results of the experiments conducted by the participants have been added to the database after review with criteria similar to that for the open literature.

Many institutes have measured properties of in-service or decommissioned components that are not always available in open literature. This group of experimental results measured by institutes and not reported in journal articles, conference proceedings, or open reports, (but highly relevant for the CRP) are usually reported with restricted distribution. The institutional experimental result reports used by the CRP have been cleared for inclusion into the database. In the context of this CRP they are considered as open sources. An example of such a type of report is surveillance data produced for the Netherlands government bodies charged with licencing. Other examples are material data produced by laboratories to guide design of components.

The third source of data for the database is measurement results from in-service components. Typical examples are deformation measurements of graphite parts known to show different amounts of growth in different directions. Another example is the measurement of Be reflector parts known to deform because of neutron flux gradients. Even for stainless steel parts such as grid plates; dimensions are measured to assure that deformation has not occurred and that supported parts could fit in the grid in line with design targets.

There is a large variety of decommissioned components from all materials available. Some provide highly useful data ranging from aluminium alloy embrittlement to graphite dimensional stability. Also these measurements were reviewed with the criteria valid for the other data in the RRMPDB.

3.1. OPEN LITERATURE

Published literature, e.g., journal articles, conference proceeding and reports from reputed laboratories, describing the irradiated behaviour of materials relevant for RR cores and related structures was gathered by the CRP participants.

This endeavour was expected to be complex and time consuming; therefore optimal use of the specific expertise and experience of the participants was considered essential. Thus, a matrix outlining the materials of interest and MSs known to have conducted relevant research was established to form the basis for the selection of contributors and referees for the literature search. This matrix defined the tasks, reduced the effort required and eliminated potential conflicts of judgement. All participants extracted data from the literature pertaining to all materials and arranged the information using templates. The details of the form and content of the templates were agreed upon during the RCMs. The templates simplified the uploading of data to the RRMPDB in an accessible form.

During the third RCM, the working group gave final approval to the literature and extracted data prior to inclusion in the RRMPDB.

3.1.1. Aluminium

The review focused on the effect of neutron irradiation on mechanical properties and microstructural changes e.g. precipitation of silicon, voids and swelling.

Quality of the collected literature data was based on the extent of details provided regarding material conditions (component, product form, heat treatment, etc.), irradiation conditions (thermal and fast fluence, displacement per atom (dpa), irradiation temperature, etc.), experimental conditions and test standards (specimen geometry, testing standard, test temperature, data validity, etc.), mechanical properties (strength, ductility, hardness and fracture toughness) and microstructure. About 40 papers/reports were reviewed, and 25 were found suitable for inclusion in the database. More details can be found in Appendix 1.

3.1.2. Zirconium alloys

Among the papers and reports reviewed, 26 were found suitable for inclusion in the database. More details can be found in Appendix 2.

3.1.3. Beryllium

The literature survey focused on the irradiation effects upon beryllium prepared using different manufacturing processes such as hot pressing, extrusion and vacuum hot pressing. Most of the reviewed papers discussed the defects generated by transmutation reaction and displacements. Sources are journal papers, international meeting papers and research institution reports.

Among the papers and reports reviewed, 23 were found suitable for inclusion in the database. More details can be found in Appendix 3.

3.1.4. Stainless steel

The literature survey focused on irradiation effects on 300 grade austenitic stainless steels regarding tensile strength, fracture toughness, fatigue strength, void swelling and irradiation assisted stress corrosion cracking properties. The sources were journal papers, international meeting papers and research institution reports. Vast amounts of data representing a range of NPP parameters exist; however, low temperature data relevant to the RR operation range is quite limited. From a total of about 60 papers, 23 papers mainly on 304 and 316 stainless steels were found suitable for inclusion in the database. More details can be found in Appendix 4.

3.1.5. Graphite

A literature survey was conducted on inputs provided to the CRP on graphite. A total 13 papers were submitted, of which 8 papers were deemed relevant to the CRP. Two eliminated papers were relevant to high temperature gas-cooled power reactors operating above temperatures experienced in research reactors. Three papers deal with decommissioning aspects of irradiated graphite. A large amount of data on Wigner energy is available on the graphite reflector of India's CIRUS research reactor. Insufficient data on physical degradation of graphite for its re-use or disposal are available. There is little information about the graphite manufacturing process, which affects its properties significantly. More details can be found in Appendix 5.

3.1.6. Concrete

Most of the information has been published in scientific journals, but also some papers presented in scientific meetings and one IAEA publication illustrating aspects of ageing management in concrete structures in NPPs were considered, as these programmes are applicable also for RRs. A total of 61 papers were reviewed and 24 were uploaded to the CRP platform. With respect to their applicability to the specific CRP subject, 9 were considered very applicable, 13 applicable to some extent and 2 not specifically applicable. More details can be found in Appendix 6.

3.1.7. Other materials

Only the data presented by MSs was reviewed. No open literature was considered at this stage.

3.2. COORDINATED RESEARCH PROJECT (CRP)

Data from aluminium experimental programs have been provided by the member states: Australia, India, Indonesia and the Netherlands. The Netherlands provided mechanical property data, including fracture toughness properties, measured from neutron irradiated specimens of aluminium type 5XXX up to a thermal neutron fluence 1.2×10^{23} n·cm $^{-2}$. Japan provided beryllium data extracted from reports regarding experimental details of the JMTR materials surveillance programme. South Korea contributed data on different neutron radiation effects on properties of zirconium alloys. Australia measured properties of zirconium alloys and submitted data that was included in the RRMPDB. Egypt quantitatively characterized stainless steel welds relevant for RR components. The resulting data was inserted in the RRMPDB. Japan also contributed stainless steel data extracted from JMTR materials surveillance reports. India contributed results on graphite, both from independent investigations and decommissioned parts. An aluminium (ALCAN 6056, equivalent to Al1050) sample was cut from one of the tubes from the CIRUS reactor vessel, after it was permanently shut down and analysed to evaluate its mechanical properties. Similarly, samples of graphite were taken from the reflector and analysed for Wigner energy.

Algeria submitted data on the changes of insulation materials from signal cables corresponding to radiation levels for some RR instrumentation. Argentina provided data on the ageing behaviour of steel and reinforced concrete in a test reactor relevance to RR environments.

In addition, several MSs such as Algeria (Zircaloy-4), Indonesia (aluminium alloys) and Australia (Zircaloy-4 and Zr-2.5Nb) will continue their experimental programmes to conduct measurements on material properties in support of their reactors' ageing and surveillance programmes. Such data, after validation, will also be added to the RRMPDB.

4. DATA REVIEW, VALIDATION AND IDENTIFICATION OF DATA GAPS

For reliable life predictions of essential components, a need exists to use a reliable and dependable material properties database. The owners of RRs usually have their sets of inspection and operating data to support safe operation. Availability and accessibility of this information will certainly strengthen the qualitative and quantitative aspects of life management and prediction. For this vast amount of data to be available for all RR operators, templates are used to facilitate sharing of data, which may otherwise be seen as station specific information.

4.1.DATA REVIEW AND VALIDATION

The first task was to determine the types of information relevant for reliable life predictions of primary components. The collection of data was organized according to the list of components and materials relevant for the RRMPDB, which are summarized in Annex 1. The collection of information began within the participating organizations and extended to open literature data suitable on first glance. After the "raw" information was gathered, it was reviewed and validated using previously established and agreed criteria. Major selection criteria are clear specifications of the materials and testing conditions. For tests on materials obtained from decommissioned components, the accuracy of operating data strongly influenced their inclusion in the RRMPDB. The review and validation will lead to the completion of parameter fields [Annex 2]. Information that is needed for life prediction and not available from the RRMPDB is identified as "data gaps".

The review process used the following as criteria throughout the project.

- Level of detail and assurance of chemical compositions and heat treatments for the base and welded materials.
- Accuracy of operating conditions such as the temperatures, irradiation parameters and water chemistry.
- Calibration of measurement equipment for determination of mechanical properties of test standards and samples.

The review of data on concrete focused on three main points: the degradation processes that take place on the cement part of the concrete structures, the corrosion of the reinforcement bars and the concrete structures' life assessment (service life durability). In the case of the bar corrosion studies, two main issues were: the mechanisms of steel corrosion and the development of corrosion sensor probes to perform real time monitoring of their corrosion status.

4.2. DATA GAPS

During the cycles of data acquisition and subsequent data review and validation. The RRMPDB took shape. A set of data, although limiting, is now available to support the prediction of the lifetime of primary components and the end-of-life condition of a RR. The data available does not match the set of data required for all predictions. Thus the missing data are called "data gaps", which will require further actions.

This CRP has focused on the effects of neutron irradiation on mechanical properties and microstructural changes, e.g., precipitation of silicon, voids and swelling. Prior to the review and validation of the collected literature and experimental data, the participants decided to identify data and information gaps and continuously upgrade the gap list throughout the period of project. It was foreseen that existing data gaps would have two main distinctions, because of the-lack-of:

– knowledges on materials that are of interested to MSs, e.g., the absence of fracture toughness data that are the most valuable for predicting the life of a component before cracks, and
– measurements in an applicable range of environmental conditions, e.g., the limited availability of data on mechanical properties for type 5XXX aluminium alloys exposed to high neutron fluence.

Qualities of the collected data from the literature were based on the extent of details provided regarding, for instance: material conditions (component, product form, heat treatment, etc.), irradiation conditions (thermal and fast fluence, displacement per atom (dpa), irradiation temperature, etc.), experimental conditions and test standards (specimen geometry, testing standard, test temperature, data validity, etc.), mechanical properties (strength, ductility, hardness and fracture toughness) and microstructure.

For each material, the responsible working group identified the data gaps. All three RCMs had devoted times to evaluate and rank the identified gaps. During the final RCM, participants agreed that gaps that were considered as high prioritises would be filled in the near future. Data gaps for all materials are identified in the subsequent sections.

4.2.1. Data gaps on aluminium alloys

A few data gaps were observed in the available data for irradiated aluminium. For example, only limited data has been published on the fracture toughness properties of irradiated aluminium alloys. The available data on 6XXX series alloys is considerably more due to their widespread use in several RR components and cold neutron sources. In particular, limited data, apart from HFR surveillance programme (SURP) data, has been published on 5XXX series alloys. Additionally, the data on corrosion behaviour of irradiated aluminium alloys at high neutron fluence in the RR environment is very limited.

In order to fill these gaps, the participating MSs agreed to:

– Continue generation of data on irradiated aluminium by experimental irradiations.
– Evaluate samples from core components of decommissioned reactors, e.g. CIRUS.
– Provide data generated from the HFR SURP 2015 [an existing obligation].
– Provide the data generated from the 5-year surveillance program for the OPAL reactor [an existing obligation].

4.2.2. Data gaps on zirconium

The literature did not provide extensive information on zirconium alloys used currently in RRs. There is limited data at high fluence on Zircaloy-4 at RR temperatures. The main data gaps are:

- Fracture toughness data for materials irradiated and tested below 100°C and high neutron fluence effects on Zircaloy-4.
- There are insufficient tensile and fracture properties data for Zr-2.5Nb alloy at RR conditions at fluence higher than 2.5×10^{21} $n\cdot cm^{-2}$.
- The lack of consistent information on the DHC behaviour of Zr-2.5Nb welds at RR operating temperatures. The DHC oriented experiments recently conducted in Algeria's NUR reactor may reduce the gap of data when the data becomes available.

In order to fill these gaps, the participating MSs will continue conducting experiments to gather more properties of zirconium alloys and generate the data needed to predict behaviour. Examples are:

- The experiments conducted in HANARO on Zircaloy-4 at a temperature of 40–60°C followed by PIE and the surveillance of Zircaloy-4 samples after 5 years operation in OPAL will certainly narrow the gap between what is now available and what will be in the database in future. The data for very high fluence on Zircaloy-4 at RR temperatures will, of course, not be completed in the short term.
- Algeria's Nuclear Research Centre of Draria (CRND) has planned irradiation of hydrided Zircaloy-4 and Zircaloy-2, and results on tensile strength and microstructure will be provided when available.

4.2.3. Data gaps on beryllium

Very little data related to beryllium behaviour under neutron irradiation at RR operating temperatures has been reported in literature. A grain size effect on the properties of beryllium is apparent, but data on the effects of grain size is scarce. For the improvement of fabrication methods, the lack of data forestalls the acquisition of better quality materials for reflectors. This gap will be difficult to fill considering the limited supplier basis and the strategic character of beryllium. Limited access to operational experience prevents distribution and analysis of valuable data. New simulation models should be developed and validated as soon as new data becomes available.

In order to fill these gaps, the participating MSs agreed to:

- Perform irradiation experiments to acquire the missing data.
- Develop and validate new simulation programs.
- Improve fabrication procedures to obtain improved materials.
- Develop new monitoring techniques.

4.2.4. Data gaps on stainless steel

There is a wealth of data on irradiation effects on austenitic stainless steel's material properties, but mostly at a much higher temperature level (250–350°C) than encountered in RRs. Particularly, there is no data available on irradiation effects at low temperatures, including corrosion; weldments among base metals, weld metals and heat affected zones; and fracture toughness.

In order to fill these gaps, the participating MSs agreed to continue:

- Proposed research work concerning the lack of fracture toughness data on austenitic stainless steel materials, especially at low irradiation and test temperatures.
- To develop a fracture toughness model with a predictive capability utilizing the available fracture toughness data on irradiated austenitic stainless steel base materials and welds.
- To obtain conservative estimates for properties at low temperatures by considering the available fracture toughness data at high irradiation/testing temperature, if it can be confidently verified that fracture toughness decreases with increasing irradiation/testing temperature.

4.2.5. Data gaps on graphite

The data available on Wigner energy release is quite exhaustive; however, a gap exists for results relevant for graphite in the operating temperature range of RRs. The data on thermal properties and irradiation-induced swelling is not as exhaustive as on Wigner energy. There is a need for supplementing this data with experimental results from RRs. The focus should be on data from RRs that have been in operation for a sufficient duration or RRs that have been permanently shut down, allowing the investigation of decommissioned graphite parts.

The data bank must be very large to account for variations in real applications, as graphite production paths strongly influence its behaviour. In order to fill these gaps, the participating MSs agreed to continue:

– Wigner energy measurements on samples of graphite from old RRs under permanent shutdown or decommissioning.
– The generation of data on thermal and mechanical properties, ^{14}C and tritium activity over a wide range of temperature and fluence.

4.2.6. Data gaps on concrete

A major gap in the field of concrete is the lack of complete and reliable information in the literature concerning relationships among the types of reinforced concrete used, the environment and the rate of the degradation process, for example, aggregate swelling. Another gap is formed by the uncertainties related to possible radiation damage in the zones most exposed in the reactor core. In some reactors boron is used to protect concrete from radiation damage; however, no experimental data indicates the efficiency of this measure. Filling these gaps would result in a database containing the information needed to make decisions about life extension.

In order to fill these gaps, it is recommended:

– To generate data on the concrete degradation rate as a function of environmental conditions for each type of concrete composition and reinforcement bar material.
– To conduct irradiation tests at higher levels than those reported in the literature to measure the effect of aggregates swelling, which can promote cracking due to expansion. Data coming from power reactors would be useful due to the similar irradiation degree.
– Monitoring of concrete surface by non-destructive examination and some actual material examination using the decommissioned reactors will also help to fill the gaps.
– The efficiency of the use of boron as shielding material to reduce the radiation impact on concrete degradation should be assessed through irradiation tests.

4.2.7. Data gaps on other materials

The data available on RR cable insulation materials is limited to input from Algeria. Since many varieties of cable insulation materials exist, this information has limited applicability.

Argentina provided information on carbon steel ageing. It is however underlined that the carbon steel varieties used in RRs have a very wide spectrum of compositions and environmental conditions, which hinders the judgement of particular carbon steel components used in RRs.

5. DATABASE STRUCTURE

The main goal of the RRMPDB is to help the RR community to understand the material behaviour of RR core components for their continued safe operation and lifetime extension of ageing RRs. The amount of relevant information is significant and presented in different forms. Organizing this information in a structured way, bringing together similar and relevant pieces of information, significantly simplifies the search and investigative process. For this purpose, the developed database, which is well-integrated with input from MSs, allows easy access to the results of research and experiments for the corresponding subjects.

This section explains the structure and functionality of the RRMPDB and provides guidance on its usage.

5.1. ARCHITECTURE OF THE RRMPDB

The RRMPDB was developed using the Microsoft SharePoint platform and made available on the IAEA single sign-on system, the Nucleus, to create a software tool for groups to cooperate on sets of data. It is comprised of scientific and experimental information on properties of irradiated materials in the form of files. These files supplied by the CRP participants constitute the main content of the database and are located in the library called RRMPDB reports.

Functionally, the database may be divided into two sections:

- The home page, where welcome and introductory information is displayed as well as registration notes. This page is open to the public.
- The main page, where a registered user can work with or search for information on a specific subject. It is located in the restricted area of the RRMPDB and can be accessed through the RRMPDB Reports button located on the home page, subject to the user's permission level.

The address of the RRMPDB is: https://nucleus.iaea.org/sites/rrmpdb/Pages/Home.aspx.

5.1.1. Users

There are three groups of users of the RRMPDB. They are distinguished by permission level and affiliation: users internal to the IAEA local area network and external users that are outside the IAEA's network. By default, the RRMPDB's library is not accessible to the public unless the owner grants user access to a person in the Visitors or Contributing Members group category.

Visitors

The targeted audience comprises this group. These are all the users external to the IAEA who are expected to benefit from the RRMPDB. Visitors can access the library and the reports, but their permission level is limited to read-only access, allowing for browsing and downloading on their local machines the reports from the database.

Contributing members

This group is comprised of those users who can view the database reports and provide files to the RRMPDB Reports library to be uploaded by the Owners group.

Owners

IAEA staff having full control over all of the features of the RRMPDB comprises the Owners group. The owners decide which access level is given to a user.

5.1.2. Metadata

The word database normally assumes a comprehensive collection of related data organized for convenient access. While the RRMPDB provides an organized collection of comprehensive data, the relational component – that is, the relationship between individual data items - is absent. Instead, a set of metadata attributed to every document is used. The metadata is a set of data that describes and gives information about other data. This approach assists users in the discovery of necessary information by specifying search results based on relevant criteria. This brings similar papers together, allowing for broader, from a scientific point of view, and deeper understanding of the issues caused by irradiation.

Figure 5.1 illustrates the metadata attributed to the information in the RRMPDB, which allows for distinguishing each paper. This example does not contain the full set of metadata used in the RRMPDB and is intended for demonstration purposes.

FIGURE 5.1: An illustration of metadata attribution.

The full list of metadata used to attribute data in the RRMPDB is comprised of four groups as follows:

1. Bibliographical data
   - Title
   - Author(s)
   - Author affiliation
   - Source title
   - Publication year

- Volume
- Funded / Supported / Commissioned by
- Peer reviewed

2. Materials
   - Aluminium
     - Al 1050
     - Al 1060
     - Al 1080
     - Al 1100
     - Al 2014
     - Al 5019
     - Al 5052
     - Al 5154
     - Al 5754
     - Al 6061
     - Al 6063
   - Stainless Steel
     - SS300
     - SS304
     - SS304L
     - SS308
     - SS316
     - SS316L
     - SS316LN
     - SS321
     - SS347
   - Zirconium alloys
     - Zircadyne
     - Zircaloy-2
     - Zircaloy-4
     - Zr-1.5Nb
     - Zr-2.5Nb
   - Beryllium
     - Unalloyed
     - Oxide
     - Alloys
   - Concrete
     - Heavy concrete
     - Normal density concrete
     - Reinforced concrete
   - Graphite
   - Others
     - Insulation
     - Polymers
     - Hafnium
     - Carbon steel

3. Components

- Beam ports/nozzles
- Beam related devices
- Bolts
- Cold neutron source
- Connecting pipes
- Core box/vessel
- Core support structures
- Fuel clamps
- Guide tubes/boxes
- Heavy water tank and accessories
- Insulators
- Lining
- Low pressure tank
- Nuts
- Poolside irradiation facilities
- Pressurized vessels
- Reflector
- Safety rod liner
- Shielding
- Shroud
- Other structural and internal support structures

4. Properties
    - Beryllium doped with neutron absorber(s)
    - Changes in crystal structure
    - Compression
    - Thermal conductivity (heat transfer)
    - Corrosion of reinforcement/steel
    - Creep
    - Creep–rupture
    - Cumulative dose
    - Density
    - Elongation
    - Fatigue
    - Fluence/dpa
    - Fracture toughness
    - Grain size
    - Growth (zero volume change)
    - Irradiation creep
    - Lithium/helium poisoning
    - Metallurgical state (heat treatment, cold work, welds etc)
    - Modulus of elasticity
    - Microstructure
    - Nil ductility transition temperature
    - Oxidation
    - Reduction of area %
    - Release of $^{14}C/^{3}H$
    - Swelling (volume increase)
    - Thermal expansion coefficient
    - Ultimate tensile strength
    - Wigner energy

- Yield stress
- Young's modulus (modulus of elasticity)

5. Tags
   - Each piece of metadata makes information in the RRMPDB easier to retrieve. By selecting the needed criterion, or combination of criteria, the user receives a reduced list of relevance papers.

5.2. USER GUIDE

5.2.1. Access to visitors

If admitted into the Visitors group, a user can proceed to the main page via the button on the home page RRMPDB Reports. The main page displays uploaded reports sorted in alphabetic order by the file name. At the top of the page, a set of filters appears. These filters are based on the metadata attributed to each report. Figure 5.2 shows the main page view with a dialogue window of metadata criteria selection open.

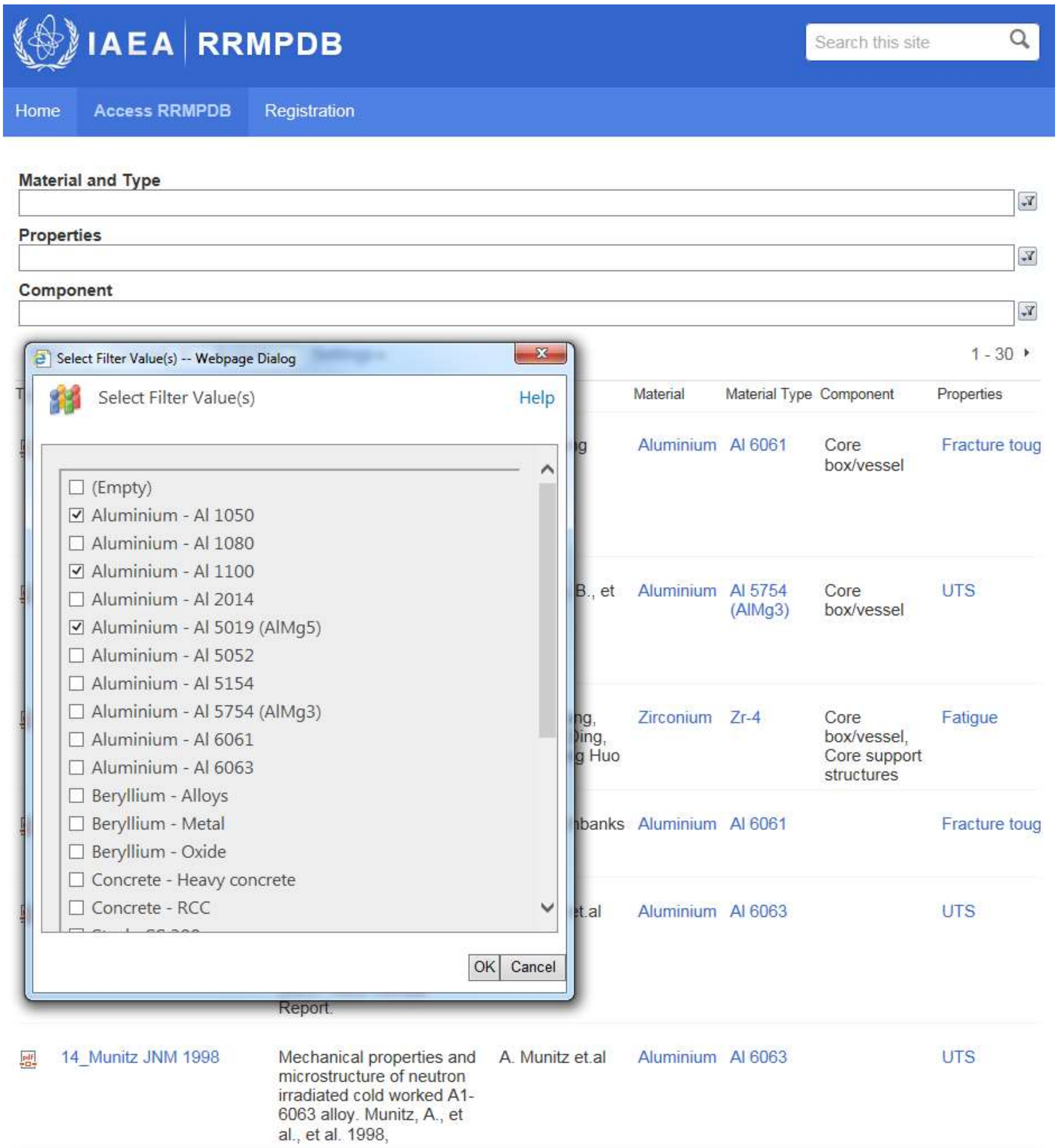


FIGURE 5.2: RRMPDB main page view.

To specify the information sought, several criteria can be chosen. All reports that satisfy the search request will be listed.

The visitor can also sort the full list of reports by clicking the corresponding button on the top ribbon.

- Title
- Author
- Material
- Material type
- Component
- Property

Another option is to select from a dropdown menu on the top ribbon, to display the relevant criteria for searching or sorting, as shown in Figure 5.3:

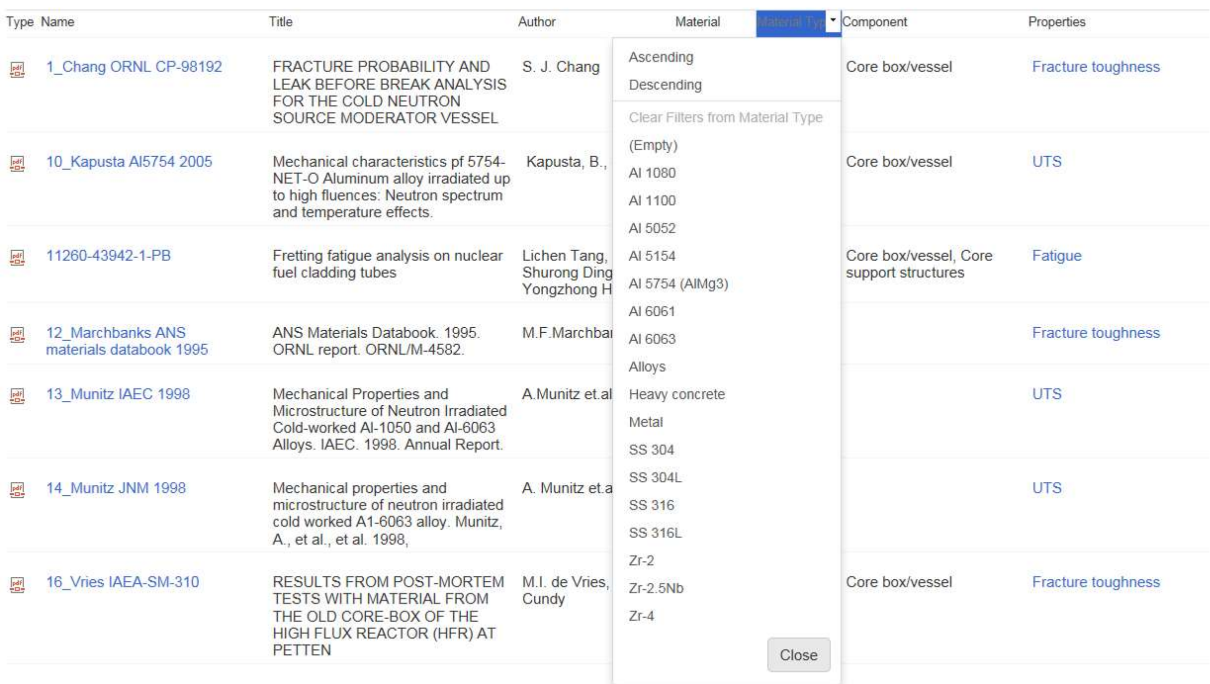


FIGURE 5.3: Dropdown menu for sorting information.

The search function at the top right of the main page is also available, and this search is conducted based on the metadata attributed during uploading.

The full article is displayed upon clicking the file name on the left hand list.

5.2.2. Uploading reports by contributing members

Contributing members can upload reports to the RRMPDB. During uploading they are requested to provide attributes for the metadata that directly describe the information in the file.

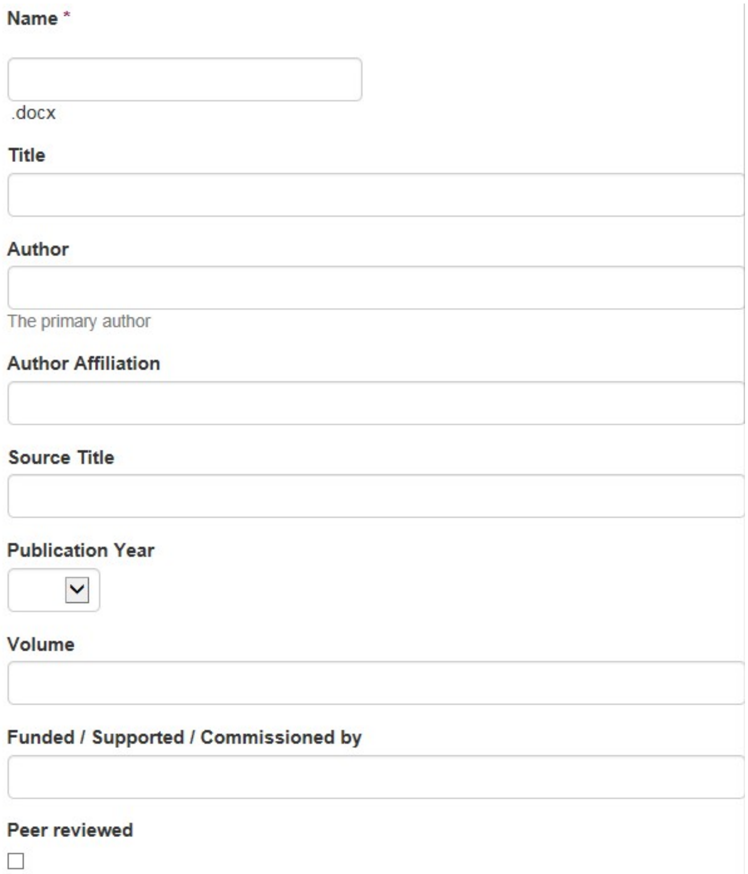
Contributing members can proceed to the main page via the button on the home page RRMPDB Reports. At the top of the list of reports appears a toolbar with several options for actions available to contributing members, subject to the given permission level (Figure 5.4).

New ▾ Upload ▾ Actions ▾ Settings ▾ 1 - 30 ▸

Type	Name	Title	Author	Material	Material Type	Component	Properties
pdf	1_Chang ORNL CP-98192	FRACTURE PROBABILITY AND LEAK BEFORE BREAK ANALYSIS FOR THE COLD NEUTRON SOURCE MODERATOR VESSEL	S. J. Chang	Aluminium	Al 6061	Core box/vessel	Fracture toughness
pdf	10_Kapusta Al5754 2005	Mechanical characteristics pf 5754-NET-O Aluminum alloy irradiated up to high fluences: Neutron spectrum and temperature effects.	Kapusta, B., et al.,	Aluminium	Al 5754 (AlMg3)	Core box/vessel	UTS
pdf	11260-43942-1-PB	Fretting fatigue analysis on nuclear fuel cladding tubes	Lichen Tang, Shurong Ding, Yongzhong Huo	Zirconium	Zr-4	Core box/vessel, Core support structures	Fatigue
pdf	12_Marchbanks ANS materials databook 1995	ANS Materials Databook. 1995. ORNL report. ORNL/M-4582.	M.F.Marchbanks	Aluminium	Al 6061		Fracture toughness

FIGURE 5.4: Main page view for contributing members.

By clicking an Upload button, a dialogue window for editing metadata opens. Metadata groups 2–4, as described in Section 5.1.2, are the controlled metadata, meaning that a user would not be able to modify the names or titles. The user can only select relevant items from the suggested list of metadata. Metadata groups 1 and 5, bibliographical information and tags, are free text fields, meaning that the member can enter any text. Figure 5.5 shows the form to enter bibliographical information, as a part of the uncontrolled metadata.



Name *

.docx

Title

Author

The primary author

Author Affiliation

Source Title

Publication Year

Volume

Funded / Supported / Commissioned by

Peer reviewed

FIGURE 5.5: Bibliographical information – uncontrolled metadata.

Figure 5.6 shows the controlled metadata, meaning the member cannot modify this attribution. One could select corresponding criteria from the suggested list. However, if a specific criterion needs to be added, a free text field is provided. For each selected material, a separate list of material types is displayed for selection. One or more components may be selected. The properties of the material to be selected are illustrated in Figure 5.7. One or more metadata on material properties can be applied.

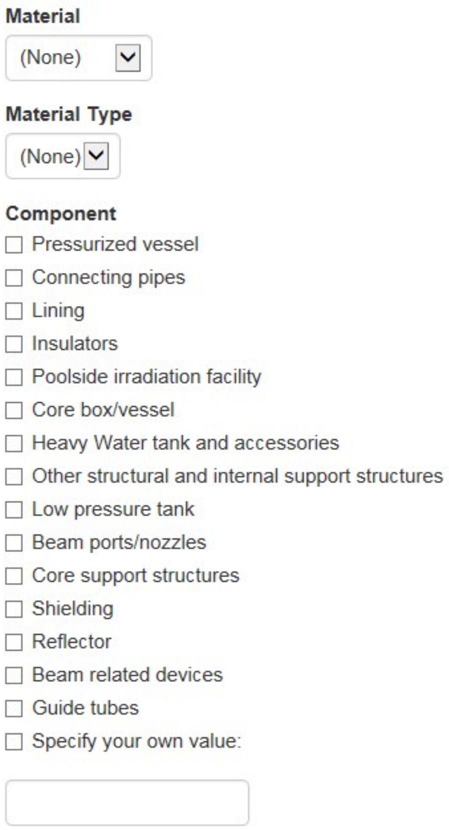


FIGURE5.6: Material, material type, and component information – controlled metadata.

FIGURE 5.7: Material properties information – controlled metadata.

The last group of metadata is tags, a set of key words that serves to identify the contents of the file more specifically (Figure 5.8). This helps users search for information via the Search field at the top of the main page.

FIGURE 5.8: Key word information – controlled metadata.

6. SUMMARY

The CRP on Establishment of Material Properties Database for Irradiated Core Structural Components for Continued Safe Operation and Lifetime Extension of Ageing Research Reactors has been completed and it has achieved its objectives. A database, comprising of 138 reports, has been launched and can be accessed at https://nucleus.iaea.org/sites/rramp/Pages/Home.aspx. The database is for reference and use by research reactor designers, operators and regulators in facilitating assessment of life of the core structural components and planning refurbishment or replacement as most of such components are difficult to approach. This publication provides guidance for use of the database.

APPENDICES

The open literature was first utilized and searched for information on the materials properties related to the normal operating conditions of research reactors. This search has provided a preliminary list of references that are comprehensive. A review of all those references was then carried out using the criteria provided in Section 4.1. The review process led to the selection of references fit for providing reliable data for the materials properties data in the RRMPDB. The appendices provide the lists of references for each material used in the database. It should be noted that these lists are based on the contents of the database that exist at the date of publication of this TECDOC. It is expected that additional information will be added to the database.

APPENDIX 1: LIST OF REPORTS FOR ALUMINIUM AND ITS ALLOYS IN THE RRMPDB

1. Fracture Probability and Leak before Break Analysis for the Cold Neutron Source Moderator Vessel, S.J. Chang, Research Reactors Division, Oak Ridge National Laboratory, Oak Ridge, USA, ASME Pressure Vessel and Piping Conference, 1998

2. Mechanical Characteristics of 5754-Net-O Aluminium Alloy Irradiated up to High Fluences: Neutron Spectrum and Temperature Effects, B. Kapusta, C. Sainte Catherine, X. Averty, G. Campioni, A. Ballagny, CEA Saclay, Gif sur Yvette, France, Joint Meeting of the National Organization of TRTR and IGORR, 2005

3. ANS Materials Databook, M.F. Marchbanks, Oak Ridge National Laboratory, Oak Ridge, USA, 1995

4. Mechanical Properties and Microstructure of Neutron Irradiation Cold-Worked Al-1050 and Al-6063 Alloys, A. Munitz, C. Cotler, M. Talianker, Nuclear Research Center-Negev, Beer-Sheva, Israel, IAEA Annual Report 1998

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7. Effects of High Thermal and High Fast Fluences on the Mechanical Properties of Type 6061 Aluminium in the HFBR, J.R. Weeks, C.J. Czajkowski, P.R. Tichler, Department of Nuclear Energy, Brookhaven National Laboratory, Upton, USA, ASTM STP1046 (1990) 441-452

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11. Response of Aluminium and Its Alloys to Exposure in the High Flux Isotope Reactor, K. Farrell, Metals and Ceramics Division, Oak Ridge National Laboratory, Oak Ridge, USA, 1983

12. Neutron Irradiation of Dilute Aluminium Alloys, R.M. Mayer, E.T. Morris, Atomic Energy Board, Pretoria, South Africa

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E. Brosh, A. Kiv, Department of Materials Engineering, Ben-Gurion University of the Negev, Beer Sheva, Israel, Journal of Nuclear Materials 306 (2002) 73–179

15. Ageing Data on Corrosion and Irradiation Induced Degradation of In-Core Components of Research Reactor Cirus, R.C. Sharma, P.V. Varde, R. Ranjan, V. Kain, J.S. Dubey, P.A. Jadhav, B.K. Tripathi, Bhabha Atomic Research Centre, Mumbai, India, IAEA CRP Code ID – T 34002

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19. On the Precipitation of Magnesium Silicide in Irradiated Aluminium–Magnesium Alloys, M. Verwerft, Reactor Materials Research Department, SCK·CEN, Mol, Belgium, Acta Materialia. 48 (2000) 1097-1104

20. Effects of Neutron Irradiation on Toughness of Aluminium and Its Alloys
A.K. Hellier, Australian Nuclear Science and Technology Organisation, Menai, Australia, ANSTO Materials Division (2001)

21. Assessment of Aluminium Structural Materials for Service within the ANS Reflector Vessel, K. Farrell, Oak Ridge National Laboratory, Oak Ridge, USA, ORNL/TM-13049 (1995)

22. Compilation of Experimental Data from Two Aluminium Safety Rod Liners Removed from HIFAR, D.R.G. Mitchell, R. Blake, A.B. Croker, A. Day, S.R. Humphries, N.R. McDonald, C.J. Moss, Australian Nuclear Science and Technology Organisation, Menai, Australia, ANSTO Materials Division, MAT/TN112 (2001)

23. Materials Selection for the HFIR Cold Neutron Source, K. Farrell, Oak Ridge National Laboratory, Oak Ridge, USA, ORNL/TM-99-208 (1999)

24. Correlation between Shear Punch and Tensile Data for Neutron-Irradiated Aluminium Alloys, M.L. Hamilton, D.J. Edwards, M.B. Toloczko, G.E. Lucas, W.F. Sommer, M.J. Boden, J.F. Dunlap, J.F. Stubbins, Pacific Northwest Laboratory, Richland, USA, STP1270 (1996)

25. Present Status on the Mechanical Characterization of Aluminium Alloys 5754-Net-O and 6061-T6 Irradiated at High Fluences, B. Kapusta, C. Sainte-Catherine, X. Averty, M. Scibetta, G.M. Decroix, M. Rommens, CEA Saclay, Gif sur Yvette, France, IGORR 2003

APPENDIX 2: LIST OF REPORTS FOR ZIRCONIUM AND ITS ALLOYS IN THE RRMPDB

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2. Hydride Formation in Zirconium Alloys, A.T. Motta, L.Q. Chen, Department of Mechanical and Nuclear Engineering, The Pennsylvania State University, University Park, USA, The Journal of The Minerals, Metals & Materials Society (2012)

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5. Blister Growth in Zirconium Alloys: Experimentation and Modelling, G. Domizzi, R. Enrique, J. Ovejero-Garcia, G.C. Buscaglia, National Atomic Energy Commission, Buenos Aires, Argentina, Journal of Nuclear Materials 229 (1996) 36-47

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12. Delayed Hydride Cracking Velocity in Zr-2.5Nb: Detection by Acoustic Emission and Theoretical Model Testing, J.I. Mieza, G. Vigna, E. Chomik, G. Domizzi, CNEA Atomic Centre Constituyentes, San Martin, Argentina, Journal of Material Science 43 (2008) 4945-4952

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15. Fracture Toughness of Zirconium Alloys, A.K. Hellier, Australian Nuclear Science and Technology Organisation, Menai, Australia, ANSTO Materials Division (2001)

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24. Phase and Texture Analysis of a Hydride Blister in a Zr-2.5%Nb Tube by Synchrotron X-ray Diffraction, MA. Vicente Alvarez, J.R. Santisteban, G. Domizzi, J. Almer, Atomic Centre Bariloche, San Carlos de Bariloche, Argentina, Acta Materialia 59 (2011) 2210–2220

25. Radiation Effects in Zirconium Alloys, F. Onimus, J.L. Bechade, French Atomic Energy Commission, Gif-sur-Yvette, France

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APPENDIX 3: LIST OF REPORTS FOR BERYLLIUM IN THE RRMPDB

1. Behavior of BeO under Neutron Irradiation, G.W. Keilholtz, J.E. Lee, Jr., R.E. Moore, R.L. Hamner, Oak Ridge National Laboratory, Oak Ridge, USA, Journal of Nuclear Materials 14 (1964) 87-95

2. Beryllium for Fusion Application – Recent Results
   A. Khomutov, V. Barabash, V. Chakin, V. Chernov, D. Davydov, V. Gorokhov, H. Kawamura, B. Kolbasov, I. Kupriyanov, G. Longhurst, F. Scaffidi-Argentina, V. Shestakov
   A.A. Bochvar Institute of Inorganic Materials, Moscow, Russian Federation

3. Beryllium Reflectors for Research Reactors: Review and Preliminary Finite Element Analysis
   P.S. Bejarano, R.G. Cocco
   INVAP S.E., San Carlos de Bariloche, Argentina

4. Beryllium Use in the Advanced Test Reactor
   G.R. Longhurst, R.D. Rohe
   Idaho National Laboratory, Idaho Falls, USA

5. Comparison of Compression Properties and Swelling of Beryllium Irradiated at Various Temperatures
   J.M. Beeston, L.G. Miller, E.L. Wood, Jr., R.W. Moir
   EG&G Idaho, Inc., Idaho Falls, USA

6. Distortion of Beryllium Reflectors Induced by Swelling
   P.S. Bejarano, R.G. Cocco
   INVAP S.E., San Carlos de Bariloche, Argentina

7. Effects of Neutron Irradiation at 70–200°C in Beryllium
   V.P. Chakin, V.A. Kazakov, R.R. Melder, Yu.D. Goncharenko, I.B. Kurpiyanov
   Research Institute of Atomic Reactors, Dimitrovgrad, Russian Federation

8. Evolution of Beryllium Microstructure under High-Dose Neutron Irradiation
   V.P. Chakin, Z.Ye. Ostrovsky
   Research Institute of Atomic Reactors, Dimitrovgrad, Russian Federation

9. Fracture Toughness of Beryllium
   D.L. Harrod, T.F. Henstenberg, M.J. Manjoine
   Westinghouse Astronuclear Laboratory, Mt. Lebanon, USA

10. Fracture Toughness of Irradiated Beryllium
    J.M. Beeston
    EG&G Idaho, Inc., Idaho Falls, USA

11. Helium Content and Induced Swelling of Neutron Irradiated Beryllium
    L. Sannen, Ch. De Raedt, F. Moons, Y. Yao
    SCK·CEN, Mol, Belgium

12. High Dose Neutron Irradiation Damage in Beryllium as Blanket Material
V.P. Chakin, V.A. Kazakov, A.A. Teykovtsev, V.V. Pimenov, G.A. Shimansky, Z.E. Ostrovsky, D.N. Suslov, R.N. Latypov, S.V. Belozerov, I.B. Kupriyanov
Research Institute of Atomic Reactors, Dimitrovgrad, Russian Federation

13. Interim Report of Post-Irradiation Results of ATR Beryllium Surveillance Program
G.E. Korth, J.M. Beeston
Idaho Nuclear Corporation, Idaho Falls, USA

14. Irradiated Beryllium Disposal Workshop
G.R. Longhurst, G.L Anderson, C.H. Mullen, W.H. West
Idaho National Engineering and Environmental Laboratory, Idaho Falls, USA

15. Mechanical properties of irradiated beryllium
J.M. Beeston, G.R. Longhurst and R.S. Wallace
Idaho National Engineering Laboratory, EC&G Idaho, Inc., USA

16. Microstructure of long-term annealed highly irradiated beryllium
A. Leenaers, G. Verpoucke, A. Pellettieri, L. Sannen, S. van den Berghe
SCKÆCEN, Reactor Materials Research, Begium

17. Pores and cracks in highly neutron irradiated beryllium
V. Chakin, R. Rolli, H.-C. Schneider, A. Moeslang, P. Kurinskiy, W. van Renterghem
Karlsruhe Institute of Technology, IMF, Germany

18. Post-Irradiation Studies of Beryllium Reflector of Fission Reactor – Examination of Gas Release, Swelling and Structure of Beryllium under Annealing
D.V. Andreev, V.N. Bespalov, A.Ju. Birjukov, B.A. Gurovich, P.A. Platonov
Russian Research Center “Kurchatov Institute,” Moscow, Russian Federation

19. Properties of Irradiated Beryllium – Statistical Evaluation
J.M. Beeston
EG&G Idaho, Inc., Idaho Falls, USA

20. Radiation Growth of Beryllium
V.P. Chakin, A.O. Posevin, A.V. Obukhov, P.P. Silantyev
Karlsruhe Research Centre, Karlsruhe, Germany

21. Stress and Deformation Analysis of Irradiation Induced Swelling
B.V. Winkel
Aerojet Nuclear Company, Idaho Falls, USA

22. TEM Investigation of Long-Term Annealed Highly Irradiated Beryllium
W. van Renterghem, A. Leenaers, S. van den Berghe
SCK·CEN, Mol, Belgium

23. Radiation Damage in Beryllium Oxide Single Crystals
J.C. Pigg, A.K. Garrison, S.B. austerman
Solid State Division, Oak Ridge National Laboratory, USA

APPENDIX 4: LIST OF REPORTS FOR STAINLESS STEELS IN THE RRMPDB

1. Analysis of Tensile Deformation and Failure in Austensitic Stainless Steels: Irradiation Dose Dependence
   J.W. Kim, T.S. Byun
   Department of Nuclear Engineering, Chosun University, Gwangju, Republic of Korea

2. Deformation Mechanisms in 316 Stainless Steel Irradiated at 60°C and 330°C
   N. Hashimoto, S.J. Zinkle, A.F. Rowcliffe, J.P. Robertson, S. Jitsukawa
   Metals and Ceramics Division, Oak Ridge National Laboratory, Oak Ridge, USA

3. Degradation of Mechanical Properties of Stainless Steel Cladding due to Neutron Irradiation and Thermal Aging
   F.M. Haggag
   Metals and Ceramics Division, Oak Ridge National Laboratory, Oak Ridge, USA

4. The Effect of Aging at 343°C on the Microstructure and Mechanical Properties of Type 308 Stainless Steel Weldments
   D.J. Alexander, K.B. Alexander, M.K. Miller, R.K. Nanstad, Y.A. Davidov
   Oak Ridge National Laboratory, Oak Ridge, USA

5. Effect of Cold Work on Tensile Behavior of Irradiated Type 316 Stainless Steel
   R.L. Klueh, P.J. Maziasz
   Metals and Ceramics Division, Oak Ridge National Laboratory, Oak Ridge, USA

6. Elevated Temperature Ductility of Types 304 and 316 Stainless Steel
   V.K. Sikka
   Metals and Ceramics Division, Oak Ridge National Laboratory, Oak Ridge, USA

7. Relationship between Hardening and Damage Structure in Austenitic Stainless Steel 316LN Irradiated at Low Temperature in the HFIR
   N. Hasimoto, E. Wakai, J.P. Robertson
   Metals and Ceramics Division, Oak Ridge National Laboratory, Oak Ridge, USA

8. Influence of Temperature and Grain Size on the Tensile Ductility of AISI 316 Stainless Steel
   S.L. Mannan, K.G. Samuel, P. Rodriguez
   Reactor Research Centre, Kalpakkam, India

9. Irradiation Damage in 304 and 316 Stainless Steels: Experimental Investigation and Modeling: Irradiation Induced Hardening. Part II: Irradiation induced hardening
   J.P. Massoud, C. Pokor, T. Brechet, P. Dubuisson, X. Averty
   EDF R&D, Moret sur Loing, France

10. Irradiation Effects on Material Properties for 304L Stainless Steel Base Metal and Welds
    M. Dadfarnia, P. Sofronis, I.M. Robertson
    University of Illinois at Urbana-Champaign, Urbana, USA

11. Mechanical Properties of Irradiated Types 304 and 304L Stainless Steel Welment Components
R.L. Sindelar, G.R. Caskey, Jr.
Savannah River Laboratory, Aiken, USA

12. Microstructure Formation and Its Role on Yield Strength in AISI 316 SS Irradiated by Fission and Fusion Neutrons
N. Yoshida
Research Institute for Applied Mechanics, Kyushu University, Kasuga, Japan

13. Neutron Irradiation-Induced Embrittlement in Type 316 and Other Austenitic Steels and Alloys
D.R. Harries
Metallurgy Division, Atomic Energy Research Establishment, Harwell, United Kingdom

14. Notch Toughness of Austenitic Stainless Steel Weldments with Nuclear Irradiation
J.R. Hawthorne, H.E. Watson
Metallurgy Division, Naval Research Laboratory, Washington, D.C., USA

15. Annual Report of JMTR, 1994
Department of JMTR Project
Oarai-machi, Oarai Research Establishment, Japan

16. Final Report about the ETRR-2 Facility and the Work Done as per Agreement
Y.E. Tawfik Ellethy
Egypt Atomic Energy Authority, Cairo, Egypt

17. Structural Integrity of Stainless Steel Componenets Exposed to Neutron Irradiation
M. Kamaya, T. Hooj, M. Mochizuki
Institute of Nuclear Safety System, Inc., Mihama, Japan

18. Tensile Properties and Damage Microstructures in ORR/HFIR-Irradiated Austenitic Stainless Steels
E. Wakai, N. Hashimoto, J.P. Robertson, S. Jistukawa, T. Sawai, A. Hishinuma
Japan Atomic Energy Research Institute, Tokai, Japan

19. Tensile Properties of Austenitic Stainless Steels and Their Weld Joints after Irradiation by the ORR-Spectrally-Tailoring Experiment
S. Jistukawa, P.J. Maziasz, T. Ishiyama, L.T. Gibson, A. Hishinuma
Japan Atomic Energy Research Institute, Tokai, Japan

20. Tensile Properties of Type 316L(N) Stainless Steel Irradiated to 10 Displacements per Atom
M.G. Horsten, M.I. de Vries
ECN-Nuclear Energy, Petten, the Netherlands

21. The Effects of Long-Time Irradiation and Thermal Aging on 304 Stainless Steel
T.R. Allen, J.I. Cole, C.L. Trybus, D.L. Porter
Argonne National Laboratory-West, Idaho Falls, USA

22. Irradiation damage in 304 and 316 stainless steels: experimental investigation and modeling. Part I: Evolution of the microstructure, J.P. Massoud, C. Pokor, T. Brechet, P. Dubuisson, X. Averty EDF R&D, Moret sur Loing, France

23. Investigation of mechanical properties of unirradiated austenitic stainless steel sample weldments, Yasser ELLETHY, ETRR2 – EAEA, Cairo - Egypt

APPENDIX 5: LIST OF REPORTS FOR GRAPHITE IN THE RRMPDB

1. A Study on the Irradiated Strength and Stress Evaluation of Nuclear Graphite Material
   Y.S. Lee, Y.M. Lee, J.H. Kim, S.J. Lee, Y.H. Kang, K.N. Choo, M.S. Cho
   Department of Mechanical Design Engineering, Chungnam National University, Daejeon, Republic of Korea

2. Ageing Data on Corrosion and Irradiation Induced Degradation of In-Core Components of Research Reactor Cirus
   R. Ranjan, S. Bhattacharya, P.V. Varde, P. Mandal, M.K. Ohja, Y.S. Rana, G.K. Mallik, V.D. Alur, J.S. Dubey, M. Altaf, J.P. Singh, S. Jeyakumar
   Bhabha Atomic Research Centre, Mumbai, India

3. Annealing of Neutron Damage in Graphite Irradiated and Stored at Room Temperature
   W.J. Gray, P.A. Thrower
   Pacific Northwest Laboratory, Richland, USA

4. Fine Structure of Wigner Energy Release Spectrum in Neutron Irradiated Graphite
   T. Iwata
   Department of Physics, Japan Atomic Energy Research Institute, Tokai, Japan

5. Studies on Cirus Graphite Reflector
   R.C. Sharma, P.V. Varde, R. Ranjan, V. Kain, J.S. Dubey, P.A. Jadhav, B.K. Tripathi
   Bhabha Atomic Research Centre, Mumbai, India

6. The Thermal Conductivity of Fast Neutron Irradiated Graphite
   R. Taylor, B.T. Kelly, K.E. Gilchrist
   Institute of Science and Technology, University of Manchester, Manchester, United Kingdom

7. Thermal and Structural Properties of Low-Fluence Irradiated Graphite
   D. Lexa, M. Dauke
   Canberra Packard Central Europe GmbH, Schwadorf, Austria

8. Thermal, Structural and Radiological Properties of Irradiated Graphite from the ASTRA Research Reactor – Implications for Disposal
   D. Lexa, A.J. Kropf
   Department of Radioactive Waste Management, Nuclear Engineering Seibersdorf, Seibersdorf, Austria

APPENDIX 6: LIST OF REPORTS FOR CONCRETE IN THE RRMPDB

1. Automatic System for Monitoring Corrosion of Steel in Concrete
   A. Poursaee
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2. Stress Corrosion Cracking of Stainless-Steel Canister for Concrete Cask Storage of Spent Fuel
   J. Tani, M. Mayuzumi, N. Hara
   Materials Science Research Laboratory, Central Research Institute of Electric Power Industry, Yokosuka, Japan

3. Reinforcement Corrosion in Concrete Structures, Its Monitoring and Service Life Prediction – A Review
   S. Ahmad
   Department of Civil Engineering, King Fahd University of Petroleum and Minerals, Dhahran, Saudi Arabia

4. Recent Durability Studies on Concrete Structure
   Z.J. Li, S.W. Tang, Y. Yao, C. Andrade
   The Hong Kong University of Science and Technology, Hong Kong, China

5. Examples of Reinforcement Corrosion Minoriting by Embedded Sensors in Concrete Structures
   I. Martinez, C. Andrade
   Institute of Construction Science "Eduardo Torroja," Madrid, Spain

6. Methods for Measuring pH in Concrete: A Review
   N. de Belie, A. Behnood, K. van Tittelboom
   Magnel Laboratory for Concrete Research, Ghent University, Ghent, Belgium

7. Implementation of Non-Destructive Techniques for the Monitoring of Rebar Corrosion of Concretes Developed for Nuclear Applications
   D. Vazquez, G. Duffo
   National Atomic Energy Commission, San Martin, Argentina

8. Sensor Systems for Use in Reinforced Concrete Structures
   W.J. McCarter, O. Vennesland
   School of the Built Environment, Edinburgh, United Kingdom

9. The Use of Permanent Corrosion Monitoring in New and Existing Reinforced Concrete Structures
   J.P. Broomfield, K. Davies, K. Hladky
   BGB Projects Ltd., London, United Kingdom

10. Experimental Study of the Effect of Radiation Exposure to Concrete
    K. Fujiwara, M. Ito, M. Sasanuma, H. Tanaka, K. Hirotani, K. Onizawa, M. Suzuki, H. Amezawa
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11. Implementation of Different Techniques for Monitoring the Corrosion of Rebars Embedded in Concretes Made with Ordinary and Pozzolanic Cements
D.R. Vazquez, Y.A. Villagran Zaccardi, C.J. Zega, M.E. Sosa, G.S. Duffo
Instituto Sabato, National University of General San Martin, San Martin, Argentina

12. Development of Solid State Embeddable Reference Electrode for Corrosion Monitoring of Steel in Reinforced Concrete Structures
V. Maruthapandian, V. Saraswathy, S. Muralidharan
Central Electrochemical Research Institute, Karaikudi, India

13. Long-Term Relative Performance of Embedded Sensor and Surface Mounted Electrode for Corrosion Monitoring of Steel in Concrete Structures
S. Muralidharan, S.P. Karthik, V. Saraswathy, K. Thangavel
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14. NDE of Thick and Highly Reinforced Concrete Structures: State of the Art
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15. Monitoring Chloride-Induced Corrosion of Carbon Steel Tendons in Concrete Using a Multi-Electrode System
M. Ormellese, A. Brenna, L. Lazzari
Polytechnic University of Milan, Milan, Italy

16. Ageing Management of Concrete Structures in Nuclear Power Plants
IAEA Nuclear Energy Series, NP-T-3.5
International Atomic Energy Agency, Vienna, Austria

17. Development of an Embeddable Sensor to Monitor the Corrosion Process of New and Existing Reinforced Concrete Structures
S.B Farina, G.S. Duffo
National Scientific and Technical Research Council, San Martin, Argentina

18. Non-Destructive Test Methods for Concrete Bridges: A Review
S. Kashif Ur Rehman, Z. Ibrahim, S. Ali Memon, M. Jameel
Department of Civil Engineering, University of Malaya, Kuala Lumpur, Malaysia

19. Development of a Galvanic Sensor System for Detecting the Corrosion Damage of the Steel Embedded in Concrete Structure – Laboratory Tests to Correlate Galvanic Current with Actual Damage
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20. Development of a Galvanic Sensor System for Detecting the Corrosion Damage of the Steel Embedded in Concrete Structure – Laboratory Electrochemical Testing of Sensors in Concrete
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21. Macrocell Sensor Systems for Monitoring of the Corrosion Risk of the Reinforcement in Concrete Structures
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22. Aging Effects on Structural Concrete and Long-Term Storage of Spent Nuclear Fuel in DCSS at ISFSIs in the USA
    B.P. Tripathi
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23. Towards a Passive Contactless Sensor for Monitoring Resistivity in Porous Materials
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24. Real-Time Monitoring Framework to Investigate the Environmental and Structural Performance of Buildings
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ANNEX I: RESEARCH REACTOR CORE COMPONENTS AND MATERIALS

In this table, the components that are generally fixed and difficult to replace are indicated by (1) and those that are more readily replaceable are indicated by (2).

COMPONENTS	MATERIALS
Pressure vessels (Tanks) + connecting pipes	
Pressure vessels (1)	Aluminium alloys Stainless steels Zirconium alloys
Low pressure tanks (1)	Aluminium alloys Stainless steels Zirconium alloys
Connecting pipes (1/2)	Aluminium alloys Stainless steels Zirconium alloys
Beam ports/nozzles (1)	Aluminium alloys Low alloy steel Zirconium alloys
Poolside irradiation facility (1)	Aluminium alloys Stainless steels
Liners (Pool type RR) (1)	Aluminium alloys Stainless steels Epoxy resins Ceramic tiles
Core support structure components	
Core support grid (1/2)	Aluminium alloys Stainless steels Zirconium alloys
Other structural and internal support structures	Aluminium alloys Stainless steels Low alloy steels
Core box/vessel (1/2)	Aluminium alloys Stainless steels Low alloy steels

COMPONENTS	MATERIALS
Guide Tubes (2) for:	
Control guide devices	Aluminium alloys Stainless steels Zirconium alloys
Experimental facilities	Aluminium alloys Stainless steels Zirconium alloys
Isotope production	Aluminium alloys Stainless steels Zirconium alloys
Reflector Devices/Systems	
Heavy water tank & accessories (1)	Aluminium alloys

	Zirconium alloys
Solid reflector (1/2)	Graphite Beryllium Aluminium-Be alloy Beryllium oxide
Sealing rings	Metals
Beam related devices, highly irradiated portion (2)	
Cold source	Zirconium alloys Aluminium alloys
Thermal column	Zirconium alloys Aluminium alloys
Neutron guide tubes	Aluminium alloys Stainless steels
Sealing rings	Metals
Shielding structures	
Shields	High density concrete (1/2) Structural concrete (1/2) Lead (2) Tungsten (2)
Cables	
Cable (2)	Organic insulation Elastomers

ANNEX II: IDEAL SET OF MATERIAL PROPERTIES

Material properties for an unirradiated material under high or very high neutron and gamma fluence are listed below for a given material and appropriate thermal-to-fast neutron ratios.

Environment:
1. Definition of thermal and fast neutron energies.
2. Thermal-to-fast ratio for each test (not fast-to-thermal).
3. Neutron flux in $n \cdot cm^{-2}s^{-1}$ and gamma flux (if required).
4. Irradiation (calculated/measured and average or maximum) and test temperatures
5. Duration of irradiation (fluence) and decay time prior to testing.
6. Coolant chemistry (e.g., for water: pH, conductivity; for NaK: mass ratio; for He: impurities)

Sample:

7. Definition of unirradiated specimen material (certificated composition and thermal-mechanical treatment, material properties, residual stresses etc.).
8. Standard used for specimen manufacture including dimensions and gauge length.
9. Location of specimens: base metal, weld metal, cross weld, etc.

The list below gives the main parameters of interest for most irradiated structural materials. Additional parameters should be provided, as appropriate, for materials for special applications.

Measurements and calculations:
10. Composition changes (chemical and isotopic) (calculated or measured).
11. Displacements per atom (with method of determination).
12. Formation of loops and dislocation density (m^{-2}).
13. Voids and bubble density and composition (1H, 3H, He) (% volume or m^{-3}).
14. Swelling (volume %).
15. Growth (% in each direction)
16. Corrosion rate (dimensional, mass changes).
17. Determine resistance to Stress Corrosion Cracking (SSR test, ECP-diagram).
18. Measure UTS, yield strength, 0.2% yield (MPa).
19. Measure ductility (uniform and total elongation) (%).
20. Measure fracture toughness ($MPa.m^{1/2}$).
21. Measure fatigue properties (threshold stress; N_f with definition).
22. Measure fatigue crack growth rate (da/dn).

Some examples:
1. Epoxy resins and adhesives: initial curing conditions and subsequent bond degradation with neutron and gamma fluence
2. Fastening and joining aspects: bolting (relaxation, galvanic), welding (heat affected zone, galvanic), compatibility of materials
3. Graphite: swelling, accumulation of Wigner energy
4. Zirconium alloys: growth, directionality of properties
5. Concrete: density, compression strength
6. Cable: insulation degradation.

LIST OF ABBREVIATIONS

bcc	body centered cubic arrangement of atoms in a metal matrix
CRP	Coordinated Research Project
DHC	Delayed Hydride Cracking
fcc	Face centered cubic arrangement of atoms in a metal matrix
hcp	hexagonally closed packed arrangement of atoms in a metal matrix
MS	Member State
NPP	Nuclear Power Plant
PKA	Primary Knock-on Atoms
RCM	Research Coordination Meeting
RR	Research Reactor
RRMPDB	Research Reactor Material Properties Database
SSC	Structure, System and Component

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3rd Research Coordination Meeting on
Establishment of Material Properties Database for Irradiated Core Structural Components for Continued Safe Operation and Lifetime Extension of aging Research Reactors,
IAEA Headquarters, Vienna, 10-13 April 2017

Consultants' Meeting (CSM) to finalize the IAEA Technical Document on Material Properties Database for Irradiated Core Structural Components for Lifetime Management for Long Term Operation of Research Reactors, IAEA Headquarters, Vienna, 12 – 14 February 2018

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19-00321

International Atomic Energy Agency
Vienna
ISBN 978–92–0–102519–7
ISSN 1011–4289