5.9 REACTOR MATERIALS

5.9.1 Introduction

The objective of this section is to describe the main structural materials to be used in the reactor core, support structures and main safety-related systems and to demonstrate that these materials are suitable for their intended purpose. The three basic structural materials used in the Reactor Facility internals and support structures are stainless steel, aluminium and zirconium alloys (Zircaloy/Zircadyne). When other materials are required, and their contact in use with water is not desirable, they will be protected by an external sealed cover made of any of the three basic materials.

Other materials are also used; for example such as polymers, in seals and protective coatings, and hafnium (Hf) as absorber material in the control rod plates. A list of the materials used for the main reactor components is shown in Table 5.9/1.

5.9.2 Stainless Steel

Low carbon stainless steel, AISI 304L, is used for the reactor pool. The chemical composition is shown in Table 5.9/2 (Metals Handbook Vol. 3, 1978). The nominal properties of these steels are given in Table 5.9/3 where AISI 304 stainless steel properties are also provided as a reference.

5.9.2.1 Irradiation Damage

Mechanical properties of stainless steel alloys change when they are subjected to irradiation. The main phenomena observed are swelling and irradiation-induced creep (Voyevodin, V. N., et al., 1999).

The swelling phenomenon depends on the operating temperature and neutron fluence. For the Reactor Facility, components will operate at temperatures below 70ºC, and are expected to see a lifetime fluence of approximately 1×10^{23} $n.cm^{-2}$. These conditions are well below the conditions where swelling becomes significant.

Irradiation-induced creep depends, for a given fluence, on the operating temperature and mechanical load. For the temperature and fluence expected for the stainless steel components of the Reactor Facility, radiation-induced creep will be negligible.

Irradiation-induced changes in the fracture toughness are not expected to be of significance under the operational conditions of the reactor. The design will accommodate any irradiation-induced changes to fracture properties.

Nevertheless, the design takes into account these effects and provides sufficient safety margins to ensure adequate component performance.

5.9.2.2 Welding

Austenitic stainless steels will be welded with fillers of similar material (ie. low carbon filler metals). Shielded Metal Arc Welding, Gas Metal Arc Welding or Gas Tungsten Arc Welding will be used with appropriate selection of protection gases and fluxes so that carburisation or an increase in the nitrogen content of the weld is avoided. This will ensure that problems such as hot-cracking or intergranular corrosion are avoided.

Degradation of corrosion resistance in heat-affected zones of the weld metal will be prevented by using low carbon steels and appropriate welding practices. The chemical composition and the temperatures during welding will inhibit the initiation of the precipitation of carbides (such as $M_{23}C_6$) in accordance with the Time-Temperature-Precipitation curve (Folkhard, E., 1987) shown in Figure 5.9/1.

The welding will be carried out with support to guarantee a minimum amount of ferrite in the weld in order to avoid weld cracking.

5.9.2.3 Corrosion

Stainless steels have strong resistance to corrosion over a wide range of environments and temperatures. The reactor pool and primary circuit water is demineralised water with controlled low conductivity of less than 100 μSm^{-1}. No failure mechanism is known under such process conditions provided the quality of the water is maintained and the temperature remains below 60°C. Analysis conducted to study these alloys suggests that the corrosion resistance of stainless steel (304 L) is good under the expected operating conditions for reactor.

Contamination of the stock material, welding consumables and completed components by harmful substances will be avoided during all handling, storage and manufacturing operations.

Intergranular corrosion and intergranular stress corrosion will not present any problem under the reactor operational environment (Folkhard, E., 1987).

5.9.2.4 Dissimilar Metal Joints

Stainless steel-to-aluminium joints are acceptable within the operating temperature range and water quality requirements for the reactor.

Stainless steel-to-Zircaloy joints are discussed in Section 5.9.6.

5.9.2.5 Conclusions

AISI 304L stainless steel has been selected for use in the Reactor Facility because of its excellent performance under the operating conditions in the reactor. Corrosion and irradiation damage do not present a safety hazard for the components to be constructed with this material. To avoid corrosion, an operational limit will apply to the reactor water quality (conductivity to be less than 100 μSm^{-1}) to ensure that the performance is as predicted.

5.9.3 Aluminium Alloys

Aluminium alloys will be used for the construction of some reactor internals. Table 5.9/4 shows the designation of these alloys with that of 1100 grade pure aluminium (Davis, J. R., 1994) given as a comparison.

5.9.3.1 Irradiation Damage

Aluminium alloys are suitable materials for use in components working in radiation environments as their properties are well understood and show predictable behaviour under such conditions.

Aluminium is extensively used in water-cooled research reactors because of its low cross section for the capture of thermal neutrons, excellent corrosion resistance and thermal conductivity. The most common alloys used are the 5000 and 6000 series. Table 5.9/5 presents data on these alloys with data for the 1100 series alloy included as a comparison.

The aluminium alloys to be used in Reactor Facility components are solid solution or precipitation hardened alloys that provide satisfactory mechanical properties for the intended service in the Reactor Facility. In alloys of the 5000 series, the increase in strength over pure aluminium is provided by Mg in solid solution. This alloy has good properties as regards to formability and resistance to corrosion in water flows. In the case of alloys of the 6000 series, the increase in strength is achieved by an ageing heat treatment that forms a fine precipitate of Mg_2Si.

Table 5.9/6 shows a range of mechanical properties for the two aluminium alloys at fluences of around 0.8 to 2.0 x 10^{23} n.cm^{-2}, the expected maximum fluence that these components will experience. Mechanical properties at room temperature and at 100°C are presented.

In the case of the 6061-T6 aluminium alloy, samples were irradiated to 0.81 x 10^{23} n.cm^{-2} at a Φ_{th}/ Φ_f neutron ratio of 2. The test samples were irradiated at 95°C and tested at 25°C and 95°C (Alexander, D. J., 1999). An increase of approximately 7% wt Si can be expected at these fluences, while swelling is expected to be below 1%. Increases of between 15 and 25% in yield strength and ultimate strength, with a reduction in the total elongation of 30% have been measured (Weeks, J. R., et al., 1990; Dunlap, J. A., et al., 1996).

Results for 5052-0 aluminium irradiated to fluences of 2 x 10^{23} n.cm^{-2} at a temperature of 55°C and tested at 50°C showed an increase in yield strength of more than 500% and an increase in ultimate tensile strength of ~200%. A reduction in the total elongation from ~25% to 7% was also observed.

The fracture toughness of 6061-T6 has been shown to be adequate at neutron fluences in the order of 0.8 x 10^{23} n.cm^{-2}. ASME Nuclear Code Case N519 recommends that a limit of 25.3 MPa√m be adopted for this alloy. All samples tested by Alexander (1999) up to 95°C were above this value using the Kj method of testing fracture toughness.

5.9.3.2 Welding

Aluminium alloys are readily welded by employing welding processes with gas protection (TIG welding). Welding has been carried out in an atmosphere of argon or helium. In addition, the shielding gas has been used to regulate the cross section of the seam and the deposition speed.

The welding process and procedure selection has ensured appropriate performance of these welds.

5.9.3.3 Corrosion

Aluminium alloys develop an oxide surface layer of the order 10 Å thick that protects the alloy from corrosion. The thermodynamic stability conditions of the oxide film are usually represented on a Pourbaix diagram of applied potential vs. pH. The conditions for aluminium passivity (that is, it is protected by the oxide) are between pH 4 and pH 8.5. In this pH range and good-quality water, the oxidation rate depends on temperature. In the temperature range between 70°C and 80°C, oxidation rates are sufficiently low that the materials can be used for extended periods. This is evidenced by the behaviour of similar materials in research reactors that operate at comparable temperatures, such as HIFAR, which has operated at 50°C over more than 40 years of service, and HFBR in the United States (Farrell, K., 1995). It is noted that the reactor will operate at temperatures below 70°C.

Aluminium alloy susceptibility to crevice corrosion, galvanic corrosion and stress corrosion will be minimised by controlling water quality. The conductivity will be controlled and maintained below 100 μSm^{-1} with neutral pH, and adequate water circulation will be ensured (a minimum velocity of 0.04 ms^{-1} is required).

Due to the quality of the water used in the reactor, corrosion-erosion is not considered to be a problem. It has been shown, in laboratory scale, that the severity of this phenomenon is dependent on the quality of the water. Tests at the Savannah River Laboratory show that erosion of aluminum would not be a problem until the velocities exceeded about 30 ms^{-1} (velocities in the core area of the reactor will be around 10 ms^{-1}).

In summary, corrosion resistance for the aluminium alloys to be used in the Replacement Reactor Facility is very good in both the welded and un-welded condition, particularly when the pH and conductivity of the water are tightly controlled. The following table shows the corrosion rates in seawater for aluminium 6061 to be used in the Reactor Facility, and for aluminium 1100 as comparison.

5.9.3.4 Dissimilar Metal Joints

Aluminium-to-Zircaloy joints are acceptable under the water quality conditions that will apply for the Reactor Facility. It is particularly important to control the presence of chlorides. Aluminium-to-stainless steel metal joints have already been discussed in Section 5.9.2.

5.9.3.5 Conclusions

AA 6061, AA 5456 and AA 5052, widely used in research reactor applications, are suitable aluminium alloys for use in Reactor Facility components. This is because of their excellent resistance to corrosion and their predictable changes in properties under irradiation.

5.9.4 Polymers

5.9.4.1 Irradiation Effects

In general, the irradiation conditions, manufacturing techniques and composition are the main factors that determine the radiation resistance of polymers. Irradiation damage of polymers strongly depends on the fluence received by the material. Figure 5.9/2 shows the relative radiation resistance of the most common thermoplastic, and elastomers (Parker Hannifin Corporation).

The damage caused by irradiation over a long period of time may be more important than damage by irradiation to the same total dose for a short time (i.e. at high dose rates). The dose rate effect is, in addition, dependent on the chemical structure of the material itself. The amount of oxygen present is a function of the sample thickness, its permeability for gases and the amount of stabilizers added to the polymer to control oxidation damage under normal ageing conditions. For example, it is known that the effect is more pronounced in polyolefins [e.g. polyethylene (PE)], but is usually not of great importance for polyvinylchloride (PVC) and ethylene-propylene rubber (EPR) (Beynel, P., et al., 1982).

The best performances in the case of rubber are those of butadiene-styrene (Buna-S), ethylene-propylene rubber – EPR, polyurethane rubber - EPDM (Nordel, Desmopan), and finally, ordered by decreasing performance, natural vulcanised rubber. The best performances of thermoplastics are those registered by polyvinyl butyral, polystyrene, polyamides (Kapton, Kinel) and polyvinyl acetate (PVA) (Mounchanin, M., and Thibault, X., 2000).

Comparing the properties under irradiation of some polymers, the elastomers show comparable resistance and greater strain than thermoplastics (Figure 5.9/3).

Irradiation effects of polymers also depend on their composition and molecular structure. In the case of natural rubber-polypropylene, if the content of natural rubber is high, irradiation induces an increase in the tensile strength. Where the content of polypropylene is high, irradiation reduces the strength. This is shown in Figure 5.9/4.

Elastomers are widely used as sealing materials in nuclear applications where they are required to withstand relatively low dose rates over long periods of time. The sealing force exerted by a compressed seal tends to decay with the time; this stress relaxation can be temporary, due to a physical rearrangement of the rubber molecules, or permanent, when cross-linking occurs. The decay of this stress with time in a compressed elastomer is a well-known phenomenon and is accelerated by high temperatures or ionising radiation.

The comparison between EPR (Ethylene-propylene rubber) and Viton (Fluorinated copolymer) o-rings is shown in Figure 5.9/5. The elongation at failure for EPR reaches the 100% end-point at 6 x 10^7 rad, while for the case of Viton the elongation at break point decreases very rapidly, reaching 100% at 2.5 x 10^7 rad (extrapolated). The tensile strength and the hardness of Viton o-rings increase with irradiation, however, EPR shows a decrease in these properties with irradiation. This suggests that chain scission and cross-linking are the dominant processes in Viton, particularly at high irradiation doses. A similar behaviour has been found for another fluorinated polymer: tetrafluorethylene-propylene rubber and behaviour with respect to sealing appears to be similar (Clegg, D. W., 1991).

The retained sealing force obtained on Viton seals (vinylidene fluoride-hexafluoropropylene copolymer) decreases with irradiation dose (γ irradiation), becoming effectively zero at 0.45 MGy at 250°C and 0.6 MGy at 200°C (Burnay, S. G., and Hitchon, J. W., 1985).

Another polymer assessed is polyurethane rubber (PUR) which is used for the fuel clamp and control rod seals. As a guideline, PUR is usable at gamma doses smaller than 10^5Gy.

Tests have been performed by placing PUR seals in the ASTRA reactor (CERN, 1982). The irradiation position has the following characteristics:

a) gamma radiation field characteristic of the reactor FA (0.5-3 MeV)

b) radiation field medium: air

c) temperature: 25-35ºC

d) dose rate: 10^4 – 10^5 Gy

e) dosimetry method: ionisation method

On the specific seals made of PUR no damage has been reported for dose smaller than 10^5Gy.

Tests have also been performed in the RA-6 reactor at the Bariloche Atomic Centre (Argentine Atomic Energy Commission) on seals identical to those to be used in the Reactor Facility, and they have confirmed the good behaviour of this material.

5.9.4.2 Conclusion

Polymers, used in o-rings or hoses, have been demonstrated to be suitable for use in the Reactor Facility.

5.9.5 Hafnium Alloy

5.9.5.1 Introduction

Hafnium was introduced in 1980 as a neutron absorber for use in control rods, to be utilized in addition to carbon tetra boride as a replacement for silver-indium-cadmium. Hafnium control rods were introduced as original equipment in several pressurised water reactors. Actually, the program developed by FRAMATOME in the new Harmoni 2G RCCA design is an example of the excellent behaviour of Hf (Mounchanin M. and Thibault X., 2000 and De Perthuis, S., 1995).

Control rods are exposed to integrated neutron/thermal flux and to temperature throughout their lifetime. The irradiation effects for the Hf absorber are low.

5.9.5.2 General Characteristics of Hf as Control Element

Hafnium metal is a superior control material for light-water-cooled reactors. A unique combination of properties, including high neutron absorption cross-section, excellent corrosion resistance, good strength, ease of manufacturing and welding capacity, resistance to irradiation damage and excellent dimensional stability, makes it an ideal material for this application. It is mainly used within the nuclear and aerospace industries.

ASTM B 776 gives two types of Hf, (Grade1: Nuclear and Grade 3: Commercial). Table 5.9/6B shows the chemical composition of Hf in accordance with ASTM, Atucha I absorbers and Wah Chang commercial products (HAFNIUM – Wah Chang).

5.9.5.3 Physical and Mechanical Properties

Table 5.9/6C shows the mechanical properties whilst Table 5.9/6D shows a summary of physical properties for the hafnium alloy.

Hf is a refractory material found in nature together with Zr, its crystalline structure being given as hexagonal compact (hcp) for temperatures below 1760^{o}C and as centred body (bcc) for higher temperatures.

Hafnium is a relatively hard metal. In appearance, the metal is somewhat darker than either zirconium or titanium products (HAFNIUM – Wah Chang).

In view of its crystallographic nature, Hf is anisotropic, i.e. in worked or rolled products, properties such as thermal expansion, mechanical resistance and ductility vary in accordance with directionality (Haygarth, J. C. and R. A. Graham, 2000).

5.9.5.4 Corrosion Resistance

The corrosion resistance of Hf is high and the electrochemical potential is nearest to Zr or Zr alloys. At RRRP operating temperature it is not expected to react with other elements. In other words, Hf initiates its reaction slowly with air and oxygen at 400^{o}C, forming Hf oxide, nitrides with nitrogen at 900^{o}C and hydrides with hydrogen at 700^{o}C. Therefore, the Hf control rod may be built without cladding (Haygarth, J. C. and R. A. Graham, 2000 and Kennard, M. W. and J. E. Harbottle, 2000).

5.9.5.5 Irradiation Effects

Hf yields better performance than Ag-In-Cd and B_4 C, for it experiences no growth, swelling or creep (Mounchanin M. and Thibault X., 2000). These phenomena are related to the material's fusion temperature and the stress such material must withstand. Hf has a fusion temperature of 2227^{o}C. Considering the RRR operation temperature and the design of the control rods, these phenomena are irrelevant.

Since its implementation as control material, its application evolved and it has been used in control rods fully built in it for nuclear submarines, PWR and BWR reactors, e.g. the Atucha I reactor and several research reactors throughout the world, such as MAPLE, ORPHEE, OSIRIS, SCARABEEN, SILOE, FRG-1, JMTR, etc.

5.9.6 Zirconium Alloys

A comprehensive analysis has been undertaken for the zirconium alloys intended for use in the Reactor Facility. A series of tests has been undertaken aimed at providing confidence in the prediction of material behaviour over time.

Nuclear characteristics, good resistance to corrosion, good mechanical properties and good ductility make zirconium alloys the most 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 (Low Hf content). Zircaloy 4 and Zircadyne (grade R60702) has a predominant α (hcp) phase rich in Sn with Fe, Cr and Ni present in the inter-metallic precipitates. Zr-2.5 Nb alloys (grade R60705) 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.

5.9.6.1 Irradiation Damage

Zirconium alloy properties are modified through the effects of radiation. 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 Reactor Facility. For example, the core chimney wall will receive a neutron fluence of the order of $4{,}9\cdot10^{22}$ n.cm^{-2} over 40 years and will operate at temperatures around 60ºC.

As a result of this 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 Zircaloy, a hardening of the material has been observed at high fluence, a phenomenon related to the dissolving of the precipitates during irradiation (Adamsom, R. B., 2000).

The fracture toughness of Zircaloy-2 increases substantially as temperature rises (from 0^{o}C to 250^{o}C) for the base material as well as for welds. On the other hand results for Zircaloy-4 show that it does not register a drop in K_{IC} (plain-strain fracture toughness) at fluences up to 10^{21} n cm^{-2} at 260^{o}C.

Non-nuclear-grade zirconium alloys - R60705 and 60702 – will be used in those places where there are no low neutron absorption cross-section requirements, such as flanges, frames, piping connections, external piping (heavy water outlet and inlet, venting pipe), skirts, CNS and HNS housing assembly, brace and brace anchoring, etc. Zirconium alloys - R60702 and R60705 – will be submitted to accumulated integrated fluxes below 10^{20} n.cm^{-2} during the 40 years of reactor operation. This integrated flux value is quite below the thresholds in which noticeably growths or modifications may be registered as to mechanical properties (Price, E. G., CANDU Pressure Tubes).

5.9.6.2 Irradiation Growth

The irradiation of zirconium and its alloys causes the component to grow. Irradiation growth is a directionally oriented increase in dimension that is determined by the preferred orientation of the grain structure or texture. The magnitude of the growth is dependent on the metallurgical state (texture and thermal treatment). The growth rate is also influenced by temperature and creep.

As stated, the growth mechanism is strongly correlated with the microstructure. For cold-worked materials, the initial growth with fluence is linear and its magnitude depends on the strain level within the material. For recrystallised materials at low integrated fluxes (approximately 10^{21} n.cm^{-2}), there is a smaller initial growth (< 0.1%) that soon saturates (Holt, R. A., et al., 1996; Adamson, R. B., 1977; Rogerson, A., 1988).

From the comparison between the irradiation-induced growth and microstructural relationships for binary and multicomponent zirconium alloys, it follows that the pre-transition growth is related to defects. The accelerated growth stage at high fluences is related to the formation of defects.

At high temperatures (typical of nuclear power plants) and high neutron fluences a phenomenon of a change in the growth rate (also called breakaway) (Shishov, V. N., et al., 1996) has been observed. Temperature has a large effect on the level of neutron fluence at which breakaway occurs; as the temperature drops, the initiation of the rise in growth rate moves towards higher neutron fluences (Holt, R. A., 1988; Gonzalez, H., and Fortis, A. M., 2000).

At low temperatures such as the ones found in the reactor the available information shows that the effects of temperature, fluence and alloying elements on microstructure is such that the vacancy mobility (below 130^{o}C) is negligible. This, together with the influence of defects on zirconium alloy microstructure, leads to the conclusion that it is unlikely that the accelerated growth phenomena would occur under the operating conditions expected in the reactor. (Griffiths, M. 1988 and Koike, M. H., T. Akiyama, K. Nagamatsu and I. Shibahara., 1994).

The experimental data on irradiation growth of zirconium alloys, under reactor operating conditions (low temperature and high fluence) does not show evidence of breakaway. As the available database only covers doses up to 2.16×10^{22} n.cm^{-2} a Materials Surveillance Program will be implemented in order to assess the behaviour of Zircaloy-4 under reactor operating conditions. Additional details on the materials surveillance program are provided in Section 5.9.7. (Alberman, A., H. Carcreff, C. Morin and J. L. Poinssot, 1996 and Gonzalez H. and Fortis A. M., 2000).

5.9.6.3 Welding

The welding process and procedures will take into account the provision of the necessary protection of the weld and heat affected zone with inert gases to prevent weld contamination from oxygen and nitrogen (Metals Handbook, 1993).

The Tungsten Inert Gas welding process will be used with argon gas acting as the protection gas. A local protection system will be used to protect the weld and the heat affected zone until the material cools down sufficiently to ensure weld stability and to avoid contamination (Coleman, Ch. E., et al.,1994).

Post-welding thermal treatments will be used to improve the toughness and reduce the local residual stresses induced by the fusion of Zircaloy-4 during welding.

5.9.6.4 Corrosion

The most important area where zirconium alloys will be used in the Reactor Facility is for the Reflector Vessel, including the chimney surrounding the core. This environment will have demineralised water on the outer side with controlled low conductivity ($< 100\mu Sm^{-1}$) and neutral pH, and heavy water with the same conductivity on the inside of the Reflector Vessel.

Various corrosion processes have been analysed; including pitting corrosion, stress corrosion, erosion-corrosion, hydrogen effects and thinning effects (Burkart, A., et al., 2000; Conference Proceedings, 1999). Under the strict chemical control regime proposed for the light and heavy water for the reactor, none of these effects will be significant. The control of the relevant parameters of both light and heavy water is the key factor to ensuring that the effects of these processes are minimised.

5.9.6.5 Dissimilar Metal Joints

Zircaloy-to-stainless steel joints are acceptable and the effects of the oxidation process will not be significant for temperatures below 70ºC.

Zircaloy-to-aluminium joints will also be suitable for the water quality conditions in the reactor.

Failures in metal joints induced by stress corrosion will not occur under the operating conditions of the reactor. Study shows that dissimilar metal joints would produce corrosion only in the event of deterioration of water quality. Water quality control is the most important factor ensuring the optimum performance with these types of joints.

5.9.6.6 Conclusions

Observations made on different components made of Zircaloy alloys in reactor applications (for example, hot cells, pools, tubes, etc.) indicate that zirconium alloys are highly durable. For the conditions existing in the Reactor Facility, it is estimated such Zircaloy alloy components will be adequate for operation periods between 50 and 100 years, well beyond the Reactor Facility lifetime of 40 years, depending on the fluence.

Studies show that the three materials are suitable for application in the Reactor Facility. Zircaloy-4 presents more advantages when compared with Non-nuclear Zirconium alloys, in the areas of welding and properties under irradiation (microstructure) as it has a more stable phase structure. Non-nuclear Zirconium alloys, however, shows better mechanical properties than Zircaloy-4 and may be used in those applications requiring higher mechanical strength and in those places where there are no low neutron absorption cross-section requirements, such as: flanges, frames, piping connections, external piping (heavy water outlet and inlet, venting pipe), skirts, CNS and HNS housing assembly, brace and brace anchoring, etc.

5.9.7 Materials Surveillance Plan

A Materials Surveillance Plan has been developed and will be implemented from the commencement of reactor operation. The materials to be used to construct the components of the Reactor Facility are widely used in other nuclear reactors. Some of them, such as Zircaloy-4, were developed for more demanding conditions (high temperature, pressure and doses) than those conditions to be present in the reactor. Despite this, the surveillance plan will be implemented in order to guarantee the integrity of the Reactor Facility components. The plan will be based on the guidelines provided in the ASTM E185 standard. Although this standard is designed for light-water power reactors, it provides useful guidelines for designing and implementing a surveillance program.

The Materials Surveillance Plan is mainly oriented towards monitoring the mechanical properties, irradiation induced growth, corrosion and swelling of components such as the Reflector Vessel, the Chimney and other structures associated with the reactor core.

The plan includes the use of the stems of depleted control rods, which will be made of Zircaloy 4, to obtain specimens for Charpy and tensile tests. The first control rod replacement will take place after 12 years of operations. During this period, the control rod stems will receive a fluence equivalent to that to be received by the core chimney wall during 40 years of operation. This will allow knowing in anticipation the behaviour of the core chimney material regarding swelling and mechanical properties.

The use of Zircaloy 4, Aluminium and Stainless Steel specimens is included in the plan as well. These specimens will be subject to the same conditions regarding temperature, pressure and irradiation conditions than the structures and components under surveillance. The specimens will be placed in containers that will allow the contact of specimens with water. Dosimeters will be placed inside the containers to gather information on irradiation parameters. The specimens will be tested gradually to evaluate the behaviour of material properties during the first years of operation. The number of containers, the number and type of specimens per container, the orientation of the specimens inside the container and the container withdrawal schedule will be guided by the ASTM E185 standard.

5.9.8 References

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End of Section

Table 5.9/2 Nominal Composition of Stainless Steel (wt%)

Type	UNS	Fe	C	Mn	Si	Cr	Ni	Mo
304	S30400	Balance	0.08 max	2 max	1 max	18-20	8-12	-
304 L	S30403	Balance	0.03 max	2 max	1 max	18-20	8-12	-

Table 5.9/3 Nominal Mechanical Properties of Stainless Steels at Room Temperature

Type	TS (MPa)	0.2% YS (MPa)	El. (%)	RA (%)	HRB (Rockwell)
304*	515	205	40	40	88 max
304L*	480	170	40	40	88 max

* Annealed, TS: Tensile Strength, YS: Yield Strength, El: Elongation (in 50mm gauge length), RA: Reduction of Area, H: Hardness.

Table 5.9/4 Wrought Aluminium Alloys Designation

Aluminium Association	UNS No.	ISO No. R209	ASTM
5456	A95456	AlMg5Mn1	SB-209 (plates)
1100	A91100	Al99.0Cu	
5052	A95052	AlMg2.5	
6061	A96061	AlMg1SiCu	

Table 5.9/5 Typical Mechanical Properties of Various Aluminium Alloys (Metals Handbook, Vol. 2, 1978)

Alloy Temper	UTS (MPa)	TYS (MPa)	El - 50 mm (%)		H (HB)	USS (MPa)	FEL (MPa)	E (GPa)
			1.6 Thick	12.5 Diameter				
1100-O	90	34	35	45	23	62	34	69
5052-O	195	90	25	27	47	125	110	70
5052-H38	290	255	7	7	77	165	140	70
6061-T651	310	275	12	17	95	205	97	69
5456-O / H	310/350	150/250	24/16	-	90	207	-	70.3
	520*	170*	33*	-	-	-	-	

UTS: Ultimate Tensile Strength, TYS: Tensile Yield Strength (0.2% offset), El: Elongation, H: Hardness, USS: Ultimate Shearing Strength, FEL: Fatigue Endurance Limit ($5x10^8$ cycles), E: Elasticity module.

(*): Properties given at cryogenic temperatures (-253 °C)

Table 5.9/6 Strength and Ductility Measurements for Al 5052 (Farrell, K., 1981) and Al 6061

Alloy	Temperature Irrad./Test (^oC)	Yield Strength (MPa)	Ultimate Strength (MPa)	Total Elongation (%)
Non-irradiated				
6061-T651	- /21	262	299	18.4
5052-O	- /50	95.5	207	25
6061-T651	- /95	264	304	19
5052-O	- /100	92	180	40
Irradiated *				
6061-T651	95/25	351	384	12.8
5052-O	55/50	579	605	7.5
6061-T651	95/95	325	349	13
5052-O	55/100	540	540	7

5052-0 was irradiated to 2 x 10^{23} n cm^{-2} at a $\Phi_{th}/\ \Phi_f$ neutron ratio of 1.7.

6061-T651 was irradiated to 0.81 x 10^{23} n cm^{-2} at a $\Phi_{th}/\ \Phi_f$ neutron ratio of 2. These values were averaged data from Alexander's work.

Table 5.9/6A: Corrosion Rates for Aluminium in Seawater

Seawater	Weight loss [gr/yr] (*)	Maximun depth of pitting [mils/yr] (*)	Weight loss [mg/m^2] (**)
6061-T6	≤ 4.4	≤ 11.6	≤ 19.1
1100 – H14	≤ 1.27	≤ 2.9	≤ 6.4

(*): Test Specimen: 6.35x305x305 mm y 1.6 kg

(**): Test specimen: 0.193 m^2

Source: Aluminum and Aluminum Alloys – ASM Handbook

Table 5.9/6B: Chemical Composition of Hafnium Alloy

Elements	ASTM - Composition, [%Wt]		Atucha I, [ppm]	Wah Chang
	Grade 1	Grade 3		
Aluminium	0.010	0.050	≤ 100	70
Carbon	0.015	0.025	≤ 150	30
Chromium	0.010	0.050	≤ 200	20
Hydrogen	0.0025	0.0050	≤ 20	-
Iron	0.050	0.0750	≤ 750	250
Nitrogen	0.010	0.015	≤ 50	10
Oxygen	0.040	0.130	≤ 300…1500	100
Silicon	0.010	0.050	≤ 50	50
Titanium	0.010	0.050	≤ 100	50
Tungsten	0.0150	0.0150	-	10
Zirconium	(1)	(1)	≤ 4.5 (%wt)	1.5 - 4.5 (%wt)
Hafnium	Balance	Balance	≥ 95.3 (%wt)	Balance

(1): Admissible zirconium levels shall be established by mutual agreement between procurer and producer and reported. Grade 1: Mo: 0.002, Ni: 0.005, Nb: 0.01, Ta: 0.02, Sn: 0.005, U: 0.0010, Va: 0.0050, Cu: 0.010. Atucha I: Cl ≤ 500, Co ≤ 10, Mg ≤ 600, Mn ≤ 20, Mo ≤ 20, Nb ≤ 100, Ni ≤ 50, U-235 ≤ 0.07. Wah Chang: Ni: 50, U: 5, Mo: 5, Ni: 50.

Table 5.9/6C: Mechanical Properties for Hafnium Alloy

Grade	Condition	Testing Temperature	TS (MPa)	YS (MPa)	EI (%)
Hf	Long. / Annealed	RT	400	151	20
	Trans. / Annealed		310	172	25

Table 5.9/6D: Physical Properties for Hafnium Alloy

Atomic Number		72
Atomic Weight		178.49
Density [kg/m^3]		13.31×10^3
Elasticity modulus [Pa]	at 21°C at 260°C at 371°C	13.7×10^{10} 10.6×10^{10} 9.5×10^{10}
Magnetic Susceptibility [per g]	at 298.2K	0.42×10^{-6}
Hall Effect [V-m/A-T]	at 250ºC	-1.62×10^{-12}
Spectral Emissivity, 1750-2300K, e,.65,		0.40
Electron Emission [A/m^2]	at 1627°C at 1727°C at 1827°C at 1927°C	4.80×10 2.62×10^2 1.23×10^3 4.85×10^3
Work Function [J]		6.25×10^{-19}
Thermal Neutron Absorption Cross Section [Barns]		104
Crystal Structure	Alpha Phase Beta Phase	HCP BCC
Transformation Temperature [K]	Alpha to Beta	1760
Melting Point [K]		2230
Boiling Point [K]		4600
Linear Thermal Expansion Coefficient [(m/m)/K]	0-1000°C	5.9×10^{-6}
Thermal Conductivity [Watt/m K]	at 25.2°C 100.2ºC	23.0 22.4
Specific Heat [J/(kg K)]	25.2°C	117.0
Vapour Pressure [Pa]	at 1767°C at 2007ºC	10^{-5} 10^{-4}
Electrical Resistivity, [Ω m]	at 25.2°C	3.51×10^{-7}
Electrical Resistivity Temperature Coefficient [Ω m/K]	at 25.2°C	3.82×10^{-11}
Latent Fusion Heat [J/kg]		1.35×10^5
Latent Vaporization Heat [J/mole]	at 25.2°C at 723ºC at 1727ºC	7.02×10^5 6.99×10^5 6.96×10^5

Table 5.9/7 Nominal Compositions of Zirconium Alloys (wt%)

Alloy	Sn	Fe	Cr	Ni	O	Nb	Hf
Zircaloy-2	1.50	0.13	0.10	0.06	0.12	-	0.01
Zircaloy-4	1.50	0.21	0.10	0.007	0.12	-	0.01
Zr-Nb	-	-	-	0.007	0.10	2.5	-

Table 5.9/8 Mechanical Properties of Zirconium Alloys

Alloy	Condition	Direction of test / Temperature	Tensile Strength (MPa)	Yield Strength (MPa)
Zirconium	Annealed	Longitudinal / Room	296	138
		Transverse / Room	296	207
Zircaloy-2	Annealed	Longitudinal / Room	400	241
		Transverse / Room	386	303
Zircaloy-4	Annealed	Longitudinal / Room	400	241
		Transverse / Room	386	303
Zr-2.5 Nb	Annealed	Longitudinal / Room	448	310
		Transverse / Room	448	344

End of Tables

Figure 5.9/1 Time Temperature Precipitation $M_{23}C_6$ Precipitation Diagram and Intergranular Corrosion Attack Area of AISI 304 [02]

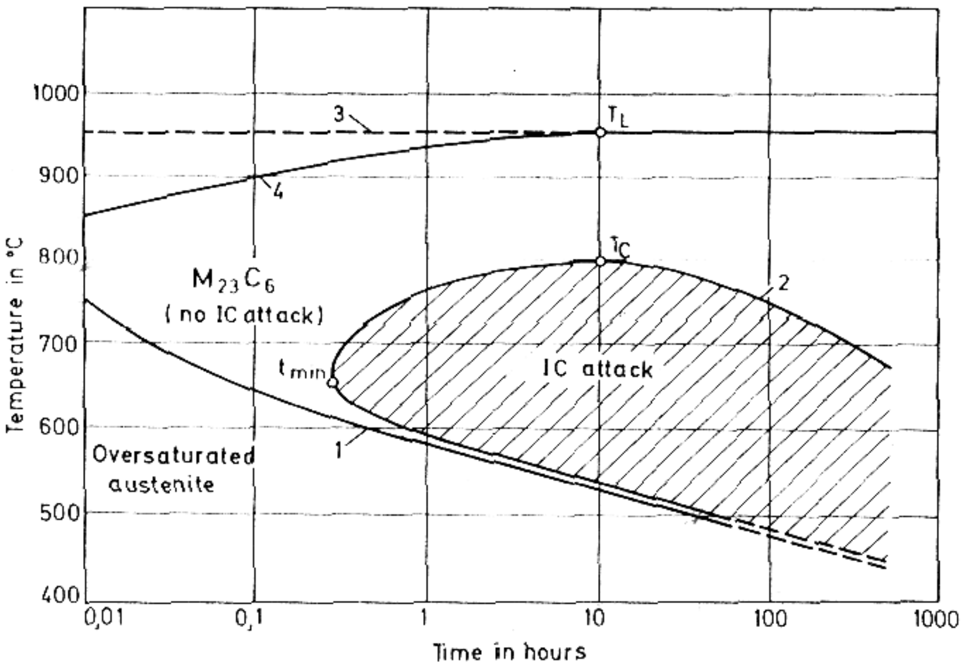


Figure 5.9/2 Relative Radiation Resistance

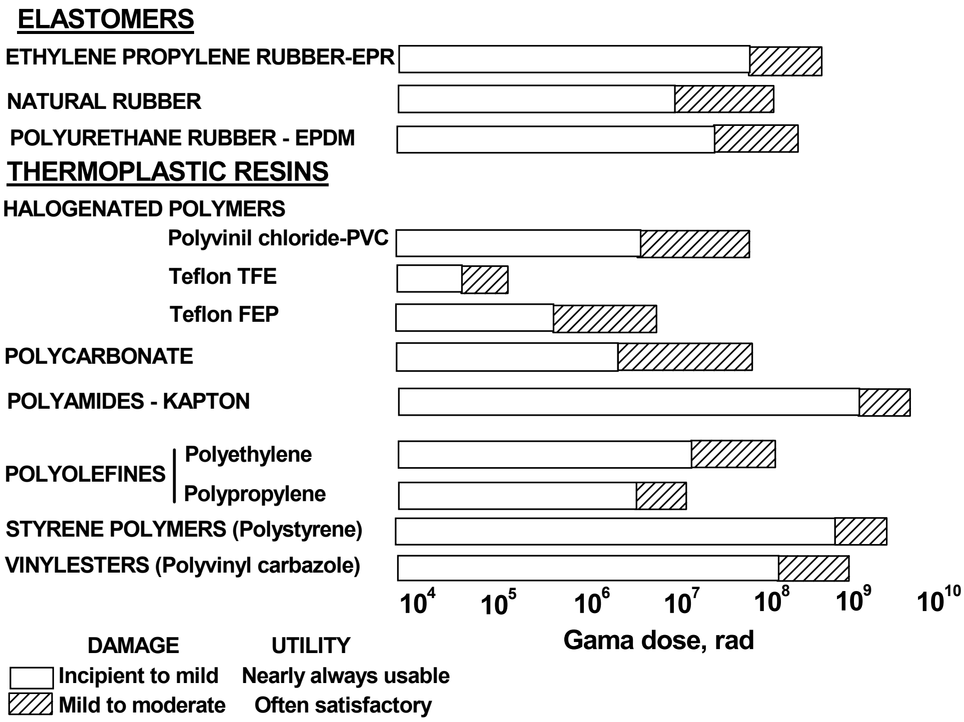
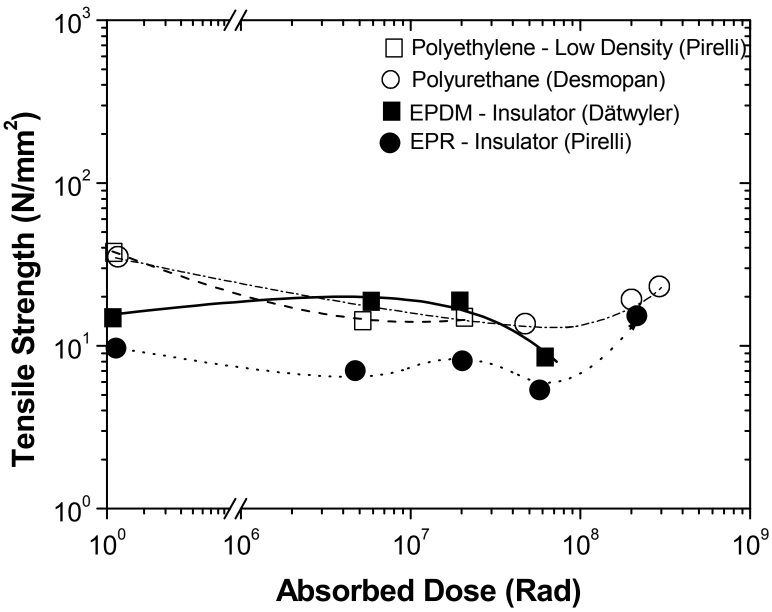


Figure 5.9/3 Tensile Strength and Elongation at Break of Elastomers and Thermoplastic Materials



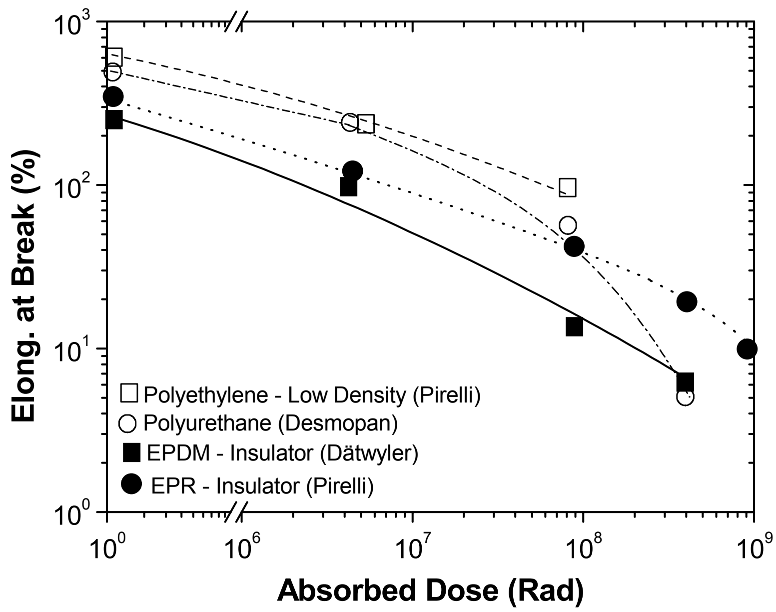
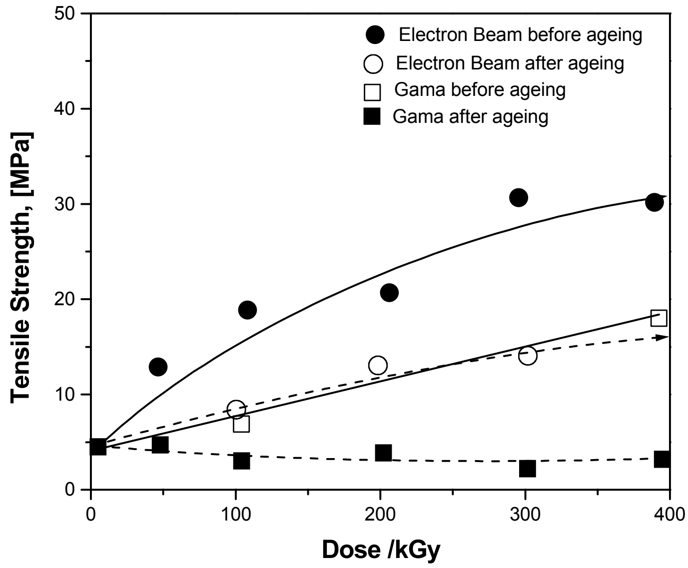
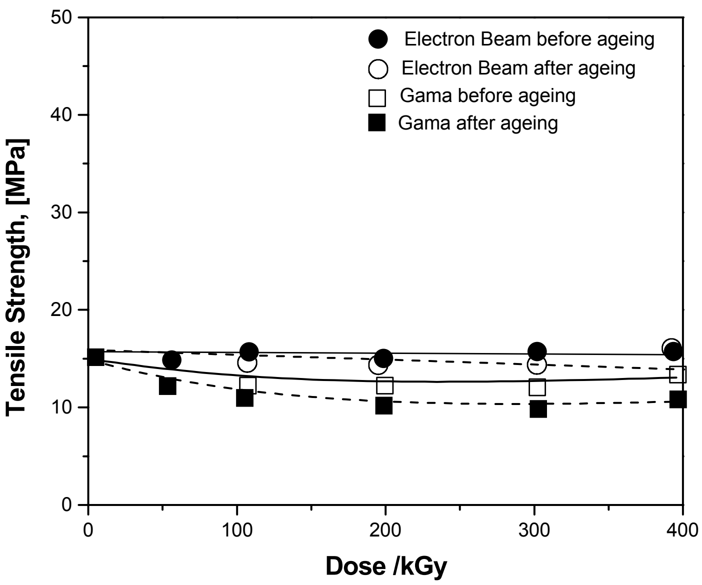


Figure 5.9/4 Effect of Ageing on the Tensile Strength of NR:PP. a) NR:PP (70:30), b) NR:PP (30:70)

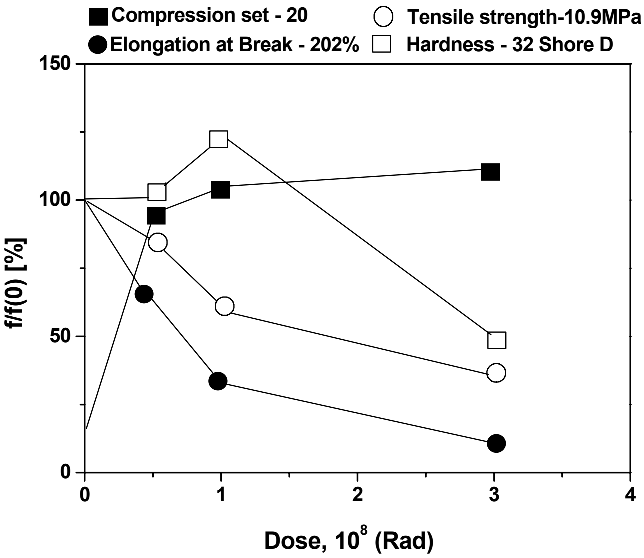


a)

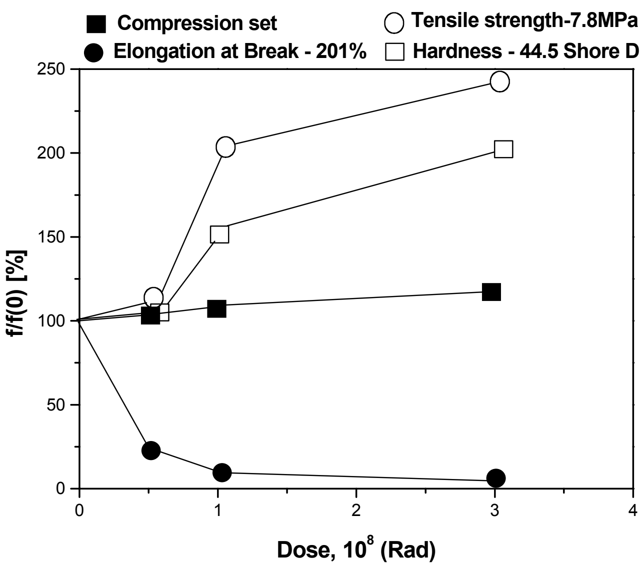


b)

Figure 5.9/5 Comparison Between a) Ethylene-Propylene Rubber and b) Fluorinated Copolymer (Viton E60C)



a)



b)

End of Figures