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No. 62

Proactive Management of Ageing for Nuclear Power Plants

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PROACTIVE MANAGEMENT
OF AGEING FOR NUCLEAR
POWER PLANTS

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PROACTIVE MANAGEMENT OF AGEING FOR NUCLEAR POWER PLANTS

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FOREWORD

Of the total number of nuclear power plants currently operating around the world, approximately 25% have been in operation for more than 30 years, and about 70% have been in operation for more than 20 years. Effective management of ageing of systems, structures and components (SSCs) is thus becoming increasingly important. To assist its Member States in managing ageing effectively, in the 1990s the IAEA developed a comprehensive set of publications on ageing management for nuclear power plant components important to safety. Recently, the IAEA issued a Safety Guide on Ageing Management for Nuclear Power Plants.

Proactive ageing management means management of the ageing of SSCs that is implemented with foresight and anticipation throughout their lifetime. Proactive ageing management is a key element of the safe and reliable operation of nuclear power plants. Section 3 of the IAEA Safety Guide on Ageing Management for Nuclear Power Plants provides recommendations on how to proactively manage ageing for the entire lifetime of a nuclear power plant.

The objective of this Safety Report is to provide information on good practices for maintaining and improving the safety and performance of nuclear power plants by facilitating proactive ageing management of SSCs throughout their lifetime. This report provides practical information supplementary to that in Section 3 of the Safety Guide on Ageing Management for Nuclear Power Plants.

The IAEA officer responsible for this publication was T. Inagaki of the Division of Nuclear Installation Safety.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Ageing is defined as a general process in which characteristics of a system, structure or component (SSC) gradually change with time or use. Examples of ageing mechanisms include curing, wear, fatigue, creep, erosion, microbiological fouling, corrosion, embrittlement, chemical or biological reactions and combinations of these processes (e.g. erosion–corrosion, creep–fatigue). Since ageing can have impacts on both the safety and the performance of nuclear power plants, effective and proactive management of the ageing of SSCs is a key element of the safe and reliable operation of nuclear power plants. For effective ageing management, ageing must be systematically taken into account at each stage of the plant life cycle; that is, during design, construction, commissioning, operation (including long term operation and extended shutdown) and decommissioning.

From the economic point of view, a nuclear power plant represents a large investment for any operating organization. Capital costs are very high, and the duration of construction and the time required for a return on the investment are quite long. Therefore, once the plant is in operation, it is important for the operating organization to keep it in service as long as possible. Thus, it will generally be cost effective to devote significant resources to systematic ageing management during plant operation, including resources for investigation, understanding and effective management of ageing.

Ageing is a major concern for operating organizations for several reasons:

(a) Most existing nuclear power plants were built 20–30 years ago. As these plants age, new ageing phenomena that develop slowly continue to be discovered.
(b) While the lessons learned from past ageing problems have been partly taken into account in the plant design, new ageing phenomena are emerging as a result of the more severe service conditions associated with increased plant performance (through implementation of the long term operating experience obtained and the application of new technologies).

(c) While many plants are approaching their design and/or regulatory limits, operating organizations would like to continue their operation. This raises the questions of how SSC ageing has been managed in the past and how the SSCs will behave in future long term operation.

To assist its Member States in managing ageing effectively, in the 1990s the IAEA began developing a comprehensive set of publications on ageing management (see Annex I for a brief description of these publications). The Safety Guide on Ageing Management for Nuclear Power Plants [1] provides a set of guidelines on and recommendations for managing the ageing of SSCs important to safety in nuclear power plants. The present publication supplements Ref. [1] and provides specific examples of good practices and proactive ageing management.

1.2. OBJECTIVE

The objective of this Safety Report is to provide information on good practices for maintaining and improving the safety and performance of nuclear power plants by facilitating proactive ageing management throughout the lifetime of a nuclear power plant.

1.3. SCOPE

This Safety Report deals with organizational and managerial means to increase the effectiveness of existing ageing management programmes in nuclear power plants, including minimizing the premature ageing of SSCs. The information provided is also applicable to the development of new ageing management programmes.

The report provides information on:

(a) Recognizing common weaknesses in ageing management for nuclear power plants, including root causes and sources of ageing of SSCs important to safety and reliability;
(b) Application of a proactive, systematic ageing management process, including continuous improvement of ageing management actions.

It is intended for use by the management (decision makers) and technical staff (engineers and scientists) of operating organizations of nuclear power plants and of technical support organizations. It will be also of interest to regulatory bodies and designers of nuclear power plants.

1.4. STRUCTURE

This Safety Report is set out in five main sections, an appendix and two annexes. Section 2 discusses the impact of ageing on plant safety and performance, and the safety and economic aspects of ageing and its management. In Section 3, the most frequently encountered weaknesses of ageing management are identified and illustrated with examples. Section 4 provides information on a proactive strategy for the management of physical ageing of SSCs important to safety throughout the lifetime of a nuclear power plant. Section 5 provides guidance on the implementation of a proactive approach in plant operation. This includes the need to adopt a clear strategy, to put into place an appropriate organization and processes, and to make available appropriate tools. The Appendix provides 12 case studies of significant ageing management issues encountered in various reactor systems, illustrating the lessons learned. These case studies provide a helpful basis for developing more effective proactive approaches to ageing management. The annexes provide supplementary information.

2. IMPACT OF AGEING ON PLANT SAFETY AND PERFORMANCE

The primary aim of most ageing management programmes is to help ensure the availability of required safety functions throughout the service life of the nuclear power plant. However, it is recognized that effective management of ageing is also essential to the achievement of the desired plant performance and profitability. This section discusses the safety and economic aspects of ageing, and its management.

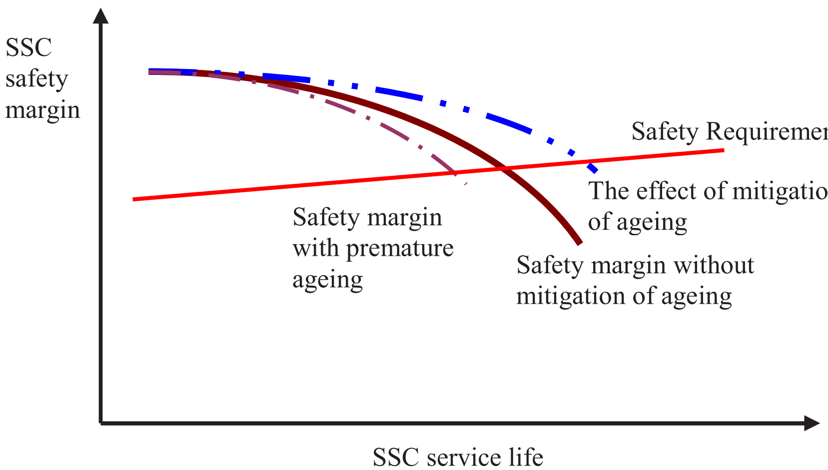


FIG. 1. Relation between SSC safety margin and SSC service life.

2.1. IMPACT OF AGEING ON PLANT SAFETY

The aim of physical or materials ageing management of SSCs important to safety is to maintain their design safety margins[1] above SSC specific requirements (see Fig. 1) and thus to minimize the risk to public health, the environment and plant safety.

Ageing increases the probability of both single component failures and common cause failures. Operating experience has shown many SSC failures that have occurred as a result of ageing mechanisms such as general and local corrosion, erosion–corrosion, radiation and thermal embrittlement, fatigue, creep, vibration and wear. These failures have affected plant safety through abnormalities of process systems and increased unavailability of safety systems. To ensure a high level of plant safety, the operating organization is advised to manage SSC ageing effectively and proactively; this includes not only physical ageing — the subject of this report — but also non-physical ageing that results in obsolescence, that is, the plant's being out of date with respect to current safety regulations, standards, practices and technology.

[1] The safety margin is the integrity and functional capability of both passive and active SSCs in excess of their normal operating requirements, including the integrity and functional capability required for operation under accident and post-accident conditions.

A periodic safety review (PSR) is a regulatory instrument used by many Member States for reviewing and maintaining the safety of plant operation throughout a plant's service life (Ref. [2], para. 2.6). PSRs are also used in the assessment of plant safety in connection with the approval of continued plant operation beyond the licensed term or the period established by the safety evaluation.

In the United States of America (USA), the effectiveness of ageing management programmes for active SSCs is verified by the regulatory body through the application of the maintenance rule. The maintenance rule requires the licence holder to monitor the performance or condition of all safety related active SSCs to ensure that they are capable of fulfilling their intended functions. The effectiveness of ageing management programmes for passive SSCs is verified in connection with the licence renewal process for operation beyond 40 years.

2.2. IMPACT OF AGEING ON PERFORMANCE AND LONG TERM OPERATION OF PLANTS

There is no doubt that ageing has a negative impact on the performance of nuclear power plants. An example is degradation causing an SSC to approach a given limit (or maximum allowable value), making it necessary to repair or replace the SSC during a planned outage. Such an event represents an additional expenditure (e.g. parts purchases, human resources costs, costs associated with a possible outage extension). A more severe impact occurs when an age related incident leads to a forced outage, which increases production and/or operations and maintenance (O&M) costs (e.g. resources to cope with the event, consequences of the incident, loss of power). Whereas the aim is to minimize outage times, ageing degradation can result in increased outage times if it is not properly managed.

While efforts to minimize ageing also have a cost, minimizing premature or unexpected ageing degradation represents avoided costs. For some components, accepting known 'normal' ageing and simply waiting for an appropriate repair/replacement time may be the most cost effective way to manage ageing. This is sometimes known as the 'run to failure' strategy, and it is often used for small components. However, even these small components can have a significant impact on nuclear safety. In many cases, ageing of larger components can be slowed by proactive measures at the design, construction, operation and maintenance stages, requiring only limited resources (e.g. selection of better materials, fabrication and installation methods; optimization or adjustment of operating and/or maintenance procedures and conditions). In addition, ongoing uprating, upgrading and/or modernization of SSCs with the aim of maintaining the nuclear power plant as new has been shown to be an effective proactive strategy for managing ageing as well as for maintaining a high level of performance [3]. Implementation of such a strategy requires a long term view and an ‘asset management’ approach (including cost–benefit analysis and assumptions about the planned plant service life), since uprating, upgrading and/or modernization of SSCs is typically resource intensive.

A good example of proactive ageing management is the way nuclear power plants and the industry have dealt with the issue of reactor pressure vessel (RPV) radiation embrittlement (see Annex II). The embrittlement of RPV steel due to irradiation was recognized as a potential degradation mechanism at the outset of nuclear power generation. The nuclear engineers who designed the early RPV surveillance programmes produced outstanding work based on a limited understanding (by today’s standards) of the detailed mechanisms of irradiation embrittlement in welded and plate material.

More recently, embrittlement mitigation measures, such as revised fuel patterns, hafnium absorbers and local shielding, have been developed and used to limit dose. The use of a low leakage core may allow a reduction of vessel irradiation of 20–30%, provided it is implemented early enough. Furthermore, the use of annealing to mitigate damage is now well established for the RPVs of WWERs. The ductility of these early RPVs can be restored to a significant degree, allowing years of continued safe operation. Only complete RPV replacement has not been attempted, although engineering studies have been undertaken.

As a result of these efforts, there have been relatively few plant closures or prolonged shutdowns directly related to RPV embrittlement. The good predictability of the radiation embrittlement mechanism, improved capability of non-destructive testing methods and fracture mechanics mean that the cost of managing the risks can be estimated with an acceptable level of confidence. However, it should be recognized that successful RPV ageing management has been based on a substantial and costly R&D effort.

An interesting example of a significant ageing issue affecting the performance of nuclear power plants is stress corrosion cracking (SCC) of steam generator tubes made of Alloy 600. In the 1980s, this ageing mechanism was a serious concern for several plant operators. Although it led only to small primary-to-secondary leakage, there was an impact on safety because of the difficulty of assessing how much it was increasing the probability of tube rupture — a serious safety concern. In addition, it had a considerable impact on plant performance. First, the implementation of extensive programmes for inspection and repair required extended outages and large expenditures (the maintenance costs for a steam generator affected by SCC could be five times those of a normal steam generator). Second, the need to shut the plant down when the leak rate was above the acceptance criteria increased forced unavailability. Finally, the cost of replacing the steam generator, which often was the only solution, was very high.

It is interesting to note that, despite the high costs associated with the problem, many operating organizations did not investigate the risk that other components made of Alloy 600 might also be affected until after the issue had become a serious problem. Thus, their lack of foresight and comprehensive investigation had a significant impact on plant performance.

The impact of an ageing phenomenon can be particularly serious in the case of an unforeseen generic ageing problem that could affect an entire fleet of plants of the same design, and thus threaten the security of electric power supply. If the ageing phenomenon is unexpected (or has not been properly handled), it is likely that all the units will have to deal with it at the same time.

One such example is the unexpected thermal fatigue that developed in an auxiliary system in a series of plants. Although the fatigue degradation did not develop at the same rate in all the plants, when it was discovered at one plant, an investigation was conducted and the same problem was found in all the units. Repairing one unit would have been easy, but repairing a large number of units was a serious problem because of the shortage of spare parts and the lack of qualified welders.

Another example is the SCC of vessel head penetrations that affected a large number of units where nickel based alloys had been used. In some series of plants, the issue was well managed; that is, there was timely detection, investigation and analysis of the problem so that vessel head replacement could be scheduled well in advance, giving the industry time to manufacture replacement heads. However, in one series of plants, the problem was not properly investigated or handled, leading to severe corrosion that jeopardized the vessel head pressure boundary of one unit. All vessel heads of the series were inspected at the same time, and many were found to be damaged; however, the industry was not able to rapidly provide spare parts, leading to a tremendous increase of the cost of services and procurement.

Although repair or replacement is technically possible for most SSCs across all nuclear power plants, ageing has an indirect impact on plant service life through extended outages, reduced availability, increased production costs, regulatory body suspicions about the level of plant safety and the operator's ability to manage ageing issues, and, possibly, insufficient industry capacity to support numerous replacements or backfitting of qualified components within a short period of time. Thus, ageing is a clear challenge to the safe and cost effective long term operation of nuclear power plants that requires systematic and proactive management.

3. COMMON WEAKNESSES OF AGEING MANAGEMENT

Effective management of ageing can be hindered by several factors that can lead to either unexpected or premature ageing (i.e. ageing degradation that occurs earlier than expected) of SSCs. These factors or weaknesses need to be identified and addressed if the proactive approach to ageing management is to be successful. In this section, the most frequently encountered weaknesses of ageing management are identified and illustrated with examples. These weaknesses are:

(a) Insufficient understanding or predictability of ageing;
(b) Lack of data for ageing management;
(c) Inadequate communication and coordination;
(d) Inadequate safety culture;
(e) Error induced ageing;
(f) Inappropriate use of reactive ageing management;
(g) Insufficient capability for dealing with unforeseen ageing phenomena.

The Appendix provides examples that illustrate these common ageing issues and their resolution.

3.1. INSUFFICIENT UNDERSTANDING OR PREDICTABILITY OF AGEING

The underlying reason for significant ageing degradation of many SSCs in existing nuclear power plants is that there was insufficient understanding, and therefore a lack of predictability, of ageing at the time of plant design and construction. There was limited relevant operating experience and R&D before the design and construction of the first commercial nuclear power plants. Therefore, although the design, fabrication and construction codes used for these plants were the best available at the time, they have proved to be inadequate for the evaluation of all possible ageing phenomena and key related factors, such as residual stresses or stresses caused by thermal stratification or mixing flows.

The probability and predictability of future age related degradation and the associated costs of required corrective actions are generally estimated on the basis of a combination of operating experience and an understanding of ageing. When there is sufficient operating experience, a statistical approach can be used to obtain estimates of likely future behaviour. Technical understanding of relevant ageing mechanisms and their effects is used to support this prediction and thus significantly reduce the uncertainty of the estimate.

The limitations of the design basis of many nuclear plant components with regard to ageing effects on materials may mean that ageing represents a significant threat to plant safety, performance and economic viability. Experience of ageing phenomena such as stress corrosion has revealed a lack of the technical understanding that would enable accurate prediction of future degradation on the basis of plant observations. Figure 2 shows the degree of technical understanding versus the quality of plant data, indicating that a high score is needed in both areas to achieve good predictability of ageing. Predictability is the key to effective ageing management. It enables engineering, operation, inspection and maintenance to be optimized and sound decisions to be made on the desirability, type and timing of ageing management actions (e.g. mitigation, preventive maintenance, condition monitoring, repair or replacement).

In practical terms, predictability of ageing depends on two elements: (i) modelling ability, that is, the degree of technical understanding, and (ii) condition monitoring to measure the progress of ageing in the component. A condition indicator could be a bulk property change or a crack detected on the component, both of which are dependent on results of specific inspection or monitoring systems. Figure 2 shows the current status of modelling and condition monitoring of some of the observed ageing processes.

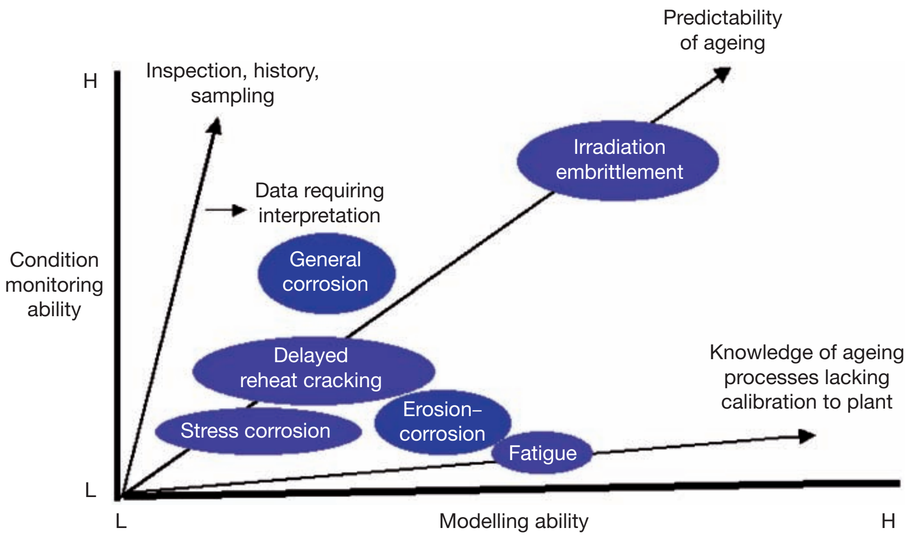


FIG. 2. Modelling ability versus condition monitoring ability for typical ageing mechanisms.

In general, there are three categories of ageing effects and associated mechanisms that can affect the ability of SSCs to meet their performance criteria or technical specifications:

(a) Changes in bulk material properties through temperature, strain or irradiation;
(b) Global or local changes in material surfaces or interfaces;
(c) Macroscopic defect propagation (e.g. crack growth).

In the first category, bulk material properties degrade and the risk of failure of the component increases if there is a pre-existing crack on the component. From an ageing management perspective, this type of effect can be modelled and measured. Irradiation embrittlement falls into this category, as do thermal embrittlement of stainless steels, and irradiation and thermal degradation of polymers in cables, seals and coatings.

Irradiation embrittlement of RPVs generally has been successfully dealt with through modelling and condition monitoring using accelerated irradiation tests, in-plant surveillance samples and, more recently, material samples from operating RPVs. If embrittlement is detected and managed at an early stage, it is possible to keep RPVs in service for long term operation. There have only been two possible cases of premature permanent plant shutdown resulting from RPV embrittlement due to high impurities in the vessel material.

Measurement technology has also advanced such that small RPV material samples can be extracted from those sites at greatest risk, for direct measurement of component embrittlement. This process has been carried out on Magnox reactors, advanced gas reactors (AGRs) and light water reactors (LWRs). Model results can be confirmed by such condition monitoring, and accurate predictions or estimations of future trends can then be made.

The second category of ageing effects and mechanisms includes the loss of surface material (e.g. due to general corrosion, erosion–corrosion, fretting or wear) and the development of small cracks (typically environmentally or cyclically driven). These effects are only modelled in some cases, and in the case of small crack formation, measurement usually is not possible. Certain types of corrosion fall into this category and are the most uncertain degradation mechanisms in terms of predictability. For these mechanisms, prevention is usually the preferred ageing management approach. In this area, there is a distinct lack of technical understanding of many corrosion mechanisms in specific plants.

The third category is characterized by rapid degradation (such as crack growth) within a timescale allowing operator intervention. Examples are thermal or vibration induced high cycle fatigue, environmentally assisted cracking (stress corrosion, corrosion fatigue) and delayed reheat cracking (a form of high temperature creep), which generate cracks from the beginning of the component's service life. These ageing mechanisms can produce crack growth that is rapid in terms of the total plant lifetime. Rapid crack growth rates make it particularly challenging for ageing management programmes to provide timely information to guide future actions. Without this capability, the time to initiation remains the dominant consideration and the overall uncertainty remains high.

However, in many cases, the driving stresses are secondary (thermal, residual), often resulting in a reduction in stress with depth through a component section, which can lead to a reduction in the crack growth rate and, possibly, crack arrest. In such cases, in-service inspections combined with fracture mechanics assessment may be used to optimize a repair or replacement strategy, or even to allow the component with the crack on it to remain in service.

Finally, where crack growth does not begin from crack initiation in service but from a pre-existing manufacturing defect, further uncertainty is introduced. This uncertainty can sometimes be bounded by assumptions relating to the original acceptance standards, supported by pre-service non-destructive examination (NDE) records. This can be difficult in older plants, where baseline data[2] often are not available or are limited.

For given ageing mechanisms, the location of occurrence can be estimated with a certain level of predictability. Where there is better technical understanding and more operating experience, predictability increases. Figure 2 illustrates a number of mechanisms that have been experienced in various reactor systems as they are currently perceived in terms of ageing predictability. Such estimations indicate where R&D investment is required for predictability to be improved. It is clear that significant R&D expenditure is needed if modelling and condition monitoring technology are to reach the levels required for effective ageing management based on predictability. Such effective ageing management should lead to an increase in the use of mitigation measures that, if applied early enough, can prevent crack initiation or deterioration. Such measures also require further development. Examples of these mitigation measures include surface treatment to counter SCC initiation, barrier coatings and environment modification (such as fluid chemistry measures to eliminate aggressive contaminants).

[2] Baseline data are the design basis data resulting from the fabrication and pre-service inspection of a component.

3.2. LACK OF DATA FOR AGEING MANAGEMENT

Ageing management of SSCs requires the availability of relevant, accurate and sufficiently complete information to enable timely and correct decisions on specific ageing management actions. These data can be divided into three categories: baseline data, operating history data and maintenance history data (Ref. [4], para. 2.1). Experience shows that the required information and data often are not sufficiently comprehensive or readily retrievable, generally because nuclear power plant organizations were not sufficiently aware of the usefulness of these data for effective ageing management.

For example, baseline data may be lacking where a plant owner no longer has the fabrication data related to the chemical content of the RPV steel, making the prediction of embrittlement very difficult to assess. Such original information about design and construction may have been lost because of changes of the plant owner, or because the original supplier or vendor no longer exists.

Another common problem is the lack of baseline data that are useful for effective ageing management but that were not originally required to be recorded. One example involves the delta ferrite content of cast stainless steel. At the time that many plants were constructed, there was no requirement to keep a record of the exact value of delta ferrite content for each component made of this material. It was only several years later that the risk of thermal embrittlement was discovered, along with the detrimental effect of delta ferrite content. In addition, assessing the risk for operating components required the plant operator to perform on-site measurement and/or sampling, both of which are complicated and costly.

A lack of operating history data may be due to improper monitoring and/or record keeping of actual operating transients. In several circumstances (e.g. because the plant operator wants to make changes in operating practices or to keep components in operation beyond their design life), it may be necessary to update design fatigue analysis on the basis of actual transients instead of postulated design transients, to take advantage of the fact that the actual transients are generally less severe than the postulated design transients. The updated analysis must demonstrate that sufficient margins remain with respect to fatigue. The availability of data required for fatigue assessments of components on the basis of actual transient histories is especially valuable at a time when many plant owners are considering long term operation.

3.3. INADEQUATE COMMUNICATION AND COORDINATION

In many cases, ageing degradation cannot be traced back to an explicit error, but rather is a result of inadequate communication and coordination between relevant functions or programmes. It must be recognized that at the various phases of a plant's lifetime, from design to operation, and for most SSCs, no person or body has explicit responsibility for achieving a specific component lifetime. Similarly, no person or body has explicit responsibility for analysing all causes and consequences of ageing. There are a few cases where such assessments are performed quite extensively (e.g. fatigue analysis of the reactor coolant system of pressurized components, embrittlement prediction for the RPV), but in most cases, a component lifetime is a target that is not translated into evaluations, rules or even guidelines. One exception is the case where equipment important to safety is covered by a well established equipment qualification (EQ) programme that includes systematic establishment of a qualified life or qualified condition for the equipment, as well as prescribed installation, service conditions and maintenance.

Thus, unexpected or premature ageing degradation of an SSC (see Section 3.5) may occur even though each organization involved has been complying with the rules in its professional area. For example, a vendor may design a system assuming that the future plant operator will use the system about 10 hours per year. Years later, the plant operator may want to use the system 1000 hours per year, which is not forbidden in the vendor specifications. In this case, even though no rules have been violated, fatigue cracking may develop before the end of the plant's service life, because the operator has unknowingly gone beyond the design intent, and this fact has not been communicated to the vendor.

Another example involves feedwater lines. To reduce the fatigue load of feedwater lines that results from thermal mixing/stratification, the steam generators of pressurized water reactors (PWRs) are often intermittently filled during startup. Process engineers have developed specific startup procedures for this phenomenon. In some cases, because of insufficient communication and coordination between engineering and operating organizations, these procedures are not implemented correctly, resulting in continuous filling of the steam generators and therefore unnecessary usage of the thermal fatigue cycles of the feedwater lines.

Another example is where the vendor uses a type of steel that may be sensitive to chloride attack, assuming that there is no reason for chloride to be in the reactor building, and NDE personnel then use chloride-containing adhesive tape to mark inspection locations, as there is no specification forbidding the use of such tape. Corrosion damage may subsequently develop, even on safety related equipment, where such adhesive tape has been used.

A case where inadequate communication and coordination led to serious ageing related damage was the secondary piping rupture event that occurred at Mihama Nuclear Power Station Unit 3 in Japan in 2004 (see Fig. 3) [5]. The pipe rupture was a consequence of wall thinning caused by flow accelerated corrosion. Whereas according to the technical standards the required minimum thickness was 4.7 mm, at its thinnest part the pipe wall had thinned to 0.4 mm. The rupture event occurred despite the fact that all licensees operating PWRs had conducted wall thickness measurements as part of a self-imposed inspection in compliance with their own guidelines for secondary piping wall thickness control at nuclear facilities (PWRs), which were issued in 1990. However, the actual wall thickness measurement work was initially performed by the unit’s manufacturer and was subsequently taken over by another company. In its interim summary, the regulatory body in Japan concluded that the direct cause of the accident was a mistake in secondary pipe thinning control involving the three companies, and that, owing to this mistake, the portion to be controlled was missing from the initial control list, and this could not be corrected until the accident.

When severe thermal fatigue cracking was detected in the residual heat removal systems of a number of PWR plants, it was clearly not the fault of a

FIG. 3. Ruptured piping at Mihama Nuclear Power Station Unit 3.

single organization: design was in compliance with codes and standards, there was no fabrication or construction abnormality, operation of the system was in compliance with procedures, in-service inspection was not required, etc. What was lacking was effective communication between the relevant organizations leading to the realization that thermal fatigue could develop under a certain combination of operating conditions. The solution to the problem was a combination of measures to be taken by each of the organizations.

Another scenario where inadequate communication and coordination could lead to problems involves the possible presence of an acceptable fabrication defect in a component. In this case, prior to construction, a fatigue assessment of the component is carried out in compliance with design codes and standards, and taking into account postulated operational transients. During fabrication, the defect is proved to be acceptable on the basis of a fatigue analysis using the same postulated operational transients. During operation, the plant owner wants to make a change in operating procedures that will cause a change in the transients. To make sure that this change is acceptable, the owner will redo the design fatigue analysis, taking into account the new transients. However, if there is a lack of communication between organizations, the owner may forget to verify the absence of an impact of the operational changes on the defect assessment.

In another example, installation personnel forgot to put waterproof grease on some through-the-floor anchor bolts in a nuclear power plant. They realized their error but did not correct it, because they did not expect that water would be present there. Years later, the spray system was inadvertently activated; however, operating personnel did not put the anchor bolts in the list of components to be inspected, since the bolts were supposed to have been protected by grease. Some years later, a bolt was found to be broken, leaving a major safety relevant component with reduced resistance to external load.

The common point of these examples is that, although no single organization committed any explicit error, there were serious shortcomings in effective communication and coordination between the various organizations involved. The are also numerous examples of premature ageing that could have been prevented by better communication between operation and maintenance staff within the same plant (e.g. bolt fatigue and surface wear associated with excessive maintenance of a component, even though functional reliability and performance are satisfactory).

3.4. INADEQUATE SAFETY CULTURE

In February 2002 the Davis Besse nuclear power plant in the USA went into its thirteenth refuelling outage. During that refuelling outage, while performing RPV head inspections, workers discovered a large cavity in the 15 cm thick low alloy carbon steel head material (Fig. 4). The cavity was about 16 cm long and 11 cm at the widest point, and extended down to the 6 mm thick type 308 stainless steel cladding. The utility promptly commissioned a root cause analysis to evaluate what had caused the observed corrosion. The nuclear power plant was out of service for over two years while the head was replaced and other wide scale evaluations and improvements were made to the plant, to programmes and to the staffing organization.

The technical root cause analysis determined that the corrosion was the result of boric acid interacting with the carbon steel on the RPV head. The source of the boric acid was a through-wall crack in a control rod drive mechanism nozzle. This crack was initiated as a result of primary water stress corrosion cracking (PWSCC). The Davis Besse operating organization believed that the boron accumulation on the RPV head was due to leakage from control rod drive mechanism flanges above the RPV head, and that such accumulation would not cause corrosion because of the elevated temperatures at that location. The accumulated boron on the top of the RPV was not fully removed during refuelling outages, and it masked the typical ‘popcorn’ type boron deposits from a nozzle containing a crack. Thus, the boric acid that was introduced through the nozzle crack corroded the carbon steel head, creating a cavity.

FIG. 4. Corrosion in the event at the Davis Besse nuclear power plant.

The most serious conclusion concerning the Davis Besse case was not the technical root cause of the problem, but the fact that for several years all the available warning signs — industry reports, abnormal coolant leaking, rust and boron on filters, the amount of dry boric acid on the RPV head, etc. — had been ignored. With signs such as these, the ongoing degradation of the RPV head should have been detected at an early stage of the process, at which point the process could have been stopped.

The root cause analysis concluded that the failure to identify and prevent the degradation of the RPV head was attributable to the following:

(a) There was an inadequate nuclear safety focus (instead, there was a production focus combined with minimal actions to meet regulatory requirements).
(b) Implementation of the corrective action programme was inadequate, as indicated by:
    (i) Symptoms rather than causes being addressed;
    (ii) Underestimation of deteriorating conditions;
    (iii) Inadequate cause determinations;
    (iv) Inadequate corrective action;
    (v) Inadequate trending.
(c) The organization failed to integrate and apply key industry information and site knowledge, and to compare new information on plant conditions with baseline knowledge.

Personnel did not comply with the boric acid corrosion control procedure or the in-service inspection programme, as evidenced by the failure to remove boric acid from the RPV head and to inspect the affected areas for corrosion and leakage from nozzles. Overall, there was an inadequate safety culture.

3.5. ERROR INDUCED AGEING

Premature ageing may be caused by pre-service and service conditions that are more severe than, or different from, those assumed in the design and that result from errors or omissions in design, procurement, fabrication, transportation, installation, commissioning, operation or maintenance. Examples of error induced pre-service or service conditions include: high temperature due to inadvertently closed vents or degraded thermal insulation of steam lines; excessive bearing pressure due to overtightened bolts; excessive vibration due to undertightened bolts; too frequent pressure/temperature (P/T) transients; and improper chemistry of plant circuits. Such excessive service conditions often result from improper maintenance or improper operation.

Too frequent P/T transients, even though they remain within design limits, may cause premature component fatigue. Small incidents, in particular those that affect the chemistry of the circuits, or a localized fire or inadvertent actuation of the containment spray system (even if correctly handled at the time), may induce premature degradation several months or years later. Excessive testing or maintenance can accelerate wear of components without additional benefit. Such incidents may not be taken into account in the usual assessment of ageing (because they are not anticipated and are not included in the list of design transients or incidents).

In some components, improper fabrication or improper installation can impart residual tensile stresses that can contribute significantly to ageing degradation, particularly SCC in austenitic and nickel based alloys in reactor environments. For example, improperly controlled fabrication processes used to bend piping or to straighten tubing after heat treatment can result in unexpectedly high residual tensile stresses in the material. Improper torquing of foundation bolting when installing large horizontal rotating equipment assemblies on building supports can contribute to operational vibrations that can lead to excessive in-service wear of the rotating components. For example, recent inspection results of the core internals and the recirculation pipes of boiling water reactors (BWRs) in Japan showed that a grinding and/or machining process after welding had caused surface hardening, and consequently SCC. Also, excessive implementation of tests can cause unnecessary degradation and is generally of limited value in assessing a component's initial capability and future resistance to age related failure.

In one plant, an error in the slope of auxiliary system piping made during installation led to thermal fatigue because of hydraulic turbulence that plant operators did not predict, since they were not aware of the installation error. In some cases, piping has been extremely strained during fit-up for welding during steam generator replacement, so that high residual tensile stresses were left in the component, thus enhancing the risk of stress related ageing mechanisms.

3.6. INAPPROPRIATE USE OF REACTIVE AGEING MANAGEMENT

Although the systematic and proactive approach to ageing management has been widely accepted by the operating organizations of nuclear power plants, there is still a tendency in some operating organizations to use reactive ageing management (i.e. repairing and replacing degraded components) as the primary means of managing ageing. Operating experience shows that when the ageing management strategy used is primarily reactive, the risk and costs can be very high, as is illustrated by the example of reactor cooling system corrosion.

Operating experience has raised the following primary circuit corrosion issues: In the 1980s, LWR systems experienced widespread corrosion problems on BWR recirculation lines and PWR steam generators. In the 1990s, corrosion issues included SCC of BWR internals, PWSCC of Alloy 600/182/82 components and some corrosion fatigue problems on PWRs. In the 1980s, erosion–corrosion was observed on the balance of plant components of PWRs.

Although the problem of PWSCC of Alloy 600/182/82 components had come to light long before, many nuclear power plants were 'surprised' to find cracks in SSCs important to the safety of their plants. In many cases, after finding the cracks, the operating organizations and the industry focused on dealing with the individual instances of the problem instead of considering the possibility that the problem was more general. As a result, some nuclear power plants were confronted with long outages and high costs resulting from unanticipated repairs of severe cracks.

In the past, the ageing management approach to primary circuit corrosion problems was reactive: typically, the initial response was to develop an inspection methodology to determine whether a plant was affected and, if so, where and how severely. When the problem was confirmed, typical responses were to repair the component (e.g. steam generator sleeving, BWR shroud clamps) and then, as the problem progressed, to replace it (steam generators, BWR piping, RPV heads, BWR internals and balance of plant piping for PWRs). Proactive mitigation to prevent or slow the rate of degradation has historically been a fallback strategy (e.g. H_2 injection for BWRs, mechanical stress improvements for piping and nozzle welds for BWRs and to a smaller extent for PWRs, relief of residual stress for steam generator tubes, changing of the secondary water chemistry for PWRs).

In summary, a typical sequence of events for each successive corrosion issue has been: inspection, more inspection, more sophisticated inspection, remediation or repair, and, finally, replacement. From a short term perspective, a reactive ageing management strategy is necessary; however, experience shows it to be costly (owing to downtime, the need to develop an engineering solution, regulatory uncertainty). Using a reactive strategy, a given problem can eventually be resolved, but the cumulative cost is usually high. A proactive strategy requires some upfront investment, but the downstream O&M costs are lower, more predictable and controlled.

3.7. INSUFFICIENT CAPABILITY FOR DEALING WITH UNFORESEEN AGEING PHENOMENA

Premature ageing has occurred in a number of reactor systems as a result of degradation of materials that was not anticipated at the design stage. This degradation has resulted from known ageing phenomena (e.g. fatigue or stress corrosion) and from phenomena resulting from the synergy between known degradation processes (e.g. corrosion fatigue, creep fatigue) and new phenomena (delayed hydride cracking, delayed reheat cracking). All these processes are time dependent and result in extensive cracking or material damage before loss of component function. They are therefore detectable by existing monitoring techniques; however, uncertainty in the rate of degradation poses a challenge in determining effective inspection intervals. Relevant materials data often are not directly accessible. New data need to be generated or accessed from a broader base of experience, including sectors outside the nuclear industry. Signs of possible unforeseen ageing phenomena are often detectable at an early stage if operation, maintenance and technical staff are vigilant in keeping the whole plant under review. In addition, technical staff must have a wider view by keeping up to date with worldwide nuclear and other relevant industry experience.

4. PROACTIVE STRATEGY FOR AGEING MANAGEMENT

A sound foundation for effective ageing management is one that properly takes ageing into account at each stage of a plant's lifetime. Proactive ageing management of SSCs covers the entire lifetime of the plant (design, construction, commissioning, operation and decommissioning), as well as all associated activities such as engineering, procurement, fabrication, transportation and installation. Ageing management entails continuous learning and improvement. It is useful for an operating organization to regularly consider lessons learned from its own operating experience as well as those learned from the operating experience of other nuclear power plants, in order to avoid recurrent problems and to improve SSC specific ageing management programmes.

Although, generally, different organizations are in charge of different stages and activities, all the organizations play an important role in managing the ageing of SSCs; therefore, each needs to be aware of its role in the proactive strategy for ageing management. In the proactive approach, the plant operator and plant supplier specify requirements relating to ageing management in their bidding and tendering documents, and finally in their procurement contracts. Communication is vital.

Regulatory requirements and guides for ageing management are established and updated to ensure that plant suppliers and operating organizations are aware of national requirements and expectations. Rule making is one of the most effective instruments for articulating the long term obligations of a plant operator, of a plant supplier and of their subcontractors.

This section provides guidance on, and examples of how the different organizations can contribute to, the proactive management of ageing throughout the successive stages and activities of a nuclear power plant's lifetime.

4.1. DESIGN

4.1.1. Scope of design and organizations involved

The conceptual design of a plant is available at the time the plant supplier and the future plant operator initiate their contractual negotiations. However, the design is not yet complete at the time a contract is signed and construction of the nuclear power plant begins. In particular, system and component level design continues long after construction has begun, and design modifications continue throughout the plant's service life.

The design process comprises different steps, all of which have a great effect on ageing management:

(a) Conceptual design of a nuclear power plant;
(b) System level design;
(c) Component level design.

A plant supplier is responsible for designing a plant and its SSCs to comply with the requirements set by the plant operator and the national regulatory body. This responsibility remains with the plant supplier, even in cases where design activities are contracted out to engineering companies and manufacturers.

The plant supplier, design organizations and manufacturers may not necessarily be available for support at future stages of the plant’s lifetime, and the transfer of knowledge and experience cannot be guaranteed at these later stages. Therefore, the plant operator needs to address ageing management even at the pre-operational stages.

The plant supplier, design organizations and manufacturers have the best available information and are the most knowledgeable about their products. To receive a product that complies with all criteria and requirements for long term operation, the plant owner or operator must clearly communicate these criteria and requirements to the plant supplier, design organizations and manufacturers.

Equipment qualification (EQ) is an important issue in designing SSCs. A proper EQ programme provides effective means for ageing management. Therefore, when the plant supplier and the plant operator select a manufacturer of equipment, they determine whether the EQ tests of the manufacturer are in compliance with their own EQ programmes. The EQ process is illustrated in Fig. 5.

Activities at the design input stage provide important information that is needed before EQ can be established for specific plant applications. The stage during which EQ is established involves all those activities necessary to

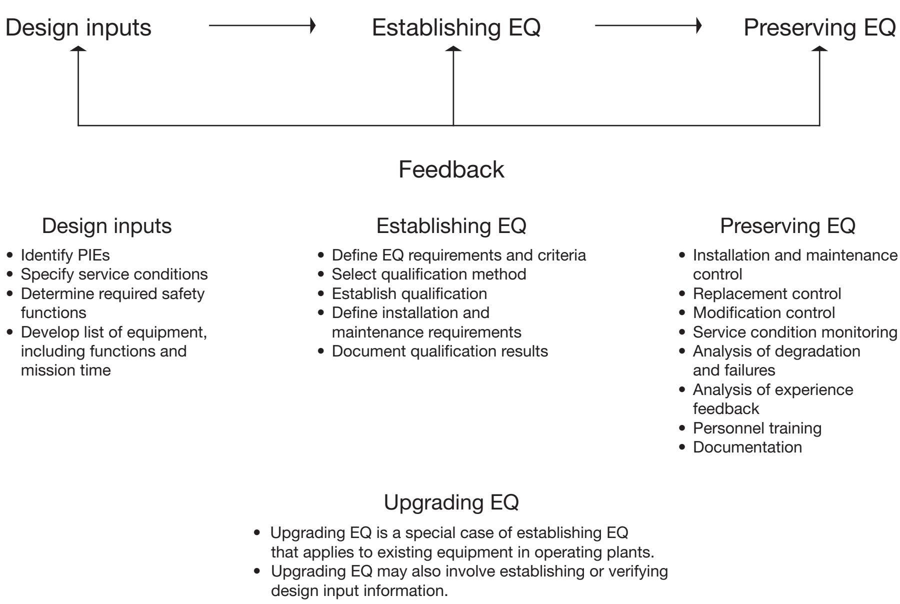


FIG. 5. The equipment qualification(EQ) process; PIE: postulated initiating event.

establish EQ for equipment design, required safety functions and service conditions. After EQ is established, a number of nuclear power plant activities have to be implemented and controlled so that the qualification of each installed item of equipment is preserved throughout the lifetime of the nuclear power plant.

Through the application for a licence, the operator demonstrates the acceptability of the design to the nation regulatory body; however, this requires the cooperation of all organizations involved. The design should include adequate arrangements for ageing management. It is useful to include this requirement in each bidding, tendering and contract document with a plant supplier, engineering company or manufacturer, as well as between the plant operator and the plant supplier. In their management system documents, each organization involved must describe the ways it contributes to ageing management. The sharing of documents and records between contracting organizations is essential for effective ageing management.

For example, for some parts of the auxiliary systems of LWRs, the design rules do not require that a complete fatigue assessment be carried out, if, during operation, the temperature differences between the main line and a branch line remain sufficiently small. This absence of a fatigue assessment may lead the plant operator to wrongly conclude that there is no risk of fatigue in this part of the facility. However, this is only the case if the temperature limits are known by the plant operator and appropriate procedures are implemented to guarantee that these limits are not exceeded.

Another example involves the choice of materials. Information concerning any potential risk that the plant operator may be taking by replacing a component with a new one made of an alternative material is important.

4.1.2. Regulations, codes and standards

Regulations, codes and standards generally require that ageing be taken into account in the nuclear power plant design. Detailed methodologies are provided for managing some of the ageing mechanisms; for others, only broad recommendations are provided. In all cases, it would be useful to clearly identify analysis and design features addressing specific ageing issues, so that the plant owner and operator can use this information to establish effective ageing management programmes. This is particularly important for ageing concerns that can lead to operating restrictions.

For example, in some cases the regulatory body requires a preliminary description of the principles of managing the ageing of the nuclear power plant as a part of the construction licence application. The description must take into account the significant ageing and wear mechanisms, and any related potential degradation, as well as provide information on the following:

- The facility's overall ageing management strategy and the prerequisites for its implementation;
- Provision for sufficient margins in the design of SSCs important to safety to ensure that they can fulfill all necessary safety functions throughout their operating lives;
- The ways that the facility layout ensures accessibility of SSCs to enable their inspection, maintenance and repair;
- The ways that the suitability and reliability of SSCs under all design basis operating and accident conditions are ensured during their acquisition;
- The ways that the availability of sufficient reference data on SSCs and on their operating conditions is ensured during construction and commissioning (testing);
- The ways that the availability of ageing management related knowledge and the expertise of facility personnel are ensured as early as during the design, construction and commissioning (testing) of the facility.

4.2. FABRICATION AND CONSTRUCTION

At the fabrication and construction stage, regulations, codes and standards provide rules or recommendations aimed at eliminating or reducing the risk of ageing degradation during subsequent operation. Again, it is important that future plant operators be informed of the measures, so that these elements can be taken into account when SSC specific ageing management programmes, including operating and maintenance procedures, are defined.

One example is the elimination of residual stress following a fabrication process. Not only must the process be properly executed by the manufacturer, but the plant operator must also be informed about the process so that attention can be paid to that point if repair work is required in the area that could negatively affect the residual stress field. The normal practice for removing residual stresses is proper heat treatment. Selection of a fabrication method that results in compressive stresses on the outer surface of pressurized equipment helps to improve SSC resistance against ageing degradation.

Another, more complex example involves RPV bottom penetrations in some PWRs. In one case, the penetration was made of a material sensitive to SCC. To eliminate the SCC risk, appropriate residual stress relief heat treatment was performed; however, after the treatment, during handling of the component, the penetration was completely bent and then straightened. The manufacturer did not perform any complementary stress relief treatment, but the future plant operator was informed of the situation so that specific measures for the ageing management of the RPV could be added.

A final example is one where strong force was required for welding together two different pieces of piping during construction. Such a use of force induces important residual stresses in the weld, thus increasing the risk of cracking. Although this method of pipe fit-up is not formally forbidden, it is important that the plant operator be informed of the increased risk so that it can be taken into account in the ageing management of that particular component.

Other situations that may be detrimental to ageing management are:

(a) Selection of inappropriate material;
(b) Loose parts falling into pipelines or vessels during installation;
(c) Insufficient play in pipelines due to thermal expansion;
(d) Too high a heat input in stainless steels due to recurrent repair welding, or to too high a temperature before welding of the next pass;
(e) Geometrical discontinuities;
(f) Improper shielding of equipment during transport and temporary storage.

As at the design stage, it is very important that the plant operator be provided with sufficient data and records, and with sufficient surveillance test specimens, needed for subsequent ageing management.

4.3. COMMISSIONING

The commissioning stage is the right time to verify the functional capability of SSCs and to collect baseline data for operation. It is also the right time to verify that environmental conditions in the different buildings and rooms are consistent with the design and qualification assumptions.

During the commissioning stage, the plant may be in an unusual situation, as final adjustments are made during the construction of some systems while other systems are being tested. Under such circumstances, there is a higher probability of incidents. Of course, if there is no fuel in the core, it is not a nuclear safety concern. However, some commissioning incidents may have long term effects that could induce premature ageing. An example is the case where resins were wrongly introduced into an auxiliary system of a plant. The incident was detected and the piping was cleaned. However, a very small amount of aggressive chemicals remained in the circuit, and although the circuit was flushed with water for months during subsequent operation, the resin contamination was sufficient to induce corrosion cracking in an elbow of this auxiliary system.

The commissioning stage is the first point at which it is possible to check SSCs in normal operating conditions. Accordingly, one important task is to check the available ‘play’ of pipelines and pipe supports at actual operating temperatures. Attention must also be paid to identification of hot spots in terms of temperature. Prevention of ageing degradation of concrete structures, polymers and cables is taken into consideration. It is important that all modifications needed to improve the prevention and mitigation of ageing be carried out at this time, before the plant enters the operating stage, because the commissioning stage is the last point at which it is possible to undertake these activities without any radiation dose to personnel.

It should also be noted that the many tests performed during commissioning — like any operating transients — induce stresses in the components that may have an impact on their ageing degradation (e.g. from a fatigue point of view). This is not a problem if the tests are conducted correctly, because performance of these tests is taken into account at the design stage. At the least, the testing history must be properly documented and recorded, to allow investigation of possible cases of subsequent premature ageing that may be explained by improper execution of some testing. An example is the hydro test of pressurized components that is required by the regulatory bodies of many countries. This test is supposed to be performed once during commissioning, and it is taken into account in the design fatigue analysis (if one is done) as one large pressure transient. However, the future plant operator may choose to increase and decrease the pressure several times because of leakage of valves or flanges that was not eliminated before the hydro test. Thus, the actual number of transients may be higher than that assumed in the design calculation. Even though the corresponding reduction in the fatigue usage factor may be quite small, it is important that plant staff in charge of this work be aware of any potential consequences of it.

The establishment of an ageing management programme is initiated well before the plant begins operation. In some cases, the regulatory body requires a plan for ageing management as a part of the operating licence application. Such a plan addresses the integration of the design and qualification of the components and structures, their operation and operating experience, in-service inspections and tests, and maintenance into a comprehensive ageing management programme. To provide a basis for the plan, all significant ageing and wear mechanisms and any related potential degradation are to be identified, and information is to be provided on the following:

— Provision for sufficient margins in the design of SSCs important to safety to ensure that they can fulfill all necessary safety functions throughout their operating lives;
— The ways that the facility layout ensures accessibility of SSCs to enable their inspection, maintenance and repair;
— The ways that the suitability and reliability of SSCs under all design basis operating and accident conditions are ensured during their acquisition;
— The ways that the availability of sufficient reference data on SSCs and on their operating conditions is ensured during construction and commissioning (testing);
— The ways that the availability of ageing management related knowledge and the expertise of facility personnel are ensured as early as during the design, construction and commissioning (testing) of the facility.

4.4. OPERATION

Almost all activities during the operation stage can have an impact on the rate of ageing degradation. The goal is not to eliminate ageing but to control it effectively. Effective ageing management requires SSC specific application of the systematic ageing management process (see Section 5), which provides a framework for coordinating all programmes and activities relating to the understanding, control, monitoring and mitigation of ageing of a plant component or structure.

Establishment of the overall nuclear power plant ageing management programme or system ensures that the operating organization is managing ageing on the basis of 'collective awareness':

(a) Each organization and each individual is aware of the potential consequences of its, or his or her, activity with respect to ageing (operation, maintenance, engineering, etc.).
(b) Open and effective communication and coordination between organizations and between individuals is applied so that they are collectively aware of the possible cross-effects. As this 'collective awareness' is not always a natural feature, strong commitment by management is necessary. As with safety, ageing management is everybody's concern, every day. Utilization of multidisciplinary teams in complex ageing management issues provides added value.
(c) The 'collective awareness' aspect is important, because many ageing issues concern different plant organizations, different programmes and different disciplines.

(d) Staff awareness of ageing, as well as motivation, training and sense of ownership, are important elements to facilitate effective ageing management. The topic of ageing management is discussed in most training courses, not just in those training courses specifically concerning ageing management. A sense of ownership increases staff motivation and responsibility.
(e) A management system for ageing management is implemented as part of the integrated management system, which is put into effect by the operating organization early in the plant's lifetime.

When the operating organization carries out extensive maintenance or refurbishment work, all design or materials changes made are to be thoroughly assessed. There may be a detrimental ageing effect if the operating organization is not aware of changes in the installation design or in the materials or components used. In such cases, it may be that some design and construction options are chosen in order to mitigate an ageing concern, but that this decision is not explicitly documented and thus the plant owner is not aware of it. One such example is the decision to use spare parts made of slightly different materials than those originally used, without proper assessment of the potential safety consequences. In the case of guide tube pins made of a nickel based alloy, very minor differences in the fabrication process could affect component sensitivity to stress corrosion.

A second important point is that the rate of ageing degradation can often be reduced by optimizing operations. An example is ensuring the smooth operation of systems, to reduce pressure and temperature transients. In the case of primary system components, which generally undergo fatigue assessment at the design stage, proper recording of transients allows the fatigue usage factor to be recalculated with actual stressors. Another example is optimizing the fuel pattern in the core to reduce fluence on the core vessel and, in turn, to mitigate embrittlement. In some plants, it has been possible to reduce the fluence by 20% while maintaining the same power output and fulfilling all technical specifications. Situations where it is possible to achieve several beneficial objectives such as safety, ageing management and performance arise frequently. These objectives are not necessarily contradictory if operation is properly optimized.

5. PROACTIVE AGEING MANAGEMENT IN THE OPERATION OF NUCLEAR POWER PLANTS

The experience to date has demonstrated the impact of unexpected ageing phenomena and the consequent degradation of SSCs. In older plants, ageing effects have often been so severe that component replacement is the only realistic option. In newer plants (which also tend to be larger in terms of capacity), improved materials, fabrication and inspection have been deployed in construction so that age related degradation may be slower, giving hope that the early implementation of a proactive ageing management strategy will make large component replacement unnecessary.

It is recognized that even the normal expected rate of ageing degradation may be controlled and reduced by taking timely and appropriate operational measures. Ageing degradation of SSCs can also be controlled and reduced by implementing proactive measures during the design and fabrication stages.

At existing nuclear power plants, it would be prudent to review current ageing management programmes employed for long lived passive SSCs and major types of active component (such as motor operated valves), to determine the potential advantages of using the proactive strategy. The review should take into account:

(a) The current condition of the SSC;
(b) The current understanding of SSC ageing, including the significant ageing mechanisms and effects, their modelling and predictability, and likely degradation sites based on both operating experience and research;
(c) Current ageing management practices and available monitoring and mitigation methods;
(d) The planned service life of the nuclear power plant.

Implementation of a systematic ageing management process facilitates the selection of appropriate strategies (proactive or reactive) and the coordination of relevant programmes and activities to minimize premature ageing.

This section provides guidance on the implementation of proactive ageing management in the operation of nuclear power plants through the application of a systematic ageing management process.

5.1. APPLICATION OF A SYSTEMATIC APPROACH TO AGEING MANAGEMENT

Effective ageing management throughout the service life of an SSC requires the use of a systematic approach to managing ageing that provides a framework for coordinating all programmes and activities relating to the understanding, control, monitoring and mitigation of ageing effects of the plant component or structure. This approach is illustrated in Fig. 6, which is an adaptation of Deming's 'Plan–Do–Check–Act' cycle to the ageing management of an SSC [6]. Understanding the ageing of a structure or component, as illustrated in Fig. 6, is the key to its effective ageing management.

The 'Plan' activity in Fig. 6 involves coordinating, integrating and modifying existing programmes and activities that relate to managing the ageing of a structure or component and developing new programmes, if necessary. The 'Do' activity aims at minimizing expected degradation of a structure or component through its 'careful' operation or use, in accordance with operating procedures and technical specifications. The 'Check' activity involves the timely detection and characterization of significant degradation through inspection and monitoring of a structure or component, and the assessment of observed degradation to determine the type and timing of any corrective actions required. The 'Act' activity aims at the timely mitigation and correction of component degradation through appropriate maintenance and design modifications, including component repair and replacement of a structure or component. The closed loop in Fig. 6 indicates the continuous improvement of the ageing management programme for a particular structure or component, on the basis of feedback of relevant operating experience and R&D results, and of results of self-assessment and peer reviews, to help ensure that emerging ageing issues are addressed.

5.1.1. Understanding SSC ageing

Adequate understanding of SSC ageing is the basis of a systematic ageing management process and the key to successful proactive ageing management. This understanding is derived from: knowledge of the design basis; the design and fabrication data (including material properties and specified service conditions); the operation and maintenance history (including commissioning

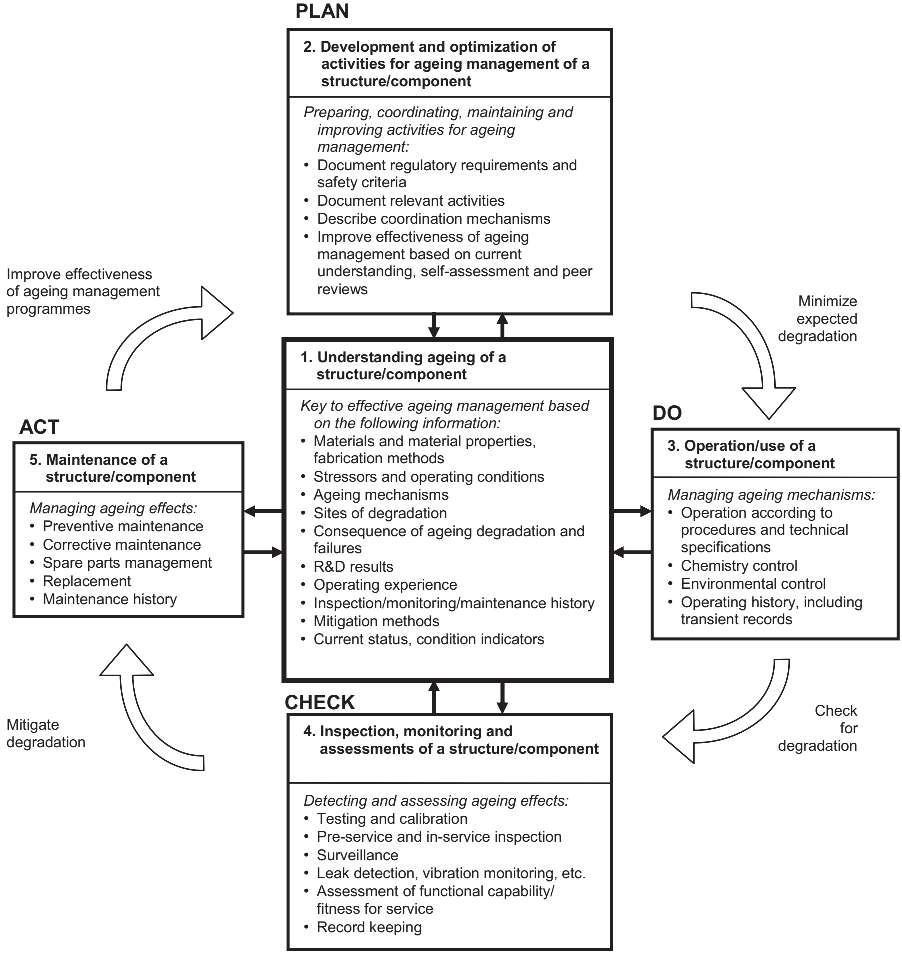


FIG. 6. Systematic approach to ageing management of a structure or component.

and surveillance); inspection results; and generic operating experience and research results.

The level of understanding of SSC ageing depends to a large extent on the degree of technical and/or scientific understanding of relevant ageing mechanisms, and on the quality and quantity of relevant data from operating experience. Understanding ageing enables the prediction of future SSC ageing degradation; this predictability in turn enables the optimization and coordination of SSC operation/use, inspection and maintenance.

In practice, predictability requires modelling the relevant ageing mechanism and SSC degradation, and measuring the progress of SSC degradation ‘along the curve’ (i.e. condition monitoring; see Fig. 2).

Radiation embrittlement is an ageing mechanism leading to changes in bulk material properties that has been successfully modelled; its future effects can be accurately predicted and measured. The predictability of thermal embrittlement of stainless steels, and of irradiation and thermal degradation of polymers (used in cable insulation and seals), which also produce changes in bulk material properties, is also generally adequate.

On the other hand, the predictability of corrosion, wear and high cycle fatigue, which produce changes in material surfaces or interfaces (loss of material or formation of small cracks), is generally low. The resulting uncertainty has caused significant nuclear power plant unavailability and increased O&M costs.

Operating experience has revealed SSC degradation and failures caused by previously unrecognized ageing mechanisms. Thus, in addition to improving the understanding and predictability of known ageing phenomena, there is a need to provide for early detection of new ageing mechanisms.

5.1.2. Development and optimization of activities for ageing management of a structure or component (Plan)

As illustrated by the examples given in Section 3.2, premature ageing degradation often can be traced back to a lack of communication, documentation and coordination between design, operation and maintenance organizations. This results from the general situation that, except for major SSCs (such as an RPV) and replaceable components covered by an EQ programme, no single person or body has the explicit responsibility to achieve a specific SSC lifetime.

The effectiveness of SSC ageing management can be significantly improved, and SSC service life can be significantly extended, through the coordination of all relevant programmes and activities, taking into account the design assumptions and limitations, applicable regulatory requirements, significant ageing mechanisms, and the impact of O&M activities on these mechanisms and on the rate of SSC degradation. The objectives of systematic coordinated ageing management are: to fulfil an overview role, to integrate and coordinate existing relevant programmes in order to identify areas for additional effort, to eliminate any overlaps or unnecessary activities, and to provide a mechanism for programme coordination and continuous improvement. Although this coordination generally requires a modest financial investment, the positive cultural and organizational implications may be considerable.

The systematic ageing management process outlined in Fig. 6 facilitates communication among all contributors to ageing management: each contributor can clearly see its, or his or her, role in the process, as well as the synergy of the team approach that stimulates creativity and amplifies the benefits of individual contributions.

5.1.3. Operation or use of a structure or component (Do)

Operation of a nuclear power plant can affect the rate of degradation of SSCs. Exposure of an SSC to operating conditions (e.g. temperature, pressure, water chemistry) outside prescribed operational limits can lead to accelerated ageing and premature SSC degradation, affecting plant safety and availability. In particular, it is prudent to attempt to control the operating environment of inaccessible SSCs where detection and repair of degradation would be difficult and costly.

Since operating practices influence SSC operating conditions, nuclear power plant operation staff have an important role in minimizing SSC ageing. They can do this by maintaining operating conditions within prescribed operational limits. Examples of such operating practices are:

(a) Operation within the prescribed pressure and temperature range and rate of change during startup and shutdown to avoid undue transient stresses and the risk of overstress.
(b) Even where design fatigue criteria are met, optimization of operating procedures may reduce transients on some components, thus increasing margins and/or service life expectancy.
(c) Performing maintenance according to procedures designed to avoid contamination of metal components with aggressive contaminants (such as boric acid or other reagents containing halogens).
(d) Maintaining plant heating, ventilation and air conditioning in a state that keeps plant environments within prescribed (design basis) conditions, including prevention of overheating of concrete structures.
(e) Maintaining the thermal insulation on high temperature lines and equipment in good order.
(f) Careful chemistry monitoring, trending and result assessment, beyond specifications compliance, may allow a reduction in the corrosion rate and an increase in the steam generator lifetime.

Moreover, a good understanding of SSC ageing facilitates optimization of operating conditions and procedures to reduce the rate of normal ageing degradation of some SSCs, hence increasing safety margins and/or service life.

5.1.4. Inspection, monitoring and assessment of a structure or component (Check)

Inspection and monitoring activities are designed to detect and characterize significant SSC degradation before safety margins are compromised. Together with an understanding of ageing degradation, the results of inspections and monitoring and the subsequent trending of degradation parameters provide a basis for decisions regarding the type and timing of maintenance actions, and the operational changes and design modifications to manage detected ageing effects. A risk informed methodology can provide the basis for targeted and more effective inspection and assessment.

For example, a proactive monitoring, inspection and trending programme can be used to detect wall thinning of steam/feedwater piping due to flow assisted corrosion, to characterize degradation rates and locations, and, if necessary, to predict when repairs or replacements need to be implemented.

Procedures for the inspection and assessment of crack-like defects (ASME XI, R5, R6) have been established and are under continuous development. However, these procedures are limited to the detection and evaluation of significant degradation phenomena (e.g. fatigue, environment assisted cracking).

Inspection and monitoring should be directed both at the whole plant and at specific SSCs, since many of the ageing degradations were discovered not with NDE techniques but with more global monitoring (e.g. visual inspection, loose parts monitoring, leak monitoring). For example, SCC of Alloy 600 steam generator tubes was discovered by leak detection, while the eddy current inspection was not capable of detecting the cracking at that time. In another case, cracking of guide tube pins made of Alloy 750 was found by the loose parts monitoring system.

5.1.5. Maintenance of a structure or component (Act)

A variety of preventive and corrective actions are available to mitigate ageing effects detected by inspection and monitoring of SSCs. Decisions on the type and timing of the maintenance actions are based on an assessment of the observed ageing effects, an understanding of the applicable ageing mechanism(s), the predictability of future degradation, the available decision criteria and the effectiveness of available maintenance technologies.

For ageing mechanisms with low predictability, such as corrosion, it may be appropriate to use preventive maintenance to prevent the onset of corrosion. For example, proactive steam generator secondary side cleaning, implemented on a regular basis before in-service degradation is detected, is preventive maintenance that can prevent or delay tubing corrosion.

For some ageing problems, modification of a component or system is the only solution. Fatigue caused by thermal stratification often can only be avoided or minimized by replacing horizontal piping with piping that has a certain slope in it. An example is the pressurizer spray system in some PWRs, which was modified to avoid thermal stratification.

5.2. IMPLEMENTATION OF PROACTIVE AGEING MANAGEMENT

The effective implementation of a proactive ageing management process that counters common weaknesses (see Section 3) requires that a clear strategy be adopted by plant owners, that appropriate organization and technical processes be in place, and that the proper tools be available. Utilities that introduced systematic ageing management early in the lifetime of the plant are demonstrating high levels of safety and performance (high capacity factor and low O&M costs), and long service life.

5.2.1. Strategy

The first priority is for the owner or operator of the plant to have a clear ageing management strategy that provides for timely detection and mitigation of ageing degradation. This strategy is dictated by the plant safety case, needed to satisfy regulatory requirements, and by the economic requirements for the plant's operation. With these two main drivers, the ageing management strategy should include:

(a) Ownership of the strategy at the highest level in the company owning or operating the plant. Ageing management is a 'destiny' issue.
(b) A clear policy statement that commits the owner or operator to implementation of the strategy.
(c) Recognition that, although plant safety and life management may be seen as separate issues, their effective implementation requires a common approach to ageing management.

(d) The use of appropriate multidisciplinary teams, which is a proven means to effectively address and resolve ageing management issues.
(e) The instilment of a strong safety culture in all staff to achieve excellence in plant material condition, safety and performance, and of a blame-free culture in which mistakes are acknowledged and acted upon.

5.2.2. Organization

As Section 4 indicates, the comprehensive nature of proactive ageing management requires a team approach and a long term partnership in which participants from several organizations are involved (e.g. engineering, manufacturing, O&M). A multidisciplinary team approach in a traditional organization with strong 'boundaries' is a challenging task that requires considerable effort. A successful team approach and long term partnership requires:

(a) Awareness on the part of management of interorganizational boundaries and of appropriate action to facilitate cooperation and a long term relationship.
(b) An ageing management coordinator with credibility, an appropriate background and broad experience covering nuclear power plant engineering/technology, maintenance and operation to oversee the ageing management programme. The coordinator must have the ability to work with people from different organizational units and having different expertise.
(c) Competent and sufficient staff (either dedicated or part-time) to assist the ageing management coordinator in the integration and coordination of existing plant and external ageing management programme activities, and in ageing management programme self-assessments on the basis of documented acceptance criteria.
(d) Sufficient numbers of qualified and well-trained operations, maintenance and technical support staff who understand the systematic ageing management process and are capable of performing required ageing management actions and tasks, in harmony with the ageing management programme.
(e) Adequate funding to maintain the ageing management programme, independent of other plant budget allocations.
(f) Transparency and good communication between and within all organizations involved. A special effort must be made to reduce the gap between people who work in the plant and people from more distant supporting engineering/technical departments (e.g. by staff rotation).

(g) Adequate and appropriate equipment and tools to perform ageing management actions.
(h) Formal communication links with external agencies and other nuclear power plants, specialized nuclear power plant teams and consultants in areas in which the ageing management programme organization is not self-sufficient.
(i) Mechanisms to incorporate industry and in-house operating experience and research relating to age related degradation of SSCs.
(j) Building up the team approach, starting with pilot projects. It should be noted that an appropriate embryonic team structure often exists as a result of the implementation of periodic safety reviews and the use of multidisciplinary task forces in reaction to earlier ageing problems.

5.3. PROCESS

The ageing management coordinator is responsible for effective implementation of the systematic ageing management process (see Section 5.1). This requires, inter alia, selection of appropriate strategies (proactive or reactive) for different types of SSC to achieve optimum cost effectiveness of ageing management. Inappropriate use of reactive ageing management has been shown to be a significant weakness in this regard (see Section 3.5). However, for active components that are not important to safety, and whose failure has limited economic impact, it may be cost effective to implement a reactive, ‘run to failure’ strategy.

It is wise to focus detailed ageing management actions on SSCs where they are most beneficial from both a safety and an economic point of view. Risk based/informed methodologies can be used to facilitate appropriate decisions.

Design is based on a deterministic methodology, whereas system safety analysis has developed on a risk basis. The development of risk based methodologies and their use in PSRs has encouraged their use in relation to nuclear power plant component reliability and integrity. Reliability centred maintenance of active components and risk based inspection and assessment of passive components are now well developed methodologies in widespread use in other industries in relation to plant ageing. By evaluating both the likelihood and the consequences of failure, they help in the prioritization of SSCs for degradation inspection and monitoring, and for corrective action (the Check and Act parts of the Deming cycle shown in Fig. 6). Such tools can also be used to indicate new design weaknesses.

Much of the work of the coordinator consists in implementing the proactive process outlined in Section 5.1. This covers a range of activities aimed at identifying and supporting appropriate operations, monitoring and inspection, and maintenance actions. These include development, in advance, of methods and tools for likely repair or replacement (including spare part purchase), and modification of components or systems to slow the rate of ageing degradation.

Early identification and mitigation of unforeseen ageing phenomena requires vigilance of operation, maintenance and technical staff in relation to the plant's condition and behaviour. The detection of anomalies can provide an indication of new ageing phenomena. The ageing management team should ensure that the required high level of vigilance is met.

Effective quality control is an essential element of the systematic ageing management process. It is one of the main factors in minimizing error induced ageing.

5.4. INFORMATION MANAGEMENT

Ageing management involves a large amount of information from different internal and external programmes (e.g. in-service inspection, maintenance, operation, water chemistry control, design, fabrication, installation/commissioning, R&D) and from the operating experience of other nuclear power plants. Information and data collection must be done globally, through bilateral relationships or through international organizations such as the IAEA, INPO, OECD/NEA, WANO and owners groups. This search for information should also be extended to other industries that may have similar components, materials and operational situations that could induce similar ageing mechanisms (e.g. power generation, process plant, offshore oil and gas).

Because of the large amount of data involved, it is important to have a good information management (data collection and record keeping) system. This is also very important for sharing knowledge about ageing among all relevant staff, including newcomers. The IAEA has established a safety knowledge base on ageing and long term operation (SKALTO), which is available on its web site and from which users can retrieve relevant information on these thematic areas.

Appendix

EXAMPLES OF AGEING MANAGEMENT ISSUES AND THEIR RESOLUTION

The following examples review specific plant ageing issues and identify their causes and corrective actions. The purpose of the examples is to demonstrate that plant ageing issues may have their origins, as well as their resolution, in any kind of activity, including procurement, design, transportation, fabrication, construction, commissioning, operation and maintenance.

FORMAT OF EXAMPLES

The following format is used for the examples provided:

1. Component/subcomponent
2. Brief description of ageing issue or concern
3. How ageing was detected or suspected
4. Contributing causes to the ageing issue:
   — Design and analysis, procurement
   — Fabrication, transportation, installation and construction
   — Commissioning
   — Maintenance
   — Operations
5. Impact of ageing on:
   — Safety
   — Reliability
   — Production
   — Cost
6. Brief description of corrective actions
7. Lessons learned, including long term ageing management actions and current performance
8. Applicability to other components
9. Effect on understanding and communications, and on development of an effective ageing management culture at the plant

EXAMPLE 1

1. COMPONENT/SUBCOMPONENT

Steam generator tubing (Alloy 600)

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

Primary side stress corrosion cracking (SCC) led to the need for tube plugging and, ultimately, replacement of the steam generator.

3. HOW AGEING WAS DETECTED OR SUSPECTED

The problem was originally detected by leakage monitoring. Non-destructive testing examination techniques were used to characterize and monitor for cracks.

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

- Design and analysis, procurement: The choice of material was inadequate.
- Fabrication, transportation, installation and construction: There was no appropriate residual stress relieving (it should be noted that in the ASME Code there is no effective requirement for stress relieving).
- Commissioning: None.
- Maintenance: None.
- Operations: None.

5. IMPACT OF AGEING ON:

- Safety: Risk of tube rupture.
- Reliability: Some plants experienced forced outages.
- Production: Could lead to degradation if there is excessive tube plugging.
- Cost: Significant costs and extra doses (inspection, maintenance and replacement).

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

- In situ residual stress relieving.
- Ongoing inspection and plugging of tubes.

— Reduction of system operating temperature (cost–benefit considerations need to be considered).
— Replacement of steam generator.

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

— Selection of materials is critical in heat exchanger components.
— Inconel 690 is the preferred material.
— Low residual stress fabrication techniques are to be used.
— There is a need for ongoing inspections and tube plugging.
— Preparations should be made to replace the steam generator, depending on the scope of the damage or degradation.

8. APPLICABILITY TO OTHER COMPONENTS

Materials specification is important for all heat exchanger applications.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

This is a well documented example of premature ageing in the nuclear power industry and is often quoted as an example of understanding the many contributors to this issue.

EXAMPLE 2

1. COMPONENT/SUBCOMPONENT

Pressurizer surge line and other pipelines

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

The pressurizer surge line and pipelines subjected to thermohydraulic ‘turbulence’ (THT) and thermal stratification have fatigue damage (see Fig. 7(a) and (b)). The design takes into account the THT that results from major load changes. However, it has been found that THT is experienced frequently and in response to many load changes.

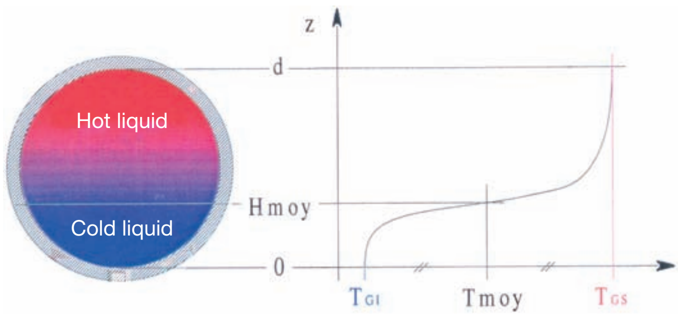


FIG. 7(a). Thermal stratification.

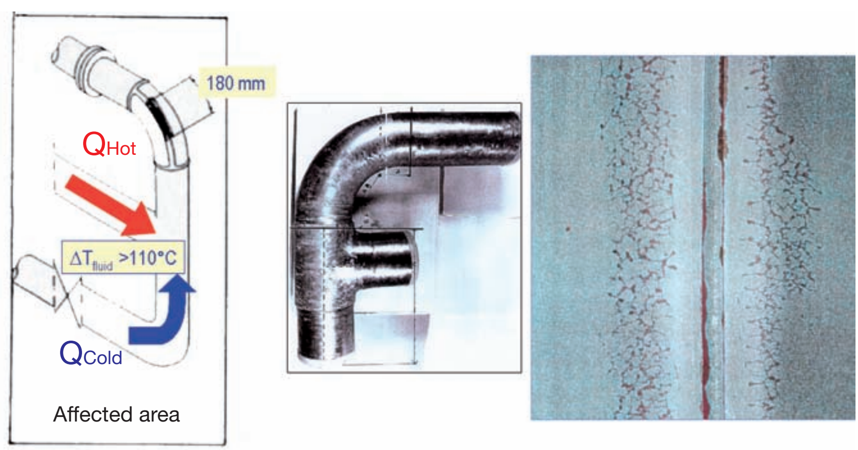


FIG. 7(b). Thermal mixing Ts.

3. HOW AGEING WAS DETECTED OR SUSPECTED

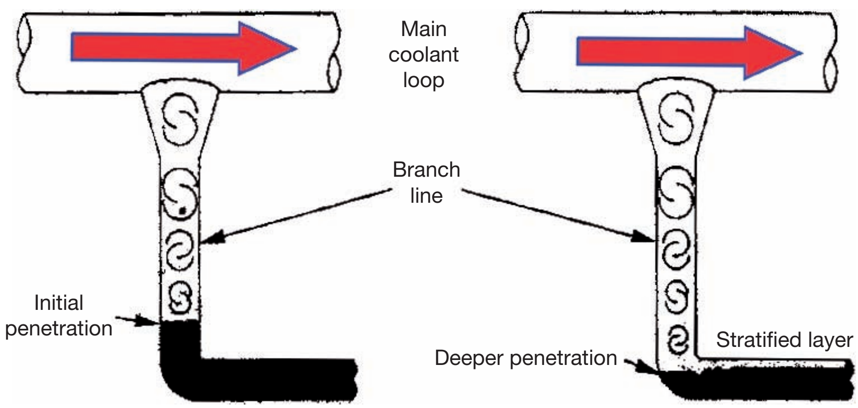
THT was detected by specially designed and installed instrumentation.

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

— Design and analysis, procurement: The piping layout determines the prime locations subject to THT. Thermohydraulic analysis predicted THT during major load changes; however, the models did not predict the rate of THT that results from less severe operating changes (Fig. 8).
— Fabrication, transportation, installation and construction: None.
— Commissioning: None.
— Maintenance: None.
— Operations: Load changes cause THT.

5. IMPACT OF AGEING ON:

— Safety: Failure of the pressurizer line would cause a major loss of coolant accident.
— Reliability: None.
— Production: If replacement of the pressurizer line were required, a significant loss of production would be incurred.
— Cost: Replacing the line would incur high costs and extra doses.



(a) Initial penetration of turbulence is confined to the vertical section.

(b) Deeper penetration causes stratification in the horizontal section.

FIG. 8. One of the mechanisms of thermal stratification.

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

— Monitor the effect of THT stresses.
— Identify operating modes to minimize THT.

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

— If it is not possible to eliminate thermal stresses and fatiguing, there may be ways to direct the site of the fatiguing to those portions of the system or piping where inspection and replacement are more expedient.
— Ongoing investigation and monitoring.

8. APPLICABILITY TO OTHER COMPONENTS

This is applicable to all piping systems that experience THT or other stressors such as water/steam hammer.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

Detailed root cause analyses have been performed for these phenomena, and mechanisms have been well analysed. Some R&D programmes are ongoing. It has been found that the current code is not appropriate.

EXAMPLE 3

1. COMPONENT/SUBCOMPONENT

Elbow of the ECCS system piping (PWR)

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

A leak occurred in 1988 in a hot leg elbow of the emergency core cooling system (ECCS) injection line during normal plant operation (see Fig. 9). Similar incidents occurred in other plants in 1987 and in 1992. In the example presented here:

— The leak was attributed to high cycle thermal fatigue.
— The thermal cycling was attributed to small quantities of chemical and volume control system (CVCS) cold water flowing into the non-isolable section of the reactor coolant system (RCS).
— CVCS cold water intrusion was attributed to a leaking isolation block valve.

3. HOW AGEING WAS DETECTED OR SUSPECTED

Ageing was detected through leak detection and laboratory examination of flawed pipe spools.

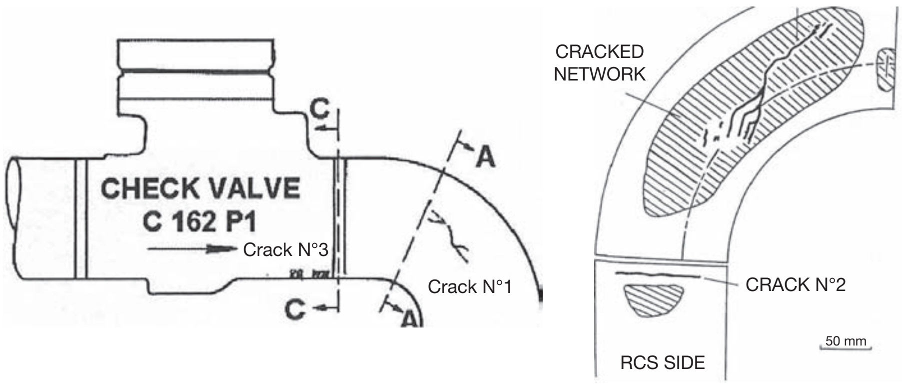


FIG. 9. Cracks on the ECCS system elbow.

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

- Design and analysis, procurement:
  - Contributing thermal transients were not included in the RCS design basis.
  - Contributing transients are difficult to characterize.
- Fabrication, transportation, installation and construction: Material analysis did not reveal any deviation from specified requirements.
- Maintenance:
  - There was no periodic verification of the leaktightness of block valves.
  - The standard for in-service inspections (ISIs) does not cover areas other than welds.
  - The use of the ultrasonic testing (UT) technique recommended by the ASME Section XI Code could not reveal unacceptable flaw indications.

5. IMPACT OF AGEING ON:

- Safety: Rupture of an elbow in the ECCS would cause a loss of coolant accident.
- Production: The leak led to unexpected outage for evaluation and repair.
- Cost:
  - Short term: Outage, repair, examinations, tests, etc.;
  - Long term: Study and implementation of definitive corrective actions.

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

- Replacement of flawed pipe spool and check valve:
  - Short term safety/reliability actions:
    - Inspection of similar RCS (ECCS) pipe sections;
    - Inspection of block valve leaktightness;
    - Temperature monitoring downstream of the ECCS check valve;
    - Pressure monitoring between ECCS block and check valves;
  - Long term safety/reliability actions:
    - Installation of a permanent pressure monitoring system with depressurization capability to avoid $P > P_{RCS}$ in pipes located between block and check valves in the ECCS;
    - No further in-service inspection required by the safety authority.

The corrective system has fulfilled its purpose since 1988.

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

— Severe thermal fatigue phenomena were not anticipated at the design stage.
— Other critical areas are affected by related phenomena.
— Efficiency of code criteria for UT inspections.
— Leaktightness is critical for some block valves.

8. APPLICABILITY TO OTHER COMPONENTS

— Near auxiliary feedwater (FW)/shutdown cooling/main FW Ts;
— Residual heat removal (RHR) lines near isolation valve;
— RHR lines near bypass T.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

The case described led to better understanding and communications, and contributed to the development of an ageing management culture at the plant. In addition, analysis of critical areas has been included in the periodic safety review programme.

EXAMPLE 4

1. COMPONENT/SUBCOMPONENT

Main and auxiliary feedwater lines in PWRs

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

Thermal fatigue was induced by thermal stratification transients. Several operation modes may be responsible for thermal stratification transients in the feedwater system.

3. HOW AGEING WAS DETECTED OR SUSPECTED

The ageing mechanism was revealed in NRC Bulletins 79-13 Rev.2 [7] and 88-11 [8]. Long term temperature monitoring was used to characterize stratification thermal transients.

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

— Design and analysis, procurement:
  • The type of steam generator (boiler or with preheater) plays an important role.
  • The type of feedwater isolation check valve (with or without pilot valve) plays an important role.
  • Feedwater piping layout may have a significant impact.
  • The presence of sharp discontinuities (stress raisers) at critical locations, such as the steam generator feedwater nozzle and the penetration in the reactor building, is a contributing cause.
  • Contributing thermal transients were not included in the feedwater system design basis.
  • Class 2 piping equations are not adequate to evaluate fatigue damage due to stratification (local effects cannot be taken into account).
— Fabrication, transportation, installation and construction: Quality control of the welds is important.
— Commissioning: None.
— Maintenance: None.

— Operations:
  - The use of the auxiliary feedwater system during hot standby plays a major role (flow, number of occurrences).
  - The use of the main and auxiliary feedwater systems after a scram also plays a major role.

5. IMPACT OF AGEING ON:

— Safety and reliability:
  - Risk of feedwater line rupture.
  - Forced outage to repair leak or rupture and related damage.

— Production and cost:
  - Higher stress and fatigue:
    - ○ Additional breaks;
    - ○ Additional whip restraints;
    - ○ Additional design and construction costs.
  - Costs related to long term monitoring and fatigue analysis.
  - Repair leads to significant costs and extra doses.

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

— Use appropriate steam generator type (only possible in very few instances).

— Use appropriate isolation valve type (replacement is expensive).

— Use appropriate piping layout (see Fig. 10).

— Use devices capable of preventing the development of stratification along large portions of piping (see Fig. 10).

— Avoid stress raisers.

— Avoid use of the auxiliary feedwater system after a scram (to avoid stratification in lower portions of the main feedwater pipe).

— Use continuous injection of auxiliary feedwater during hot standby (to avoid stratification cycling in the upper portion of the main feedwater pipe).

— Use temperature monitoring to define stratification transients and Class 1 type fatigue analysis to justify the feedwater line as is.

Temperature measurements confirm that no stratification occurs in the lower part of the main feedwater lines.

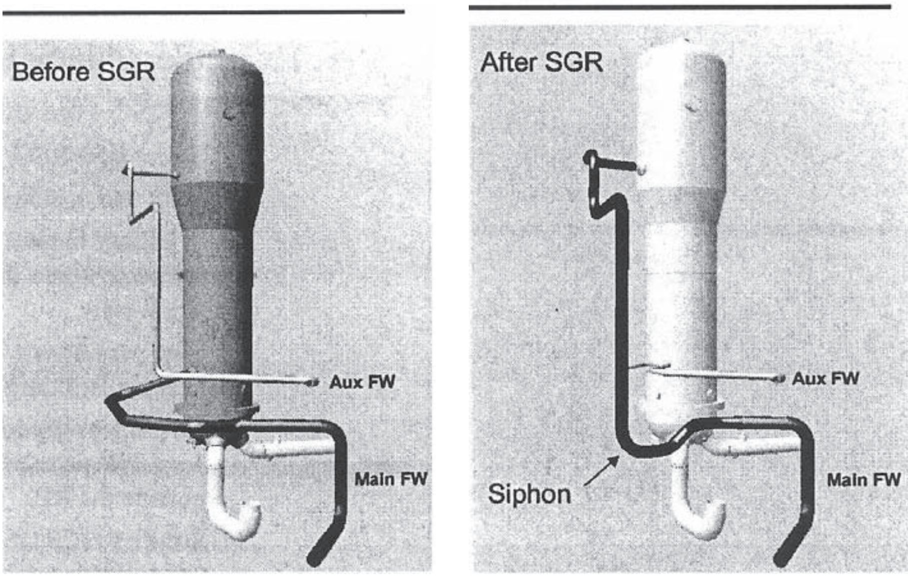


FIG. 10. Piping layout before and after steam generator replacement (SGR); FW: feedwater line.

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

— Several design options may have a significant impact on the development and severity of the thermal stratification transients.
— Operation of the auxiliary feedwater system is also a key factor.

8. APPLICABILITY TO OTHER COMPONENTS

This may be applicable to other systems where thermal stratification is possible.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

This example shows the need to maintain long term fatigue monitoring at critical sections of the feedwater lines.

EXAMPLE 5

1. COMPONENT/SUBCOMPONENT

Pipes, valves, T junctions in primary side (WWER)

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

This case involved thermal fatigue, with thermal stratification in horizontal low-flow pipes and high cycle fatigue in T junctions (see Fig. 11).

3. HOW AGEING WAS DETECTED OR SUSPECTED

The problem was known from existing reports from other nuclear power plants, and from scientific studies. Two leaks in the plant were exacerbated by thermal fatigue.

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

— Design and analysis, procurement: In the design, small velocities and big differences in temperatures (dT) in T junctions should be avoided; difficult to analyse.
— Fabrication, transportation, installation and construction: Material defects.
— Commissioning: Transients that were unplanned and not documented.
— Maintenance: None.
— Operations: Unplanned, improper operations.

5. IMPACT OF AGEING ON:

— Safety: Loss of integrity of the primary system.
— Reliability: None.
— Production: None.
— Cost: Unplanned shutdowns and additional doses.

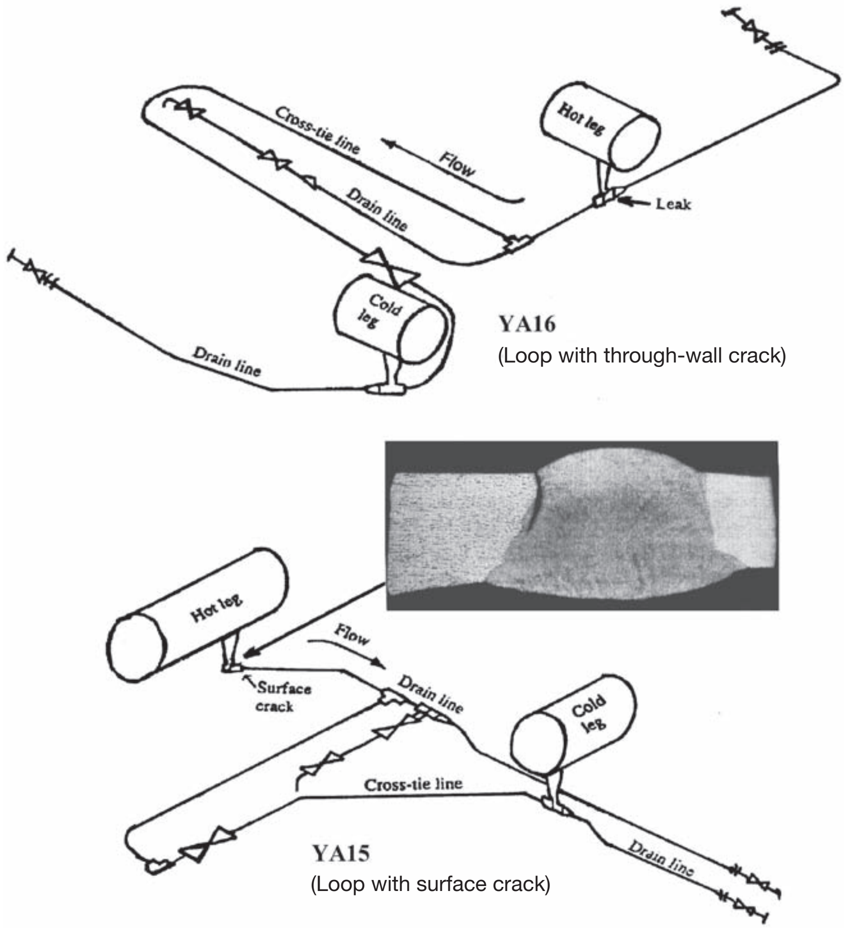


FIG. 11. Layout of cross-tie lines and drain lines in the primary circuit of a WWER 440 reactor.

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

— Permanently closed valves should be avoided. Difficulties include the large number of potential locations and the development of screening criteria.

— Sensitive systems were studied and suspect locations were listed.
— Leak detection systems were increased.
— The scope of in-service inspections was increased.

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

Leaking (permanently ‘closed’) valves are suspect and should be avoided in the design. The operating history, including the commissioning stage, should be well documented.

8. APPLICABILITY TO OTHER COMPONENTS

This case is basically applicable to all pipes, T junctions and other components in the primary circuit.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

The mapping of potential areas of stratification and the increasing frequency of in-service inspections have increased understanding of the phenomena.

EXAMPLE 6

1. COMPONENT/SUBCOMPONENT

Core internals and recirculation piping made of austenitic stainless steel (BWR)

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

Intergranular stress corrosion cracking (IGSCC) was found on many core internals and recirculation pipes (Fig. 12), even on those made of nuclear grade stainless steel, which has very low sensitivity to IGSCC (Figs 13 and 14).

3. HOW AGEING WAS DETECTED OR SUSPECTED

The IGSCC was detected through in-service inspections (ISIs) and other inspection activities based on experience in several countries.

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

— Design and analysis, procurement: IGSCC was not taken into account in the original design of BWRs; materials susceptible to IGSCC were used.

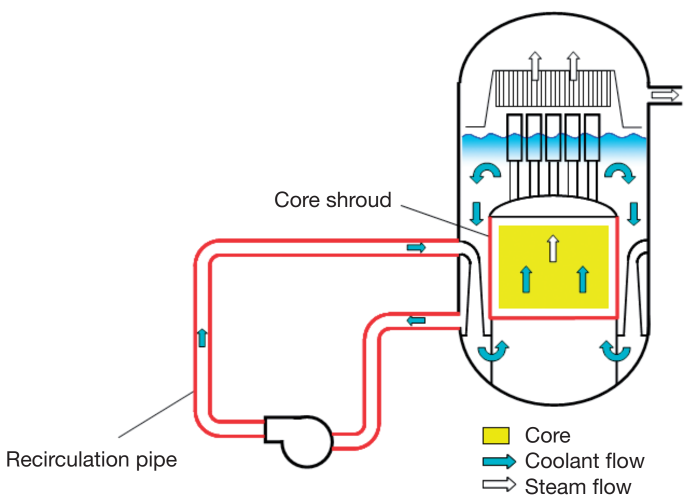


FIG. 12. BWR core shroud and recirculation pipe.

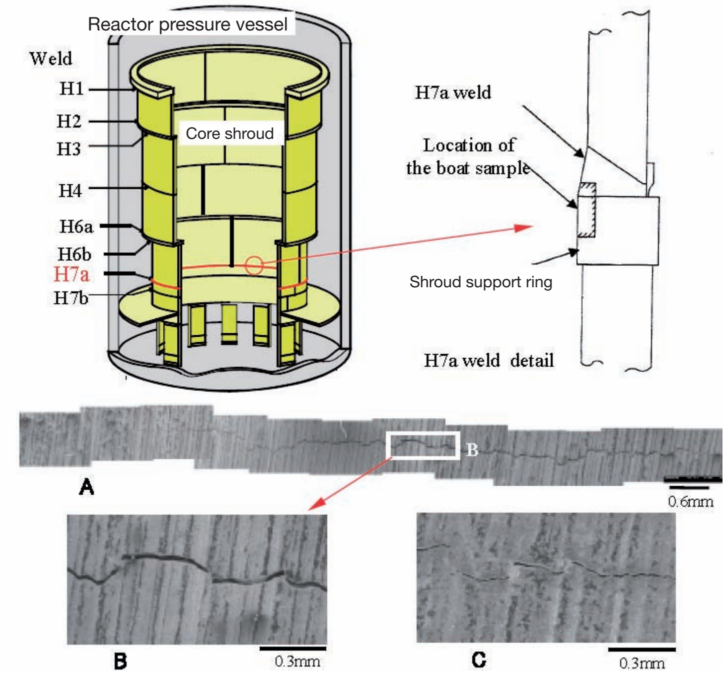


FIG. 13. IGSCC on the core shroud.

- — Fabrication, transportation, installation and construction: There was no or insufficient post-weld heat treatment to release residual stress.
- — Commissioning: None.
- — Maintenance: Overreliance on the nuclear grade stainless steel (replacement).
- — Operations: None.

5. IMPACT OF AGEING ON:

- — Safety: Increased risk of LOCA and core damage;
- — Reliability: Significant impact on component integrity and reliability;
- — Production: Unplanned shutdowns, long outages for repair and replacement;
- — Cost: Significant costs for repair, replacement and mitigation.

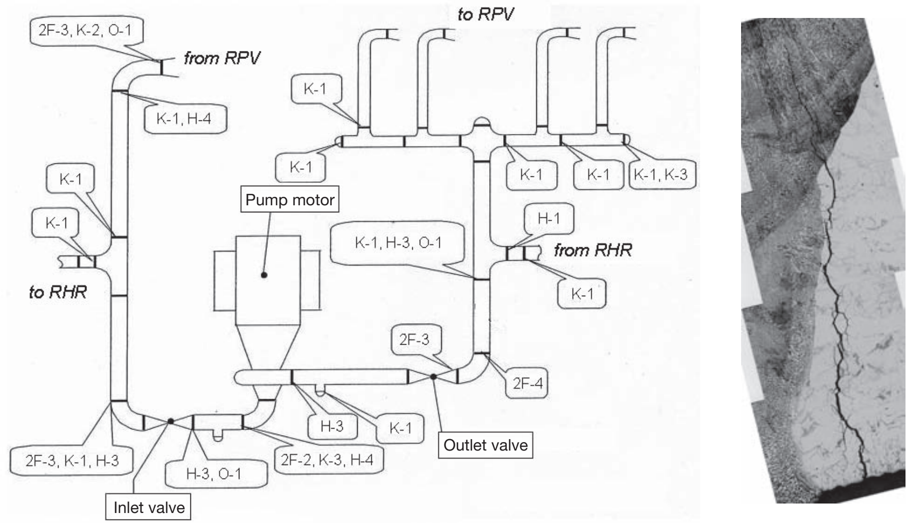


FIG. 14. IGSCC on the recirculation pipe welding.

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

— Replacement with nuclear grade stainless steel (original countermeasure);
— Crack removal by electric discharge machining (EDM) and post-EDM surface treatment by peening;
— Surface improvement using the following technologies:
  - Water jet or laser peening (Fig. 15);
  - Buttering (corrosion resistant cladding (CRC);
  - Removal of hardened surface layer;
— Introduction of welding technologies that improve residual stress (Fig. 16):
  - Induction heating stress improvement;
  - Heat sink welding;
— Enhanced inspection programme (100% of the recirculation piping welding every 5 years);
— Introduction of hydrogen water chemistry and noble metal chemical addition (some plants) (Fig. 17).

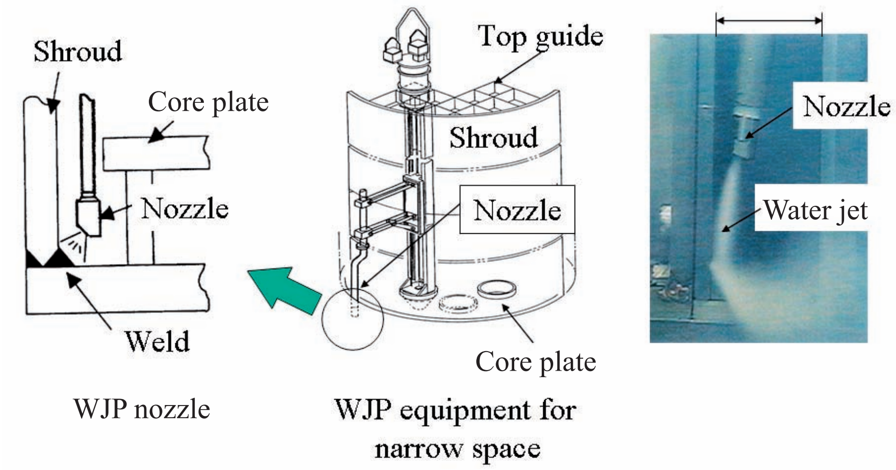


FIG. 15. Surface improvement using water jet peening (WJP).

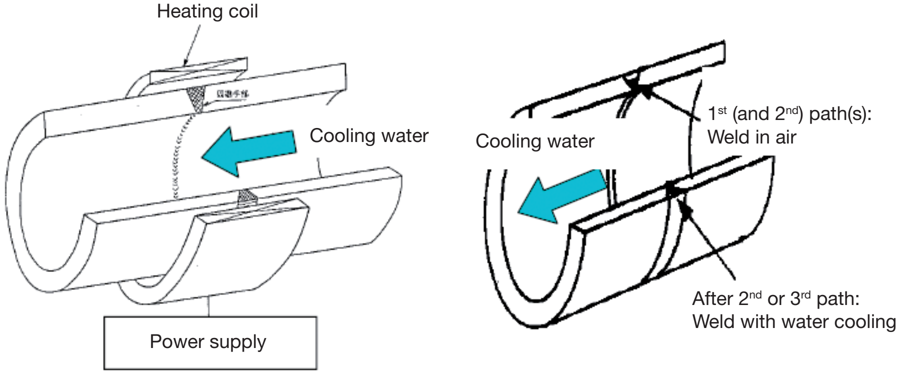


FIG. 16. Induction heating stress improvement (left) and heat sink welding (right).

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

Elimination of one of the three factors of SCC (material, stress and environment) was not sufficient. In particular, overreliance on nuclear grade stainless steel was not an appropriate measure. Comprehensive assessment and measures that cover all the factors are necessary.

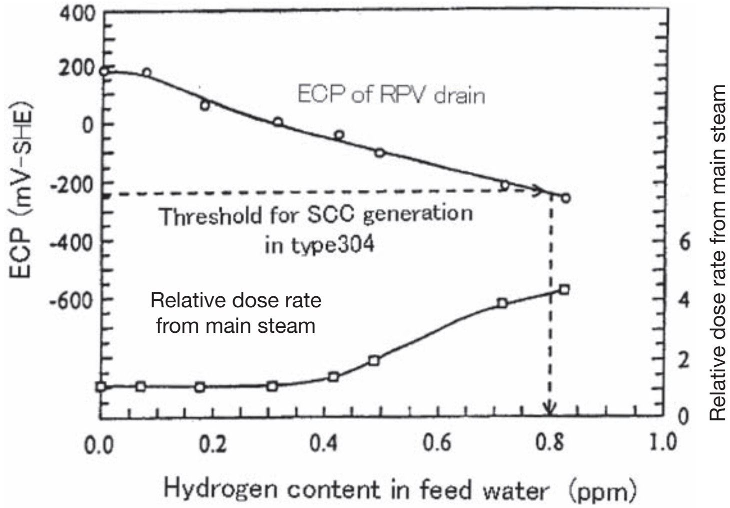


FIG. 17. Variation of eletrochemical corrosion potential (ECP) as a function of hydrogen content in feedwater.

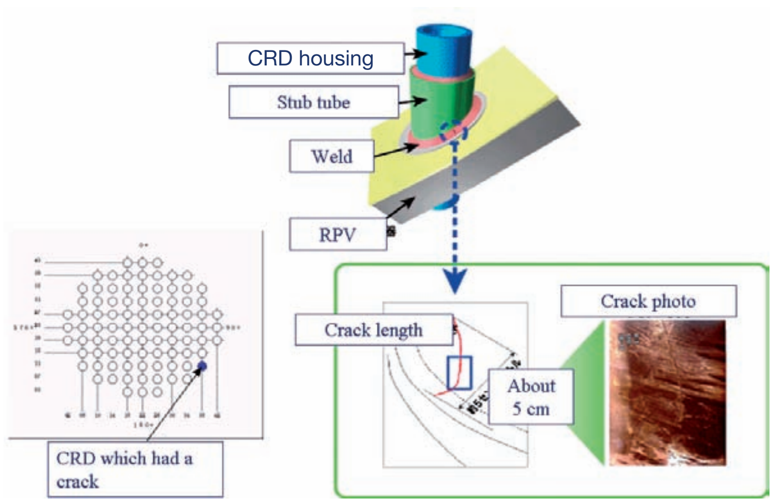


FIG. 18. IGSCC on the RPV penetration of the control rod drive (CRD) housing.

8. APPLICABILITY TO OTHER COMPONENTS

Potential IGSCC of core internals and some parts of the RPV made of nickel based alloy (especially Alloy 1821 welding material) (Fig. 18) is now being thoroughly investigated using an approach similar to that described here.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

Comprehensive and systematic ageing management programmes aimed at addressing IGSCC are now being established.

EXAMPLE 7

1. COMPONENT/SUBCOMPONENT

Carbon steel liner of containment

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

The boric acid spray was inadvertently activated during normal on-power testing. As a consequence, some of the water with boric acid was caught behind the carbon steel liner of the containment. Boric acid is highly corrosive, and any corrosion on the steel liner cannot be readily observed in the areas between the liner and the containment concrete wall.

3. HOW AGEING WAS DETECTED OR SUSPECTED

All accessible surfaces were cleaned; however, water was detected on the operating floor, indicating the presence of boric acid behind the liner, and some corrosion was assumed to have taken place.

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

— Design and analysis, procurement: The design had not adequately considered the requirement to prevent leakage of boric acid behind the liner in the event of a spurious spray. It also had not adequately considered requirements for access to this space in the case of spurious sprays.
— Fabrication, transportation, installation and construction: None.
— Commissioning: None.
— Maintenance: None.
— Operations: Operator error precipitated this accelerated ageing phenomenon.

5. IMPACT OF AGEING ON:

— Safety: The carbon steel liner is part of the pressure boundary of containment.
— Reliability: None.
— Production: The repair requires a long outage.
— Cost: Repair of the liner due to corrosion can be very expensive.

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

— Changes to operator test procedures and to control panel design;
— Samples of the liner taken to establish the degree of corrosion;
— Installation of an elastomeric seal at the top of the liner to prevent any additional leakage into the cavity.

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

— Provide a protective barrier to prevent ingress of boric acid into cavities that are difficult to access.
— Consider access/maintainability requirements during the design phase.
— Monitor the liner for corrosion due to boric acid.
— Inspect the seal for deterioration and replace it as required.

8. APPLICABILITY TO OTHER COMPONENTS

This is applicable to all components that are subject to normal or accelerated ageing and are inaccessible.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

This example demonstrates the inherent risks of accelerated corrosion in those parts of the plant that are not visible and are not easily monitored or inspected.

EXAMPLE 8

1. COMPONENT/SUBCOMPONENT

Electrical generator stator bars

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

Corrosion products caused generator cooling system blockages, resulting in increased temperatures. High temperatures could lead to generator trips and ultimately to the need to rewind the stator.

3. HOW AGEING WAS DETECTED OR SUSPECTED

The problem was detected by temperature monitoring (it should be noted that the water chemistry was within the specifications).

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

— Design and analysis, procurement: The root cause was determined to be an inadequate understanding of operating conditions and chemistry requirements for the cooling systems of an extended (new) design set of electrical generators.
— Fabrication, transportation, installation and construction: None.
— Commissioning: None.
— Maintenance: None.
— Operations: None; the generator was operated within manufacturer specifications.

5. IMPACT OF AGEING ON:

— Safety: None.
— Reliability: Forced outages and derating due to high generator temperatures.
— Production: Loss of production during outages.
— Cost: The costs to rewind the stator were very high.

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

Modify chemistry control of the cooling water.

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

— Design of equipment should consider the impact of long term operation.
— Chemistry is to be monitored in line with the new specification.

8. APPLICABILITY TO OTHER COMPONENTS

All intermediate cooling systems should be reviewed for sources of leakage and contamination which affect component ageing and performance due to unexpected or inappropriate chemical or thermal activities.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

Input to solving ageing problems needs to be managed to find optimum solutions. In this case, rewinding the stators would be a very expensive solution. A solution involving the water chemistry was more effective and inexpensive.

EXAMPLE 9

1. COMPONENT/SUBCOMPONENT

Reactor pressure vessel (WWER)

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

Radiation embrittlement.

3. HOW AGEING WAS DETECTED OR SUSPECTED

The first surveillance test results showed significant embrittlement.

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

— Design and analysis, procurement: The RPV diameter is too small for the size of the core.
— Fabrication, transportation, installation and construction: Impure critical core weld.
— Commissioning: None.
— Maintenance: None.
— Operations: None.

5. IMPACT OF AGEING ON:

— Safety: Increased risk of brittle fracture.
— Reliability: Significant impact on component integrity and reliability.
— Production: None.
— Cost: None.

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

— In 1980, 36 fuel bundles were replaced with dummy (steel) elements to reduce neutron flux at the edge of the core.
— Later in the 1980s, low leakage core refuelling was adopted (no new bundles are loaded at the core edge).
— The temperature of two low pressure emergency cooling accumulators was raised to 100ºC.

— The temperature of the high pressure emergency cooling tank was raised to 55°C. The head of the high pressure pump was decreased (one impeller was removed) to avoid potential opening of the pressurizer safety valves in some pressurized thermal shock (PTS) transients.
— The main steam line and feedwater line isolation criteria were modified to smooth the cooling transients caused by main steam line leaks.
— Low pressure safety valves were installed to avoid a cold pressurization of the RPV.
— Operating procedures were modified, and the operators were trained to take care of PTS transients.
— Improvements were made to volumetric testing methods.
— Improvements were made to fracture mechanical analyses and codes.
— Annealing was performed.
— A new surveillance test programme was introduced.

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

Materials science and welding technology are both extremely important. Understanding of radiation embrittlement, as well as of re-embrittlement, has increased significantly.

8. APPLICABILITY TO OTHER COMPONENTS

The degradation mechanism is unique, but materials science and welding technology have general importance.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

This case led to a comprehensive research programme to clarify the material properties of the RPV base and welding materials. It also had a large effect on developing fracture mechanical analysis methods, analysis of pressurized thermal shocks and qualification of in-service inspection methods.

EXAMPLE 10

1. COMPONENT/SUBCOMPONENT

Original feedwater manifolds (WWER)

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

Erosion–corrosion in carbon steel pipes and components (Fig. 19).

3. HOW WAS AGEING DETECTED OR SUSPECTED

Thickness measurements.

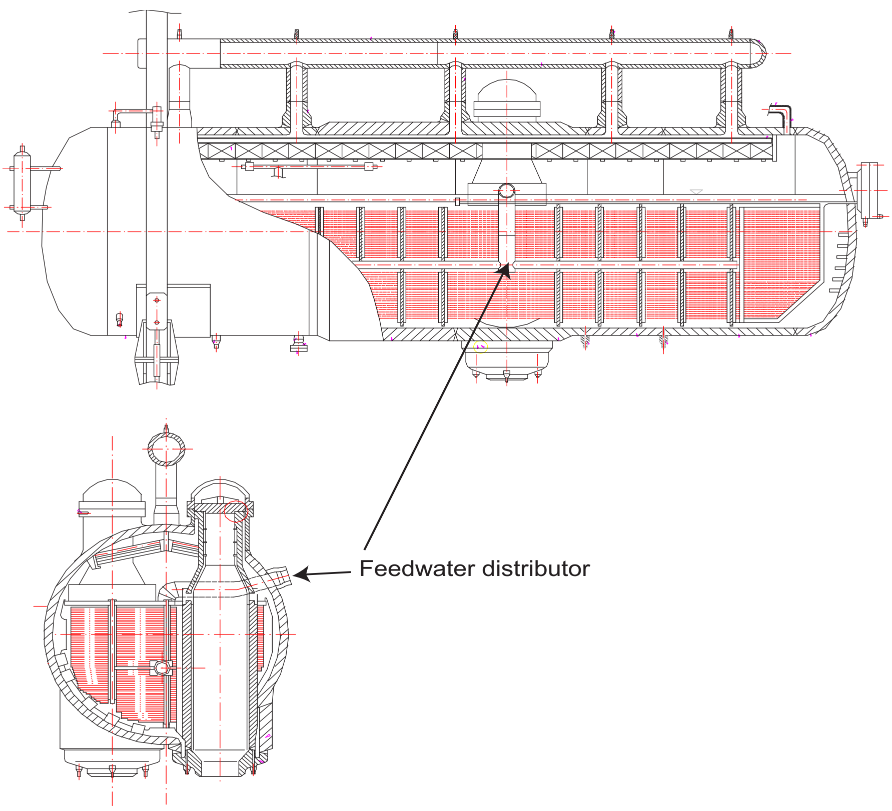


FIG. 19. The feedwater distributors of steam generators in WWER-440 nuclear power plants were reconstructed by installing them above the tube bundle.

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

- Design and analysis, procurement: Carbon steel proved to be the wrong material for use with neutral water chemistry.
- Fabrication, transportation, installation and construction: There are circumferential grooves near the welds that wake up the turbulence and erosion–corrosion.
- Commissioning: None.
- Maintenance: Power uprating has accelerated erosion–corrosion and focused it at different locations.
- Operations: Periodic blowdowns.

5. IMPACT OF AGEING ON:

- Safety: Pipe breaks may endanger the lives of personnel.
- Reliability: Unexpected shutdowns decrease reliability.
- Production: Production losses have occurred. The sensitivity of flow measurement flanges is decreasing owing to erosion–corrosion, and there has been an impact on plant efficiency.
- Cost: Numerous pipes must be replaced.

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

The water chemistry was changed from neutral to alkaline. The pipe material was changed from carbon steel to low alloy or austenitic steel. A more comprehensive inspection programme was introduced. A computer code was purchased for weak point analysis. A special group of experts was established to coordinate and provide advice on corrective actions. The case is currently being managed through the comprehensive inspection programme and material replacements.

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

Erosion–corrosion may be very local in nature. Inspections should be performed according to proper procedures and instructions.

8. APPLICABILITY TO OTHER COMPONENTS

This degradation mechanism is present almost everywhere in the secondary circuit in areas made of carbon steel.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

A special interdisciplinary group of experts was established to coordinate and provide advice on corrective actions. The inspection and pipe replacement programme and the results of the inspections are discussed regularly within the group and with plant personnel. The mechanism is now better understood.

EXAMPLE 11

1. COMPONENT/SUBCOMPONENT

Retaining rings on the rotor of a hydrogen cooled turbogenerator (Fig. 20)

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

— Stress corrosion cracking (SCC) of X55MnCr(N)18K (see Fig. 21);
— High stresses at shrink fit interfaces;
— Possibility of moisture during outages;
— Risk of rupture of the rings.

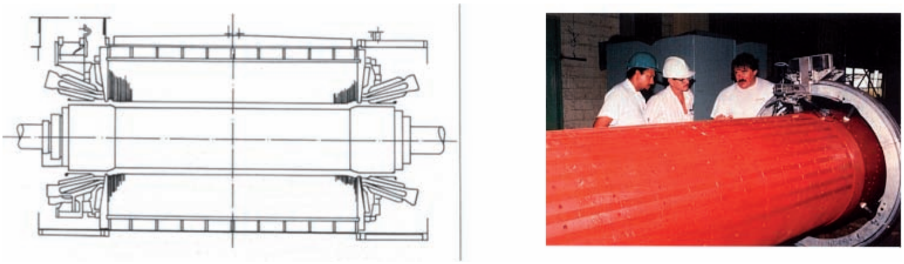


FIG. 20. Retaining rings on the rotor of a hydrogen cooled turbogenerator.

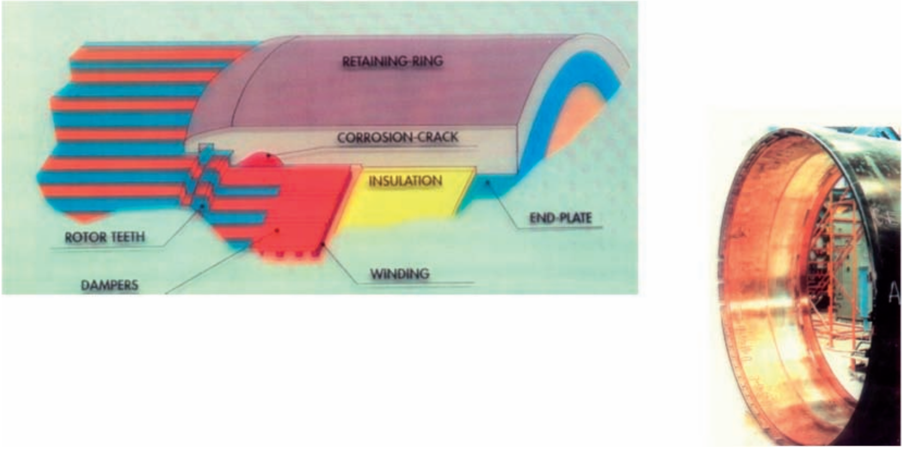


FIG. 21. SCC of retaining rings on the rotor of a hydrogen cooled turbogenerator.

3. HOW AGEING WAS DETECTED OR SUSPECTED

There were event reports (from the manufacturer) of observed cracks and rupture.

> **Note**: This material's sensitivity to SCC was known but was thought to be under control because the environment around the rings was assumed to be free of moisture.

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

- Design and analysis, procurement: High stresses as a result of shrink fit and rotation are incorporated into the design.
- Fabrication, transportation, installation and construction: Defects caused by manufacturing can increase the rate of crack growth.
- Commissioning: None.
- Maintenance: Indirect maintenance is performed; during the outage when maintenance is performed, the generator is out of operation. There is a risk of moisture condensation on the rings (conservation is essential).
- Operations: The rotation speed influences the rate of crack growth. Silica gel is used to dry the cooling gas; silica gel and dew point temperature are monitored.

5. IMPACT OF AGEING ON:

- Safety: Risk of injury to personnel in the case of a rupture.
- Reliability: Expected long outage in the case of a rupture.
- Production: Loss of production during forced outage in the case of a rupture.
- Cost: The cost of replacing the rings (not including production loss) is high (around $1 million).

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

- Joint and integrated approach.
- All relevant operating organizations participated in a multidisciplinary study performed by a consultancy company.

- All aspects were incorporated:
  - Manufacturing processes of the rings;
  - Materials research;
  - Operating experience;
  - Non-destructive testing (NDT), experience and techniques;
  - Ring specific stress and crack growth analyses;
  - Maintenance experience;
  - Monitoring.
- Programme to manage SCC of rings without the need to replace the rings.
- Focused inspection plan and interval.
- Appropriate technique for NDT defects.
- Measures to avoid moisture.
- Special device to perform NDT without having to remove the rotor from the generator.

Currently there are no plans to replace the rings. The first inspection after the study revealed a defect, but a defect assessment concluded that the defect was not a problem. Additional assurance was provided by temporarily shortening the NDT inspection interval. Since confirmation, the plant has been following the original programme.

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

- Good ageing management can help to reduce costs.
- The benefits of an integrated approach were evident.

8. APPLICABILITY TO OTHER COMPONENTS

The approach used is applicable to all types of ageing phenomenon, particularly SCC problems.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

As a result of this case, understanding was improved and better methods were developed.

EXAMPLE 12

1. COMPONENT/SUBCOMPONENT

Safety Class 1E cable

2. BRIEF DESCRIPTION OF AGEING ISSUE OR CONCERN

This example involves the unanticipated failure of a buried, 23 year old safety related power cable (4160 V safety related Class 1E, ethylene propylene rubber insulation with a neoprene jacket). The buried cable was in a mild environment supplying power to component cooling water (CCW) pump motors. Similar types of cable are also used for the service water system. Moisture intrusion over a long period of time resulted in corrosion of the copper conductor and insulation embrittlement.

3. HOW AGEING WAS DETECTED OR SUSPECTED

There was an electrical line to ground fault that affected the functionality of the cable system.

4. CONTRIBUTING CAUSES OF THE AGEING ISSUE

Over time, water with impurities penetrated the cable conduit throughout the thickness of the cable jacket and main conductor insulation, disturbed the normal electric field distribution and resulted in localized stress enhancement and electrical failure.

- Design and analysis, procurement: The basic design was sound for the 4160 V application. This was evident from the fact that similar cables in similar applications at the same plant were not damaged over their 23 year operating history.
- Fabrication, transportation, installation and construction: As stated, there was no design defect, nor was any deficiency noticed in the fabrication process. However, there could have been a deficiency in the overall installation of the cable in the buried conduit.
- Commissioning: Since two other identical cables did not experience an anomaly, improper commissioning was ruled out in the failure analysis.

— Maintenance: In general, no routine maintenance is carried out on buried cables. System functionality tests are routinely conducted to ensure system availability. These types of ‘go–no go’ test may not be adequate for detecting defects in an incipient state prior to failure.
— Operations: No anomalies were observed or detected during normal operations prior to the electrical failure.

5. IMPACT OF AGEING ON:

— Safety: There was no adverse impact on overall plant safety. There was no evidence of common mode types of anomaly/failure, as two other identical cable systems were not affected.
— Reliability: The overall reliability of the plant’s CCW system remained high because of the availability of redundant systems.
— Production: Production of electricity was not affected.
— Cost: There were costs associated with (i) removal, (ii) laboratory testing and analysis, (iii) field testing and (iv) component replacement.

6. BRIEF DESCRIPTION OF CORRECTIVE ACTIONS

The root cause of moisture intrusion should be determined. In addition to the routine functional tests, diagnostic tests should be performed to detect anomalies prior to failure. Currently, the CCW cable system involved is performing its required function. An ageing management programme to address water intrusion is in place because of diagnostics and condition monitoring.

7. LESSONS LEARNED, INCLUDING LONG TERM AGEING MANAGEMENT ACTIONS AND CURRENT PERFORMANCE

Buried cables are susceptible to moisture intrusion, which could degrade the integrity of insulating materials over a long period of time. Periodic inspection and condition monitoring are suggested to ensure that water does not migrate into the pipes housing the cables.

8. APPLICABILITY TO OTHER COMPONENTS

Many buried SSCs including pipes, tanks, reinforced concrete and metal structures are susceptible to corrosion, which could affect their long term properties and performance.

9. EFFECT ON UNDERSTANDING AND COMMUNICATIONS, AND ON DEVELOPMENT OF AN AGEING MANAGEMENT CULTURE AT THE PLANT

This case led to better understanding and communications, and contributed to the development of an ageing management culture at the plant, especially with respect to buried cables.

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Annex I

IAEA PUBLICATIONS ON AGEING MANAGEMENT

The present report has been developed within the framework of the IAEA programme on safety aspects of nuclear power plant ageing. The programme and its main products are summarized in this annex.

In 1985, the IAEA initiated information exchange on the safety aspects of nuclear power plant ageing to increase awareness of the emerging safety issue relating to physical ageing of plant systems, structures and components (SSCs). In 1989, a systematic project was begun, aimed at assisting Member States in understanding ageing of SSCs important to safety and in effective ageing management of these SSCs in order to ensure their integrity and functional capability throughout their service life. This project integrates information on the evaluation and management of safety aspects of nuclear power plant ageing generated by Member States into a common knowledge base, derives guidance, and assists Member States in the application of this guidance. The main publications of the project [I–1 to I–15] fall into four groups.

Awareness: Following the first International Symposium on Safety Aspects of Ageing and Maintenance of Nuclear Power Plants [I–2], which was organized by the IAEA in 1987, increased awareness of physical ageing of SSCs and its potential safety impact was achieved by the development and wide dissemination in 1990 of an IAEA-TECDOC on Safety Aspects of Nuclear Power Plant Ageing [I–3]. In the 1980s, most people believed that classical maintenance programmes were adequate for dealing with the ageing of nuclear plants. However, in the 1990s the need for ageing and lifetime management of nuclear power plants became widely recognized.

Ageing management programmes: The following reports have been developed using the experience of Member States:

— Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing [I–16] provides information on the baseline, operating and maintenance data necessary and a system for data collection and record keeping.
— Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety [I–4] provides guidance on screening (selecting) SSCs to make effective use of limited resources and on performing ageing management studies to identify or develop effective ageing management actions for the selected components.

— Implementation and Review of Nuclear Power Plant Ageing Management Programmes [I–5] provides information on the systematic approach to managing ageing and an organizational model for its implementation.
— Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving and Reviewing [I–6] documents current methods and practices relating to upgrading and preserving equipment qualification in operational nuclear power plants and reviewing the effectiveness of plant equipment qualification programmes.

Component specific publications: The guidance in Ref. [I–4] has been used to implement coordinated research projects on the management of ageing of concrete containment buildings and in-containment instrumentation and control cables, and to develop comprehensive technical documents on Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety for the following components:

— Steam generators [I–7];
— Concrete containment buildings [I–8];
— CANDU pressure tubes [I–9];
— Metal components of BWR containment systems [I–10] ;
— In-containment instrumentation and control cables [I–11];
— CANDU reactor assemblies [I–12];
— PWR primary piping [I–13];
— BWR pressure vessels [I–14];
— BWR pressure vessel internals [I–15];
— PWR pressure vessels [I–17];
— PWR vessel internals [I–18].

Ageing management review guidelines: AMAT Guidelines [I–19] is a reference publication for IAEA Ageing Management Assessment Teams (AMATs) and for utility self-assessments; these reviews can be programmatic or problem oriented.

The focus of the project work has progressively shifted from developing awareness to preparing programme related publications and then to providing component specific guidelines. In the future, the focus will be on providing services to assist Member States in the application of the guidelines. Facilitation of information exchange will continue through the preparation of additional guidelines and the updating of existing guidelines.

Guidelines on Safe Long Term Operation

The Safety Report on Safe Long Term Operation of Nuclear Power Plants [I–20] provides information on key technical considerations and activities to ensure safe long term operation (LTO) of nuclear power plants in accordance with regulatory requirements. This Safety Report takes into account approaches, practices and the experience of Member States pursuing LTO. Specifically, it provides information on a step by step approach that may be taken by operating organizations and regulatory bodies considering LTO.

Recently, the IAEA launched a peer review service on safe long term operation (SALTO), which is a new comprehensive engineering safety review service that directly addresses strategy and the key elements for safe LTO of nuclear power plants, and which includes AMAT objectives and complements OSART reviews. The SALTO review guidelines [I–21] provide a basic structure and common reference across the various areas covered by SALTO peer review missions.

Guidelines on plant life management

Useful information on ageing management of instrumentation and control equipment in nuclear power plants [I–22] was prepared under the IAEA project on nuclear power plant control and instrumentation.

The IAEA Technical Report on Plant Life Management for Long Term Operation of Nuclear Power Plants [I–23] describes plant life management from the technological, regulatory, economic and human standpoints, as well as research requirements and international good practices.

The TECDOC on Nuclear Power Plant Life Management Processes: Guidelines and Practices for Heavy Water Reactors [I–24] deals with organizational and managerial means to implement effective plant life management for existing operating HWRs. The guidance provided is applicable also to future HWRs.

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Annex II

AGEING MANAGEMENT FOR RPV EMBRITTLEMENT

Since the introduction of RPV radiation embrittlement management, the existing programmes have been improved and new methods have been introduced. Knowledge concerning the mechanism has increased, and better prediction formulas have been developed. Instead of indirect but conservative determination of RT_{NDT} by testing Charpy-V specimens, it is now possible and acceptable to directly determine fracture toughness on irradiated small specimens (three-point bending). This allows for a less conservative but more realistic determination of the toughness of the RPV. A new methodology for the direct measurement of embrittlement from samples taken directly from the irradiated vessel wall is under development.

In addition to material properties, detailed analyses were conducted to clarify the various loadings and transients, as well as the sensitivity of in-service inspection methods. Plenty of measures were adopted to mitigate transients in order to improve the reliability of in-service inspections and fracture mechanical analyses.

Based on a conservative determination of the ‘brittle to ductile fracture temperature’ (RTNDT), safety assessments were made in which the sufficient ductility of the RPV for the design life could be proved. A very important step was the implementation of RPV surveillance programmes for the specific RPVs. During fabrication of the RPV, extra material (from the same charge) of the highly irradiated beltline zone was produced, to have appropriate specimens for surveillance. Capsules with these specimens (Charpy-V, etc.) and fluence detectors were put in the RPV near the vessel wall at the height of the reactor core so that the specimens received a high dose of neutron fluence. Capsules were taken out after being irradiated for a limited number of cycles. Depending on the position of the specimens relative to the wall, evidence of a high fluence after only a few refuelling cycles could be obtained. By keeping specimens in the RPV long enough, a fluence level comparable with that at design life was reached. By testing the irradiated specimens for several fluence levels, the change in RTNDT and the actual RTNDT at design life could be determined with enough conservatism.

The proactive ageing management strategy for RPV irradiation embrittlement has been in place since the outset of nuclear power. The process is not perfect: for example, material monitoring is not continuous, and there is always a need to interpolate material properties to that part of the vessel where they are to be estimated (generally one quarter wall thickness of beltline welds). Also, in some cases no extra beltline material was available for surveillance, or the surveillance programmes did not represent the condition of the vessel wall. Nevertheless, a long term proactive programme remains in place and is based on early recognition of a major safety related ageing issue and timely development of the necessary technology.

The IAEA has published information on this topic [II–1 to II–5].

REFERENCES TO ANNEX II

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Of the nuclear power plants currently operating around the world, approximately 25% have been in operation for more than 30 years, and about 70% have been in operation for more than 20 years. Effective management of ageing of systems, structures and components (SSCs) is thus becoming increasingly important. This Safety Report provides information on good practices for maintaining and improving the safety and performance of nuclear power plants by facilitating proactive ageing management of SSCs throughout their lifetime.

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