

# **The Future of Nuclear Power: from Ageing Plant to New Reactors**

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## **Introduction**

Civil nuclear power is not new. The first nuclear power station at Calder Hall opened in 1956 and operated successfully until its closure in 2003; a lifetime approaching 50 years. Throughout the 1960's and 70's there was a rapid expansion of nuclear power in the developed world with an increasing number of reactors, of increasing capacity from the 60MW per reactor at Calder Hall to over 1000MW per reactor. However, this rapid expansion, leading to around 200 reactors either built or under construction by the late 1970's, was brought to an instant halt in the United States by an accident to a 900MW reactor at the Three Mile Island Plant in Pennsylvania in 1979. This arrest in new US orders was compounded worldwide in 1986 following a second major accident, this time at the Chernobyl nuclear power plant in the Ukraine, then part of the former Soviet Union. The Three Mile Island accident, following a shortage of cooling water to the core of this pressurised water reactor (PWR) resulted in partial meltdown of the core, but there was no significant release of radioactive material to the outside. This was in contrast to the Chernobyl accident where operational errors resulted in a runaway nuclear reaction in the core of the RBMK reactor which blew the top off the reactor, resulting in a widespread release and distribution of radioactive material. Neither accident was the result of structural/material failure and in fact the PWR accident demonstrated the high integrity of the core pressure boundary.

The lack of new build of plants over the last twenty years or so has resulted in a switch within the industry from design, construction and the development of new systems to the strengthening of safety systems, both through engineered safeguards and the implementation of rigorous safety assessment, and the life extension of existing reactors. The earli-

est reactors, such as those at Calder Hall, were not expected to operate beyond 20 years or so, being overtaken by newer systems, such as the advanced gas cooled reactor (AGR), already under development. The fact that this first power station operated successfully for almost 50 years is indicative of the success of the development and use of life extension procedures, with associated new technology for ageing management. Because it was the first country to be seriously involved in nuclear plant life extension, the UK has been at the forefront of these developments, which have involved effective interaction between plant operators, original equipment designers and manufacturers, regulatory authorities and technology support organisations. An engineer who was responsible for the lifetime management of the oldest operating nuclear reactor in Spain once told me, “We look to the British as a lighthouse in the management of reactor life”.

So the changed foci of safety and life extension/management, together with a trickle of new plant build, particularly in Japan and Korea, have maintained nuclear engineering and technology through a long period of drought for the industry. Now, perhaps, with the new driver of climate change associated with greenhouse gas emissions from fossil power plants, a renaissance for nuclear power is near. This paper examines both the contribution of the extension of life of existing plants, along with possible new build, to the future of nuclear power.

Although the Calder Hall experience is rightly seen as a success for the understanding and management of structural integrity for the Magnox reactor system, operators of nuclear power plant worldwide have been surprised by the extent of materials degradation problems arising in other nuclear reactor systems. Whilst not threatening safety, these have inhibited plant reliability, and hence had a significant effect on economic viability.

### **Surprised by Ageing**

Table 1 identifies some of the most significant materials ageing issues for the major reactor types. The age given is that at which the issue required significant, and often costly, action. It can be seen that

on average a major issue has occurred at around 20 years, typically half of the life now expected from the newer nuclear power plants and a third of that expected for future plants. Most of these degradation issues were not anticipated at the design stage, for which there are several reasons (1.).

1. From the outset it was recognised that the design and development of the fuel and core of a nuclear reactor was a new and difficult engineering task utilising both new and conventional materials. New materials were required for the nuclear fuel itself, its containment in a thin metal cladding, and other core components such as the graphite neutron moderator. The use of prototype reactors was a success in the development of high reliability fuel elements (better than  $10^4$ ). However, once outside the nuclear core, it was judged that good quality conventional power plant engineering was adequate for the reactor coolant circuits, and steam and electricity generation systems. It is in this area that most of the materials integrity issues have been encountered because the performance demanded of these components has proved often to be significantly beyond previous experience.
2. Early success with prototype reactors in demonstrating the reliability and effectiveness of nuclear heat generation encouraged the rapid construction of nuclear power plants around the world with steadily increasing capacity. This resulted in a large strategic and financial commitment to civil nuclear power in the developed world well in advance of the time needed to demonstrate long term reliability and integrity of the major components and systems. As a result, prototype reactors, which continued to run on in to the 1980s began to suffer significant age related materials degradation problems only just ahead of the larger operating plants.

3. The effect of the Three Mile Island and Chernobyl accidents was to focus new design and engineering work on safety, whilst it was component reliability which was proving to be the major factor in plant availability for nuclear electricity generation. The proactive approach to component reliability embodied in the design base and ongoing maintenance was overtaken by a reactive approach when significant problems arose with ageing of major components. The connections between design, operations, maintenance and technology support were stretched, so that age related materials degradation issues were not dealt with as effectively as they might have been.
4. The compactness of the heat source and associated irradiation environment within a nuclear reactor make accessibility to some major components and structures difficult. This in turn makes inspection and early detection of degradation more difficult and repair/replacement of components both complex and expensive.
5. The primary coolant, which removes heat from the reactor core, is usually contained in a pressurized primary circuit comprising pressure boundary components such as vessels and pipe work. In light water reactor systems and some gas cooled reactor systems, the primary circuit involves a large steel pressure vessel around the core, whose integrity must be extremely high. This component is also subject to irradiation by neutrons generated in the core throughout the lifetime of the reactor, which can induce deterioration of mechanical properties, particularly fracture resistance. The fact that these and other pressurised components lack structural redundancy heightens integrity concerns.

6. Although the design and construction codes used for nuclear plant have always been the best available, they have proved to be limited in relation to materials degradation. Of particular significance has been the lack of appreciation of the importance of secondary stresses in driving significant degradation processes such as fatigue (thermal) and stress corrosion (residual). The source of significant cyclic thermal stressing has been rapid coolant movements around reactor circuits, particularly in water reactors. Residual stresses have largely been the result of welded fabrications, particularly those involving stainless steels and Inconel alloys.

Table 2 lists the significant material degradation mechanisms due to ageing encountered by components in various reactor systems. As can be seen, most of the major operating reactor systems have been affected. Although the operation of these mechanisms has created reliability problems and a challenge to life extension, most have been understood efficiently to monitor their development and deploy effective counter-measures. This process, plant life management, has evolved over the past 15-20 years with the result that a relatively small number of older reactors have shut down prematurely.

## **Materials Degradation Mechanisms**

Some of the mechanisms listed in Table 2 were identified at the design stage but others have been revealed only during long-term plant operation. These include mechanisms which have not been observed before and are the result of an interaction of materials degradation processes. New and interactive mechanisms are difficult to evaluate, particularly with regard to their rate of development and the threat which they pose to component integrity.

## **(i) Irradiation induced embrittlement**

It was recognised from the outset that iron and steels are subject to irradiation induced hardening and embrittlement. Neutron irradiation increases the tensile yield strength, reduces the tensile ductility, and increases the ductile-brittle transition temperature. This type of radiation induced embrittlement poses a threat to all reactor systems which have a steel pressure vessel surrounding the reactor core. These include light water reactors (PWR, BWR, VVER) and the earlier UK Magnox reactors. The causes of irradiation induced embrittlement and understanding and evaluating its extent in the range of steels used has resulted in extensive research programmes over many decades. A recent review (2) gives a succinct picture of the results of these programmes to-date

The degradation in mechanical properties, tensile and fracture, occurring in RPV steels during service is known to be related to micro structural changes on a nano scale. These basic micro mechanisms of irradiation embrittlement have been identified:

1. Matrix damage due to radiation produced point defect clusters and dislocation loops.
2. Irradiation enhanced formation of copper-enriched clusters/precipitates.
3. Irradiation induced/enhanced grain boundary segregation of embrittling elements such as phosphorus.

The first two mechanisms, matrix damage and solute clustering, give rise to obstacles which impede the motion of dislocations, thus hardening the steel matrix

with a consequent effect on the transition temperature. The third mechanism however results in non-hardening embrittlement by reducing the strength of grain boundaries and so inducing inter-granular rather than trans-granular fracture as failure mode. This mechanism has been observed in Magnox RPV steel samples.

The discovery of these mechanisms has been helped considerably by the inclusion of material and weld samples inside operating reactors so that they would be naturally irradiated but could be removed from time to time to track embrittlement. This surveillance practice has been most successful in revealing vessels which had excessive embrittlement, and the most susceptible locations, as well as revealing new mechanisms such as the third one noted above. Both this mechanism and that involving copper precipitation are particularly related to welds. Countermeasures on older reactors have included core reshaping to reduce radiation levels and vessel annealing. For newer reactors, it has been possible to minimise embrittlement through use of forgings, improved steel composition with respect to trace elements, and vessel design which precludes weldments from the most highly irradiated regions. Also considerable effort has gone into minimising the size of crack like defects produced during vessel fabrication. The result of continued technology support and international collaboration over many years has made the control of irradiation embrittlement of RPV materials a major success story in the integrity management in nuclear power plant.

## (ii) **Fatigue**

Whereas neutron irradiation can reduce the fracture toughness of a material, this does not of itself produce

failure, which requires the presence of a crack or crack like defect. Degradation processes which produce cracks, such as fatigue and stress corrosion, are much more likely to produce component failure albeit not of a catastrophic nature. Despite the application of design procedures for fatigue, cracking has occurred in nuclear power plants mainly as a result of the flow of reactor coolant which induces vibration and thermal transients. The cracking produced by these mechanisms, particularly in steam generators and pipe work is a design issue rather than one of materials selection. However, one new materials related fatigue issue was highlighted in the 1970s as a result of investigations in Japan(3), where fatigue crack growth tests were performed on pressure vessel steels in reactor coolant water (typical of PWR and BWR) under simulated (low frequency) reactor stress transient conditions. A significant increase in fatigue crack growth rate was observed over that from tests in air caused in effect by small increments of stress corrosion cracking enhancing the fatigue cracking rates. Although this form of corrosion fatigue has not been observed in RPVs, it has been found in steam generator shell welds following ingress of impurities in the secondary circuit.

### (iii) **Stress Corrosion**

Most of the world's operating civil nuclear power plants are water reactors, where water is used as the main reactor coolant as well as the medium for steam generation. This has resulted in a wide range of structural materials/water environment combinations in a wide range of components operating at relatively high temperature (~300 °C). The result is that despite best efforts in material selection and design and construction, stress corrosion cracking has been, and continues to be, the major structural integrity problem in these systems.

1997, the US Electrical Power Research Institute reported that, “corrosion has reduced plant average of 5% during this last decade” (4).

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For stress corrosion cracking to occur, a critical combination of material, environment and tensile stress must be present. On the one hand this makes the incidence of stress corrosion cracking in a given plant difficult to predict because the critical combination of stress, material and environment can arise through variation in just one of the three parameters. For example, difficulty in water chemistry control, which might arise during operation could trigger the mechanism, or inadequate control of residual stresses as a result of welding during fabrication, or minor variations in material conditions even within specification. On the other hand, the need for a critical combination of parameters means that stress corrosion susceptibility can be mitigated by suitable modification of one of the parameters. For example, a tightening of water chemistry control or introduction of surface compressive stress, or modification to the surface material chemistry could stop the process from occurring. The range of components affected is given in Table 2. Stainless steels and Inconel alloys have proved most susceptible. For more detailed discussion see a recent review (5).

#### (iv) **Delayed Hydride Cracking**

The CANDU and RBMK water-cooled reactors do not have a single pressure vessel surrounding the reactor core containing the fuel elements. Rather the elements are housed in zirconium alloy pressure tubes which offer significant redundancy in the overall pressure boundary around the core, a feature which is advantageous in terms

of safety. However, pressure tubes are subject to a more arduous environment; a much higher neutron irradiation fluence and hydrogen ingress into the zirconium alloy matrix following any corrosion on the inner surface of the tube. Hydrogen has a limited solubility in zirconium and its alloys, which if exceeded results in small platelets of brittle zirconium hydride precipitate within the microstructure. The hydrogen solubility limit decreases with decreasing temperature and the effect of hydride precipitation combined with irradiation is to reduce the toughness of the zirconium alloy making the pressure tube less resistant to fracture and prone to a phenomenon known as delayed hydride cracking.

#### (v) Delayed Re-Heat Cracking

The main materials integrity threat to the UK's Advanced Gas Cooled Reactor (AGR) has come from a high temperature degradation process which occurs in higher carbon stainless steels, particularly at thick section weldments, known as delayed re-heat cracking (DRC). DRC is a variant of the better known phenomenon of re-heat cracking which can occur in a range of materials on cool-down following welding in thick sections. Here the creep ductility of the material is insufficient to accommodate the strains accompanying relaxation of the welding induced residual stresses. Delayed re-heat cracking occurs in weldments, often after several years of operation at elevated temperature, where the operating temperature is not high enough to induce rapid stresses of relaxation of residual stresses and very low creep ductility is present in the material particularly in regions of high constraint which induce significant multi-axial tensile residual stress patterns. High carbon stainless steels (304, 316, 321) are particularly susceptible at temperatures in the region 475 °C to 550 °C, and particularly in repair weld locations where constraint is

very high. In AGRs thick section steam pipework and headers have been particularly susceptible to this cracking mechanism at weld locations (6). A similar process was observed in the late 1980s at repair welds in the thick section (50mm) shells of steam generator vessels in the UK prototype Fast Breeder Reactor.

## **The Management of Ageing and Plant Lifetime**

At the heart of successful ageing and life management is the understanding and predictability of relevant materials ageing degradation mechanisms. It was noted earlier that the plant operators have been surprised by the variety and extent of such mechanisms. In the light of the discovery of new and interactive mechanisms it is not surprising that many of these were not recognised at the stage of design and construction. On the basis of what is now wide and lengthy operating experience across several reactor systems, the challenge has been to quantify adequately new knowledge and experience so that the proactive approach to structural integrity embodied in the design process can be restored for operating plants. This can be used both to extend lifetime, as well as provide feedback for new designs in order to prevent or limit ageing. But understanding and predictability of ageing mechanisms is not simply a matter of operating plant experience. The physical process involved must be adequately understood in order to define the thermodynamic conditions necessary for their operation and the kinetics of the mechanisms, in order to predict how quickly and extensively resulting damage, particularly cracking, may develop. Adequate thermodynamic and kinetic understanding results in the generation of quantifiable models of damage development under conditions pertaining in the plant structures and components. Such models are not simply theoretical, based on laboratory testing and simulation, but are pragmatic models taking account of field observations and measurements taken from operating plant. Predictability of ageing is dependent upon the quality of operating experience data and degree of technical understanding. In

practical terms, for a given degradation mechanism, predictability of ageing is a function of the ability to measure and monitor the degree of damage in a plant structure or component, along with pragmatic modelling ability. Most of the degradation mechanisms identified earlier can be mapped in terms of these two parameters as shown in Figure 1. The high degree of predictability for irradiation embrittlement is on account of its resulting in a change of bulk material property, providing accurate condition evaluation through material sampling combined with strong modelling ability through micro-structural analysis. In contrast the time dependent mechanisms which result in localised cracking are much more difficult to detect by inspection or monitoring until they are well developed. Accurate modelling of these processes is variable. For example, for fatigue, modelling is good if the local cyclic stress conditions are well known. This is particularly helpful for thermally driven fatigue which often results in crack arrest or at best steady crack growth through a component section. On the other hand stress corrosion cracking is particularly difficult to predict because of its complex dependence on the three parameters of local stress, material condition and environment.

Understanding and predictability, however, provide only the basis for action against ageing mechanisms which are known to be active in a given structure or component. Several possible courses of action are possible; no action, because any degradation can be demonstrated to be self limiting; mitigation, by changes to plant operation; repair; component replacement. For successful proactive ageing management all relevant knowledge, understanding and possible operation and maintenance action should be set within an effective systematic ageing management process such as that proposed by the International Atomic Energy Agency (7) and shown in Figure 2. All actions relate to an adequate understanding of ageing processes operating in systems, structures and components (SSCs), and are then set within four separate connected zones. Firstly, a zone which involves co-ordination of all relevant information sources and appropriate links between potential parties involved; design, fabrication/construction, operation, maintenance, and engineering/technology support. The output from this activity is a “plan” to minimise anticipated degradation by defining what needs to be

“done” in terms of operation and maintenance to effectively manage the operating ageing mechanism. The output from this zone is a “check” for anticipated degradation and its extent through inspection, monitoring and assessment. As a result of the observations of this activity any unacceptable degradation should be revealed enabling the most appropriate maintenance “action” for its removal and restoration of the component to an acceptable condition. The process is continuous throughout life, responding to new information and technical advances.

Development of the tools to quantify and track materials degradation, however, is only part of the ageing and lifetime management process. The whole process requires a systematic, periodic review of all major components and systems in a plant. The development and use of such an integrated process has been key to the successful extension of life for the older plants and has been adopted in most countries with significant nuclear programmes. It has come to be known as the Periodic Safety Review (PSR) process. Its origins lay in the operation of the early Magnox reactors as they approached their design life of 20 years in the late 1970s. Then, it was agreed between operator and regulator that a detailed Long Term Safety Review (LTSR) should be carried out on each plant. These reviews were in effect design reviews and focussed on continued safe operation following the requirement of new health and safety legislation for rigorous safety cases following the ALARP principle, that risk should be “as low as reasonably practicable”. The reviews therefore tracked change, both in ageing effects on the plant and new engineering and technology developments which could be incorporated to satisfy ALARP. One new tool which was used in the LTSRs, and has become the cornerstone of the successor PSRs is Probabilistic Safety Assessment methodology (PSA). PSA is a systems approach, using fault tree analysis, which enables both components, structures and engineering systems to be evaluated in terms of an acceptable failure probability. The PSR process is in effect a regular, usually ten yearly, update of the design safety review to ensure that the design intent of the plant continues to be realised.

## **New Reactor Options**

The 21st century is setting new ground rules worldwide for electricity generation and therefore a renaissance of nuclear power. On the one hand, demand is rising dramatically, particularly in China, so that estimated world capacity is expected to double by 2020 from that installed in 1995. On the other hand, most of the new capacity will be fossil fuelled (coal/gas) in a world increasingly aware of the perils of global warming associated with carbon emissions. This picture is accelerating the reintroduction of new nuclear build in the developed world, alongside a push for renewable generation, particularly wind power, and energy conservation.

As regards new nuclear build, after a near-moratorium of 20-30 years, the question of what type and size of reactor arises. In turn, this decision depends on the desirable criteria for new plant and the availability of adequately proven designs and construction capability. The design criteria issue is one which has been evolving over the years and focuses on four areas:

- (i) Economic competitiveness. This has been a clear criterion for many years since the privatisation of the electricity supply industry in a number of countries including the UK. Low capital costs, comparable with alternatives, and minimum fuel cycle and operation/maintenance costs are the prime driving forces.
- (ii) Safety and Acceptability. Safety came to the fore following Three Mile Island and Chernobyl with a severe impact of add-on systems and their associated costs on plants under construction. A clear goal for new build now is public acceptability of safety arguments best achieved through passive and inherent built-in safety systems. An additional, important factor is a low proliferation risk in the light of the increased terrorism risk.

- (iii) Environmental Impact. Waste minimisation and effective short and long term management of waste continue to be key demands.
- (iv) Market Requirements. Flexibility in the deployment of new plants is coming to the fore. This is bound to increase the demand for smaller capacity plants which allow for both localised and multireactor siting.

In terms of technology these requirements have resulted in the following trends for next generation systems:

- a. smaller, modular reactors
- b. passive/inherent safety features
- c. a greater interest in gas cooled systems
- d. higher temperature and direct cycle operation
- e. proliferation resistant fuel cycles

In addition, lessons from the operation of existing reactors to long lifetimes have introduced ageing countermeasures to new designs. These include ageing resistant materials (eg Inconel 690, Alloy 800 and Nuclear Grade (NG) 316 stainless steel); improved fabrication of components to reduce residual stresses, severe weld details and weld defects; improved monitoring of component ageing and inspection; good control of corrosion related processes by improved water chemistry.

The evaluation of designs in relation to customer and industry demands for new reactor systems has resulted in the following main choices. The first relates to short/medium term and the second to medium/long term.

For short/medium term (5 - 10years) the choice is an advanced water reactor, which for the UK would likely mean an advanced pressurised water reactor (APWR). The two APWR options are a Westinghouse design at 600 or 1000MW capacity (AP600, AP1000) or a European design (EPR). These advanced designs either are already licensed or are at an advanced stage of licensing by key regulatory bodies (e.g. the US Nuclear Regulatory Commission). However, the UK Nuclear Installa-

tions Inspectorate would need to satisfy itself on the safety of these designs before new build could proceed in the UK. The AP600/AP1000 designs incorporate “passive” safety features, which reduce the core damage frequency by a factor of nearly 4 over that accepted for the Sizewell B PWR. “Passive” safety involves a number of key engineering features which enable safety systems to rely on natural forces such as gravity and natural circulation for their operation, rather than on pumps and valves. In addition, these advanced designs have simpler engineering systems, resulting in considerable savings by reducing by 50% or more the number of valves, amount of piping, cabling etc.

For the medium/long term (10 - 20 years) the most interesting development is the high temperature gas cooled reactor (HTR). This is not a new concept as small prototype HTRs ran successfully in the early days of nuclear power as the logical successors to advanced gas cooled reactors (AGRs). Their aim was to increase efficiency through a high gas outlet temperature ( $>900\text{ }^{\circ}\text{C}$ ), using all ceramic fuel elements. Currently new prototype HTRs are being developed in China, Japan and South Africa. The most commercially targeted of these is the pebble bed modular reactor (PBMR) proposed by the South African utility Eskom, and now being developed as an international project. PBMRs are small (100 - 125MW) modular reactors utilising 60mm diameter fuel spheres (or “pebbles”), comprising 0.5mm diameter uranium dioxide particles coated in silicon carbide and embedded in a graphite matrix. These pebbles then pass slowly under gravity through an annular graphite moderator, whilst helium coolant gas passes over the pebbles and is heated to  $900\text{ }^{\circ}\text{C}$  before passing through turbo-compressors and on to a gas turbine/generator. PBMR technology is well proven as regards fuel performance and a small German prototype ran for over 20 years in the 1970/80s. The new feature of the PBMR system design is the direct cycle using modern gas turbine technology.

There are a number of attractions today for the PBMR system. These include:

- (a) “Inherent” safety where the physics of the system means that Engineered safety systems are not required to

prevent dangerous conditions arising. For example, no credible faults can lead to a loss of fuel integrity, by ensuring containment of fission products within the fuel pebble. A natural post-loss-of-coolant maximum fuel temperature of about 1400 °C is much lower than the 2000 °C plus required for significant fuel failure.

- (b) The modular nature of the reactor units and simple direct cycle system makes for significantly reduced capital costs over current reactor systems.
- (c) The high outlet temperature means that this reactor can be used as a heat source for reforming, needed for hydrogen production linked to a future “hydrogen economy”.
- (d) The all-ceramic pebble fuel is extremely resistant to plutonium extraction and so reduces proliferation concerns.

So, the new reactor options for the first quarter of the 21st century are sufficiently well established for reasonably rapid deployment. Along with the lessons already learned from the operation of 200 plus commercial reactors worldwide in the latter half of the 20th century, this should provide confidence in an appropriate new nuclear power programme designed for the challenges of this century. Meanwhile economic life extension of many existing plants should be possible, so encouraging confidence in a long life for any new plant build.

## References

- (1) B Tomkins, “A historical perspective of materials related structural integrity issues in the nuclear industry”. Fracture in Plastic Flow and Structural Integrity, eds PB Hirsch and D P G Lidbury, IOM Communications Ltd, 2000

- (2) C English, J Hyde and S Ortner. “Microstructural development in RPV Steels”, in Plastic Flow and Structural Integrity, eds P B Hirsch and D P G Lidbury, IOM Publications Ltd, 2000
- (3) T Kondo, T Kikuyama, H Nakajima, M Shindo, R Nagasaki, in “Proceedings of Conference on Corrosion Fatigue” NACE, 1971, pp 536-556
- (4) Electrical Power Research Institute, “Major Component Reliability”, 1977,  
<<http://www/epri.com/npg/brochure/mcr.htm>>, 1997
- (5) Tomkins Elsevier SI
- (6) P Neumann, D A Miller and R A Ainsworth, “Materials factors which influence remaining life of components subject to reheat cracking”, in Fracture, Plastic Flow and Structural Integrity, eds P B Hirsch and D P G Lidbury, IOM Publications, 2000
- (7) International Atomic Energy Ageing, “Implementation and Review of Nuclear Power Plant Ageing Management Programme”, Safety Reports Services No 15, IAEA Vienna, 1999

## Tables

<u>Reactor Type</u>		<u>Issue</u>
	Age* (Yrs)	
Magnox	RPV Embrittlement	30
PWR	Corrosion/Stress Corrosion	20
BWR	Stress Corrosion	15
VVER	RPV Embrittlement	20
CANDU	Hydride Cracking	20
AGR	Reheat Cracking	20

\* Approximate age when issue required significant action

Table 1 - Significant ageing issues for major operating reactor types

Degradation Mechanism	Component Affected
Irradiation Embrittlement	All steel reactor pressure vessels (LWR, Magnox)
Fatigue (vibration) Steam Generator Shell	Pipework (thermal) Steam Generator Tubing (corrosion fatigue)
Stress Corrosion Steam Generator Tubing (PWR)	Pipework (BWR) Reactor Internals (BWR, PWR) Steam Generator Headers (VVER) CRDM Penetrations
Delayed Hydride Cracking	Pressure Tubes (CANDU)
Delayed Reheat Cracking	Boilers/Steam Generators (AGR, FBR)

Table 2 - Significant degradation mechanisms encountered by nuclear reactor components

Figure 1. Mapping of degradation mechanisms in reactor plant materials in terms of monitoring and modelling ability.

Figure 2. A systematic ageing management process