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SAFETY ASPECTS OF NUCLEAR POWER PLANT AGEING

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FOREWORD

The nuclear community is facing new challenges as commerical nuclear power plants (NPPs) of the first generation get older. At present, some of the plants are approaching or have even exceeded the end of their nominal design life. Experience with fossil fired power plants and in other industries shows that reliability of NPP components, and consequently general plant safety and reliability, may decline in the middle and later years of plant life. Thus, the task of maintaining operational safety and reliability during the entire plant life and especially, in its later years, is of growing importance.

Recognizing the potential impact of ageing on plant safety, the IAEA convened a Working Group in 1985 to draft a report to stimulate relevant activities in the Member States. This report provided the basis for the preparation of the present document, which included a review in 1986 by a Technical Committee and the incorporation of relevant results presented at the 1987 IAEA Symposium on the Safety Aspects of the Ageing and Maintenance of NPPs and in available literature.

The purpose of the present document is to increase awareness and understanding of the potential impact of ageing on plant safety; of ageing processes; and of the approach and actions needed to manage the ageing of NPP components effectively.

Despite of the continuing growth in knowledge on the subject during the preparation of this report it nevertheless contains much that will be of interest to a wide technical and managerial audience. Furthermore, more specific technical publications on the evaluation and management of NPP ageing and service life are being developed under the Agency's programme, which is based on the recommendations of its 1988 Advisory Group on NPP ageing.

A special acknowledgement is due to Messrs. J. Pachner of Canada and J.P. Vora of the United States for their contribution and interest in the production of the document.

EDITORIAL NOTE

In preparing this material for the press, staff of the International Atomic Energy Agency have mounted and paginated the original manuscripts and given some attention to presentation.

The views expressed do not necessarily reflect those of the governments of the Member States or organizations under whose auspices the manuscripts were produced.

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1. INTRODUCTION

Experience with large fossil fuel fired generating units and in process industries shows that plant availability begins to decline owing to age related processes after about ten years of operation. It is reasonable to postulate, and it has been found during the operation of nuclear power plants (NPPs), that ageing phenomena will occur for nuclear units that not only affect availability but that potentially also affect the safety of NPPs if appropriate measures are not taken.

It is evident from Fig.1.1 that the number of older power reactors in IAEA Member States is increasing. The age related deterioration occurring as plant operation continues must be properly managed to ensure that the safety of the plants is maintained.

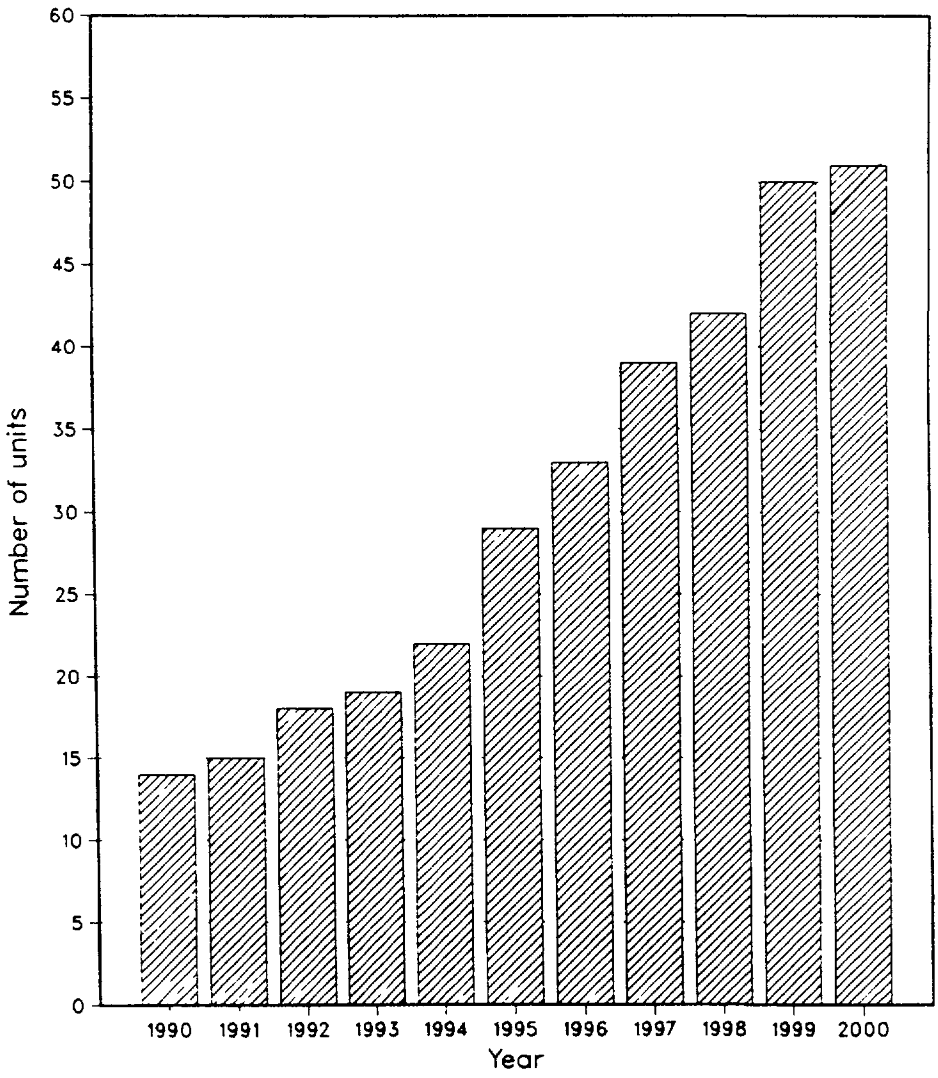

Fig.1.1. Power reactors of 30-40 years age (Source: IAEA PRIS).

NPP operating organizations have been using different programmes or methods to cope with potential or actual failures. Design provisions, surveillance, preventive maintenance programmes, periodic review of NPP performance, including reliability based assessment of the performance of safety related systems, and feedback of operating experience are all methods used to prevent, detect, remedy and mitigate the effects of failures of NPP systems and components due to any cause, including the effects of ageing.[1/] These methods have in general been quite effective in dealing with ageing effects encountered up until now.

However, because of the increasing average age of plants, it is prudent to assess the current provisions for dealings with the ageing deterioration of NPP components and to determine whether and what enhancements should be made to ensure the continued safe and economical operation of NPPs. Owing to the complexity of ageing phenomena, a systematic and proactive approach should be taken.

The general safety concern related to NPP ageing is that plant safety could be impaired if degradation of key components and structures is not detected before loss of functional capability and timely corrective action is not taken. In particular, it is possible that degradation may not be revealed during routine operation and testing, but may lead to failure or even multiple failures of redundant components under upset or accident conditions. Thus, if not controlled, ageing degradation could erode the safety margins afforded by defence-in-depth and therefore increase risk to public health and safety.

During the past few years, various organizations, including utilities, regulatory bodies and other institutions concerned with or responsible for the operation of NPPs, have worked on significant ageing-related issued (for example pressure vessel embrittlement, stress corrosion cracking of reactor coolant piping, degradation of electrical insulation materials) and some have initiated systematic and proactive ageing related programmes.

The IAEA, in response to Member States requests, commenced activities concerned with the safety aspects of NPP ageing four years ago. A working group was convened, which at its first meeting held in 1985 discussed general topics related to ageing, and agreed that more attention should be devoted to NPP ageing issues by all Member States. To stimulate and facilitate the enhancement or establishment of relevant activities in the Member States, the group decided that pertinent available information should be collected and published in the form of an IAEA TECDOC (Technical Document). The first draft of the TECDOC was prepared by the working group at this meeting, and was finalized at its second meeting held in autumn of 1987 using recommendations of the Technical Committee, additional information collected at is meeting of September 1986, as well as the papers presented at the 1987 IAEA International Symposium on Safety Aspects of NPP Ageing and Maintenance. The document is directed at a wide technical and management audience engaged in the design, construction, operation, maintenance and regulation of nuclear power plants. Its goal is to promote the understanding of the potential impact of ageing on NPP safety, of the ageing processes, and the approach and actions needed to effectively manage the ageing of NPP components.

1/ The term components is used in a broad sense and includes plant structures and systems.

Section 2 of the document gives the definition and explanation of the ageing, Section 3 deals with the relationship of ageing to NPP safety, which is further expanded in Section 4 and component qualification approach is discussed. Methods for determining components susceptible to ageing degradation whose failures could have a significant adverse effect on plant safety are outlined in Section 5. Section 6 is the most extensive one although it deals only with the fundamental aspects of ageing of materials used in NPPs. General principles of the methods for timely detection of ageing effects are discussed in Section 7, and finally activities of some Member States aimed at managing the impact of ageing on plant safety are presented in Section 8.

Since ageing affects all kinds of plant components and materials there is a very large body of relevant literature. As a result the scope of the document had to be limited to general safety aspect of NPP ageing. Although much of the presented information is relevant to the issues of plant life assurance and extension, these issues dealing with the major NPP components are not specifically addressed in this document.

The document is based primarily on US literature as can be seen from the list of references. This reflects the composition of the source documents which were available to the working group. Follow-up reports dealing in greater detail with managing of ageing of different materials and equipment are planned, and it is hoped that a better balance of the source documents and the resulting reports will be achieved.

The Secretariat invites comments and additional information on progress in the management of ageing in NPPs. Comments should be addressed to the Department of Nuclear Energy and Safety, International Atomic Energy Agency, P.O. Box 100, Wagramerstrasse 5, A-1400 Vienna, Austria.

2. DEFINITION OF AGEING

Ageing is defined as the continuous time dependent degradation of materials due to normal service conditions, which include normal operation and transient conditions; postulated accident and post-accident conditions are excluded [1, 2, 3].

All materials in a nuclear power plant can undergo ageing and can lose, partially or totally, their designed function, (see Fig. 2.1). If not managed, ageing is of concern not only for active components, for which the probability of malfunction increases with time, but also for passive components, since their safety margin is thereby reduced towards the lowest permissible level.

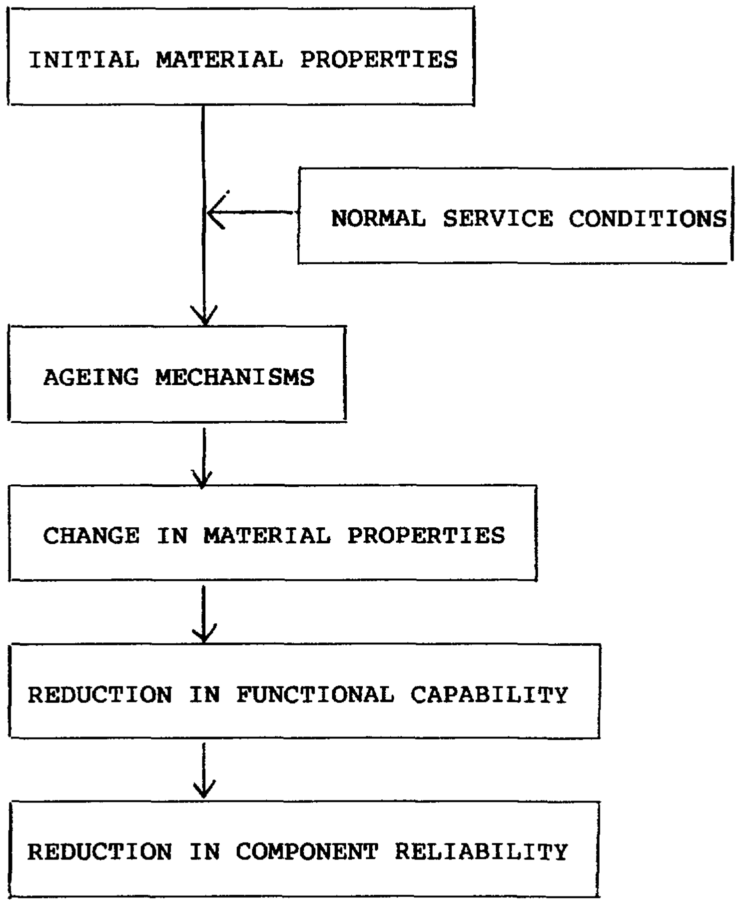

Fig.2.1. Impact of service conditions on material degradation and component reliability.

The service conditions which contribute to ageing act mainly in two different ways.

(1) Through chemical and physical processes that affect material properties, which are caused by:
- stress and/or strain
- temperature
- environmental factors such as radiation, high humidity, or the presence of chemically active liquids or gases (before or during operation).

(2) Through factors that can lead to the degradation of functional capability, which are caused by:
- service wear and corrosion, including changes in the dimensions and/or the relative position of individual parts or assemblies
- excessive testing
- improper installation or maintenance.

The main ageing effects of concern are:
- changes in physical properties (e.g. electrical conductivity)
- irradiation embrittlement
- thermal embrittlement
- creep
- fatigue
- corrosion, including corrosion erosion and corrosion assisted cracking
- wear (e.g. fretting) and wear assisted cracking (e.g. fretting fatigue).

The term 'ageing' then represents the cumulative changes over time that may occur within a component or structure owing to one or more of these factors. From this perspective, it is clear that ageing is a complex process that begins as soon as a component or structure is produced and continues throughout its service life. Ageing is certainly a significant factor in determining the limits of nuclear plant lifetime or life extensions. No nuclear plant, including those still under construction or being mothballed, should be considered immune from its effects.

The rate of ageing depends strongly on both the service conditions and the sensitivity of materials to those conditions. Therefore consideration must be given to the ageing of a component or of equipment in the design phase by selecting adequate materials capable of withstanding operating stressors and environment and must be continued throughout its complete life cycle.

3. AGEING AND ITS RELATIONSHIP TO THE SAFETY OF NUCLEAR POWER PLANTS

As discussed in Section 2, with time, changes gradually occur in the properties of materials. These changes can affect the capability of engineered components, systems or structures to perform their required functions. Not all changes are deleterious, but it is a commonly observed experience that the so-called ageing processes normally involve a gradual reduction in performance.

The safety of nuclear power plants (NPPs) could be affected by the age related degradation of key components or structures if it is not detected prior to loss of functional capability and if timely corrective action is not taken. The loss or even a reduction of functional capability of the key plant components could cause the impairment of one or more of the multiple levels of protection afforded by defence in depth, and in this way reduce the plant safety.

This section briefly describes the defence in depth concept and the ways in which the ageing degradation may reduce its actual effectiveness.

3.1. Effects of ageing on the integrity of defence in depth

Defence in depth is the fundamental safety principle underlying the safety technology of nuclear power. All safety activities, whether organizational, behavioural or hardware related are subject to layers of overlapping provisions, so that if a failure should occur, it would be compensated for or corrected without causing harm to NPP staff or the general public. This idea of multiple levels of protection is the central feature of defence in depth.

If an undetected erosion in the defence in depth occurs during normal operation due to ageing, it may lead eventually to a serious reduction in design safety margins or in the effectiveness of the installed safety systems.

The defence in depth concept provides an overall strategy for safety measures and features of nuclear power plants. When properly applied, it ensures that no single human or mechanical failure would lead to injury to the public, and even combinations of failures that are only remotely possible would lead to little or no injury. The principle of defence in depth is implemented primarily by means of series of barriers which should in principle never be jeopardized, and which must be violated in turn before harm can occur to people or the environment. These barriers (fuel matrix, fuel cladding, primary coolant circuit boundary and containment structure) are physical, providing for the confinement of radioactive material at successive locations. The barriers may serve both operational and safety purposes, or may serve safety purposes only. Power operation is only allowed if this multi-barrier system is not jeopardized and is capable of functioning as designed.

The reliability of the physical barriers is enhanced by applying the concept of defence in depth to them in turn, protecting each of them by a series of measures. Each physical barrier is designed conservatively, its quality is checked to ensure that the margins against failure are retained, its status is monitored, and all plant processes capable of affecting it are controlled and monitored in operation. Design provisions for normal operating and safety systems help to ensure that the three basic safety functions (controlling the reactor power, cooling the fuel and confining the radioactive material) are preserved to prevent undue challenges to the integrity of the physical barriers, to prevent the failure of a barrier if it is jeopardized, and to prevent consequential damage of multiple barriers in series.

All of the elements of defence in depth must be available at all times that a plant is at normal power. Appropriate levels of defence should be available at other times. The existence of several elements of defence in depth is never justification for continued operation in the absence of one component. Severe accidents in the past have been the result of multiple failures, both human and equipment failures, due to deficiencies in several elements of defence in depth that should not have been permitted.

Competent implementation of the defence in depth strategy is indicated by a smooth and steady NPP operation with little or no need to call on safety systems.

As noted above, ageing degradation of NPP components may have an adverse impact on the effectiveness of the defence in depth. Various ageing processes occurring in NPPs gradually degrade characteristics of plant components, may reduce design safety margins and cause failures of both process and safety system components. In particular, it is possible that degradation may not be revealed during normal operation and testing, but may lead to failure or even multiple common cause failures of redundant components under transient conditions (e.g. high pressure, vibration, steam, electrical pulse) associated with an operational upset or accident. The following subsections describe in more detail the ways in which different elements of defence in depth may be affected by ageing.

3.2. The potential increase in the probability of component failures due to ageing

Experience shows that the probability of a component failure and therefore the component failure rate increases as the component ages or wears out. This increase in failure rate may be due to one or more ageing phenomena discussed in Section 2, and is represented by the wear-out region of the familiar 'bathtub' curve. The increase in failure rate can have a negative impact on defence in depth by increasing the probability and frequency of process system transients or failures and by increasing the probability of mitigating component failures (mitigating components include both the standby components, such as auxiliary feedwater pump or emergency diesel generator and the components of the safety systems, i.e., the emergency core cooling, shutdown and containment systems).

Probability of transient or accident initiators

The increased failure rate of certain components of normal operating systems can cause an increase in the probability and frequency of the system transients or failures jeopardizing the integrity of the physical barriers. At the same time, this increases the number of times the mitigating components and system must operate to ensure that the three essential safety functions (controlling the reactor power, cooling the fuel and confirming the radioactive material) and the physical barriers are preserved. Increased demand on the operation of mitigating systems presents more opportunities for a malfunction in their operation and thus increases the plant risk.

An example of a component whose ageing degradation could cause an increase in the probability of transients or accidents is a steam generator tube. Should steam generator tubes be allowed to degrade significantly, the likelihood of steam generator ruptures would increase. The ruptures could be caused by a large pressure transient in the primary system or by a severe earthquake. This might result in lifting of the secondary side relief valves and a loss of coolant accident with a direct pathway to atmosphere; a core damage could occur should the necessary safety systems fail to operate correctly. Also, active components, such as the main feedwater pumps, which are subject to ageing degradation, can increase the probability of transient or accident initiators.

Probability of failures of mitigating components

The increased failure rate of mitigating components increases the component unavailabilities and thus the probability of failure of the mitigating systems. These failures of the mitigating systems may occur when the systems are called upon to operate or during their operation required to mitigate the consequences of a key process system component failure, for example a break in the primary cooling system boundary. In this case, the emergency core cooling system would be called upon to provide the required core cooling, however, its operational readiness could be affected by ageing degradation of its components, e.g. coolant injection pumps.

3.3. Common cause failure

A particular concern relating to ageing degradation of NPP components is the potential wide-scale degradation of physical barriers and redundant components which leads to a higher probability of common cause failures. For example, age-degraded redundant components of safety systems could fail simultaneously when exposed to abnormal conditions associated with a transient or an accident.

The associated issue is how long it might take for ageing degradation to place equipment into a state when its performance under harsh environments resulting from postulated accidents can no longer be assured. To deal with this issue, equipment qualification programmes have been established. Their fundamental role is to provide reasonable assurance that the equipment can perform its safety function(s) during postulated service conditions without design, manufacture and age-related common cause failures. The effects of ageing must be addressed in the equipment qualification programme. This then leads to a programme of systematic replacement of the equipment prior to its reaching a degraded condition where material and functionality margins are eroded to such an extent that its survival of an accident cannot be reasonably assured. (Section 4 deals with the subject of common cause failure and equipment qualification in greater detail.)

3.4. Analysis of the effect of component ageing on system performance

Nuclear power plants are designed to accommodate the effects of single component failures. While such individual component failures have an adverse impact on plant reliability and safety, it is the multiple component failures which are of a significant safety concern. Since the performance of all plant components may be affected by ageing, there is a need to evaluate the effect that aged components have on system performance and plant safety.

Mathematical models of ageing processes and probabilistic safety assessment techniques can be used to determine how ageing is affecting component and system unavailability; to identify plant components and systems that have a propensity to ageing and are risk significant; to optimize testing, surveillance and maintenance activities with respect to managing ageing, and to provide a quantitative assessment of the effects of ageing on plant safety. Section 5.3 provides some details on the development of probabilistic safety assessment techniques applicable to ageing.

4. EQUIPMENT QUALIFICATION: DEFENCE AGAINST COMMON CAUSE FAILURES

4.1. Equipment qualification and types of common cause failures

In a nuclear power plant, the defence in depth strategy is implemented primarily through a series of physical barriers which should in principle never be jeopardized. Nuclear safety systems are designed to protect the barriers and prevent the loss of their integrity. Design features include redundancy, diversity and physical separation of parallel components, where appropriate, to enable the safety systems to withstand isolated component failures without the loss of general protective function. In addition, individual safety related components/equipment must meet their functional safety requirements throughout their installed life. Such safety requirements can be specified, in general, as maintaining:

- the integrity of passive components, and

- the performance of assigned function(s) of active components when they are operated within their design limits.

These requirements are generally ensured by a thorough programme of quality assurance, design, qualification, production, transportation, storage, installation, operation, periodic testing, surveillance, maintenance, and replacement of the components.

The safety analysis for nuclear power plants, in part, considers the station and its safety system design in terms of postulated service conditions, including events such as submersion, hydrogen burn, radiation plate-out, etc. Inherent in each such analysis are two aspects to be evaluated. First, designs must be such that components/equipment can actually perform designated safety functions in postulated service environments. Second, in-service ageing must not degrade NPP components/equipment to the extent that they cannot perform designated safety functions when required.

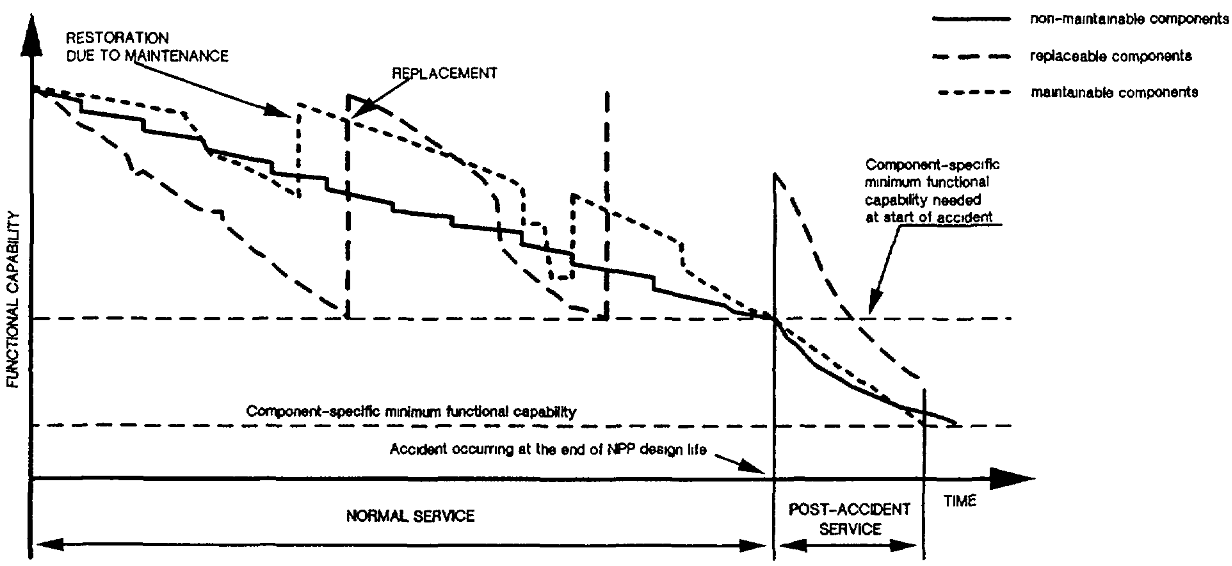

Fig.4.1. Qualitative picture of functional capability changes of different types of components during normal operation and following an accident occurring at the end of NPP design life.

The ageing degradation and other common causes of component/equipment failures can originate from many kinds of shared similarities. The commonality can occur at any point in the history of a device, from the original concept through the various stages of design, manufacturing, testing, installation, checking, and operation. A listing of main common cause types is given in Table 4.1. At each stage in the life of a component the fault peculiar to the stage can occur, be discovered and be rectified. The common fault of one stage may also be discovered in a later stage, but by then it is often difficult to rectify or eliminate.

Table 4.1 Common Cause Types of Equipment Failures

Type 1	Conceptual or engineering design error or inadequacy
Type 2	Manufacturing error, shortcoming or poor practice
Type 3	Testing or qualification error or omission
Type 4	Installation error or omission, or lack of validation of proper installation
Type 5	In-service ageing or deterioration due to operational stressors and environments and DBE and post-DBE conditions
Type 6	Operational misuse

Production testing, normal service testing and surveillance may not be able (or may not be designed) to determine whether the equipment is vulnerable to failure, as a result of either inadequate design, manufacture or ageing degradation. It is the ageing degradation, followed by exposure to environmental extremes of temperature, pressure, humidity, radiation, vibration, or chemical spay resulting from design basis events which presents a potential for common cause failures of safety related components/equipment. This concern is addressed by equipment qualification whose fundamental role is to provide reasonable assurance by means of documented evidence, with due recognition given to the established technology, that common cause failures related to component/equipment design, manufacture and age do not occur, and that the design and manufacture are adequate to permit the components/ equipment to perform their safety function(s) during postulated service conditions, throughout their installed lifetimes. During the selection and procurement of equipment, such evidence needs to be generated and documented: proper qualification should be initially achieved. Then during plant operation, qualification must be maintained through surveillance and maintenance of the equipment. The qualification documentation originally generated should be updated continuously during this stage.

4.2. Equipment qualification process and standards

The fundamental objective of equipment qualification is to uncover deficiencies that could have the potential for causing simultaneous failures of redundant components/equipment. This objective is achieved through the qualification process which consists of generating, documenting, and maintaining evidence that equipment can perform its safety functions whenever necessary during its installed life.

Potential ageing processes are identified, assessed and accounted for in the qualification process by establishing qualified life for equipment with significant ageing processes unless ageing is adequately addressed by in-service surveillance/maintenance. Well executed equipment qualification process thus provides a defence against age-related component/equipment failures.

An example of qualification process used in USA is illustrated in Fig. 4.2 and outlined below [91].

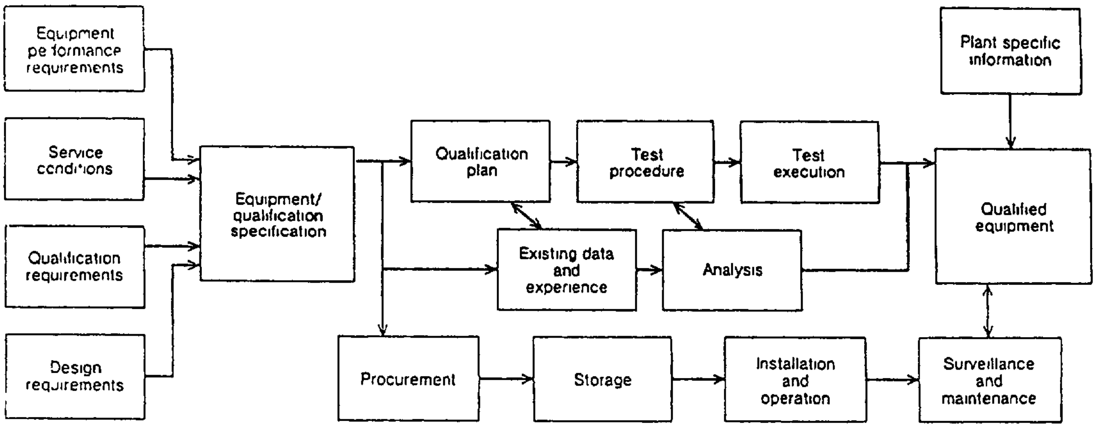

Fig.4.2. Equipment qualification process - block diagram.

Even before the equipment for a nuclear power plant is selected, the conditions of the service environment - and the safety functions that must be demonstrated by the qualification process - are established. The utility or its architect-engineer then develops a qualification specification for each type of equipment that is to be used. The qualification specification may state only the specific goals of the qualification process for the particular type of equipment, or it may also specify the method of qualification to be used (type testing, operating experience, analysis or combined methods).

During manufacture and delivery of equipment, the utility, its architect-engineer, the equipment manufacturer, and the qualification testing organization (if one is involved) develop a qualification plan for each type of equipment, using the requirements designated in the specification. Depending on the method of qualification chosen and incorporated in the qualification plan, one or more organizations may be involved in its implementation.

If the method chosen is analysis, the qualification plan is in most cases executed by the utility or its architect-engineer, but a testing organization may also be involved. If testing (either alone or in combination with analysis) is chosen, then a test plan and specification are developed. Specific test procedures are developed, and the testing is then performed. The results are compiled in a test report, which is used in documenting qualification of that type of equipment. However, if generic test data are used, they must be supplemented with plant-specific information to establish qualification. If analysis and/or operating experience is used in combination with testing, information from all these sources is used in documenting qualification.

In all cases, plant-specific qualification documentation must be generated.

When equipment is stored at the site and then installed, and when the plant is first started up, qualification may entail requirements for the storage, handling, installation, and inspection of equipment. When the equipment is operated in its final configuration in the plant, qualification affects surveillance; maintenance; repair; replacement; and procurement, storage, and handling of spare parts.

In developing and implementing equipment qualification program, the utilities and other involved organizations are guided by pertinent regulatory requirements and industrial standards. In general, the regulations pertaining to nuclear power plants require that utilities quality all safety related components/equipment to ensure safe operation under all anticipated service conditions, including normal operating and design basis accident conditions.

The main regulatory requirements and industrial standards adopted in USA and France for the qualification of NPP components/equipment are listed in Tables 4.2 and 4.3.

The Appendix contains a conceptual discussion of the relationship of the risk of common cause failure, the ageing degradation and design basis event stresses, and an outline of some issues relating to ageing simulation.

Table 4.2 Major US Standards for Equipment Qualification

	ENVIRONMENTAL	SEISMIC
Electrical/ Instrumentation	- 10 CFR 50.49 - NUREG-0800 (SRP 3.11) - IEEE 323/83 - RG 1.89	- NUREG-0800 (SRP 3.10) - IEEE 344/87 - RG 1.100
Mechanical	- NUREG-0800 (SRP 3.11) - IEEE-627/80 - ASME	- NUREG-0800 (SRP 3.9) - IEEE 627/80 - ASME
Structures		- NUREG-0800 (SRP 3.7, 3.8)

Table 4.3 French Standards Applied for Electrical Instrumentation, Mechanical Equipment and Structure Qualification

	ENVIRONMENTAL		SEISMIC	
	AGEING	ACCIDENT	AGEING	ACCIDENT
Electrical and Instrumentation Equipment	1 CAB 300MZ & 1121MZ RFS III IV.1a RFS IV.2.b RFS V.2.a RFS V.2.d Arrêté 10.08.84 RCC-E CEI 944	2 CAB 900MZ & 1121 M2 RFS IV.1.a RFS IV.2.b RFS V.1.a RFS V.2.d RFS V.2.f RCC-E CEI 944	3 CAB 900MZ & 1121 MZ RFS I.3.b RFS IV.1.a RFS V.2.b RFS V.2.d RCC-E UTE-C 20.420	4 See No. 3
Mechanical Equipment	5 CAB 900MZ & 1121MZ RPS IV.1.a RPS IV.2.a RFS V.2.a Arrêté 10.08.84 CEI 544 RCC-M	6 CAB 900MZ & 1121MZ RFS IV.1.a RFS IV.2.a RFS V.2.f. CEI 344 MFT 30 900 MFT 30 903 RCC-M	7 CAB 900MZ & 1121MZ RFS IV.1.a RFS IV.2.a UTE C20.420 RCC-M	8 See No. 7
Piping and Vessel	9 See No. 5	10 See No. 6	11 See No. 7	12 See No. 7
Concrete Structures	13 RCC-G 1985 BPEL 1983 BPEL 1983 Arrêté 10.08.84 AFNOR std	14 NV 65 FB 80 RFS I.2.a RFS I.2.d RFS I.2.e	15 See No. 13	16 RCC-G 1985 PS 69 RFS I.2.c RFS I.2.g

5. METHODS OF DETERMINING SAFETY SIGNIFICANT COMPONENTS AND SYSTEMS SUSCEPTIBLE TO AGEING DEGRADATION

Experience indicates that ageing mechanisms, which result in the reduction of functional capabilities of NPP components and systems, are operative to different degrees throughout nuclear power plants. Measures must be taken to detect ageing degradation of safety significant components and systems before their failure, and to mitigate it by appropriate maintenance and operational actions. Because of the large number of NPP components, the variety of their applications, the complexity of ageing processes, limited knowledge and limited resources, there is a need to concentrate the effort on the understanding and managing the safety impact of ageing on key ageing problems and components.

The purpose of this Section is to outline practical methods for investigating ageing problems, i.e. determining, on the basis of existing knowledge, the components and systems susceptible to ageing degradation whose failures could have a significant adverse effect on plant safety.

The methods discussed are:

- analysis of operating experience
- expert opinion
- probabilistic techniques for prioritization and for determining risk significance of ageing.

The methods are complementary and for the best results should be combined. The review of original design basis concerning materials, stressors and environment should be provided.

5.1. Analysis of operating experience

Analysis of operating experience data is a valuable method of identifying key components and systems susceptible to ageing degradation. Two such specific studies have been carried out in the USA [15, 51]. Similar reviews of operating experience (using significant event reports and reliability databases) are being conducted periodically in other Member States, for example France [53] and Canada [52], to identify, correct and mitigate system and component failures from any cause, including the effects of ageing degradation.

This section outlines the two US studies aimed specifically at the ageing problems.

5.1.1. Survey of LWR Operating Experience from LERs

To evaluate the effect of ageing degradation on the performance of LWR safety systems and components performance, a survey of licensee event reports (LERs) was conducted by Oak Ridge National Laboratory (ORNL); for details see Ref. [15].

Objectives

(a) To identify the extent to which the performance of LWR systems and components has been affected by ageing, and the ageing mechanisms responsible.

(b) To identify methods of failure detection and the severity of the failures.

Scope

The work included the review of 35000 LERs and LER predecessor abstracts covering a period from 1969 to 1982 which were obtained from the Nuclear Operations Analysis Centre (NOAC).

Data Source

An LER is a regulatory instrument generated by a licensee upon a deviation from the plant's Technical Specifications, and therefore generally only includes failures that affect safety related components or systems. It should be noted that the LER system is not an engineering data collection system and is not suitable for any definitive collection or analysis of statistical engineering data. LERs do not generally give detailed information about specific component ageing mechanisms, causes or required corrective actions. This limits the utility of the LER system for identifying ageing related failures.

Survey methodology

The NOAC maintains a keyworded file of LER abstracts. Keywords are assigned to identify the specific plant, reactor type, affected systems and components, and proximate cause(s) of an event, if the information is available in the LER.

For this study 16 keywords most likely to yield age related failure events were selected. This process was necessary to reduce the number of LERs to be examined in the first phase of the study. Table 5.1.1.1 gives the number of failure events obtained for each selected keyword. These keywords extracted a major part of the LER events that involved age related failures.

The review process involved the elimination of non-ageing effects, the consolidation of information from multiple abstracts concerning a single event and the discarding of events at other than commercial nuclear power plants. For each event judged to be an age related failure, the reviewer prepared an input record for entry into a data file established in the ORNL computer for the ageing study project. Data collected for each event included the system, component, subpart, the age related failure mechanism, the severity and the method of detection of the failure.

Survey results

The keyword search of the NOAC file yielded a total of 7256 LER abstracts. Of these, 2795 abstracts identified instrument drift as the cause of a reportable event. Events identified by the other fifteen keywords totalled 4461. Their detailed review resulted in 3098 events considered age-related.

Drift

Of the 7200 events extracted from the NOAC LER file for this study, 39% of the failures (2795) were identified as being due to drift. Drift was the reported cause of a system or component failure whenever a safety related device set point or calibration was found to be beyond the acceptance criteria delineated in the plant Technical Specifications. No apparent failure trend of instrumentation can be inferred from the data collected. The diversity of data indicates that further study is required to determine more accurately the statistics of true instrument drift (ageing) as opposed to the reported events

Table.5.1.1.1 NOAC Keywords Used in Ageing Study LER Searches

Keyword	Number of abstracts identified
Aged; effect, age	326
Corrosion	581
Stress corrosion	151
Vibration	616
Wear	533
Crud	1128
Erosion	115
Fatigue	28
Failure, fatigue	113
Oxidation	8
Friction	35
Hardening	7
Crack	411
Flow blockage	409
Sub-total	4461
Drift	2795
Total	7256

Note: The order of the keywords shown in Table 5.1.1.1 affects the number of event descriptions identified for each keyword. The numbers given are only the additional abstracts identified for each succeeding keyword because the computer keyword search process eliminates events previously selected under previously used keywords. For example, an event yielded by the keyword 'vibration' would not appear again under subsequent keywords. For this study, the keyword 'vibration' was reviewed as one of the 616 'vibration' accessions and not repeated as one of the 'fatigue' accessions.

due to administrative requirements (Technical Specifications) or device surveillance, testing or maintenance practices. The discussion in the following sub-sections addresses 3098 age related events other than drift.

Systems

The 3098 events mentioned were all associated with one of 68 system classifications. The most frequently reported systems are listed in Table 5.1.1.2; these ten systems account for 53.4% of the events.

The emergency core cooling system (ECCS), emergency diesel generator, containment isolation, chemical volume control system (CVCS) and liquid poison systems were responsible for 28% of the LERs listed as being caused by an age related failure. This result is not unexpected since plant Technical Specifications are predominantly concerned with maintaining the operability of these key systems. Consequently almost all failures in these areas are reportable; thus a large percentage of LERs originate within these five systems.

Table 5.1.1.2. Age Related LERs by System

System	Number of age related LERs	Percentage
ECCS and controls	227	7.3
Emergency generator system and controls	222	7.2
Containment isolation system and controls	215	6.9
CVCS and liquid poison system and controls	212	6.8
Station service water systems and controls	144	4.6
Coolant recirculation systems and controls	138	4.5
Containment heat removal systems and controls	134	4.3
Main steam supply system and controls (other than BWR steam supply)	130	4.2
Residual heat removal systems and controls	129	4.2
Reactor containment systems	104	3.4
Subtotal	1655	53.5
Other systems (56 systems)	1443	46.5
Total	3098	100.0

The next largest percentage of reported events occurred due to failure of the service water system. Such failures occurred most often at a component interface with one of the five systems and usually involved heat exchangers or control valves for heat exchangers.

Coolant recirculation system failure events were mostly reactor coolant pump seals and associated cooling water or leakoff piping and controls, as well as reactor-coolant-system piping degradation. Containment heat removal system failure events mostly involved coolers and associated devices affected by silt or foreign material in the cooling water. Main steam valves (isolation and control) and their controls were the cause of most age related failure events of the main steam supply system.

The residual heat removal system events were mainly caused by the failure of valves and pumps and associated control devices. The predominant types of failed devices for reactor containment systems were isolation valves, vacuum breakers and airlock doors and seals.

Components and parts

Tables 5.1.1.3. and 5.1.1.4. present a ranking of the ten components and parts most frequently identified as failed in the event reports. On a component basis, valves made up 20% of the total number of failed components, and over one-third of these (7.8%) were containment isolation valves. Most of the reported events for valves of this type resulted from failure to pass leakage tests required during periodic surveillance testing. In most cases, foreign material or wear on the valve seat was identified as the root cause of leakage. Other types of valves, such as check, control and drain valves, comprised slightly over half of all failed valves. Most failures were internal leaks and packaging leaks.

Table 5.1.1.3. Age Related LERs by Component

Rank	Component	Number	Percentage[a/]	
1	Pipe	446	14.4	
2	Valves, other		277	8.9
3	Monitor	273	8.8	
4	Valve, isolation	243	7.8	
5	Pump	239	7.7	
6	Diesel	151	4.9	
7	Valve, check		101	3.3
8	Steam generator	87	2.8	
9	Heat exchanger		83	2.7
10	Snubbers	69	2.2	
11	Others	1129	36.5	
	Total	3098	100.0	

a/ Based on 3098 age related events

Table 5.1.1.4. Age-related LERs by Part

Rank	Part	Number	Percentage[a/]	
1	Weld	324	10.5	
2	Miscellaneous subcomponent	266	8.6	
3	Pipe or tubing		233	7.5
4	Valve seat		212	6.8
5	Contacts	192	6.2	
6	Packing, seal		163	5.3
7	Wall (pipe)		137	4.4
8	Shaft	113	3.6	
9	Housing	102	3.3	
10	Bearing	69	2.2	
11	Others	1287	41.6	
	Total	3098	100.0	

a/ Based on 3098 age related events.

Pipe failures comprised over 14% of the age related incidents reported. The predominant associated failures occurred in the pipe welds and pipe walls. Most pipe weld failures appeared to be caused by: (1) vibration induced fatigue of inadequately supported piping; (2) temperature recycling stresses on improperly fitted welds; and (3) weld defects. Pipe wall failures were caused mostly by erosion by the process fluid (wet steam, borated water) or heat mechanical stress. On a part basis, tubing and pipe failures combined to make up 7.5% of the failed parts, ranking third after welds and miscellaneous subcomponents. Most of these failures were due to vibration induced cracking at pipe threads or tube fittings.

Pump failures, not including events involving atmosphere monitoring pumps, were responsible for 239 events. Of pump failures, 42% involved impellers, wear rings, shafts, bearings, housing or couplings; 21% resulted from failure of seals or packing. The balance of pump events were failures of belts, mounting bolts, gaskets and miscellaneous associated parts.

One notable component that appeared to fail frequently was radiation monitors for the building atmosphere, used in containments, dry wells, auxiliary buildings, etc. The vane type air pumps used in these monitors appeared to fail from wear caused by foreign material in the sampled air stream.

Failure mechanisms

Among the failure mechanisms identified in Table 5.1.1.5, 'wear' was identified in almost 9% of the reported events as that responsible for the component or part failure. Corrosion of components made up 7% of the LERs, not including electrical contact corrosion in relays and switches, which accounted for 3.5% of the identified failure causes. Debris or contamination originating within a system was termed 'contamination, internal' (6.5%); externally generated contamination (e.g., marine life and construction dust) was termed 'contamination, external' (5.8%). The distinction was made to identify age related degradation of components that generate contamination (internal) as opposed to infrequent or one-time environmentally induced contamination (external), which is not necessarily time related. For example, many failures of smoke detectors were attributed to construction dust, not related to a normal operating situation. 'Fatigue' was identified in 5.5% of the abstracts as the failure mechanism of the affected part. No further definition of such failures was possible from the reports since laboratory determination of thermal fatigue or mechanical fatigue on a microscopic basis is not generally made, except in the case of major components.

Table 5.1.1.5. Failure Mechanism Percentages

Mechanism	Number	Percent[a]
Drift	2795	47.4
Wear	522	8.9
Corrosion	414	7.0
Contamination (internal)	382	6.5
Contamination (external)	331	5.6
Fatigue	324	5.5
Crack	259	4.4
End of life	226	3.8
Corrosion contacts	204	3.5
Vibration	165	2.8
Stress corrosion 110	1.9	
Erosion	102	1.7
Miscellaneous (oxidation, friction, stress hardening high temperature)	59	1.0
Total	5893	100.0

a/ Percentage of events examined in this study.

Method of detection

Timely detection of age related degradation is a key factor in maintaining the readiness of safety related systems to perform their function when required. Of the 3098 events determined to be age related (not including drift), over 64% of the failures were detected by routine testing and surveillance performed in accordance with the plant Technical Specifications or maintenance programme. Operating personnel detected 28% of the reportable failure events during normal operational checks and inspections.

For LERs identifying 'drift' as a failure cause (2795), a survey of a representative number of abstracts indicated that about 80% of the events were detected by scheduled surveillance testing, and about 20% of the 'drift' events were detected by plant operators as operational abnormalities.

These percentages indicate that detection of age-related degradation of many components is accomplished, for the most part, by plant surveillance testing and maintenance programmes.

Severity of failure

Review of the 3098 age related LER abstracts indicated that about 62% of the failure severities were judged degraded; 38% were deemed catastrophic (for definitions see Table 5.1.1.6). No events could be judged to indicate incipient failure because such events are not required to be reported by the LERs.

Table 5.1.1.6. Ageing Study Codes for Failure Severity

Code	Description
C--catastrophic	Component is completely unable to perform its function
D--degraded	Component operates below its specific performance level
I--incipient	Component performs within its design envelope but exhibits indications that, if left unattended, it would probably undergo a degraded or catastrophic failure

When applying a degraded or catastrophic severity to 'instrument' or 'relief valve drift' failures, it should be noted that a reportable failure of such devices may not necessarily be classified as catastrophic. The degree of drift is administratively defined in the Technical Specification acceptance criteria; deviation beyond these criteria is cause for an LER, but it does not necessarily count as a device failure.

General Conclusions

The following general conclusions can be made:

- Surveillance testing and maintenance programmes are effective techniques for detecting degradation.

- The obtained data support recent ASME code emphasis on pump and valve testing, and the IPRDS (In-Plant Reliability Data System) study emphasis on reliability of pumps, valves and diesel generators.

- More research is needed to understand instrument drift causes and piping failures.

- Data reporting system needs improvement. Reports of failure events lack some key information needed to evaluate ageing concerns. These include types of materials involved, the discussion on environmental conditions contributing to failure, and an assessment of the capability to detect component degradation in the incipient state before failure.

5.1.2. Survey of LWR Operating Experience from NPRDS

An ageing failure survey was conducted by the Idaho National Engineering Laboratory using data from the US Nuclear Plant Reliability Data System (NPRDS), a component failure data system owned by the US Institute of Nuclear Power Operations (INPO) and generated on a voluntary basis by the INPO member utilities. For details see Ref. [51].

Objectives

(a) To identify which LWR safety systems and components have been significantly affected by ageing phenomena.

(b) To identify specific ageing failure causes for selected systems and components.

Tasks

The first task was an ageing survey analysis which consisted of a computerized sorting of the as-reported NPRDS component failure records from nine different LWR safety and support systems. The data surveyed covered the period of 1977-86. NPRDS cause-category/cause-code combinations were collected into five major failure categories. These categories were failures due to: ageing, design and installation errors, testing and maintenance, human related and other. The resulting data allowed an identification of LWR safety and support systems which have been significantly affected by ageing effects, as reflected in operational data. The data also allowed the relative magnitude of ageing effects between systems to be evaluated.

The second task was an analysis of ageing related reported causes of failure. This analysis examined the NPRDS failure event records for a boiling water reactor (BWR) and a pressurized water reactor (PWR) service water system and for a PWR Class 1E power distribution system. The resulting failure cause data were analysed to identify the dominant reported ageing mechanisms and the components most affected.

Data source

The NPRDS data source has several strengths and some limitations for ageing evaluations of plant safety systems, support systems and components that reflect on the applicability of the data for certain uses and interpretations. Some of the strengths of the NPRDS are:

- The NPRDS is a large computerized database containing multiple entries for all the safety significant systems and components.

- The component failure records provide detailed information on the component design, the service life of a categorization of the failure by the utility personnel and a failure description. Dates of events, discovery methods, plant conditions and corrective actions are also reported.

- The database contains sufficient information to allow a reasonable determination of the relative number of failures attributable to various causes: dirt, corrosion, cyclic fatigue, mechanical damage/binding, normal/abnormal wear, contact burned/pitted/corroded, loose parts, etc.

Some of the limitations include:

- Complete maintenance histories not available.

- Plant specific effects of maintenance or the environment or other effects are masked.

- Approximately 50% of the reported data are placed in the 'unknown' or 'other' categories.

- Component populations, i.e., total numbers of different components in service, are not available in the database, so component failure rates due to ageing cannot be derived.

In view of these strengths and limitations, the data can supply only relative information regarding which LWR safety systems and components have been significantly affected by ageing and the root or underlying cause of that ageing. The magnitude of ageing problems is not indicated since no figures for the component populations (i.e. number of component-years) are not available in the NPRDS. Accurate determinations require analyses of plant records which were beyond the scope of this study. However, relative information is valuable as one of the inputs for defining future in-depth engineering studies of related systems and components.

Survey Results

Component and systems affected by ageing

NPRDS failure data were compiled for the following PWR and BWR systems and their major subsystems:

Class 1E power distribution system: PWR, BWR
Auxiliary feedwater system (AFW): PWR
Component cooling water system (CCW): PWR, BWR
High pressure injection system (HPIS): PWR
Main feedwater system (MFW): PWR, BWR
Reactor protection trip system (RPS)

Residual heat removal system (RHR): BWR
Service water system (SWS): PWR, BWR
Standby liquid control system (SBL): BWR.

The failure fractions presented in the results are defined for components or systems. These are calculated by dividing the total number of failures for a given system or component within a given category by the total number of failures for that system or component.

The analyses performed to extract ageing related failure data from nine light water reactor systems provided the following insights into the effect of ageing failures. For the system studied, approximately 31% of the failures reported were attributed to ageing (Fig.5.1.2.1). The normally operating fluid systems, such as the service water system (SWS), main feedwater system (MFW) and component cooling water system (CCW), exhibited the highest ageing failure fractions, with pumps and valves being the components most affected. Components in some standby systems, such as pumps in the standby liquid control system (SBL) and valves and pumps in the diesel subsystems of the Class 1E power distribution system (1E), also displayed failure with high ageing fractions (Fig.5.1.2.2).

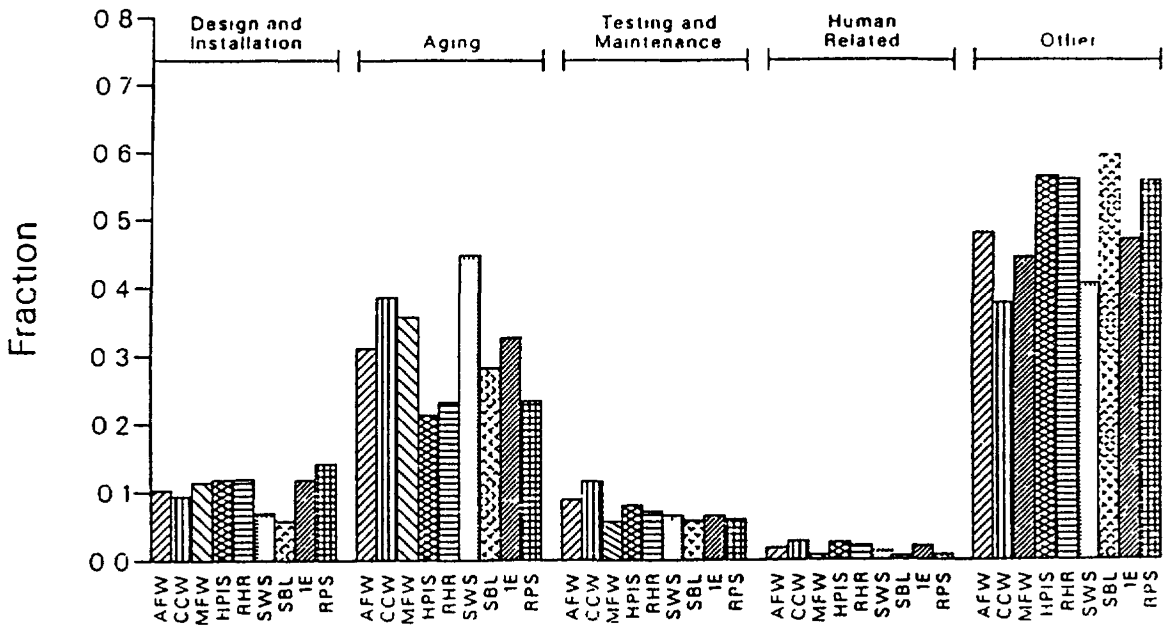

Fig.5.1.2.1. Relationship of failure category fractions by system.

Figures 5.1.2.2 and 5.1.2.3 illustrate the relationship of ageing fractions for selected components in different systems. These figures represent only a subset of the data evaluated. The components selected for Fig. 5.1.2.2 were chosen because they tend to have high ageing fractions in some or all of the systems. The four instrumentation components illustrated in Fig. 5.1.2.3 were selected because they appeared in all nine systems and exhibited ageing failures. The effect of ageing on solid state component failures is minimal in comparison with its effects for other components. The numbers shown with each bar are the number of failures associated with that component in that system.

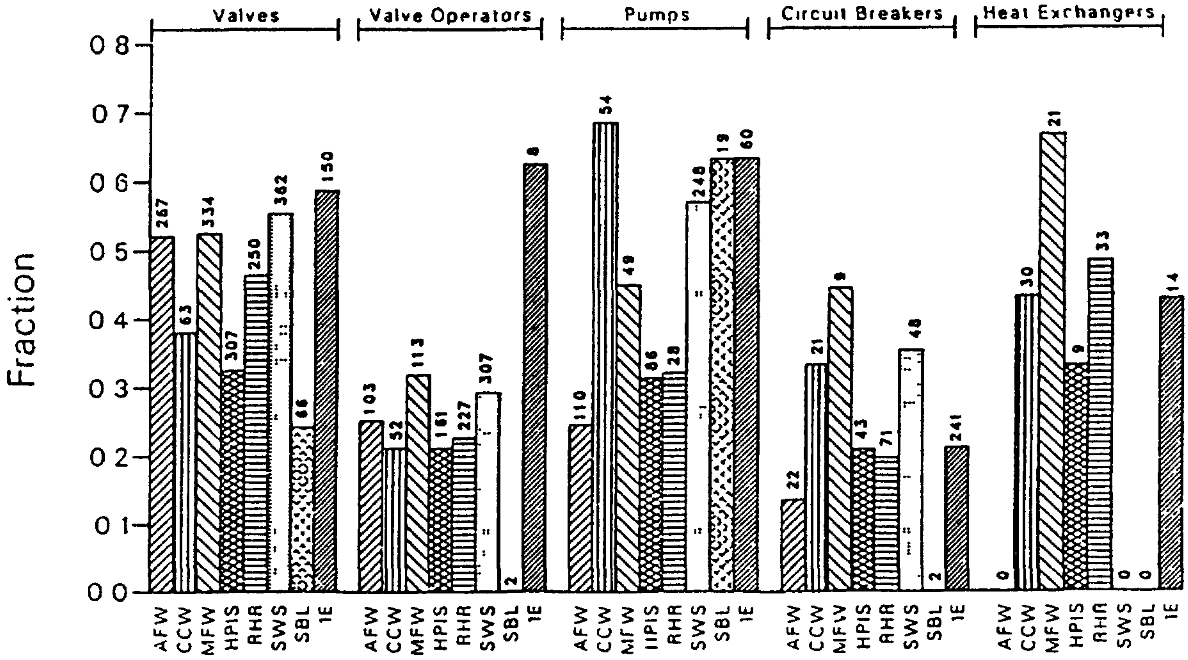

Fig.5.1.2.2. Relationship of component ageing fractions by system.

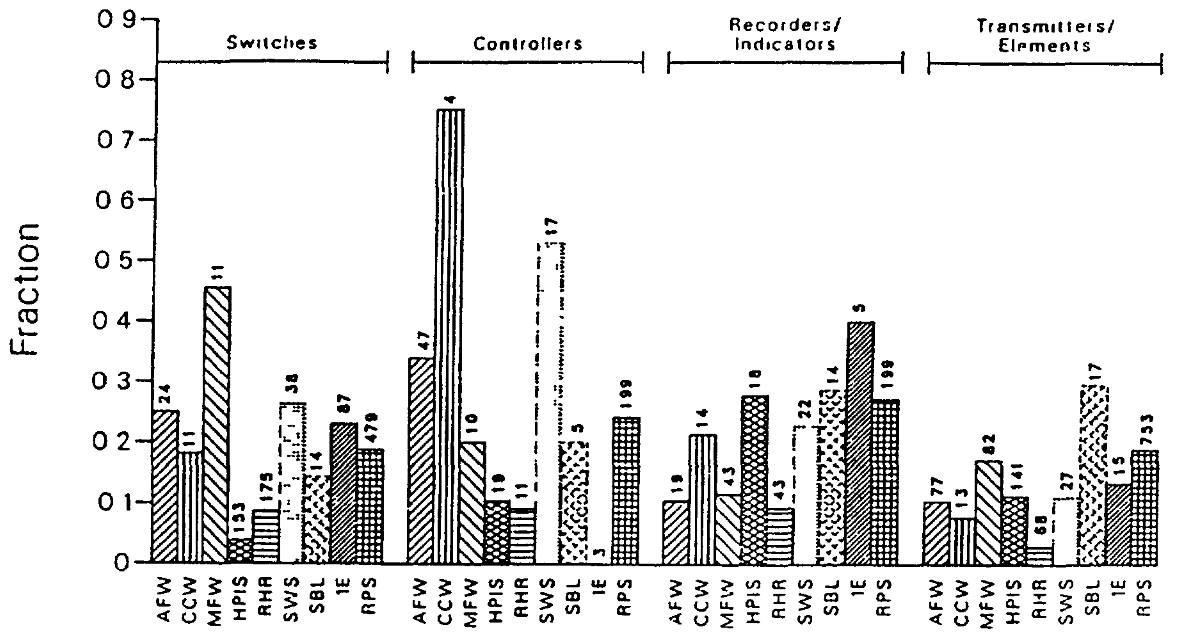

Fig. 5.1.2.3. Relationship of instrumentation component ageing fractions by system.

Ageing Failure Causes

The analysis of the reported causes of failure used a cause categorization scheme to identify and characterize the failure information for ageing related and non-ageing related failures. Results of the analysis (see Tables 5.1.2.1 and 5.1.2.2) indicated that, within the two systems studied, two components in the SWS and one component in the Class 1E power distribution systems dominated the failure contributions. Motor operated valves and motor driven pumps contributed the largest number of failures within the SWS, with the dominant ageing failure cause being 'wear'. The SWS components most affected by ageing were check valves and strainers, with 'wear' and 'corrosion' being the dominant mechanisms reported. The failure of emergency diesel generators dominated the Class 1E power distribution system failures. The dominant reported cause was 'wear'.

Table 5.1.2.1 Table of SWS and Class 1E Component Failures and Ageing Fractions as Determined by the Reported Causes of Failure Study[a]

Components	Total Failures	Lower Bound Aging Total	Upper Bound Aging Total	Lower Bound Aging Fraction	Upper Bound Aging Fraction
		Service Water System			
Check valves	31	27	28	0.87	0.90
Strainers	21	17	18	0.81	0.86
Motor-driven pumps	167	129	140	0.77	0.84
Pneumatic-operated valves	47	36	45	0.77	0.77
Hand control valves	17	11	12	0.65	0.71
Circuit breakers	17	9	12	0.53	0.71
Motor-operated valves	111	43	93	0.39	0.84
TOTAL	411	272	348		
System aging fraction				0.66	0.85
		Class 1E Power Distribution System			
Battery chargers	35	18	34	0.51	0.97
Batteries	10	5	10	0.50	1.00
Diesel generators	113	55	75	0.49	0.66
Circuit breakers	10	3	6	0.30	0.60
Inverters	63	18	38	0.29	0.60
TOTAL	231	99	163		
System aging fraction				0.43	0.71

a. Components have been ordered by lower bound aging fraction.

Table 5.1.2.2 Reported Failure Cause Identification Summary

System/Components	Failure Mode	Total Counts	Dominant Failure Cause	Failure Cause Fraction	Lower Bound Aging Fraction	Upper Bound Aging Fraction
Essential Service Water						
Check valves	Internal leakage	25	Wear	0.480	0.480	0.480
			Corrosion	0.240	0.240	0.240
Motor-operated valves	External leakage	7	Wear	0.714	0.714	0.714
	Fails to open	27	Binding/out of adjustment	0.296	0.037	0.222
	Fails to close	43	Wear	0.140	0.140	0.140
			Binding/out of adjustment	0.535	0.000	0.488
	Fails to opr. as req.	17	Binding/out of adjustment	0.294	0.059	0.235
Pneumatic-operated valves	External leakage	5	Wear	1.000	1.000	1.000
	Fails to close	15	Wear	0.333	0.333	0.333
	Fails to opr. as req.	13	Wear	0.385	0.385	0.385
Motor-driven pumps	Fails to run	91	Wear	0.319	0.319	0.319
			Binding/out of adjustment	0.110	0.011	0.066
			Foreign materials intrusion	0.253	0.176	0.176
	External leakage	64	Wear	0.703	0.703	0.703
			Foreign materials intrusion	0.141	0.125	0.125
Strainers	Loss of function	13	Wear	0.615	0.615	0.615
	Plugged	8	Foreign materials intrusion	0.750	0.500	0.500
Inst. & Uninter. Pwr.-Class 1E						
Inverters	Loss of function	63	Design error or inadequacy	0.079	0.000	0.016
			Wear	0.079	0.079	0.076
			Electrical overload	0.143	0.000	0.016
			Faulty module	0.286	0.079	0.222
			Short circuit	0.127	0.016	0.079
DC Power - Class 1E						
Battery chargers	Loss of function	35	Faulty module	0.657	0.400	0.657
Emergency On-Site Power						
Diesel generator	Fails to run	46	Wear	0.304	0.304	0.304
	No failure	49	Water intrusion	0.184	0.041	0.061
			Wear	0.122	0.122	0.122
			Cyclic fatigue	0.102	0.102	0.102

The failure cause fractions, as used in this analysis, were derived for use in a probabilistic analysis and are therefore specific to component and failure mode. The cause fraction is simply the fraction of the total number of failures due to all causes represented by the component and failure mode of interest.

General Conclusions

The following general conclusions can be made:

- Component ageing contributed to approximately 30% of the reported failures.

- The normally operating fluid systems experienced more ageing related failures (pumps and valves being most affected) than the standby systems. However, pumps and valves in some standby systems (e.g. in the diesel generator subsystem of Class 1E power) also exhibited high number of ageing failures.

- A substantial number of failures are classified as "other" which reflects a common practice of replacing failed or degraded components without determining or reporting the true cause of failure.

- Data reporting system needs improvement. Reports of failure events lack information on root causes of failures including materials and service conditions contributing to failure.

5.1.3. Comments

Limitations of analyses of operating experience

Although both studies of the operating experience described in the foregoing provided valuable information, it should be noted that the databases used were not designed to provide the data needed for the proper evaluation of ageing effects. LERs, for example, were not required to include data on equipment age, service life or service conditions. Although NPRDS data include these, maintenance histories of the failed components and the populations of the components are not required inputs.

Since the evaluation of data on operating experience is an important mechanism for identifying ageing problems, the Member States should review their existing databases to ensure that data (e.g. materials, service conditions and their interactions) needed to evaluate the effects of the ageing of an actual system and component performance are collected.

Periodic analysis of operating experience

One aspect of database analysis is that it may yield different results, depending on the period of time over which the database is sampled, especially for those components with moderately long lifetimes. As plants get older, these types of components may become more prominent in the population of failures occurring owing to ageing.

In a recent investigation [25] soon to be published, the LER database from 1981-1986 was evaluated. In this study, bistables, switches, heat exchangers, relays, invertors/power supplies and indicators/recorders lead piping, valves, valve operators, etc., in ageing percentage of the total age related failures reported over the period 1981-1986. This study tended to elevate engineering safety features actuation systems (ESFAS) and reactor protection trip systems (RPTS) ahead of water systems such as ECCS, MFW, SW, which were found to have the most failures due to ageing in the LER database for 1969-1982.

The importance of this observation is in realising that periodic assessment of these databases can yield information on increasing component failure rates, thereby giving vital information for focusing maintenance and surveillance activities.

5.2. Expert Opinion (Workshops)

A further method of identifying safety significant systems and components which may be subject to age related degradation is to consult members of plant personnel, engineers and scientists working in the nuclear power industry, research and regulatory organizations who have a deep knowledge and experience of NPP performance and behaviour.

The organization and the main results of two workshops on NPP ageing problems that were conducted in the USA [16] are discussed below.

The primary objectives of the workshops were:

(1) To identify LWR ageing problems.

(2) To determine the relative importance of the ageing problems identified for plant safety.

5.2.1. Organization of the workshops

The workshops were organized by Sandia National Laboratory for the US Nuclear Regulatory Commission (USNRC). Representatives of utilities, national laboratories, architect/engineers, vendors of nuclear steam supply systems, research firms and a university with background in materials science, power plant operations, and electrical, mechanical and structural engineering were chosen to participate in the workshops.

Because of the volume of material to be covered, two separate workshops were held. The first workshop addressed two basic ageing questions:

What are believed to be potential ageing problems?

What is the relative ranking of the problems in terms of their implications for safety and what is the basis of the ranking?

The second workshop extended the findings of the first workshop and addressed the following questions:

What has been or could be done to detect, prevent and cope with significant ageing problems?

What is the best mechanism to deal with and solve each problem?

To address the above questions and achieve the workshop's objectives efficiently a structured approach was adopted as outlined below.

Before the first workshop, a questionnaire intended to identify components believed to be susceptible to ageing was completed by each of the participants and the answers were complied by the workshop organizers. (A sample questionnaire is included: see Table 5.2.1.1). One hundred and twelve components were identified.

During the first workshop, each participant used a set of ten questions (Table 5.2.1.2) as a guide in rating the overall importance of the 112 components identified. These potential ageing problems were then ranked using the individual ratings assigned by the participants. Some generic types of components (Table 5.2.1.3) were selected for further review at the second workshop. In addition, discussions were held on the importance of other component ageing problems not listed with the original 112 components. This resulted in a supplementary group of new problems for consideration at the second workshop.

Before the second workshop, the compiled information on the top 14 generic component ageing problems was sent to each workshop participant to consider the second pair of basic ageing questions.

Table 5.2.1.1. Sample Questionnaire

System	Component	Actual or Potential Failure Mode	Manner of Discovery	Observed or Suspected Fundamental Cause of Failure	Observed or Suspected Aging Environment or Aging Problems	Comments
HVAC	High pressure injection pump room unit cooler	Insufficient output	Failure during operation	Air flow blockage through cooler	Dirt/dust	----
Component Cooling Water	Piping	Pressure boundary	Routine walk through	Wall thinning	Liquid Erosion	High flow rate
Component Cooling Water	Heat exchanger	Insufficient output	Operational parameter change (T)	Poor heat transfer coefficient	Corrosive service water	Organic growth buildup
Emergency DC	MCC's for low pressure injection valves	Delayed response	Routine testing	Binding of switches	Corrosive vapors	Salt moisture in air
Service Air	Air compressor foundations	Foundation failure	Special surveillance	Cracking of concrete	Vibration	----
Emergency AC	Cabling	Insufficient Fire Protection	Routine maintenance	Cracking of fire retardant coating	Insufficient moisture and high temperatures	Coatings separated from cabling
HVAC	Fire damper	Insufficient fire protection	Special surveillance	Binding of damper	Dirt/dust	----

Table 5.2.1.2. Component Ranking Questions

1. Have examples of the problem been observed?
2. Is the problem potentially widespread?
3. Does or could the problem involve safety system components?
4. Can the problem jeopardize an entire safety function?
5. Is the resulting component degradation rapid?
6. Can the problem occur with little or no warning?
7. Can the problem escape current T&M practices?
8. Can a frequently challenged safety function be affected?
9. Can the problem result in common-mode failure during design-basis events
10. Is little or no work being done to address the problem?

Table 5.2.1.3. List of Top 14 Generic Component ageing problems

Component	Actual or Potential Failure Mode	Observed or Suspected Fundamental Cause of Failure	Observed or Suspected Aging Environment or Aging Problems
Pressure/temp sensors	Decalibration	Mech. aging of bellows, springs	Vibration, connector cabling degradation
	Insufficient/ no output	Binding	Waterhammer
		Electronics drift or sensor degradation	Thermal degradation, voltage transients, impurity introduction
		Brittle connector	High temperatures
	Open circuits	Set point drift	High temperatures
	Decalibration	Moving parts wear	
Electrical connectors/	Open circuit	Oxidation of contact surface	Normal cabinet environment
Terminal blocks	Spurious response	Tracking (carbonizing)	Dirt/dust/salt
	Open circuit	Worn screws and parts	Too much surveillance
Valves/solenoid valves	Seat leakage	Wear and wire drawing	Normal design environment
	Hampered operation	Flow blockage	Oil in airline, failure of seals

Table 5.2.1.3. (continued)

Component	Actual or Potential Failure Mode	Observed or Suspected Fundamental Cause of Failure	Observed or Suspected Aging Environment or Aging Problems
Valve operators	Function impaired	Hardening of lubricant, pneumatic seal failure	Normal design environment
	Loosening of components	Spring type lock washers allow chafing of surfaces and loosening bolts	Vibration
	Excessive torque	Packing too tight	Overtightening to handle leaks
	Failure to operate	Lubricant hardens	Temperature variations
Switch/relay/ circuit breaker	Open circuit	Fatigue of spring	Vibration
	Failure to trip	Grease binding	Normal design environment
		Wear-induced friction	Lack of periodic lubrication
	Opening/clogging wrong contacts	Cam wear and coupling wear	Normal design environment
	Failure to operate in required time	Fatigue of spring	Wear/dirt impartment
		Pitting/thinning of contacts	Environment corrosion of voltage areas
	Spurious response	Binding	Dirt/dust
Diesel generator	Piping failure	Cracking	
	Structure failure	Wear	
Motors/pump motors	Bearing failure of pump	Wear	High temperature wear
	Insulation failure	Turn-to-turn short	Thermal/voltage degradation
Transformers	Insulation failure	Turn-to-turn short	Thermal/voltage degradation
Cables	Insulation failure	Short to ground	Corrosive fluids Voltage stress
	Strand breakage	Open circuit	Vibration, corrosion at interface, temperature cycles, radiation
Snubbers	Leakage or nonfunction	Seal embrittlement or blockage	Thermal/radiation/overstress
Piping	Leakage	Wall thinning	Erosive silt in water
Steam generator tubes	Leakage	Denting, cracking	Chemistry-induced corrosion
Relief valves	Leakage	Erosion	Normal design conditions
Concrete/anchors tendons	Loss of pretension	Inadequate torque, grout creep	Vibrations, excess stress

At the second workshop, a table was developed for each of the 14 generic problems, listing firstly, how one detects each problem, and secondly, how one prevents/copes with each problem (Table 5.2.1.4). In addition, for each of the speculative problems identified during the first workshop as going beyond the 14 generic component types, a five minute 'brainstorming' session was held to solicit comments and recommendations. Using the results of these sessions, plus the written input provided by some participants a list of potential problems was developed which also gives some pertinent comments on each problem and an assessment of the perceived importance of the problem.

Table 5.2.1.4. Sample of the table of generic component ageing problems (How to detect and how to prevent/cope/handle them)

PRESSURE SENSORS

FAILURE MODE AND/OR MECHANISM OF FAILURE	DETECT	PREVENT/COPE/HANDLE
1. Insufficient/No Output or Open Circuits	· Anomalous signals · Comparative outputs · Known signal (devise check) · Loop check · Annunciators	· Protect against entry of moisture and chemicals · Use higher quality electronics with more fatigue-resistant materials · Use drift information with historical data to recalibrate or replace · Preventive maintenance · Failsafe design · Use redundancy
2. Decalibration by Mechanical Aging of Bellow Springs	· Trending analysis · Anomalous signal · Comparative outputs · Known signal (devise test) · Loop check · Comparative Channel	· Less exercise · Recalibrate · Change springs · Change material
3. Decalibration by Binding	· Same Method as 2	· Avoid extreme ranges in cycles by flow limiting orifice or accumulators

VALVE OPERATORS - MOTOR

FAILURE MODE AND/OR MECHANISM OF FAILURE	DETECT	PREVENT/COPE/HANDLE
1. Function of Motor Impaired due to Lubrication Hardening	· Trips due to high torque · Stroking time increased · Analyze lubricant · Electrical current draw on motor · Use torque wrench compare new to aged	· Integrate maintenance of electrical/ mechanical parts · Periodic lubrication change · Use a different lubricant
2. Function of Motor Impaired due to Loosening of Parts	· Check operation of valve · Electrical current checks · Stroking time increased	· Monitor vibrating systems for signs of wear · Periodic torque check
3. Function of Motor Impaired due to Excessive Torque	Trips due to high torque · Stroking time increased · Electrical current draw on motor · Use torque wrench compare new to aged	· Packing too tight · Design to prevent over-tightening · Integrate maintenance-electrical/mechanical · Check packing - so does not go due to "over-torque" · Educate operator · Redesign packing materials · Follow manufacturer's instructions

As a final step at the second workshop, each participant was asked to rate the safety importance of each of the 14 generic safety problems as high, medium or low. For those problems which a participant rated highly, he/she was asked to state a reason and to suggest who should do the R&D to resolve the problem. Typical results of that rating are shown in Table 5.2.1.5.

5.2.2 Findings and observations from the workshops

The workshops identified a wide range of components whose ageing may affect plant safety.

(1) Although the participants did not feel much knowledgeable about NPP safety, most felt that ageing problems relating to safety system components are the most important. Despite this feeling, however, there is a concern that participants, knowledgeable about components and component problems, may tend to rate particular components as important simply because of an awareness of component troubles, not necessarily because of the safety significance of the components.

(2) Most participants considered ageing in terms of how it can affect the performance of a component's normal operation. However, during any off-normal conditions such as in a loss of coolant accident or an earthquake, aged components that may meet performance specifications for normal conditions may fail. The major concern here is that a common mode type failure could occur.

(3) Component ageing issues identified as important also appear to be well known, as evidenced by the fact that some work on ageing could be cited for most of the 14 components listed in Table 5.2.1.3.

(4) The priorities were identified by a method of voting by the participants. A real concern is that a problem which has been seen only once or twice and thus is not now of general concern may not receive a proper rating because the participants do not realize the potential significance of this one failure.

5.2.3. Comments

The workshops described are a useful approach for identifying ageing problems to be addressed by research or by improvements in design, operation and maintenance. They could have been more useful if some of the participants had been more knowledgeable in reactor safety and also if the workshops had reviewed and interpreted findings from the analysis of operating experience outlined in subsection 5.1. Furthermore, a consensus approach should be considered as an alternative or a supplementary method to the voting method used to establish the ranking of the ageing problems identified.

Table 5.2.1.5. Sample of the rating of generic component ageing problems

COMPONENT	R&D PRIORITY High	Medium	Low	REASONS FOR HIGH PRIORITY	WHO SHOULD DO R&D?
Pressure/Temp Sensors	8	1	0	a. Need to fundamentally understand drift limits, need historical data	a. EPRI and National Labs
				b. Need to assess if they will function under additional adverse conditions that come from TMI and Appendix R	b. Industry with EPRI assistance
				c. LER experiences and common-mode failure potential	c. NRC/National Labs/ sometimes owners or users groups
				d. Wide use in plants and safety systems	d. EPRI or NRC with vendor/ industry involvement or utilities by use of incentives and/or INPO
Valve Operators	7	4	0	a. Need to develop packing standards-specs./quality control	a. EPRI
				b. Concerned about heat and temperature degradation of lubricant, packing, etc.	b. Utility/manufacturer
				c. Large numbers of systems sensitive to their failure and failure not apparent until too late	c. Manufacturers with industry group or vendor with EPRI that could result in IEEE Standards
				d. Significant number of problems seen	d. Utilities

COMPONENT	R&D PRIORITY			REASONS FOR HIGH PRIORITY	WHO SHOULD DO R&D?
	High	Medium	Low		
Diesel Generator	4	3	4	a. Problem real and related to safety	a. DG manufacturers and owners group/EPRI or industry/NRC
Motors/Pump Motors	2	6	3	a. Need to establish standards for lubrication and relationship to wear	a. EPRI/pump vendors
				b. Need to quantify and specify testing	b. ASME with EPRI
Transformers	0	3	8	a. No reason given	
Cables	3	4	4	a. Need test specifications to predict failure	a. NRC driven
				b. Potential of common-mode failure and ubiquitous use	b. NRC/National Labs/ manufacturers
Snubbers	5	6	0	a. Cause of failures not established	a. EPRI
				b. Potential for common-mode failure	b. EPRI
				c. Need to establish technical basis for seal replacement or select new seal materials	c. National Labs
				d. Significant known problems	d. Snubber manufacturers

5.3. Probabilistic Techniques

The information presented in a standard probabilistic safety assessment (PSA) does not include time dependent effects. However, ageing is a time dependent phenomenon. In determining the risk level at a plant, PSAs generally use a time averaged unavailability which limits the utility of the information that can be extracted from a PSA. More information would be available if ageing effects were included and would not require modification of the PSA. Such an adaptation of PSA results enables an identification of the components that have the greatest effect on risk if their failure rates increase owing to ageing or service wear effects.

This section outlines the development of such probabilistic methods which have been sponsored by USNRC [8, 26]. Two types of methods are presented. A basic method which does not consider time dependent effects of ageing on the component failure rate and a more comprehensive method which does.

5.3.1 Overview of probabilistic safety assessment

PSA is a method of estimating mathematically the likelihood and consequences of potential accidents at nuclear power plants. In the process of performing a PSA, the potential accident initiators (LOCAs, transients, loss of off-site power, etc.) are identified and their likelihood quantified. The systems that must function to shut down the reactor safely and cool the reactor and prevent unacceptable releases of radionuclides from a plant are then identified for each initiator using event tree and fault tree methodology. The systems generally considered in a PSA are the reactor protection system, main and auxiliary feedwater systems, high pressure and low pressure coolant injection systems, residual heat removal systems, containment sprays, containment coolers and accumulators, electric power, service water and the associated instrumentation and control (activation and annunciation) circuits. Operator actions and errors of omission are also included in the models.

The event tree and fault tree model solutions determine the combinations of component and system failures that lead to a core melt for each of the initiators. The combination of an accident initiator and the system failures that result in core damage is referred to as an accident sequence. The combinations of individual component failures that cause the required systems to fail is referred to as a cutset.

The probability of each individual component being unavailable is referred to as its unavailability. The probability of the cutset is the product of the unavailabilities of the individual events. The frequency of an accident sequence can be approximated by the sum of all the cutsets that result in failures of the same set of safety systems. The overall plant risk is similarly approximated by the sum of the accident sequences, or equivalently, the sum of all the accident cutsets.

In addition, a probability of containment failure can be assigned to each accident sequence. In some PSAs, the consequences of accident sequences are evaluated in terms of dose, fatalities, or economic impact.

The scopes of PSAs differ greatly. Some consider internal events only; others include seismic events, floods and fires etc. The depths of the analyses of the systems and sequence consequences also vary considerably. Thus, the scope of the PSA, as well as the level of detail considered, limits the information that can be extracted from the analysis.

PSAs generally concentrate on finding the components most significant for risk; for example, those system, component and human failures which make significant contributions to the probability of core damage. In many cases passive components such as the containment building, the reactor vessel and storage tanks are considered to have negligible failure rates and are excluded from the risk analyses. In most PSAs, wires and piping segments are considered to have failure rates that are associated and are omitted from further analysis. However, the risk significance of a particular wire or piping segment can be inferred from the PSA by determining the effect of failure of the wire or pipe on the component to which it is connected.

5.3.2. Ageing analysis

It is necessary to define what is meant by ageing phenomena before evaluating their risk significance. For our purposes, 'ageing phenomena' are phenomena that have one or both of the following two effects:

(1) causing the failure rate of a component to increase as a function of time; or

(2) causing a component that was designed to meet certain standards to degrade such that it no longer fulfils those design requirements.

<u>Effect of increases in failure rate</u>

The first ageing effect considered causes the failure rate of a component (or a set of components) to increase with time as the component ages or wears out. Fig. 5.3.1 shows a sample plot of the failure rate λ as a function of time for a typical component. This is the familiar 'bathtub' curve common to many components. This curve has three distinct regions: (1) the burn-in period; (2) the period of normal operation (where the failure rate is essentially constant); and (3) the wear-out period.

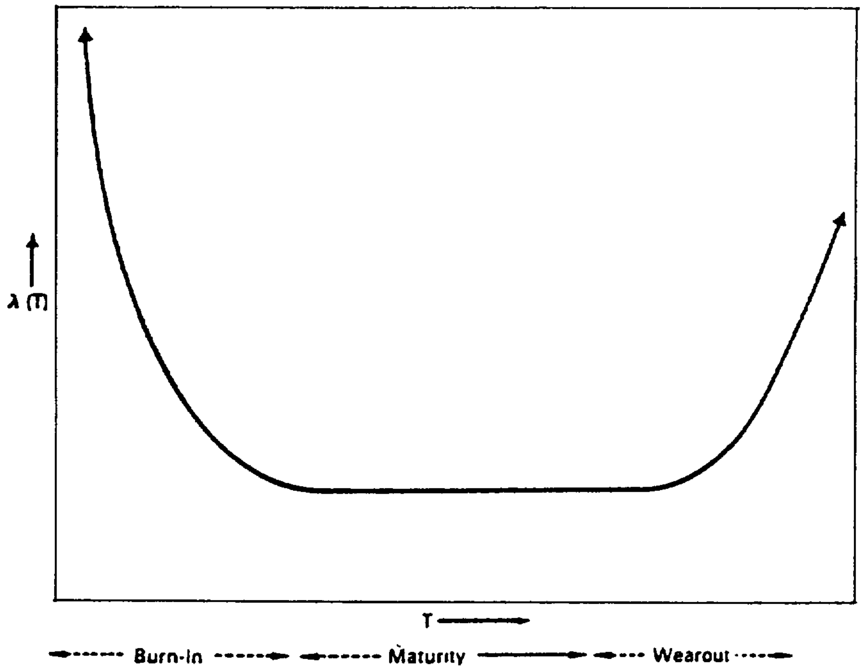

Fig.5.3.1. Example of a failure rate curve.

Ageing phenomena occur in the wear-out period where the failure rate is increasing. The root cause of this increase in failure rate can be any of a number of ageing phenomena; fatigue or corrosion, for example. The increase in the failure rate with time can have two effects on risk:

(1) The increase in failure rate increases the unavailability (decreases the reliability) of a component important to safety.

(2) The increase in failure rate of certain components could cause an increase in the frequency of process system upsets, including the frequency of accident initiators. This effectively increases the number of times safety systems must operate and proportionally increases the risk.

An example of a component whose unavailability increases with time is a pump in the low pressure coolant injection system of a PWR. Normally the pump is in the standby mode and is tested at regular intervals. If the failure rate is increasing with time (as in the wear out region), the unavailability history may look like that of Fig. 5.3.2. In this example, the test interval remains constant but the fraction of tests in which failures are detected is increasing as the component ages. The unavailability of that component, and therefore the risk associated with it, is increasing with time and may be substantially higher at the end of the period of interest than at the beginning.

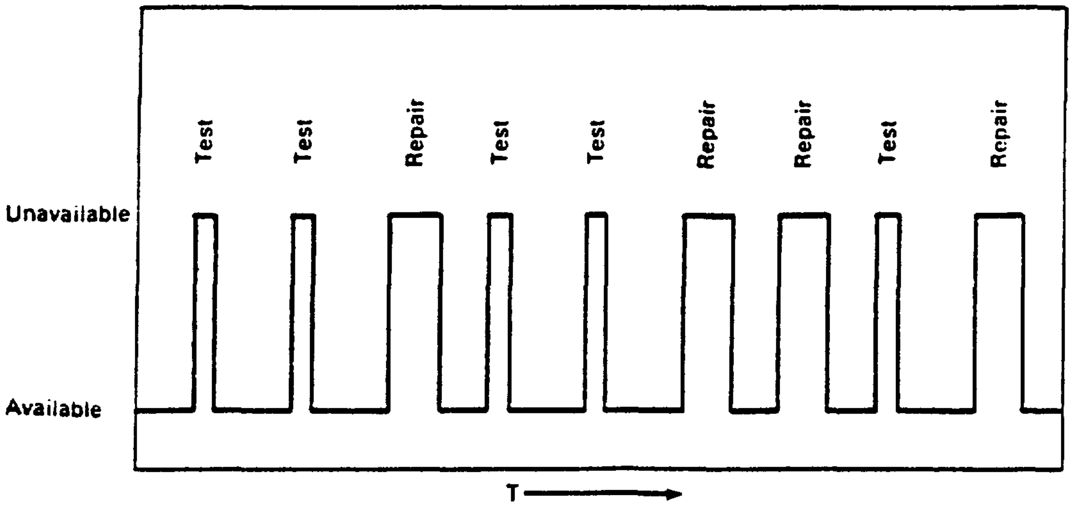

Fig.5.3.2. Component unavailability history.

An example of a component that could cause risk to increase by causing the initiator frequency to increase is a steam generator tube. If the failure rate of tubes is increasing, the likelihood of a steam generator tube rupture increases. Should this event occur, the necessary safety systems have to operate correctly to prevent core damage. Another example of components that increase risk by increasing the frequency of initiators is the reactor coolant systems piping. Also, components on the secondary side of the plant, such as the main feedwater pumps, whose failure rates increase with time have the effect of increasing the frequency of transient initiators and thus the risk.

Effect of degraded characteristics

The other type of ageing phenomenon that is of interest is processes that gradually degrade characteristics of the component. This could cause a component that is designed to meet certain design requirements to degrade such that it no longer fulfils its design requirements. Examples of this type of component are snubbers that lose their damping capacity as the fluid leaks through the seals or heat exchangers that lose heat transfer capacity as an oxidation layer is formed on the tubes. The reactor vessel can also be treated as a component of this type since its pressure capacity decreases as a function of fluence and the number of power transients (for example scrams) that it is subjected to. Determining the risk significance of this degradation is more complex than for components described in the previous section since it generally involves combining a probabilistic load distribution with fragility curves and considering the impacts of the different failure modes. It will therefore be difficult to use PSAs directly to evaluate the risk significance of components of this type. However, bounding calculations can be performed.

5.3.3. Basic time independent approach for evaluating risk sensitivity to component ageing

5.3.3.1. Methodology

In order to characterize the risk impact of component ageing and service wear effects, it is necessary to characterize the time dependent nature of the change in plant risk. That is,

$$I_A = \frac{\partial R}{\partial t} \qquad (1)$$

where

I_A is the risk impact of ageing, and

R is the plant risk.

As stated earlier, the plant risk is a function of the component unavailability q_j and the component unavailability is a function of the component failure rate λ; For the study of ageing, the failure rate is a function of time t. By the chain rule, changes in plant risk are expressed as:

$$\frac{\partial R}{\partial t} = \frac{\partial R}{\partial q_j} \cdot \frac{\partial q_j}{\partial \lambda_j} \cdot \frac{\partial \lambda_j}{\partial t} \qquad (2)$$

The risk impact due to ageing can now be separated into two distinct parts:

(1) The effects of changes in the component failure rate (the first two terms of the right hand side of Eq. 2)

(2) The time dependent effects of ageing and service wear on the component failure rate (the third term of the right hand side of Eq. 2).

The report [8] concentrates on the first part, the change in risk due to changes in component failure rate. The second part, changes in this failure rate due to ageing and service wear, was beyond the scope of the study and is the subject of another study [26] and is discussed in the following subsection 5.3.4.

We define the risk ageing sensitivity to failure rate as:

$$G_j = \frac{\partial R}{\partial \lambda_j} = \frac{\partial R}{\partial q_j} \cdot \frac{\partial q_j}{\partial \lambda_j} \qquad (3)$$

where the first term on the right hand side of Eq. 3 is the partial derivative of risk with respect to component unavailability and the second term is the partial derivative of the component unavailability with respect to the component failure rate.

The first term, the partial derivative of risk with respect to component unavailability, can be shown to be equivalent to the Birnbaum measure [9]. This is a measure of the impact of a component failure on risk and can be computed by changing the unavailability of the component in the risk equation to unity and determining the change in risk. Vesely et al. [10] have calculated values of the Birnbaum measure in recent work. The second term, the partial derivative of component unavailability with respect to component failure rate, is presented in Table 5.3.1. The expressions in Table 5.3.1 are derived from the component unavailability equations and this second term is related to the time a component is unavailable when it is failed.

Table 5.3.1 also includes the sensitivity of risk to ageing for components with negligible failure rates. This type of component unavailability is dominated by constant contributions, for example human error, and represents an essentially time independent unavailability. In this case the risk ageing sensitivity factor G_j is zero, since $\partial q_j/\partial \lambda_j=0$.

The risk ageing sensitivity measure is used to rank components based on their potential for changing the risk. The measure makes no assumptions about the rate of component ageing; the ranking results are strictly valid only when all the components age at the same rate. Differences in ageing rates between different component types were beyond the scope of the study and the time dependent behaviour of component failure rates was considered in Ref. [26].

5.3.3.2. Applications of the risk ageing sensitivity measure at selected plants

The above outlined methodology was used to calculate risk ageing sensitivity for three U.S. nuclear power plants which were analyzed by limited-scope PSAs [11, 12, 13]. The risk ageing sensitivity (a measure of the risk sensitivity to changes in component failure rates) was calculated for individual components at each of the three NPPs. Individual components were then grouped by component type and system at each plant, and the component groups were ranked from highest to lower potential risk impact.

Results

The combined results of the analysis of the two PWRs giving an overall PWR ranking are presented in Tables 5.3.2 and 5.3.3. They indicate that many of the potentially most risk significant components are in the auxiliary

Table 5.3.1 Rate of Change of Competent Unavailability with Respect to Failure Rate

Component Type	Average Unavailability	Rate of Change of Component Unavailability with Respect to Component Failure Rate
Periodically Tested Component	$\bar{q}_s = \frac{\lambda_s T}{2} + q_o \frac{\tau}{T} + \lambda_s T_R + \frac{d_M}{T_M} + C$	$\frac{\partial \bar{q}_s}{\partial \lambda_s} = \frac{T}{2} + T_R$
Periodically Tested Component with Negligible Failure Rate	$\bar{q}_s = q_o \frac{\tau}{T} + \frac{d_M}{T_M} + C$	$\frac{\partial \bar{q}_s}{\partial \lambda_s} = 0$
Continuously Monitored Component	$\bar{q}_o = \lambda_o T_R$	$\frac{\partial \bar{q}_o}{\partial \lambda_o} = T_R$

where:

λ_s = constant standby failure rate

λ_o = constant operation failure rate

T = interval between tests

T_M = average interval between maintenance

T_R = repair duration time

q_o = override unavailability (the probability that the component is inoperable during the test)

τ = test duration time

d_M = average maintenance duration time

C = human error probability

$\bar{q}_o$ = average unavailability of continuously monitored components

$\bar{q}_s$ = total average unavailability of the periodically tested component

Table 5.3.2 Ageing Sensitivity of Component Groups in Two PWRs

Rank		System	Ageing Sensitivity
1	Check Valves	Auxiliary Feedwater	5.5×10^{-3}
2	Circuit Breaker Contractor	Reactor Protection	3.2×10^{-3}
3	Trip Relay/Trip Module	Reactor Protection	2.2×10^{-3}
4	Control Valves (air operated)	Auxiliary Feedwater	1.4×10^{-3}
5	Motor Operated Valves	Auxiliary Feedwater	1.4×10^{-3}
6	Pumps	Auxiliary Feedwater	1.4×10^{-3}
7	Motor Operated Valve	High Pressure ECC	4.7×10^{-4}
8	Motor Operated Valve	Service Water	2.9×10^{-4}
9	Pumps	Service Water	2.6×10^{-4}
10	Actuation Channels	Safeguards Actuation	2.1×10^{-4}
11	Check Valve	Low Pressure ECC	1.8×10^{-4}
12	Motor Operated Valve	Low Pressure ECC	1.7×10^{-4}
13	Turbo Generator/ Diesel Generator	Emergency Power	1.6×10^{-4}
14	Check Valve	High Pressure ECC	1.0×10^{-4}
15	Batteries	Emergency Power	7.3×10^{-5}
16	Pumps	High Pressure ECC	5.3×10^{-5}
17	Room Coolers	Service Water	3.3×10^{-5}
18	Pumps	Low Pressure ECC	2.0×10^{-5}
19	Relief Valves	Reactor Coolant Pressure Boundary	1.5×10^{-5}
20	Check Valves	Service Water	1.3×10^{-5}

Table 5.3.3. Ageing Sensitivity of Component Types in Two PWRs

Rank	Type	Ageing Sensitivity
1	Check valves	5.8×10^{-3}
2	Circuit breaker/contact	3.2×10^{-3}
3	Trip module, relay/actuation channel	2.4×10^{-3}
4	Motor operated valves	2.3×10^{-3}
5	Pumps	1.7×10^{-3}
6	Control valves (air operated)	1.4×10^{-3}
7	Turbo generator/diesel generator	1.6×10^{-4}
8	Batteries	7.3×10^{-5}
9	Room collers	3.3×10^{-5}
10	Relief valves	1.5×10^{-5}

feedwater system, the reactor protection system, and the service water systems. Pumps, check valves, motor operated valves, circuit breakers, and actuating circuits are the component types that have the greatest impact on plant risk if their failure rates increase substantially. However, it should be remembered that the results do not indicate in any way which components are most susceptible to ageing degradation. The ranking is based on the assumption that all the components age at the same rate. To characterize, fully, the risk impact due to component ageing, these results must be coupled with time dependent failure rate models.

5.3.4. More comprehensive approach for determining risk sensitivity to component ageing

5.3.4.1. Methodology

The risk sensitivity to ageing is determined more comprehensively from the equation for the rate of risk change in terms of the component unavailability changes

$$\frac{dR}{dt} = \sum_j \frac{dR}{dq_j} \frac{dq_j}{dt} \tag{4}$$

where:

$$\frac{dq_j}{dt} = \text{the rate of change with time of component unavailability } q_j \tag{5}$$

However, each rate of change $\frac{dq}{dt}$ is not decomposed as

$$\frac{dq}{dt} = \frac{dq}{d\lambda} \frac{d\lambda}{dt} \tag{6}$$

as done in the previous approach. Instead, appropriate equations must be substituted for q into Equation (4). (Where convenient subscripts are deleted to simplify the notation.)

To distinguish risk sensitivity to <u>ageing-caused</u> time dependencies, we will express the component unavailability of q_j is expressed as:

$$q_j = q_{o,j} + (q_j - q_{o,j}) \tag{7}$$

where $q_{o,j}$ is the component unavailability with the ageing contribution not included.

Substituting equation (7) into equation (4) yields

$$\frac{dR}{dt} = \sum_j \frac{dR}{dq_j} \frac{dq_{o,j}}{dt} + \sum_j \frac{dR}{dq_j} \frac{d\ (q_j - q_{o,j})}{dt} \tag{8}$$

The first term on the right hand side of equation (8) is the contribution to the risk change from the normal time dependence in the component unavailability which is not due to ageing. The second term on the right hand side of equation (8) is the contribution from the additional change due to ageing. Thus, we will associate the rate of risk change due to ageing with this second term and denote it as

$\left(\frac{dR}{dt}\right)_A$:

$$\left(\frac{dR}{dt}\right)_A = \sum_j \frac{dR}{dq_j} \frac{d \quad (q_j - q_{o,j})}{dt} \tag{9}$$

If there is no ageing $q_j = q_{o,j}$ and Equation (9) gives:

$$\left(\frac{dR}{dt}\right)_A = 0 \quad \text{(no ageing)} \tag{10}$$

The basic equation for ageing risk sensitivities is thus equation (9). Each term in the sum on the right hand side of equation (9) gives the contribution to the risk change from the ageing of each specific component. Each term may be called the component ageing risk sensitivity contribution, denoted by r:

$$r = \frac{dR}{dq} \frac{d \quad (q - q_o)}{dt} \tag{11}$$

The total rate of risk increase due to ageing $\left(\frac{dR}{dt}\right)_A$ given by equation (9) is then simply the sum of the component ageing risk sensitivity contributions r:

$$\left(\frac{dR}{dt}\right)_A = \sum_j r_j \tag{12}$$

Equation (11) defining the component ageing risk sensitivity contribution r can also be expressed in words as:

$$\begin{pmatrix}\text{the component}\\\text{ageing risk}\\\text{sensitivity}\\\text{contribution}\end{pmatrix} = \begin{pmatrix}\text{the risk}\\\text{importance}\\\text{of the}\\\text{component}\end{pmatrix} \times \begin{pmatrix}\text{the extra}\\\text{rate of}\\\text{change of}\\\text{the component}\\\text{unavailability}\\\text{due to ageing}\end{pmatrix} \tag{13}$$

that is

$$r = \frac{dR}{dq} \frac{d \ (q - q_o)}{dt} \tag{14}$$

It should be noted that in general the risk importance of the component $\frac{dR}{dq}$ is also time dependent and can change with time and with age.

5.3.4.2. Specific formulas for the component ageing risk sensitivity

The equations for the component ageing risk sensitivity derived in subsection 5.3.4.1 are quite general and apply to any ageing behaviours. The specific equation for the ageing risk sensitivity r is again

$$r = \frac{dR}{dq} \frac{d\,(q - q_o)}{dt} \tag{15}$$

Specific formulas must now be used for q and q_o to obtain specific expressions for the component ageing risk sensitivity r. To consider first the case where the component is unrepairable.

Then,

$$q = 1 - \exp(- {}_o\int^t \lambda(t')\, dt') \tag{16}$$

and

$$q_o = 1 - \exp(- \lambda_o t) \tag{17}$$

where $\lambda(t)$ is the time dependent component failure rate with ageing and λ_o is the constant failure rate assuming no ageing.

Therefore,

$$\frac{d\,(q - q_o)}{dt} = \lambda(t)\exp(- {}_o\int^t \lambda(t')\, dt') - \lambda_o \exp(- \lambda_o t) \tag{18}$$

For many applications the quantities ${}_o\int^t \lambda(t')dt'$ and λ t will be small. When ${}_o\int^t \lambda(t')dt'$ and $\lambda_o t$ are small, for example less than 0.1, which corresponds to the unavailabilities q and q_o being less than 0.1, then the exponentials in equation (18) are approximately unity. Hence,

$$r \simeq \frac{dR}{dq} (\lambda(t) - \lambda_o) \quad (q, q_o < 0.1) \tag{19}$$

This means that for q, $q_o < 0.1$, to the first order the risk sensitivity r to an ageing component that is unrepairable is simply the risk importance of the component $\frac{dR}{dq}$ times the increase in failure rate due to ageing $(\lambda(t)-\lambda_o)$;

$$\begin{pmatrix}\text{risk sensitivity}\\ \text{to an ageing,}\\ \text{unrepairable}\\ \text{component}\end{pmatrix} \simeq \begin{pmatrix}\text{the risk}\\ \text{importance of}\\ \text{the component}\end{pmatrix} \times \begin{pmatrix}\text{the increase}\\ \text{in failure}\\ \text{rate due to}\\ \text{ageing}\end{pmatrix} \tag{20}$$

In extreme ageing cases where for large t, q is near 1 (i.e., q > 0.1) but where q_o is still less than 0.1, we have

$$\frac{d\,(q - q_o)}{dt} \simeq \lambda\,(t)\,\exp(-\int_o^t \lambda(t')\,dt') - \lambda_o \tag{21}$$

$$\leq \lambda(t) - \lambda_o \tag{22}$$

Hence, the appropriate expression for the component risk sensitivity given by equation (19) serves not only as an accurate approximation for small to moderate ageing effects but also serves as a conservative bound for extreme ageing effects.

If failures are detectable by the test and maintenance, then the degree to which the failures are repaired must be considered. The component may simply be restored to an operational status without replacing the degraded sub-components. For this situation the component may be modelled as being restored to "as good as old". If the component is replaced at the test or maintenance, then the component may be modelled as being restored to "as good as new". If the ageing mechanism affects various sub-components and some are simply restored to an operational status and others are replaced, then each sub-components needs to be considered as a separate component.

The component ageing risk sensitivity r will be evaluated for the case where the component was last checked at t_c and was last restored at t_o where $t_c \geq t_o$. It will be shown that the results obtained are interpretable for general test and maintenance situations.

For a test at t_c and a renewal at t_o, where $t_c \geq t_o$, the nonageing unavailability q_o is:

$$q_o = 1 - \exp\,(-\lambda_o\,(t-t_c)) \tag{23}$$

The ageing unavailability q is given by:

$$q = 1 - \exp\,(-\int_{t_c-t_o}^{t-t_o} \lambda(t')dt') \tag{24}$$

where $t_c \geq t_o$.

The component ageing risk sensitivity r is thus:

$$r = \frac{dR}{dq}\,\left(\frac{dq}{dt} - \frac{dq_o}{dt}\right) \tag{25}$$

$$= \frac{dR}{dq}\,(\,\lambda\,(t-t_o)\exp(-\int_{t_c-t_o}^{t-t_o} \lambda(t')dt') - \lambda_o\exp(-\lambda_o(t-t_c)) \tag{26}$$

Again for many applications, the terms $\int_{t_c-t_o}^{t-t_o}\lambda(t')dt'$ and $\lambda_o(t-t_c)$ are significantly less than unity. This situation applies when the ageing and nonageing unavailabilities are less than approximately 0.1. For this small exponent situation,

$$\frac{dq}{dt} - \frac{dq_o}{dt} \simeq \lambda(t-t_o) - \lambda_o; \qquad (27)$$

and the risk sensitivity is then approximately

$$r \simeq \frac{dR}{dq} (\lambda\ (t-t_o) - \lambda_o) \qquad (28)$$

Thus, to a first order the component ageing risk sensitivity depends only on the restoration time t_o and does not depend on any other test time t_c which does not renew the component.

5.3.4.3. Application Using the Linear Ageing Model

For application, specific ageing models for $\lambda(t)$ need to be substituted into the equations for the risk sensitivities (equations 19 and 28) to obtain specific numerical results which can then be used to rank the influence of ageing of specific components on the plant risk.

A variety of time dependent component failure rate models exist for treating potential ageing effects. These models include Weibull distribution, the gamma distribution, and the truncated normal distribution. They require, however, a knowledge of detailed time to failure data and cannot utilize gross data such as LER and NPRDS data which often do not include detailed information on causes of failure and ages of failed components. The linear ageing failure rate is the simplest time dependent failure rate that can be derived from gross data.

The Linear Ageing Model

The linear ageing model has a phenomenological basis which is developed in NUREG/CR-4769 [26]. As a specific application, the linear ageing failure rate is used for $\lambda(t)$ to obtain results for the ageing risk sensitivities.

The linear ageing failure rate $\lambda_a(t)$ describes the ageing portion of the total failure rate and is given by:

$$\lambda_a(t) = at \qquad (29)$$

where the constant "a" on the right hand side can be termed the ageing acceleration rate, or simply the ageing rate. The total failure rate $\lambda(t)$ with the linear ageing contribution added to the constant random failure rate λ_o is then

$$\lambda(t) = at + \lambda_o \qquad (30)$$

The linear failure rate can be simply viewed as a straight line fit to the wear-out portion of the standard bathtub curve (Fig. 5.3.3). The linear ageing model is suitable for modelling ageing mechanisms which cause a cumulative degradation in the component so as to continually increase its failure rate. The model is thus applicable to such ageing processes as linear wear, linear material buildup and linear elastic related phenomena. It can also serve as a first order linear approximation, even where the degradation buildup is not linear or independent of the previously-accumulated damage, such as that due to vibration. The inaccuracies due to using the linear model will often be overshadowed by the data uncertainties.

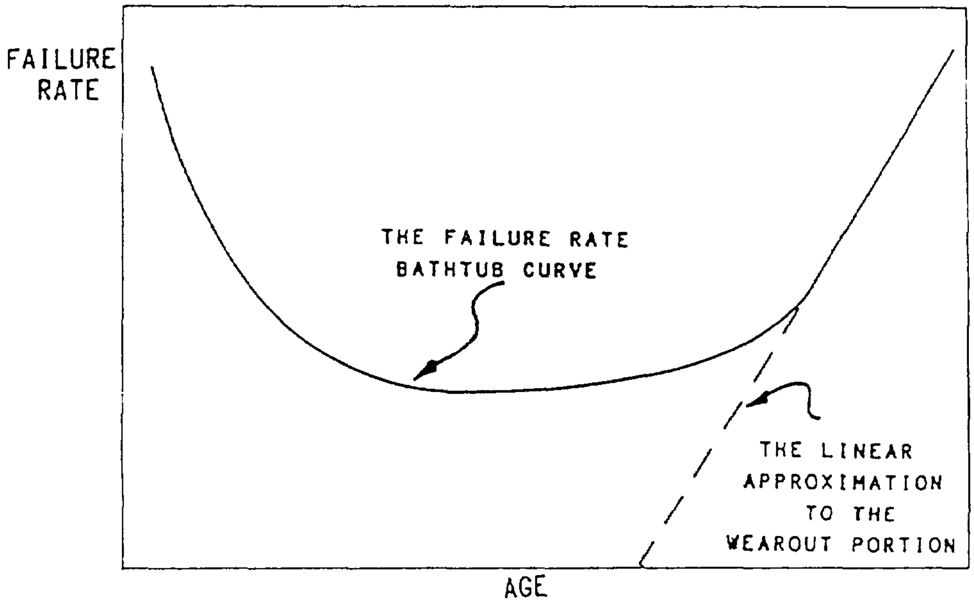

Fig.5.3.3. The bathtub curve and the linear failure rate fit.

Ageing occurring only after some minimum age τ can be modelled using a threshold time τ as follows:

$$\lambda(t) = a(t-\tau) \qquad t \geq \tau \tag{31}$$

The linear ageing failure rate applies to a specific ageing mechanism. The total time dependent failure rate $\lambda_T(t)$ of a component having random failures described by a constant failure rate λ_o and acted upon by several ageing mechanisms is given by:

$$\lambda_T(t) = \lambda_o + \lambda_1(t) + \ldots. + \lambda_K(t) \tag{32}$$

where $\lambda_1(t)$, ..., $\lambda_K(t)$ are the ageing rates for the individual ageing mechanisms. The ageing rates can also be combined into one aggregate ageing rate if they are all linear or have the same non-linear power.

Incorporating a threshold into the failure rate and summing contributions from individual ageing mechanisms can accurately model a variety of complex ageing processes. This linear modelling is analogous to fitting a segmented straight line to a curve. As shown in NUREG/CR-4769 [26], the linear ageing model can be straightforwardly extended to incorporate non-linear and dependent ageing behaviours.

Ageing Rate Estimates from Gross Failure Data

Estimates of the component ageing rate "a" can be obtained from available failure data, such as ageing fractions and other gross or relative data, using the following formula:

$$a = f_A \frac{\lambda_o}{T_A}$$

where:

f_A = the fraction of ageing failures
λ_o = the total failure rate
T_A = the average age to failure.

Generic ageing rates were determined for selected components from the ageing failure data obtained from the Nuclear Plant Reliability Data System [51]:

Table 5.3.4 - Example of Generic Component Ageing Rates

Component	Ageing Rate (1/year^2)
Motor-driven pump	8.35E-03
Turbine-driven pump	4.55E-02
Check valve	3.78E-03
Motoroperated valve	6.74E-03
Airoperated valve	3.43E-03
Manual valve	3.88E-05
Diesel generator	6.49E-02
Air conditioner	9.82E-05
Circuit breaker	1.83E-04
Relay	5.99E-05
Battery	3.36E-04

These are aggregate ageing rates resulting from the various dominant ageing mechanisms for the specific components (Table 5.3.4). Estimates of generic failure rates for different ageing mechanisms (e.g., corrosion or foreign material buildup) were also determined.

Statistical tests of the adequacy of the linear ageing model performed to date (approx. ten) generally show that the model is consistent with the data. The linear modelling approach thus appears to be more than adequate for risk calculations where order of magnitude accuracy is generally sufficient.

Even though the linear ageing failure rate is rather simple in form it gives rise to a variety of reasonable shapes for the density function $f_a(t)$ for times (ages) to failure from the ageing mechanisms where

$$f_a(t) = \lambda_a(t)\exp(-\int_o^t \lambda_a (t')dt') \qquad (33)$$

$$= a\ t\ \exp - \frac{at^2}{2} \qquad (34)$$

Fig. 5.3.4 illustrates the various shapes of $f_a(t)$; as the ageing rate "a" decreases, the density is shifted toward higher ages and becomes more diffuse. The most likely time to failure is $1/\sqrt{a}$.

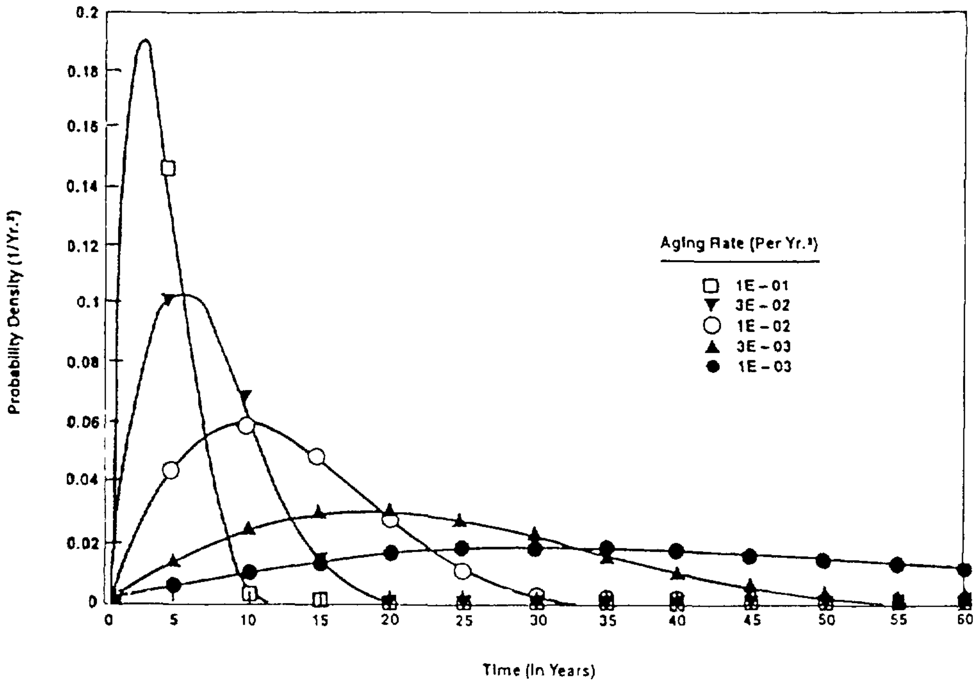

Fig.5.3.4. Time to first failure probability density functions resulting from different ageing rates for the linear ageing model.

Fig. 5.3.5 illustrates the component ageing risk sensitivities (r) calculated using the linear ageing model in Ref. [26] and the Calvert Cliffs PRA model described in Ref. [12]. The risk sensitivity calculated is the sensitivity of the core melt frequency. The sensitivities are grouped by type of component with the individual points showing the

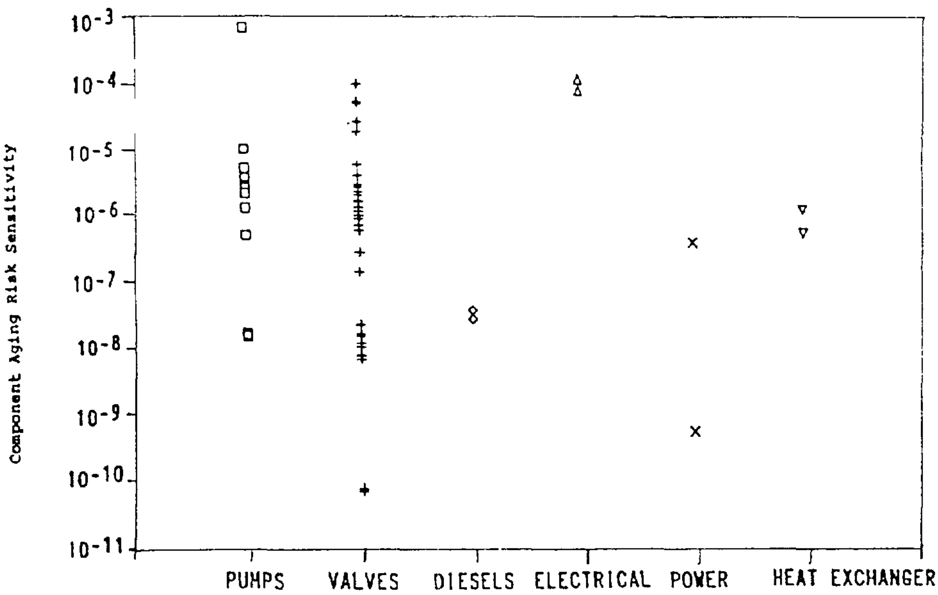

Fig.5.3.5. Component ageing risk sensitivities for the Calvert Cliffs PRA model and the linear ageing model.

individual component sensitivities. Since the PRA is an older one, Fig. 5.3.5 should be interpreted as showing general patterns and not precise numbers. As may be observed, there is a great variation in the component sensitivities, allowing important components to be meaningfully identified with regard to ageing risk effects, even with PRA associated uncertainties.

5.3.5. Comments

The applications of probabilistic techniques described in subsection 5.3 enabled determination of the relative importance to plant risk of the ageing of various plant components. In the United States, these evaluations in combination with the evaluations of operating experience and expert opinion provided information needed to establish general programme direction and to select component types to be addressed by research or design, operation and maintenance improvements.

6. MATERIAL AGEING MECHANISMS

6.1. Introduction

The components and structures used in nuclear power plants cover a broad range of materials and design which function in different applications and are exposed to different environments. For convenience this section concentrates on three major groups of materials:

- metals
- concrete structures
- non-metals (including plastics, elastomers, lubricants, etc.)

Before understanding the impact of ageing on component reliability one has first to understand the basic ageing mechanisms of constituent groups of materials resulting from service conditions, which are the cause of all ageing phenomena. Main examples of these conditions in the sense of ageing parameters are mechanical, thermal and residual stresses, temperature, electrical field, irradiation, humidity, chemically active gases and liquids, and relative motion. These are the driving forces for basic physical, chemical and mechanical processes and finally for the degradation of materials, components and equipment (Fig. 6.1).

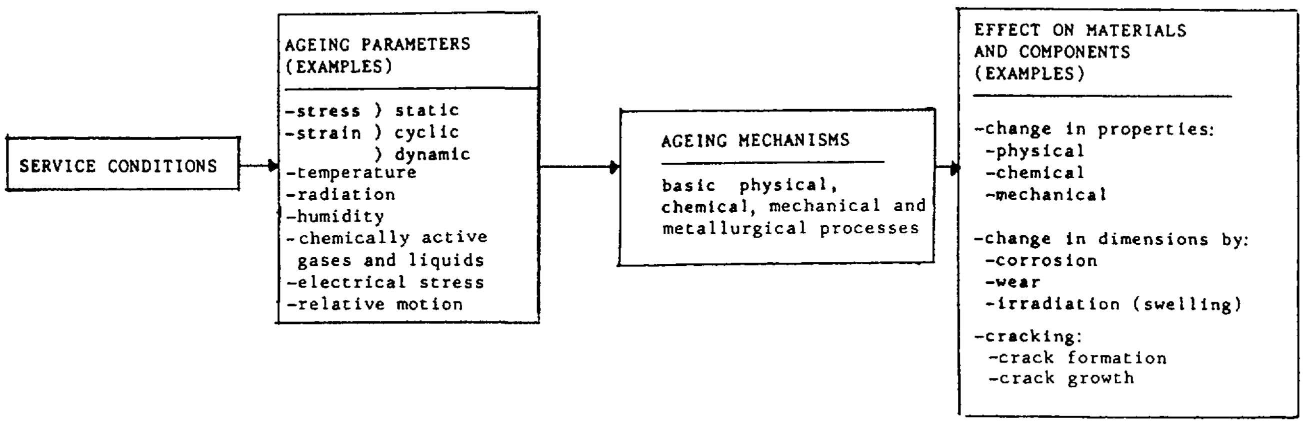

Fig.6.1. Examples of interaction between service conditions and effects on materials and components.

The present section deals only with the fundamental aspects of ageing of material itself and gives typical examples of the components on which they predominantly act, either as a single mechanism or in combination. When more than one ageing mechanism is operative the individual effects may not act independently but may combine to produce synergistic effects that cause greater degradation.

In follow-up reports, experience concerning the ageing of components and equipment will be compiled in more detail, and in particular with direct relation to safety and reliability.

6.2. Metal ageing

Metals, owing to their variety and their wide range of properties, are applied in all areas in a nuclear power plant. This is the reason why practically all potential ageing mechanisms have to be considered. These are shown in Fig. 6.2.1 and include:

- irradiation
- temperatures (thermal ageing and creep)
- complex loading, static, cyclic and dynamic (relaxation, creep, fatigue)
- aggressive environments such as gases, water, steam, liquid metals (corrosion)
- relative motion (wear).

EFFECT ON COMPONENT	POTENTIAL METAL AGEING MECHANISMS										
	6.2.1 IRRADIATION	6.2.2 THERMAL AGEING	6.2.3 CREEP	6.2.4 FATIGUE HCF LCF thermal fatigue	6.2.5 CORROSION corrosion fatigue	stress/strain corrosion cracking	intergranular corrosion	uniform corrosion attack	local corrosion attack	corrosion erosion	6.2.6 WEAR FRETTING FRETTING FATIGUE
REDUCTION IN THOUGHNESS	●	●									
CRACKING			●	●	●	●	○				●
GRAIN DISINTEGRATION							●				
SWELLING	●										
THINNING								●		●	●
DENTING								●			
PITTING									●	●	

Fig.6.2.1. Potential ageing mechanisms for metals and resulting effects on materials and components.

Depending on the materials selected and the intensity of the external stressors, there may be a high potential for metal ageing. The processes of ageing can result in material changes of which the most important are:

- reduction in toughness (embrittlement)
- cracking
- swelling
- thinning
- denting
- pitting.

Some typical examples of the occurrence of the aforementioned ageing mechanisms, from experience with LWRs and Pressurized Heavy Water Reactors are compiled in Figs. 6.2.2a, b, c. The figures illustrate that for most of the components more than just one mechanism is acting. It also shows that alternating loading (mechanical and thermal) in combination with the presence of a corrosive environment has to be considered for a great number of components.

COMPONENT	MATERIAL (EXAMPLE)	POTENTIAL METAL AGEING MECHANISMS: 6 2 1 IRRADIATION	6 2 2 THERMAL AGEING	6 2 3 CREEP	6 2 4 FATIGUE HCF LCF THERMAL FATIGUE	6 2 5 CORROSION: corrosion fatigue	stress/strain corrosion cracking	intergranual corrosion	uniform corrosion attack	local corrosion attack	corrosion erosion	6 2 6 WEAR FRETTING FRETTING FATIGUE
1 Reactor Vessel												
1 1 LWR Reactor Pressure Vessel (RPV)												
1 1 1 Vessel Shell	ferritic fine grained high strength steel	■	□		■	■						
	austenitic cladding	■			■	■		□				
1 1 2 RPV Internals												
1 1 2 1 Grid Assembly Core Support	austenitic steel	■			■	■	□					
1 1 2 2 Bolts	austenitic steel nickel-chromium alloy	■			■	■						

Fig.6.2.2.a. Examples of NPP components and their potential ageing mechanisms.

COMPONENT	MATERIAL (EXAMPLE)	POTENTIAL METAL AGEING MECHANISMS: 6 2 1 IRRADIATION	6 2 2 THERMAL AGEING	6 2 3 CREEP	6 2 4 FATIGUE HCF LCF THERMAL FATIGUE	6 2 5 CORROSION: corrosion fatigue	stress/strain corrosion cracking	intergranual corrosion	uniform corrosion attack	local corrosion attack	corrosion erosion	6 2 6 WEAR FRETTING FRETTING FATIGUE
1 2 PHWR Reactor Vessel												
1 2 1 Calandria Vessel	austenitic steel	■		■	■	■	■	■				
1 2 2 Pressure Tubes	Zirconium-Niobium alloy	■		■	■	■						
1 2 3 Calandria Tubes	Zirconium alloy	■		■	■	■						
2 Steam Generator												
2 1 Vessel Shell	ferritic fine grained high strength steel				■	■	■		■	■		
2 2 Tubes	Inconel 690 Incoloy 800				■	■	■	■	■	■		■
2 3 Grid	austenitic steel				■	■	■					
	ferritic steel				■	■	■		■			

Fig.6.2.2.b. Examples of NPP components and their potential ageing mechanisms (cont.).

COMPONENT	MATERIAL (EXAMPLE)	6.2.1 IRRADIATION	6.2.2 THERMAL AGEING	6.2.3 CREEP	6.2.4 FATIGUE HCF LCF THERMAL FATIGUE	6.2.5 CORROSION: corrosion fatigue	stress/strain corrosion cracking	intergranual corrosion	uniform corrosion attack	local corrosion attack	corrosion erosion	6.2.6 WEAR FRETTING FRETTING FATIGUE
3 Steam and Water Piping Vessels Valves	austenitic steel				■	■	■	■		■		
	ferritic steel with austenitic cladding		□		■	■	■	□		■		
	ferritic steel				■	■	■		■	■	■	
4 Main Coolant Pump												
4.1 Casing					■	■						
4.2 Shaft Impeller	alloyed steel				■	■						
5 Turbine												
5.1 Casing	cast steel								■		■	
5.2 Blades	alloyed steel				■	■	■				■	■
5.3 Shaft	alloyed steel				■	■						
6 Condensor Tubes	austenitic steel brass titanium				■	■	■		■	■	■	

Fig.6.2.2.c. Examples of NPP components and their potential ageing mechanisms (cont.).

In the following subsections a short description of the physical, chemical and metallurgical processes and their consequences for material degradation and the reduction of component reliability and safety are presented.

6.2.1. Irradiation

Neutrons from the fissions process which cover an energy range up to 15 MeV interact with the material which results in the formation of atomic displacements and, depending on the energy of the neutrons, in the formation of displacement cascades. This creates areas of low atomic density (high vacancy concentration) and of high atomic density (interstitial atoms). The contribution of the associated gamma irradiation to the ageing phenomena amounts to less than 2%. At service temperature some of the lattice damage created can anneal in a diffusion process, with the result that the residual damage tends to an equilibrium between the creation of new defects and annealing (recovery). Neutron flux and irradiation temperature, therefore, play an important role. In addition to the lattice defects caused by the interaction of fast (high energy) neutrons with the material atoms, helium is produced as a result of nuclear [n, alpha] reactions. This damage mechanism comes into play at high fluences above 10^{22} cm^{-2} and high temperatures usually above 400°C. Helium diffuses under elevated temperature easily and forms voids. Accompanied by the high temperature irradiation, irradiation assisted creep and microstructural changes like precipitation can occur.

The effect of neutron irradiation on metals is mainly to increase the yield and the ultimate strength and to reduce the toughness. Helium production not only leads to a change in material properties but is also combined with an increase in volume (swelling).

Examples of materials used in areas where they are exposed to neutron irradiation are:

- zirconium alloys for fuel cladding and CANDU pressure tubes
- austenitic stainless steel for reactor pressure vessel internals and reactor pressure vessel cladding
- fine grained high strength carbon steel for the reactor pressure vessel shell and CANDU feeder pipes.

The ageing sensitivity of the materials depends among other things on:

- the type of material
- its chemical composition
- heat treatment
- initial mechanical properties.

In the case of ferritic steels for reactor pressure vessels, irradiation embrittlement causes most concern about the integrity of the vessel. The effect of embrittlement is commonly determined with Charpy-V notch specimens in impact tests, Fig. 6.2.3. Drop in upper shelf energy and increase in ductile/brittle transition temperature are the main-criteria to describe the degree of ageing. The copper and phosphorus contents of the steel are considered to be the main factors responsible for neutron embrittlement.

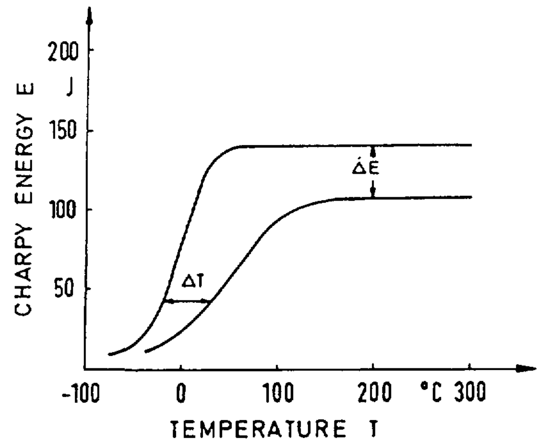

Fig. 6.2.3. Change in material toughness due to neutron irradiation measured in Charpy-V notch impact test

Recent analysis of data indicates that copper and nickel have a synergistic effect. Trend curves are available to predict the amount of embrittlement in terms of the increase in the nil ductility transition temperature (T_{NDT}) and the drop in Charpy upper shelf energy, Figs. 6.2.4 and 6.2.5.

For steels neutron irradiation is considered to be a significant ageing factor for metals only at fluence exceeding $10^{17}cm^{-2}$. The degradation of materials of pressure retaining systems (base metal, welds and heat affected zone) exceeding the fluence limit of $10^{17}cm^{-2}$ are usually investigated in in-reactor irradiation surveillance tests, where irradiation occurs under slightly accelerated conditions.

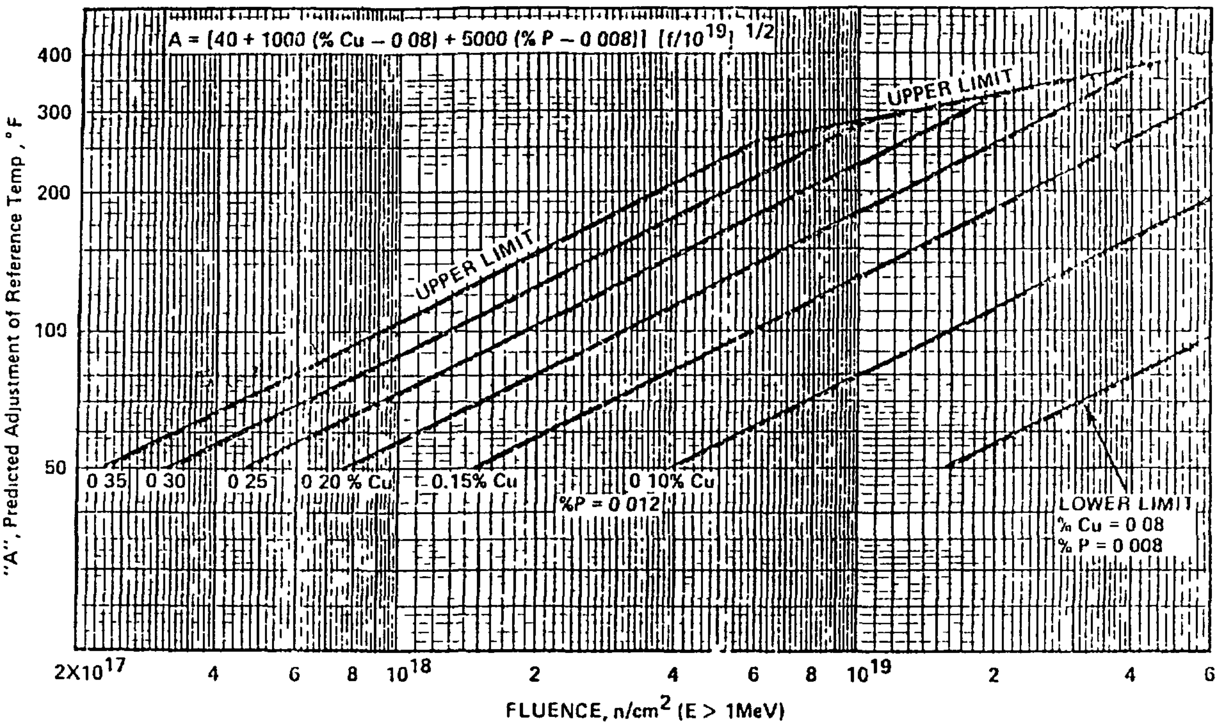

Fig. 6.2.4. Example of design curve to predict transition temperatur shift determined with Charpy-V notch specimens as a function of fluence (acc. to US Reg.Guide 1.99 Rev. 1)

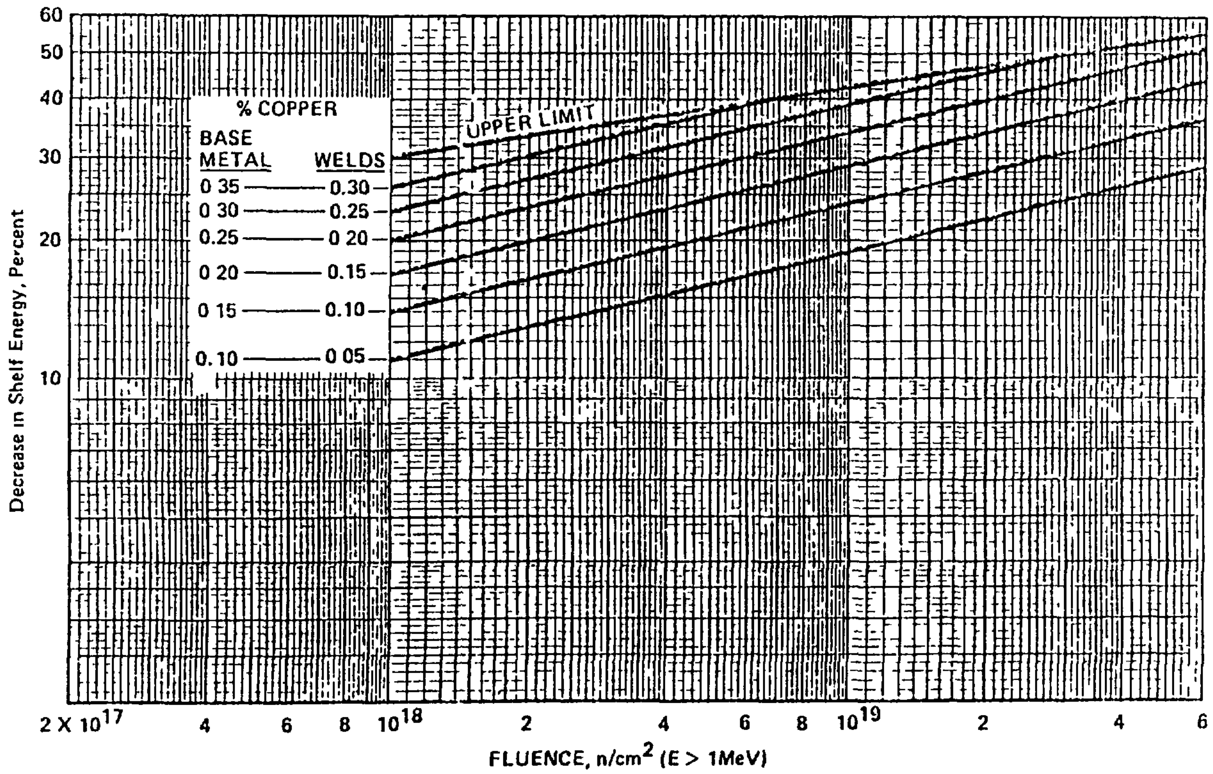

Fig. 6.2.5. Example of design curve to predict upper shelf energy drop determined with Charpy-V notch specimens as a function of fluence (acc. to US Reg.Guid 1.99 Rev. 1)

6.2.2. Thermal ageing

Thermal ageing refers to all thermally activated movements of the lattice atoms during a long time range which can occur without external mechanical load. Changes of material properties are the consequence of these diffusion processes. The dominating parameters responsible for these ageing processes are

- level of temperature
- material state (micro structure)
- time.

In addition to these parameters the environment can have effect on the changes in material state in case of superimposed oxidation and de-oxidation processes. The combination of different parameters can lead to a variety of ageing mechanisms and phenomena.

Thermal ageing in general is a degrading process commonly causing decrease in strength properties, hardness, ductility and toughness. Under certain conditions, however, an increase in ductility can be observed, too. One of the dominating factors is the chemical composition of the material and its thermo-mechanical pre-service treatment.

The mechanisms occuring in the micro structure are of the following types

- precipitation of particles
- transformation of phases
- growth of precipitated phases and the grains of the matrix
- dissolution of precipitates.

The rate of those reactions is strongly dependent on the level of temperature. Also threshold temperatures can exist below which some of the processes are not being activated. The local chemical composition and the diffusion coefficient of the different phases and atoms play an important role. They depend on the degree of over saturation as it is the case e.g. for growth of carbides.

More complex mechanisms occur in case of superimposed mechanical loading because of the interaction of diffusion processes and dislocation movement (see paragraph 6.2.3 creep). The precipitation of carbides can restrain the movement of dislocations (embrittlement) and the first precipitates can act as nucleus for accelerated precipitation.

A variety of thermal ageing processes occur when the material was subjected to cold working or plastic deformation prior to the temperature exposure. Examples are crystal recovery and recrystallization. Those processes can already occur during the production stage, e.g. during welding, where a wide temperature range is applied. Nitrogen containing steels tend to temper embrittlement caused by the precipitation of nitrides in the temperature range between 200°C and 500°C and has its origin in the competing diffusion of carbon nitrogen and of the dislocations which occur with similar velocity. The result on the material is a strong reduction in toughness combined with an increase in strength.

Some of the thermal ageing processes like temper embrittlement and recrystallization, are of the short time range type and only controlled by temperature. Thermal ageing effects of advanced steels are to be expected only after an extremely long period of time. A prerequisite for steels to be unsusceptible to thermal embrittlement is a stable precipitation state and e.g. for ferritic steels an adequate annealing treatment at temperatures higher than the service temperature prior to the application in service.

6.2.3. Creep

Creep is defined as a time and load dependent plastic deformation of the material. Creep is an ageing process because, in addition to changes in dimensional stability, changes in material properties can occur which lead to a decrease in loadability and remaining toughness. For metallic materials creep becomes a matter of technical concern when the operating temperature exceeds about 40% of the absolute melting temperature of the material. The parameters affecting creep depend on the material itself and also on characteristics of loading such as the temperature and the mechanical load. The following mechanisms can occur in metallic materials and are responsible for the deformation and material damage process:

- movement of dislocations
- diffusion processes
- grain boundary slip processes.

These three mechanisms usually occur simultaneously. The process which is finally dominant depends on a variety of factors such as stress, time, temperature, grain size, impurities, precipitations, secondary phases, etc. The nucleation of voids and pores is a consequence of the aforementioned acting deformation mechanisms.

Basically, two processes occur in the microstructure that affect the material properties:

- plastic deformation processes
- changes in the microstructure.

Both these processes are time dependent and they show certain interactions. The changes in microstructure can be caused by high temperatures alone, since they are the consequence of thermally activated mechanisms.

Changes in the microstructure can be characterized by changes in the structure due to precipitation or dissolution of foreign atoms, coagulation of carbides and oxides, or grain growth, and for some materials by internal oxidation processes. For ferritic bainitic steels, for example, a complete dissolution of the carbides can be observed owing to the disintegration of the bainite after a long exposure time.

These thermally activated processes in the microstructure result in the degradation of the material's characteristics, mainly its mechanical properties. No general quantification of these effects can be given owing to the strong dependence on the material. Qualitatively a decrease in strength occurs for steels and in some cases a reduction in toughness.

The creep damage processes - mainly the time dependent deformation - occurs according to the following steps:

- Deformation due to creep divided into three different regimes (see Fig. 6.2.6) controlled by different mechanisms. Components are usually designed to be operated only in the primary and secondary regimes. In either one of any two regimes any damage is still not irreversible;

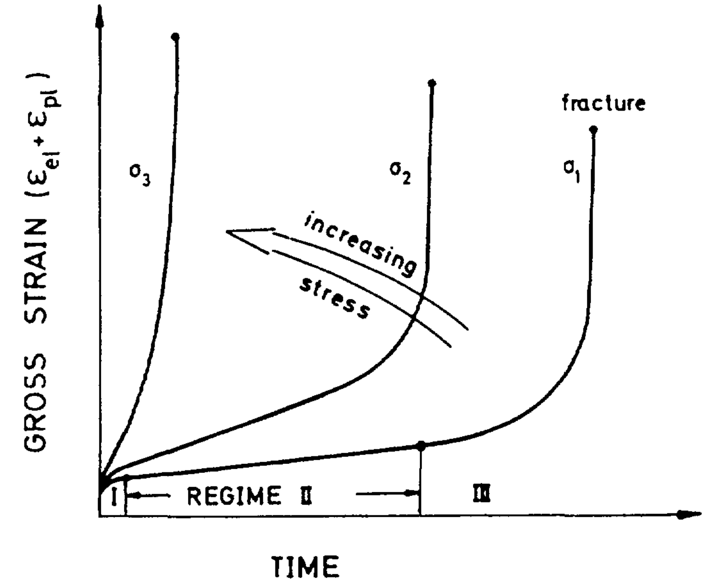

Fig. 6.2.6. Typical creep behavior of steels governed by different mechanisms in regimes I through III

- Nucleation of creep pores near the end of the secondary regime;
- Coagulation of the creep pores to form microcracks;
- Crack growth due to creep. In this regime spontaneous failure can occur when large sections of the material have already suffered from this process;
- Creep rupture. With increasing exposure time and accumulation of microstructural damage (grain boundary slip) the ability for creep deformation decreases and thus ruptures occur in the part that is only a little ductile.

Superimposed processes

An aggressive environment can have an influence on the creep behaviour and the degree of damage resulting from it. In metallic materials an internal oxidation process can occur and accelerate the creep process.

The initial creep strength depends on the initial strength of the material at the envisaged temperature. If the thermally induced process (thermal ageing) leads to a decrease in strength, then more unfavourable conditions have to be expected concerning the long range creep behaviour.

In practice the loading situation does not generally represent a steady state condition owing to startup and shutdown processes. This results in a complex loading situation ranging from creep to fatigue.

For this case, the lifetime assessment has to account for the stationary situation which leads to creep and the nonstationary condition also leading to fatigue. For such cases of creep fatigue interaction special constitutive laws have to be considered in respect of damage mechanism and the integral lifetime.

6.2.4. Fatigue

Components subjected to repeated loading can degrade by cracking and can finally come to fracture even when the stresses resulting from each load peak are far below the material's ultimate strength.

The damage process is initiated by locally limited plastic deformation in the microrange of individual crystallites at low nominal net stresses. Depending on the lattice orientation and the dislocation configuration, slip bands can be activated causing irreversible slip processes. Repeated loading leads to microdamage accumulation (fatigue). After a saturation of energy accumulation has occurred at which no further slip process can take place in the crystallite, microcracks are produced. With an increasing number of cycles these microcracks grow and additional regions undergo the microdamage process. The combination of a more intense stress field around the microcrack and the progressive local decrease in the ability of the material to respond by plastic deformation, the microcracks grow into predamaged regions and finally reach macroscopic size. As the crack grows the stress intensity at the crack tip usually increases so that crack growth is accelerated until the remaining ligament can no longer bear the load peak and complete failure occurs.

The fracture surface features a mainly smooth region with no macroscopic deformation (the region of fatigue cracking) and a rough region which results from spontaneous failure. This portion of the fracture surface can show macroscopic and microscopic deformation in the case of tough materials.

The behaviour of a material under fatigue loading is described by the Woehler Diagram, (see Fig. 6.2.7). The lower the macroscopic stress peaks, the higher is the number of cycles to failure. Below a certain material dependent stress no failure occurs. This stress level is called the fatigue strength. For steels which are stressed at the level of fatigue strength an infinite number of cycles can be applied. To check out this threshold, it is sufficient to perform tests up to 10^7 load cycles. For aluminum alloys, however, there exists no such threshold. Similar behaviour has also to be considered in case of superimposed corrosion effects (see subsection 6.2.5.2).

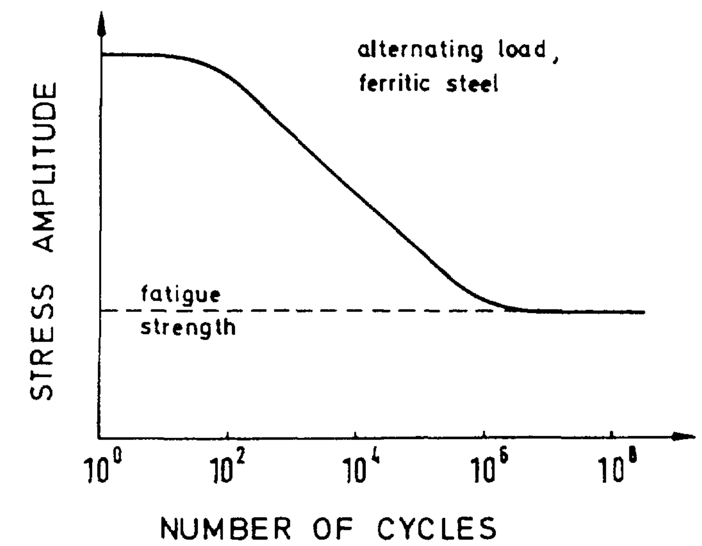

Fig. 6.2.7. Correlation between applied stress amplitude and the number of load cycles to failure (Woehler Diagram)

Besides the level of stress and the number of cycles, the following parameters are of influence for the failure behaviour.

- The loading situation:
  - mode of loading - tension, compression, torsion;
  - level of loading - mean stress, stress amplitude.
- The material:
  - with increasing ultimate strength the fatigue strength increases;
  - material inhomogeneities, internal imperfections and surface flaws cause a decrease in fatigue strength.
- The design and surface quality:
  - the absolute dimensions of the cross-section have to be considered with regard to technological effects and stresses and constraints;
  - spontaneous (abrupt) changes in cross-section, cracks and notches act as stress raisers and reduce the fatigue strength;
  - a rough surface includes the effect of micronotches and thus reduces the fatigue strength.
- Residual stresses:
  - compression stresses have a positive effect on the fatigue strength, tension stresses have a negative effect.
- Environment:
  - temperature affects the fatigue strength as the tensile properties (yield and ultimate strength) decrease;
  - corrosion gases or liquids can drastically reduce the fatigue strength.

According to the number of cycles after which failure occurs, two regimes have to be distinguished:

- high cycle fatigue;
- low cycle fatigue.

In high cycle fatigue, where machine parts in motion are usually operated like rotors and axles or structures which are excited by motional parts, the design has to take into account the high number of cycles during the lifetime. The dimensions and materials have to be designed according to the high cycle fatigue rules (fatigue strength). If there has been a complex loading history a damage accumulation model has to be applied.

In the low cycle regime machine parts are designed for a limited number of cycles and individual load peaks may be applied that exceed locally the yield strength of the material. Such conditions can occur as a result of transients which cause inhomogeneous mechanical and thermal stresses and strains. In the low cycle regime with load peaks exceeding the yield strength there is not a linear correlation between stress and strain.

Material characteristics are usually determined in tests with a constant strain amplitude. During the test the stress amplitude can either increase owing to strain hardening or decrease owing to strain softening.

If during the lifetime of component only a limited number of cycles has to be taken into account, the components have to be designed not according to the fatigue strength but only to the fatigue strength for a finite lifetime, e.g. for a finite number of load events.

For reactor pressure vessel steels the ASME III design curve represents a limit curve from which the allowable number of cycles can be taken for a given stress amplitude (Fig. 6.2.8). By comparing design stress and in-service strain measurements with the allowable stress/strain, the degree of utilization (ratio of the number of cycles that have occurred to be designed number of cycles) can be determined.

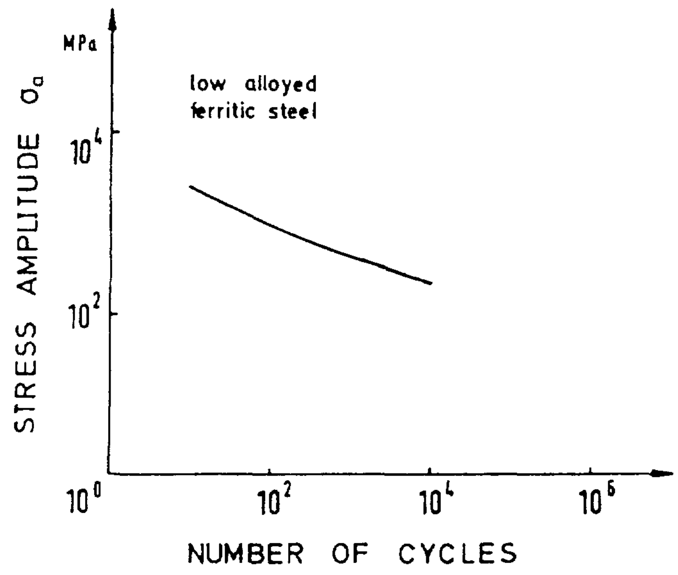

Fig. 6.2.8. Design curve for allowable strain amplitude as a function of the number of cycles in the low cycle regime

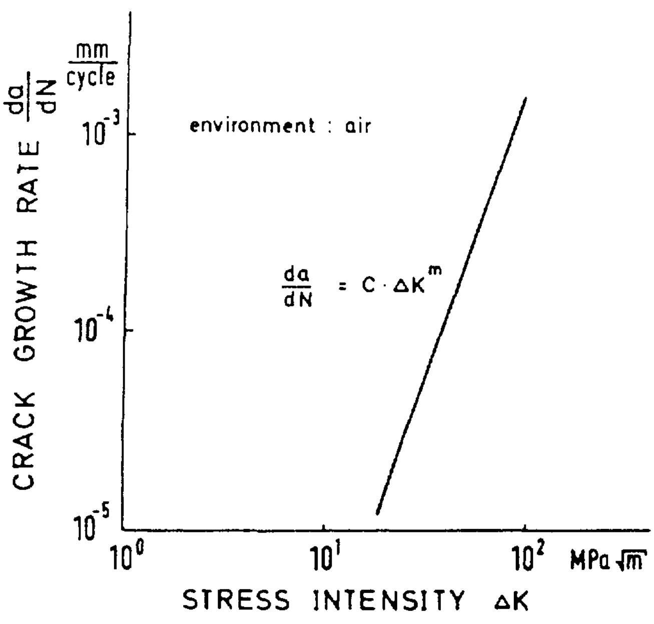

Fig. 6.2.9. Cyclic crack growth rate as a function of stress intensity amplitude

If a crack has already been initiated or has to be assumed to be initially present, crack growth occurs under the cyclically applied load during each load cycle depending on the stress intensity amplitude ΔK. The crack growth can be described by a power law in the form;

$$\frac{da}{dN} = C \cdot \Delta K^{m}$$

da/dN is the crack growth during one load cycle and C is a material constant depending on the material's state and temperature (see Fig. 6.2.9). The environment can have a strong effect on the cyclic crack growth rate by the mechanisms described in subsection 6.2.5.2. The crack growth rate can be increased by environmental effects by an order of magnitude or even more.

6.2.5. Corrosion

Corrosion is the reaction of a metal with its environment that causes a detectable change in the material and can lead to a deterioration in the function of a structure or of a complete system. In most cases this reaction is of an electrochemical nature. In some cases, it can also be of a chemical (but not an electrochemical) or of a metal physical nature. An example for a chemical reaction is the oxidation of iron in hot and dry gas (scaling). Most of the technical corrosion processes, however, are electrochemical in nature and are accompanied by hydrogen production.

The appearance of the corrosion is governed by the so-called corrosion system consisting of the metallic material and the corrosive medium (the environment) with all the participating matter that can influence the chemical and physical behaviour and the corrosion parameters.

The variety of chemical and physical variables of environments and materials leads to a large number of types and appearance of corrosion. It can be subdivided into:

- Corrosion without mechanical loading:
    - uniform corrosion attack;
    - local corrosion attack;
    - selective corrosion attack, especially intergranular corrosion.
- Corrosion with additional mechanical loading:
    - stress/strain corrosion cracking;
    - corrosion fatigue.
- Corrosion erosion.

In the case of electrochemical processes, corrosion occurs in several steps. Metal ions dissove in liquid electrolyte (anodic dissolution) and hydrogen is produced in the process. This is the process of material loss from the corroded component and of the creation of corrosion products. The free hydrogen can metallurgically interact with the metal (adsorption, absorption or interstitial bonding) and can produce secondary damage, e.g. hydrogen embrittlement.

When mechanical stresses or strains are acting in addition to the corrosion impact, the anodic dissolution of the metal can be stimulated, protection layers can rupture or hydrogen absorption can be promoted. The combined action of a corrosive environment and mechanical loading can cause cracking even when under either the chemical or the mechanical conditions alone no material degradation would occur. Ageing effects through corrosion are not only of concern with regard to material dissolution and cracking but also if the function of a machine part is affected by the build-up of reaction products.

6.2.5.1. Types of corrosion without mechanical loading

Corrosion without mechanically superimposed loading can commonly lead to uniform material loss, shallow pit formation, pitting or selective attack at the surface.

Uniform corrosion attack

In this case the metal is macroscopically uniformly removed. However, microscopically the corrosion attack is not as uniform. It can range from an increase in surface roughness, including local shallow pit formation and selective dissolution of particular material phases. Examples of uniform corrosion attack in light water reactors with the potential for functional limitations are the oxidation of fuel element cladding and the reduction of the wall thickness of steam generator tubes, with additional formation of wastage on the tube sheet.

All hot water and steam pipes of unalloyed or low alloyed ferritic steel are subjected to the formation of magnetite protection layers as a consequence of the reaction between water and iron (Schickor reaction) at operating temperature. Since these layers are insoluble and adhere well to the surface, the reaction is slowed down and even comes to a standstill. This process, which also includes a small amount of material loss, is not considered an ageing process since it saturates and the material loss amounts to only a few tenth's of a millimetre.

When the magnetite layer is locally ruptured or removed by an erosion process, the corrosion attack can be exacerbated and the process of material removal can be accelerated (e.g. in corrosion erosion, see subsection 6.2.5.3).

In general, corrosion products occupy a larger volume than that of the metal itself. In crevices or for machine parts (guiding parts) with a given clearance, the formation of reaction products can fill the clearance with the consequence of buildup of pressure. In the case, for example, of valve internals which only have to be moved occassionally, the building up of reaction products at the internal guiding parts or at the valve seat can increase the internal friction of the valve so that in extreme situations the valve cannot carry its the required function.

At ferritic spacers in steam generator tube holders, the building up of reaction products can result in such a high pressure that the thin walled tube is dented.

Local corrosion attack

In case of inhomogeneities at the metal surface and/or local differences in the electrochemical reactivity of the environment the creation of local cells is possible which results mostly in local corrosion attack. This causes locally different corrosion rates or corrosion attack only at certain locations. The corresponding manifestation of corrosion are shallow pit formation or pitting. Pits can start with a small diameter and can extend beneath the surface. A well defined differentiation between pitting and shallow pit formation is not possible in all cases.

Shallow pits occur primarily at unalloyed steels exposed to a water environment. The corrosion cells are mainly formed in conjunction with local wastage and protection layers. Shallow pit formation is usually combined with more or less uniform material loss.

The typical pitting is not combined with uniform material loss in the vicinity of the pit. For chromium alloyed steels or chomium nickel alloyed steels pitting occurs mainly owing to the action of chloride ions. For unalloyed or low alloyed steels pitting can often be observed as a consequence of idle corrosion in water containing oxygen.

Local corrosion can also have its origin in design features. The corrosion in crevices is well known (crevice corrosion) and is caused by corrosion cells which are nucleated by differences in concentration in the corrosive environment. The corrosion manifested in the crevice can be uniform, shallow pitting or selective.

Selective corrosion

In selective corrosion the attack is concentrated on distinct material phases, regions adjacent to the grain boundaries or specific alloying elements.

The corrosion proceeds into the depth of the material without changing the shape or geometry of the part. As long as the selective corrosion attack is limited to a certain depth, usually no material loss from the surface occurs.

The selective attack itself within a range of only a few grains in depth does not generally affect the function of the part. However, if the attack proceeds along the grain boundaries into its depth the structure is weakened and the material from regions near to the surface can be removed, leading to a reduction in wall thickness. A well known type of selective corrosion is the dezincification of brass (such as in heat exchanger tubes in condensators and preheaters) and the intergranular corrosion (IC) of sensitized austenitic stainless steels. Since the corrosion attach takes place along the grain boundaries, the material damage proceeds from the surface into the bulk according to a grain disintegration process.

The occurrence of IC depends on the following conditions:

- a susceptible (sensitized) material state;
- a corrosive environment (electrolyte).

Non-stabilized austenitic stainless steels with a carbon content greater than 0.04% can develop a sensitized material state when slowly cooled from temperatures above 1000°C or during a long holding time at temperatures between 450°C and 850°C. Mixed carbides with high chromium contents are formed, leading to a chromium depletion along the grain boundaries (sensitization by chromium depletion). This process can occur during heat treatment (furnace sensitization) or during welding (welding sensitization). In the presence of an electrolyte a corrosion cell can be formed and the chromium depleted regions are dissolved by an anodic process.

Corrosive environments (liquids) that can cause intergranular corrosion in practice can be due to, among others, oxygen, chlorides, sulphates and hydroxides. Dust from industrial pollution which contains one or more of these compounds has to be considered with regard to corrosion attack in combination with dew or condensate. The process of intergranular corrosion is not usually associated with significant corrosion products.

Not only austenitic stainless steels but also nickel alloys can be affected by intergranular corrosion. As well as the carbide precipitation, the formation of intermetallic phases is also considered to be responsible for the process.

6.2.5.2 Corrosion with additional mechanical loading

If there are static or cyclical mechanical stresses or strains in addition to the corrosive environment the corrosion attack can be enhanced owing to local disruption of the protection layer, and crack formation and crack growth can be accelerated.

According to the parameters responsible for the corrosion process, the following mechanisms have to be differentiated:

- Stress corrosion cracking (SCC)
  - anodic SCC
  - cathodic (hydrogen induced) SCC;

- Strain induced corrosion cracking (SICC);

- Cyclic crack growth.

Stress corrosion cracking (SCC)

Crack formation caused by stress corrosion cracking can be of the transgranular or intergranular type and occurs under static tension stresses. A characteristic of SCC is an almost brittle fracture mode without detectable corrosion products. The tension stresses often result from residual stresses in the component, or at least they can be superimposed on the load stresses. Even in the case of static load in combination with a low cyclic loading the cracking mechanisms can be referred to SCC. The change from SCC to cyclic crack growth is not easy to define, however.

The special case of anodic stress corrosion cracking occurs as a combination of the material, the environment and mechanical loading, in critical boundary conditions:

- the material must be sensitive to SCC;
- tension stresses must be sufficiently high;
- the environment must lead to a specific material reaction as, it is the case for the following combinations:
  - carbon steel with nitrides or sulphide stringers
  - austenitic steel with chlorides
  - brass with ammonia.

For some combinations of materials and corrosive environments even very low stresses are sufficient to cause SCC. The phase of first macroscopic cracking is often preceded by a long incubation phase.

If cracking occurs, whether it is intergranular or transgranular in type depends strongly on the system material/corrosive environment. Severe cases of intergranular stress corrosion cracking have occurred in piping of non-stabilized austenitic steel under the water conditions in a boiling water reactor (BWR).

Concerning the corrosion mechanism, the electrolytic (anodic) and the metal physical (hydrogen induced) cracking processes must be distinguished.

In case of hydrogen induced SCC the additional effect beyond the simple SCC exists in an interaction of hydrogen with the metal lattice. This type of corrosion attack requires a sensitive material state in which hardness and microstructure play an important role. The level of mechanical stresses resulting from both active load stresses and residual stresses is of minor importance compared with the reaction of hydrogen. Any corrosive environment that produces hydrogen as a result of a corrosive surface attack can initiate this process. A prerequisite for hydrogen induced SCC is a small uniform or local corrosion attack of the material or a reaction between reaction layers and the electrolyte with subsequent hydrogen production.

Stress corrosion cracking occurs in systems that usually have good resistance to uniform corrosion, such as austenitic stainless steel with passive layers. Nevertheless, SCC can be observed in such steels in combination with pitting.

Strain induced corrosion cracking (SICC)

This type of environmental assisted cracking can occur under monotonously increasing load or low cyclic load, causing plastic deformation at low strain rate. This mechanism comes into action particularly under a critical strain rate and the presence of an electrochemical environment (see Fig.6.2.10). Transgranular cracking with only a small amount of deformation is the typical manifestation of this mechanism.

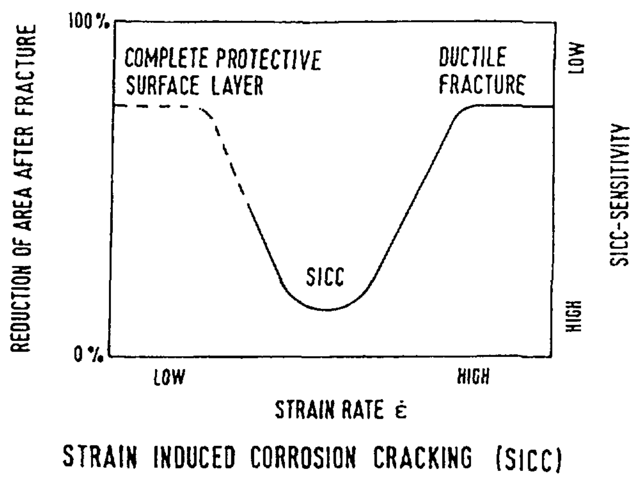

Fig.6.2.10. Effect of strain rate in uniaxial tension test on the reduction of area in corrosive environment

Strain induced corrosion cracking was found in piping and vessels of unalloyed or low alloyed ferritic steels in high purity water and water in boiling water reactors. Cracking occurs under the simultaneous action of:

- temperatures above 150°C
- an oxygen content in the water greater than 50 ppb
- a strain rate of the order of $10^{-4}\ s^{-1}$.

The susceptibility to SICC is high, particularly in stagnant water in which the oxygen content is highest. At high flow rates the conditions are less likely to lead to SICC.

High local stresses and limited plastic strains in components with thin walls are mainly produced by stress raisers in weld joints such as misalignment, inadequate root of the joint, formation of root ridges and initial cracks from the production stage. Pits resulting from corrosion often act as crack initiators for SICC. Complex thermal stresses and mechanical loading situations, as in regions of pipe fixtures or in elbows under external bending, can provide the critical strain level and strain rate for SICC.

Corrosion fatigue

In the case of fatigue loading, additional environmental effects can cause the number of cycles to crack initiation to be drastically reduced from that for an inert atmosphere. Even low stress amplitudes in the regime below the fatigue strength can cause failure after a high number of cycles, depending on the frequency. Instead of a fatigue strength which is independent of the number of cycles, a finite number of cycles has to be considered for a given stress amplitude. The interaction of mechanical alternating stresses and corrosion attack usually leads to transgranular cracking with a low associated deformation. In contrast to stress corrosion cracking, no threshold conditions exist for corrosion fatigue with respect to the corrosion system and the stress amplitude. Corrosion fatigue is not only initiated by a specific environment, as is the case for stress corrosion cracking, but can also occur in non-specific media, e.g. pure water, humid air.

According to the corrosion conditions, two types of corrosion fatigue have to be distinguished:

- corrosion fatigue under active conditions in combination with a detectable surface corrosion attack, mainly in the form of pitting and the formation of corrosion products;
- corrosion fatigue under passive conditions where no corrosion attack at the surface or in the crack channel is to be detected.

Since the corrosion mechanism is mainly time dependent and the fatigue mechanism is controlled by the number of cycles, a complex interaction of both mechanisms occurs in the regime of corrosion fatigue which leads to the frequency dependence. A decreasing frequency can drastically reduce the number of cycles to crack initiation and failure (see Figs. 6.2.11.a and b).

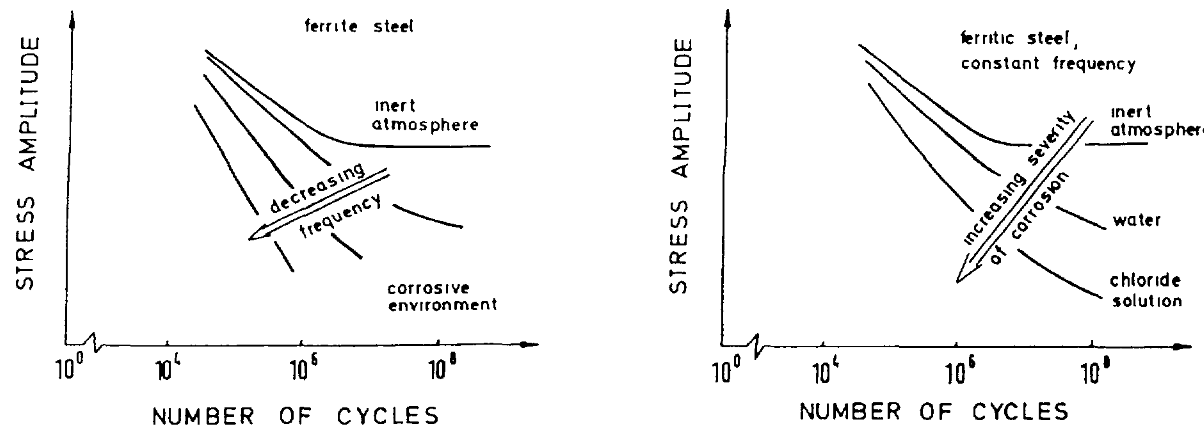

Fig. 6.2.11 a and b. Effect of frequency on the number of cycles to failure in corrosive environment

It is not only the number of cycles to crack initiation which is reduced by corrosive environments. The simultaneous corrosion attack also influences the cyclic crack growth rate. For the same stress intensity amplitude ΔK the subcritical crack growth per load cycle increases in a corrosive environment.

Different mechanisms may be acting depending on the stress intensity, the frequency and the intensity of the corrosion attack, with the result that the crack growth rate as a function of ΔK may not be described by a simple power law (see Fig. 6.2.12).

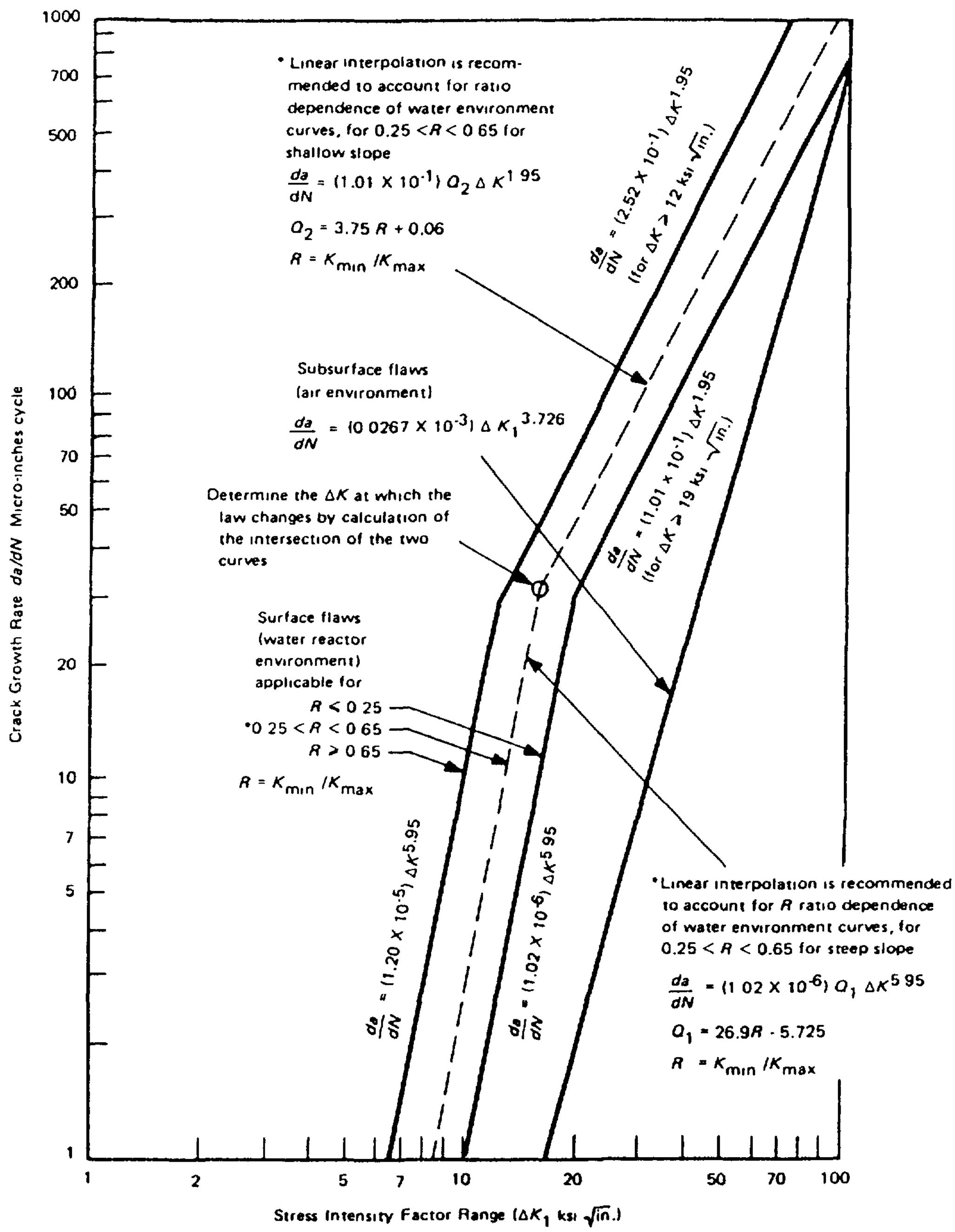

Fig. 6.2.12. Influence of corrosion on the cyclic crack growth rate in comparison to inert environment

The crack growth rate may be influenced by the following:

- characteristics of the corrosive environment, such as the pH value, the temperature, the electrochemical potential or the oxygen content in water;

- the loading frequency and wave form;

- the mean load and the load amplitude as expressed by the R ratio ($R = K_{min}/K_{max}$);

- sulphur content of steel.

Regions of concern with regard to reduced lifetimes in corrosive environments under high and low cycle fatigue or cyclic crack growth are

- parts in motion, such as the shaft of the main coolant pump;
- piping affected by motional parts or flow vibration;
- structures subjected to thermal transients caused by start-up and shut-down, cold water, etc.

6.2.5.3. Corrosion erosion

Corrosion erosion is a material removal process resulting from a corrosive process supported by a mechanical component (flowing liquid or gas) whereby the flow removes the passivation layer or prevents the buildup of such layers. The corrosion process would not occur or would take place at a significantly lower rate if the mechanical component (erosion) was not acting. Another interaction between erosion and corrosion is described in subsection 6.2.6, 'wear', in the paragraph on 'erosion/corrosion'. In this case the process of material removal by abrasive particles (particle erosion) is predominant and may be supported by superimposed corrosion processes.

The corrosion resistance of unalloyed or low alloyed ferritic steels in water and steam depends on the formation of oxidic protection layers. When these layers are removed, the corrosion attack at the metal surface can occur in the form of metal dissolution. The flow conditions shift the equilibrium between protection layer removal and formation which determines the material consumption. The severity of erosion depends on the amount of contaminants in the water and its velocity.

The protection layer removal occurs by the turbulent fluid layer at the surface, on which macroeddies can be superimposed, created at flow obstacles such as edges, deepenings and projections.

Changes in flow direction and changes in flow cross-section can also create eddies and higher local stresses. In a field with bent streamlines, as in elbows and pump casings, secondary streams can occur and lead to an increase in the mechanical stresses resulting from the stream. The flow pattern, which in practice is mostly very complex is responsible for the diversity in the appearance of erosion and in the local material loss. The process in the microrange is of the abrasion type, caused by contaminants in the form of particles with some surface fatigue, especially in connection with cavitation. The corrosion mechanism is of the uniform corrosion attack type.

Consequences of corrosion erosion processes in nuclear power plant systems are thinning of walls until leakage or rupture occurs, depending on the total stress as well as the deposition of the eroded particles at locations where there is a low flow velocity (wastage). There they can have negative effects on the stream and the thermal conductivity and last not least they can cause wastage corrosion.

Corrosion erosion phenomena in the coolant systems of light water reactors were mainly observed in vessels, pipings and armatures made of unalloyed ferritic steels. Systems with two phase flow (steam/water) are of particular concern since the impinging water drops increase the mechanical surface stress. In feedwater pipes (one phase flow) corrosion erosion effects were found at locations where there were high flow velocities (v greater than 3.5 m/s) together with inadequate stream conditions (e.g. branching and a subsequent narrow elbow).

Factors which can effect corrosion erosion processes are:

- Material: unalloyed carbon steels are sensitive; steels with a chromium content greater than 2% are most resistant.

- Water chemistry: in alkaline water the corrosion erosion process is less severe than in acidic water. At pH values greater than 9.5 practically no corrosion erosion occurs. If this pH value is reached by means of ammonia dosage, secondary effects have to be considered, such as stress corrosion cracking in condensator tubes made of copper alloys.

- Temperature: a maximum susceptibility has been observed at a temperature of about 150°C.

- Flow velocity: the material removal process increases exponentially with increasing flow velocity. The design and fabrication have great influence on the micropattern of flow and thus on the local increase on the critical threshold velocity.

6.2.6 Wear

Wear is defined as the loss of material from interacting surfaces under relative motion, resulting from a tribological process. This also includes erosion processes, irrespective of whether it is solid particles (abrasives), flowing liquids and gases, or any combination of these acting. Because of the interaction of several materials and the environment, the wear performance of machine parts and components is not a material property but describes the behaviour of a system (tribological system) consisting of:

- the material structure and the environment, and
- loading conditions (load, velocity, temperature) determining the energy input into the system.

The types of motion can be:

- sliding
- rolling
- impinging
- flowing

In some cases relative motion occurs on a microscale so that tribological effects are not always obvious. To this category belong most of the components that are fitted by shrinkage or have low tolerances. Under an applied cyclic load (vertical or parallel to the surface) sliding occurs in the micrometer range, mostly at high frequency. This process is termed fretting. When cyclic mechanical stresses are acting, even at a level below the fatigue limit stress, surface damage caused by fretting can drastically reduce the lifetime under fatigue. This mechanism of accelerated fatigue is referred to as fretting fatigue.

The process of particle removal by wear is caused by different mechanisms which depend on the characteristics of the interacting materials, the type of relative motion and the chemical environment. The wear mechanisms are:

- adhesion (microwelding of asperities with subsequent decohesion due to continued motion);

- surface fatigue by repeated transmission of normal or shear forces;

- abrasion, erosion (microgrooving by hard particles or hard asperities of a counter body).

As a consequence of the removal of wear particles, fresh surfaces are being created mostly by plastic deformation in the micro-range. In chemically active environments, chemical reactions with these highly excited surface areas can occur. This leads either to the formation of surface layers which reduce the adhesion tendency, slow down the wear process and simultaneously reduce the friction coefficient, or to chemical reactions through which the process of material removal is accelerated together with a simultaneous corrosion process.

Under certain circumstances hard oxide layers and oxidic particles can be produced. They can inhibit the adhesion wear mechanism but drastically enhance the abrasive component, especially when the particles cannot be removed from the contact area owing to the geometric arrangement. Since oxidization has an increase in volume associated the clearance of fitted parts can be reduced and make further motion impossible. This can lead to the overstressing of parts, e.g., axles, and subsequent mechanical failure (spontaneous fracture or fatigue fracture).

Adhesion processes

The most efficient way to reduce wear and friction is by the application of lubricants. By building up an adequate static or dynamic pressure (hydrostatic, hydrodynamic or elastohydrodynamic), solid contact can be prevented completely. Adhesion then no longer plays a role and wear occurs only as a consequence of fatigue processes resulting from the load transmitted through the lubricant film, or by erosion caused by solid contaminants. Under unfavourable fluid dynamic conditions, damage in a bearing can also occur by cavitation.

Complete hydrodynamic lubrication cannot always be sustained under the applied loading conditions, however, so that partial or temporary boundary lubrication has to be considered. In this regime the additions of the lubricant plays an important role since they form reaction layers which reduce friction and wear.

Parts that cannot or cannot sufficiently be lubricated must be adapted to each other with respect to their adhesion tendency. Ferritic and austenitic steels have high adhesion tendencies whereas materials with high non-metallic phases, such as carbides, nitrides and borides and non-metallic materials such as ceramics are more suitable.

Surface fatigue processes

Material surfaces stressed by Hertzian contact or by impact need to have a high fatigue limit strength to be sufficiently resistant to surface fatigue. High hardness is one of the major requirements; however, the resulting brittleness of these materials is of particular concern, and optimal compromises need to be found. The damage process starts with the accumulation of energy in the form of an increasing dislocation density due to local deformation until saturation occurs. In a second stage, further input of energy leads to crack initiation and crack growth and finally to wear particles breaking off. This wear mechanism generally shows an incubation phase with a subsequent more or less sudden increase in damage and the occurrence of failure.

Measures to enhance the resistance of material to surface fatigue, such as surface hardening or coating with hard layers, have to take into account the stress distribution in depth, e.g., under Hertzian conditions. Layers of sufficient thickness therefore have to be provided.

Abrasion and erosion processes

In the event of attack by hard erosive particles, the aim is to apply materials with a hardness higher than that of the particles. This reduces the depth of penetration of the particle and thus the amount of material worn-off owing to microgrooving.

Materials which gain their integral hardness through hard phases and a tough and mainly soft matrix, however, can suffer from selective erosion. Particularly small particles can erode the soft matrix between the hard phases. Therefore matrix hardness on the one hand and the hardness, size and distribution of hard phases on the other hand are the main factors that determine the wear resistance to abrasion and erosion.

When erosion is caused by corrosive fluids bearing a small amount of solid particles, it is difficult to classify the process. With corrosion dominating, the process might be referred to as erosion assisted corrosion, or when erosion dominates the process might be termed corrosion assisted erosion. Corrosion erosion is usually classified as a corrosion mechanism (see subsection 6.2.5.3.) in which the corrosion rate is accelerated owing to the mass transport by the flowing medium.

Depending on the fluid dynamic conditions, cavitation can occur, which is the process of creating gas bubbles in the liquid and their subsequent implosion. The effect on the material is impact stress, leading to surface fatigue which can be enhanced by corrosion processes.

Regions of concern

In a nuclear power plant almost all wear mechanisms can occur with various materials such as plastics, steels, non-ferrous metals, hard metals, ceramics and various coatings. Areas of concern are those with the following conditions:

- Designed relative motion (sliding, rolling), e.g.:
    - guiding tubes
    - valves
    - engine parts of different types
    - electrical relays and contacts

- Flowing liquids, gases or two phase flows with or without abrasive contaminants, e.g.:
    - pipes, tubes
    - valves
    - steam turbine blades and casings
- Vibration of fitted parts:
    - sleeves
    - steam generator tubes
    - turbine blades

6.3. Degradation of Concrete Structures

6.3.1 Degradation of concrete

6.3.1.1 Concrete and its application to structures related to Nuclear Safety

Concrete

Concrete is a general term for a class of ceramic materials that vary widely in their properties and application [17]. The American Concrete Institute defines concrete as a composite material that consists essentially of a binding medium within which are embedded particles or fragments of aggregate; in Portland cement concrete, for example, the binder is cement and water. By varying the constituents and their relative proportions in the mixture, one can obtain concretes of widely differing properties can be obtained.

Since aggregates generally occupy 60% to 80% of the volume of concrete, their characteristics significantly influence its properties, as well as its mix proportions and economics. The most commonly used aggregates, such as sand, gravel, crushed stone and slag from air cooled blast furnaces, produce normal weight concretes (2160 to 2560 kg m^3). Expanded shale, clay, slate and slag are used as aggregates to produce structural lightweight concretes having unit weights ranging from 1360 to 1840 kg m^3. Other light materials such as pumice, scoria, perlite, vermiculite and diatomite are used to produce insulating concretes having unit weights from 240 to 1440 kg m^{-3}. Heavyweight materials such as barites, limonite, magnetite, ilmenite, iron and steel particles are used for producing heavyweight concrete having unit weights of 6410 kg m^{-3}. Normal weight, structural lightweight and heavy aggregates should meet the standard specifications.

Properly prepared concrete has actually been found to improve with age. At one nuclear power plant site, strict control of the constituents, mixing, installation and curing, and a non-extreme environment yielded a general improvement in compression tests results during the first five years and a stabilizing during the next five years. The results of the compression test for the test sample were as follows:

530 bars for 28 day samples
620 bars for 90 day samples
810 bars for 5 years plus 10 year samples.

Yet despite this excellent behaviour there are potential ageing mechanisms to be aware of. A description of the use of concrete in safety related structures and its potential ageing mechanisms are given in this section.

Application of concrete to nuclear safety related structures

The principal applications of concrete in nuclear safety related structures include its use in containment buildings, containment base mats and biological shield walls. Other applications include balance of plant (BOP) walls and auxiliary safety related buildings. Pertinent USA codes, standards and specifications related to concrete are presented in Table 6.3.1.1.

Table 6.3.1.1
US Codes, Standards and Specifications Referenced in Codes for Nuclear Safety Related Structures and Concrete Reactor Vessels and Containments

American Society for Testing and Materials

A 82-72	C 109-80	C 173-78	C 496-71 (1979)
A 184-65 (1972)	C 114-82a	C 183-78	C 512-76
A 185-73	C 115-79b	C 191-79	C 535-81
A 416-80	C 117-80	C 192-81	C 566-78
A 421-80	C 123-69 (1975)	C 204-81	C 586-69 (1981)
A 496-72	C 127-81	C 227-81	C 595-82
A 497-72	C 128-79	C 231-82	C 618-80
A 615-81a	C 131-81	C 260-77	C 637-73 (1979)
A 706-81	C 136-82	C 266-77	C 642-81
A 722-75 (1981)	C 138-81	C 289-81	C 937-80
C 31-69 (1980)	C 142-78	C 295-79	C 938-80
C 33-82	C 143-78	C 311-77	C 939-81
C 39-81	C 144	C 342-79	C 940-81
C 40-79	C 150-81	C 430-79	C 941-81
C 42-77	C 151-77	C 441-81	C 942-81
C 78-75 (1982)	C 157-80	C 469-81	C 943-80
C 88-76	C 172-82	C 494-81	C 953-81
C 94-81			

American Concrete Institute

207	214-77	305-72	309-72 (1978)
211.1-81	304-73 (1978)	306-66	347-78

American National Standards Institute

ANSI A58.1

U.S. Army Corps of Engineers

CRD-C 36-73	CRD-C 44-63	CRD-C 621-82a
CRD-C 39-81	CRD-C 119-53	

Containment buildings

Within nuclear power plants one unique application of concrete is in the containment buildings which enclose the entire reactor and the reactor coolant systems, and serve as the final barrier to the release of radioactive fission products to the environment under the postulated conditions of a design basis accident. The containment is designed to withstand loadings associated with a loss of coolant accident resulting from a double ended rupture of the largest pipe in the reactor coolant system. The containment is also designed to retain its integrity under low probability ($<10^{-4}$) environmental loadings such as those generated by earthquakes, tornados and other site specific environmental events such as floods, seiche and tsunami.

Additionally, it is required to provide biological shielding under both normal and accident conditions and to protect the internal equipment from external missiles such as tornados or missiles generated by tornados or turbines and aircraft impact (where postulated).

The functional requirements for containments are satisfied by various types of composite and hybrid steel-concrete constructions. The use of concrete in these structures evolved as the sizes of nuclear power plants increased, making the use of stress relieved steel plate uneconomical. The first concrete containments were fabricated from reinforced concrete and were cylindrical with a hemispherical dome and a flat base slab. Later, the concrete was partially prestressed in the vertical direction only, with mechanically spliced reinforcing steel in the hoop direction and in the dome. Fully prestressed concrete containments were first built in the late 1960s, being cylindrical in shape with a shallow dome and resting on a reinforced concrete slab.

Maintaining the structural and pressure integrity of the containment throughout the life of the nuclear power plant is essential.

Containment base mats

The function of the containment base material (mat) is to support the reactor's internal structures and the containment building. The base mats are fabricated from reinforced concrete and generally have a thickness of 3 m (10 feet) or greater and a diameter of 40 m (130 feet) or greater. The base mat is generally concreted in vertical sections, horizontal layers or a combination of both. It is generally sealed at the top with a steel liner and rests on either bedrock or concrete fill extending to bedrock or to deep soil.

Biological shield walls

Biological shield walls are fabricated from either standard weight reinforced concrete or heavyweight concretes. Commercial reactors use reinforced concrete as the shield material. Thicknesses of the shield walls typically range from 3 m to 4 m (10 feet to 13 feet) and the walls can support either part or all the weight of the reactor vessel. Desirable properties of shielding concretes include: reasonably high and consistent density, reasonably high strength (34 to 41 MPa compressive strength), sufficient hydrogen content materially unaffected by operating temperature, low heat of hydration, high specific heat, high thermal conductivity, low thermal expansion, low drying shrinkage, resistance to radiation and reasonable cost.

6.3.1.2 Potential degradation modes of concrete and its reinforcement in structures related to nuclear safety [17]

Degradation modes of concrete components in nuclear safety related structures vary somewhat according to the particular application. Primary degradation modes of reinforced concrete containments would be those related to deterioration of the concrete and reinforcing steel.* These would include such items as those noted in Table 6.3.1.2, which result from interactions of

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* Although an extreme load condition is a degradation mode, this is not considered in the following discussion since it is felt that this would be a readily discernible event and would require a detailed structural inspection and evaluation.

Table 6.3.1.2

Causes of Concrete Cracking

Component	Type	Cause of distress	Environmental factor(s)	Variables to control
Cement	Unsoundness	Volume expansion	Moisture	Free lime & magnesia
	Temperature cracking	Thermal stress	Temperature	Heat of hydratation rate of cooling
Aggregate	Alkalisilica reaction	Volume expansion	Supply of moisture	Alkali in cement, composition of aggregate
	Frost attack	Hydraulic pressure	Freezing and thawing	Absorption of aggregate, air content of concrete maximum size of aggregate
Cement paste	Plastic shrinkage	Moisture loss	Wind and temperature	Temperature of concrete protection of surfaces
	Drying hrinkage	Moisture loss	Relative humidity	Mix design, rate of drying
	Sulfate attack	Volume expansion	Sulfate ions	Mix design, cement type, admixtures
	Thermal expansion	Volume expansion	Temperature change	Temperature rise, rate of change
Reinforce-ment	Electro-chemical corrosion	Volume expansion	Oxygen moisture	Adequate concrete cover

the materials of concrete, material deterioration as a result of aggressive environments, and exposure to extreme environmental conditions. Prestressed concrete containments have deterioration of the prestressing steel due to aggressive environments as an additional degradation mode. Degradation modes of the containment base mat would essentially be the same as for the reinforced concrete containment. Relevant deterioration modes for biological shield walls would be a loss in strength or shielding efficiency resulting primarily from thermal or irradiation effects.

<u>Concrete Degradation</u>

Deterioration of concrete results primarily from cracking, aggressive environments (freezing and thawing, wetting and drying, chemical attack), corrosion of embedments or extreme environment exposure (elevated temperature, pressure and irradiation).

Cracking of concrete

Cracking of concrete can occur when the concrete is in either a plastic or a hardened state. Cracks in many instances do not affect the ability of the structure to bear load, but they can expose the concrete to attack in hostile environments which can affect structural durability.

Cracking of plastic concrete

Cracking of concrete while plastic results from two primary causes: plastic shrinkage cracking and settlement cracking.

Plastic shrinkage cracking occurs most commonly in horizontal surfaces where rapid evaporation occurs. The cracks form owing to differential volume changes which induce tensile stresses. Such cracks, which are usually shallow, can be fairly wide at the surface and can extend from a few millimetres to several metres in length. Although not a threat to structural integrity, they are important in that they can permit access to the concrete by hostile environments.

Settlement cracking occurs after initial concrete placement while the concrete is still consolidating. The cracking (or the occurrence of voids) results owing to constraints imposed by obstacles such as reinforcing bars or other embedments, formwork or a prior concrete placement. As these cracks or voids develop they can provide access for hostile environments into the concrete.

Cracking of hardened concrete

Cracking of hardened concrete results from shrinkage, thermal effects, chemical reactions and reinforcement corrosion. Hardened concrete also cracks owing to freezing and thawing; this is discussed in a subsequent section on aggressive environmental effects.

Drying and carbonation shrinkage

As concrete loses water it tends to contract or shrink. If the concrete were free to deform as it contracted the volume change would be of little consequence; however, concrete is normally constrained owing to the presence of foundations, reinforcement or adjacent concrete. The combination of shrinkage and constraint leads to crack formation when the tensile stresses that develop in the member exceed the tensile strength of the concrete.

Hardened cement paste in the presence of moisture reacts chemically with carbon dioxide to cause a decrease in volume. The $Ca(OH)_2$ carbonates to $CaCO_3$ and other cement compounds are also decomposed; that is, hydrated silica, alumina and ferric oxide. In addition to the shrinkage, the weight of the concrete also increases. The effects of carbonation are significant in intermediate relative humidities (25% to 100% RH). Surface crazing has also been observed to result from carbonation shrinkage, but is confined to a thin layer near the surface. Since carbonation is primarily active near the concrete surface, it can lead to severe shrinkage cracking when combined with drying shrinkage.

Thermal effects*

When Portland cement is mixed with water, its constituent compounds undergo a series of exothermic chemical reactions. The temperature rise within the concrete is dependent on how rapidly the heat can be dissipated to its surroundings. If the heat is not permitted to escape (adiabatic conditions), the temperature rise can be 40°C or more. Although the buildup of heat occurs while the concrete is relatively plastic (in the first few days), the problem develops as the concrete starts to cool and has acquired a certain amount of rigidity. During the cooling process the outer surface cools and tends to shrink, setting up compressive stresses in the centre of the structure which is still at a higher temperature. The centre thus restrains the cooler outer surface, setting up tensile stresses. When the magnitude of the tensile stresses exceeds the tensile strength of the concrete, cracking results. Thermal effects resulting from cement hydration are primarily a problem in massive concrete structures.

Chemical reactions**

A number of detrimental chemical reactions can produce concrete cracking. These generally are related to the concrete aggregate and include: alkali-aggregate reaction, cement-aggregate reaction and the reaction of carbonate aggregates. Sulphate bearing waters also present problems.

Alkali-aggregate reaction

Cracking in concrete can occur as a result of expansive reactions between aggregates containing active silica and alkalis derived from cement hydration, admixtures or external sources. The adverse effects of the alkali-aggregate reaction led to many failures of concrete structures built during the late 1920s to early 1940s. Problems in recent years with alkali-aggregate reactions have been significantly reduced through proper selection of aggregate materials (by means of petrographic examination to identify potentially active materials), the use of low alkali cements (<0.6% equivalent Na_2O), and the addition of pozzolonic materials (which help to limit expansion).

Cement-aggregate reaction

Concrete deterioration attributable to cement-aggregate reaction has been observed with certain sand-gravel aggregates in Kansas, Nebraska and Wyoming in the USA. These materials are highly siliceous and cause map cracking in concrete. This type of distress is not prevented by the use of pozzolans or low alkali cements. Special tests have been devised to indicate potential damage due to this phenomenon. A recommended remedy, if these materials must be used, is to replace 30% of the aggregate with crushed limestone.

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* This covers thermal effects as related to cement hydration. The effects of exposure to elevated temperatures are considered in a subsequent section.

** The actions of aggressive chemical agents are considered in a later section.

Carbonate aggregate reaction

Certain dolomitic limestone aggregates containing some clay, found in a few places in the United States and Canada, react with alkalis to produce expansive reactions. The alkali-carbonate reaction can be limited by keeping the alkali content of the cement low ($< 0.4\%$) or by diluting the reactive aggregate with a less susceptible material.

Sulphate bearing waters

The sulphates of sodium, potassium and magnesium, present in alkali soils and waters, have caused deterioration of many concrete structures through chemical reaction of the sulphates with the hydrated lime and hydrated calcium aluminate in the cement paste. Calcium sulphate and calcium sulphoaluminate are formed accompanied by considerable expansion which disrupts the concrete. Improvement of the resistance of concrete to sulphate attack can be provided by the use of admixtures such as pozzolans and blast furnace slag or of special sulphate resisting cements (Types V, VP, VS, IP or IS).

Effects on concrete of aggressive environments

Aggressive environmental factors which could lead to the deterioration of concrete would be freezing and thawing, wetting and drying, aggressive chemicals and the corrosion of embedments.

Freezing and thawing

The repeated subjection of wet concrete to cycles of freezing and thawing can lead to severe damage or to the destruction of concrete structures. The principal force responsible for concrete damage resulting from this phenomenon is internal hydraulic pressure created by the expanding ice-water system during freezing. The production of highly frost resistant structures requires: (1) the use of air entrainment; (2) the selection of aggregate with adequate durability for the exposure to be encountered; (3) use of concrete with a low ratio of water to cement, properly handled, placed and cured; (4) design of the structure to minimize exposure to moisture and to facilitate drainage; and (5) avoidance of materials or construction practices which might lead to other disintegrative processes.

Wetting and drying

Water may enter and issue from structures with cracks, improperly treated construction joints, or areas of segregated or porous concrete. As the water passes through the concrete it dissolves (leaches) some of the calcium hydroxide and other solids so that in time serious disintegration of the concrete may occur. As the concrete dries, salts (sulphates and carbonates of sodium, potassium or calcium) are deposited through evaporation of the water or interaction with carbon dioxide in the atmosphere. Extensive leaching causes an increase in concrete porosity, which leads to reduced strength and increased vulnerability to hostile environments.

Aggressive chemicals

Concrete which is mixed in appropriate proportions, and is properly placed and cured is relatively impervious to most waters, soils and atmospheres. However, there are some chemical environments in which the useful life of even good quality concrete is shortened. In order for these chemicals to attack concrete significantly, they must in general be in solution and above a minimum concentration. These chemicals include sulphates of sodium, potassium or magnesium such as are found in soils, and products of the combustion of fuels which produce sulphurous gases which combine with moisture to produce acids which attack concrete.

Corrosion of embedments

Cracking and spalling of concrete can result from the corrosion of embedments, with mild steel reinforcing bars being the primary embedment. Concrete cracking and spalling result from the buildup of corrosion products which tend to act as a wedge to split and spall the adjacent concrete.

Spalling of concrete can also result from the corrosion of embedments other than reinforcing bars. Continuous exposure of aluminium to moist concrete produces severe corrosion due to the destruction of its passive film at high alkalinities. As is true with conventional reinforcing steel, the corrosion process is aggravated in the presence of chloride ions. Lead and zinc also corrode in moist concrete, zinc being more resistant than aluminium or lead.

Extreme environmental conditions

Disregarding an overload condition resulting from an unusual event such as an earthquake or a serious accident which would result in the evaluation of the plant's structural integrity, the primary extreme environmental conditions to which a nuclear safety related structure might be subjected would be elevated temperatures, pressures and irradiation.

Elevated temperature effects

Thermal gradients are important to concrete structures in that they affect concrete's compressive strength (load-carrying capacity) and stiffness (structural deformations and loads which develop at restraints). The American Society of Mechanical Engineers (ASME) Code for concrete reactor vessels generally limits the bulk concrete temperature to 65°C. (Under certain environmental conditions the concrete temperature is permitted to rise above this value.) The selection of the 65°C limit by the Code was due to the lack of a design procedure and a limited knowledge of the material's behaviour under elevated temperature conditions. Higher limits on concrete temperatures are allowed if tests are provided to evaluate the reduction in strength and this reduction is applied to design limits. (Evidence should be provided which verifies that the increased temperatures do not cause deterioration of the concrete either under or not under.)

A review of methods which have been used in various investigations for elevated temperature testing of concrete indicates that in general the test can be categorized according to cold testing or hot testing. In cold testing, the specimens are gradually heated to a specified temperature, then permitted to reach thermal stability at that temperature for a prescribed period of time, then permitted to cool slowly to the ambient temperature, and then tested to determine mechanical properties. In hot testing, the specimens are gradually heated to a specified temperature, permitted to stabilize thermally at that temperature for a prescribed period of time, and then tested at that temperature to determine their mechanical properties. During testing, specimens are maintained in either an open environment where water vapour can escape or a closed environment where the moisture is contained. The closed environment condition models conditions for mass concrete where moisture does not have ready access to the atmosphere, and the open environment models conditions where the element is either vented or has free atmospheric communication. During heating and cooling the speciments may be either loaded or unloaded.

An overview of the effects of elevated temperature on the properties of concrete lists the following general observations:

- Specimens lose less strength if moisture is permitted to escape under heating than if it is not.

- Specimens heated and then permitted to cool before testing than those tested while hot.

- Concrete specimens loaded during heating lose less strength than unloaded specimens.

- The longer the duration of heating before testing, the greater the loss in strength. The reduced strength stabilizes, however, after a period of long isothermal exposure.

- The decrease in the modulus of elasticity due to exposure to elevated temperatures is more pronounced than the decrease in the compressive strength.

- The mix proportions and the type of aggregate influence the strength of heated concrete as follows:

  - mixes with a low cement/aggregate ratio lose less strength on heating than do richer mixes;
  - concrete made with limestone aggregate degrades less as a result of heating than concrete made with siliceous aggregate.

- The water/cement ratio has a limited effect on the strength degradation of heated concrete.

- Small test specimens generally incur greater losses of strength than larger specimens.

- Specimens subjected to several cycles of heating and cooling lose more strength than those not subjected to thermal cycling.

- The strength of concrete before testing has little effect on the percentage of the strength retained at elevated temperatures.

Pressure effects

The containment is designed to withstand loadings associated with a loss of coolant accidents resulting from pipe breaks. Resultant from this break is a pressure excursion that can reach levels of 0.4 megapascals ($\approx$ 45 lbf/in^2 (g)).

Tests are periodically required to verify the integrity of containments. These tests apply near design pressure loadings and introduce potential fatigue mechanisms. The frequencies of these tests vary between Member States from every 5 to every 10 years. Optimum frequency of testing must be balanced between the need to know that containment integrity is intact and the probability of the occurrence of significant degradation. Over-testing could increase the risk of a fatigue ageing mechanism becoming significant.

With 20 years of experience with prestressed concrete containment, a cycle of five to ten years for testing appears to be reasonable; ageing due to fatigue loading does not appear to be significant.

Irradiation effects

Concrete has traditionally been used as a shielding material since a reasonable thickness attenuates radiation, it has sufficient mechanical strength, can be constructed in virtually any size and shape at reasonable cost, and it requires minimal maintenance. Irradiation, however, in the form of either fast and thermal neutrons emitted by the reactor core, or gamma rays produced as a result of the capture of neutrons by members (particularly steel) in contact with the concrete, can affect the concrete. The fast neutrons are mainly responsible for the considerable growth, caused by atomic displacement, that has been measured in the aggregate. Gamma rays induce radiolysis of water in the cement paste which can affect its creep behaviour and whether it is 'shrinkable' to a limited extent and also result in the evolution of gas. Most concrete structures are not affected as the exposure is low.

However, for concrete reactor vessels, operation of the reactor over several decades may subject the concrete to considerable fluxes fast and thermal neutron. Estimates of the maximum radiation to which a prestressed concrete reactor vessel (used for a gas cooled reactor) will be exposed after 30 years of service are:

thermal neutrons: 6×10^{19} n/cm^2

fast neutrons: 2 to 3×10^{18} n/cm^2

gamma radiation 10^{11} rad

This level of radiation is greater than would be expected for concrete in light water reactors (LWRs). Section III, Division 2, of the ASME Boiler and Pressure Vessel Code gives a radiation exposure level allowable to 10×10^{20} nvt. The British Code for prestressed concrete pressure vessels states that the maximum permissible neutron dose is governed by the effects of irradiation on the concrete properties and the effects are considered to be insignificant for doses up to 0.5×10^{18} n/cm^2. It is important to note, however, that these criteria are based on a very limited amount of data and that it is not possible to quantify the extent to which irradiation will change the properties of concrete as this is dependent on many factors, such as: variation of material properties; material state of testing; neutron energy spectrum; and neutron dose rate.

The presently available data on the effects of irradiation on concrete properties is misleading because of technical and experimental difficulties in conducting meaningful tests. In addition, data which are available are generally not comparable because: (1) different materials were used; (2) mix proportions varied; (3) specimen size was inconsistent; (4) temperatures varied or (5) cooling and drying conditions were different. However, the following generalizations may be made:

(1) For some concretes, neutron irradiation of more than 1×10^{19} n/cm^2 may cause some reduction in compressive strength and tensile strength;

(2) The decrease of tensile strength due to neutron irradiation is more pronounced than the decrease of compressive strength;

(3) The resistance of concrete to neutron irradiation apparently depends on the type of neutrons (slow or fast) involved, but the effect has not been explained;

(4) The resistance of concrete to neutron irradiation depends on the mix proportions, the type of cement and the type of aggregate;

(5) The effect of gamma radiation on concrete's mechanical properties requires clarification;

(6) The deterioration in concrete's properties associated with a temperature rise resulting from irradiation is relatively minor;

(7) The coefficients of thermal expansion and conductivity of irradiated concrete differ little from those that would result to concrete exposed to high temperature;

(8) The modulus of elasticity of concrete exposed to neutron irradiation decreases with increasing neutron fluence;

(9) Creep in concrete is not affected by low level radiation exposure, but for high levels of exposure it is likely that creep would increase with exposure because of the effects of irradiation on the concrete's tensile and compressive strengths;

(10) For some concretes, neutron irradiation with a fluence of more than 1×10^{19} n/cm^2 can cause a marked increase in volume;

(11) In general, concrete's irradiation resistance increases as the irradiation resistance of aggregate increases; and

(12) Irradiation has little effect on shielding properties of concrete beyond the effect of moisture loss due to a temperature increase.

Concrete reinforcing steel degradation

Mild steel reinforcing bars are provided to control the extent of cracking and the width of cracks at operating temperatures, to resist tensile stresses and computed compressive stresses for elastic design, and to provide structural reinforcement where required. Potential causes of degradation of the reinforcing steel would be corrosion, exposure to elevated temperatures and irradiation.

Corrosion

For steel embedded in concrete to corrode, the following conditions must all be met: (1) there must be an anode-cathode couple with at least part of the steel acting as an anode; (2) an electrical circuit must be maintained; (3) moisture must be present; and (4) oxygen must be present. In a good quality, well compacted concrete, reinforcing steel is not prone to corrosion because the highly alkaline conditions (pH > 12.0) pertaining within the concrete cause a passive oxide film to form on the surface of the iron which prevents corrosion, and concrete has a relatively high electrical resistivity. Reinforcement corrosion problems occur where the concrete's pH is reduced to less than 11.0, destroying the passive iron oxide layer. This can result from improper construction techniques (honeycomb formation, inferior concrete patching), or the presence of cracks that permit access by hostile environments. The primary effect of reinforcing steel corrosion is to crack or spall the concrete; however, under extreme conditions the load carrying ability of the steel reinforcement can be reduced owing to a reduction in the cross-sectional area.

Effects of elevated temperature

The properties of reinforcing steel used in design are generally a function of the yield stress. Like the compressive strength of concrete, the yield stress increases as the temperature is reduced. However, below the nil ductility temperature (NDT) the steel ductility substantially decreases. The NDT for carbon steels including normal reinforcing bars is approximately -51°C.

The physical properties of steel are affected by elevated temperatures. The relationships between temperature and yield strength and between temperature and modulus of elasticity of structural steels is approximately linear and decreasing from 200°F to 650°F. The relationship between the temperature and the thermal expansion of steel is also operative in reinforced concrete structures. In general, temperature induced variations in structural reinforcing steel in nuclear power plants are not permanent ageing effects.

Irradiation effects

Neutron irradiation produces changes in the mechanical properties of structural steel, in particular an increase in the material's yield strength and a rise in the ductile to brittle transition temperature. Neutron levels above 10^{17} n/cm^2 are required before any measurable effects can be found. Structural reinforcing steel do not experience these levels in nuclear power plants, so this ageing mode is of little concern for structures. Levels above this threshold will occur for concrete reactor vessels and this ageing mode may be significant. As mentioned in subsection 6.2, reduction in toughness (embrittlement) and swelling may affect the vessel's integrity. Monitoring for any effects would be appropriate.

<u>Steel degradation in concrete prestressing</u>

A post-tensioned prestressing system consists of a prestressing tendon in combination with methods of stressing and anchoring the tendon to hardened concrete. To attain satisfactory performance, prestressing systems are designed to have: (1) consistently high strength and strain at failure; (2) serviceability throughout its lifetime; (3) reliable and safe prestressing procedures; and (4) the possibility of retensioning and replacement (non-grouted systems). Prestressing systems may be grouped into three major categories, depending on the type of tendon utilized: wire, strand or bar. In the United States, the 8.9 MN systems which are approved for use in containments include: (1) BBRV (wire), (2) VSL (strand), and (3) stress steel S/H (strand). Potential degradation modes for these prestressing systems include corrosion, to exposure to elevated temperatures and irradiation.

Corrosion

Corrosion may be either highly localized or uniform; however, most related to prestressing corrosion failures have been the result of localized attack produced by either pitting, stress corrosion or hydrogen embrittlement, or by combinations of these. Pitting is an electrochemical process that results in local penetrations into the tendon that can reduce the cross-sectional area until it is incapable of supporting its load. Stress corrosion cracking results in brittle fracture of a normally ductile metal or alloy under stress (tensile or residual) in specific corrosive environments.

Hydrogen embrittlement, frequently associated with hydrogen sulfide, occurs when hydrogen atoms enter the lattice of the metal and significantly reduce its ductility. The prestressing systems are protected by filling the ducts containing the post-tensioned tendons either with microcrystalline waxes (petroaltums) compounded using organic corrosion inhibitors (nongrouted tendons) or with Portland cement grout (grouted tendons). The need for the possibility of inspection, retensioning and replacement has made non-grouted post-tensioned steel tendons the dominant prestressing system used in containments.

Reviews of the performance of prestressing tendons contained in both structures for nuclear power and conventional civil engineering structures indicate that corrosion related incidents are extremely infrequent. The evolution of corrosion inhibitors and the use of organic petrolatum based compounds designed especially for corrosion protection of prestressing materials have virtually eliminated the corrosion of prestressing materials. The few incidents of corrosion that were identified generally occurred early on in the use of prestressed concrete for containment structures, and resulted either from the use of off-the-shelf corrosion inhibitors that had not been specially formulated for prestressing materials or from poor construction practices. The problems were subsequently identified and corrected during either the construction phase, the initial structural integrity test or subsequent in-service inspections.

Temperature effects

The effect of temperature ($\approx$ 40°C) on the relaxation of a low-relaxation strand stressed to 70% guaranteed ultimate tensile strength (GUTS) varies from 1% to 4% over a period of many years. Relaxation studies for stress relieved strand stressed to 75% GUTS for this same temperature yield relaxations of from 5% to 6.4% after 30 years. As normal average temperatures increase for extended periods of time, larger allowances would be required to take account of the relaxation of the steel. Also, tendons composed of stress relieved wires have relaxation losses of about the same magnitude as stress relieved strands but relaxation of the strand is greater than that of its straight constituent wire owing to the combined stress relaxation in the helical wires.

It is known that the effect of very high temperatures on heat treated and drawn wires can be significant, and on cooling they may not regain their initial strength since severe heating destroys the crystal transformation achieved by the heat treating process. However, short term heating, on the order of three to five minutes, even to temperatures as high as 400°C, may not do any harm. Test specimens which were exposed to longer term heating, to a temperature of 400°C for one hour, and cooled to 20°C, resulted in heat treated and drawn wires exhibiting permanent losses of strength.

It is conceivable that extremely low temperatures could cause embrittlement of the anchorage and that portion of the tendon which is exposed to the low temperature.

Irradiation effects

Irradiation of steel affects its mechanical properties because atoms are displaced from their normal sites by high energy neutrons to form interstitials and vacancies. These defects can group together and effectively both strengthen the steel and reduce its ductility or, at higher temperatures, they can recombine and annihilate one another and, for a given neutron dose, reduce the irradiation damage. Results on the effects of irradiation on prestressing are somewhat limited; however, in one study samples of wire of diameter 2.5 mm from two manufacturers were irradiated in a test reactor to a total dose of 4×10^{16} n/cm^2 (flux of 2×10^{10} n/cm^2/s) while stressed to 70% of their tensile strength. Sample 1 was irradiated at 26°C and Sample 2 at 31°C. Since the basic concern of the investigation was stress relaxation, the wire stress was monitored until breakdown of the rigs. Results for a control specimen (non-irradiated) were also obtained for both samples. Relaxation curves obtained from non-irradiated specimens showed similar relaxation behaviour; thus irradiation had no effect at this flux level. Further tests of the irradiated samples showed that there was also no change in tensile strength or metallographic structure. Since the flux level was higher than that likely to be experienced by prestressing wires in containments, it does not appear that irradiation will have a harmful effect on prestressing in a containment, especially since the prestressing is not placed in positions where it will receive a harmful irradiation dose.

6.3.1.3. Conclusions

In general, the following conclusions can be drawn for concrete structures:

- Performance of concrete structures in NPPs have been very good;

- Concrete constructions in good condition predate NPPs by several mellenia and attest to its durability as a structural material;

- Techniques exist for detecting the effects of environmental stresses and can provide qualitative data;

- Remedial measures have been developed to restore structural integrity completely for most degrading mechanisms;

- Known ageing mechanisms can either be prevented by initial design and construction or progress at a very slow rate and can be monitored to provide more exact quantitative data. Table 6.3.1.3 gives tolerable crack widths in reinforced concrete for different exposure conditions.

Table 6.3.1.3

Tolerable Crack Widths in Reinforced Concrete [54]

Exposure Condition	Maximum crack width (mm)
Dry air or protective membrane	0.40
Humidity, moist air, soil	0.30
De-icing chemicals	0.175
Seawater and seawater spray; wetting and drying	0.15
Water-retaining structures	0.10

6.4 Non-metals

6.4.1. Introduction

In addition to concrete, there are many applications of non-metals in nuclear power plants - primarily in electrical and mechanical equipment. For example, non-metals (such as elastomers) perform important sealing functions in major access doors to the containments and in other structural applications at pressure boundaries. Special coatings serve sealing functions for porous structures and in corrosion protection .

Although not by any means exhaustive, this section provides brief descriptions of the properties and uses of non-metals in nuclear power plants. The degradation or ageing mechanisms in most of their applications are also discussed.

The non-metalics that are dealt with in this section are electronic components, polymers, lubricants, luminating resins, protective coatings and low temperature solder glass.

6.4.2. Electronics [24]

There are wide variations in commercial nuclear power plants (e.g. plant design, equipment and applications) which complicated the task of determining what electronic components should be discussed in this subsection. The subsection, therefore, focuses on the components in most general use and their ageing characteristics. Although important, the effects of electrostatic discharge, electromagnetic pulse and electromagnetic interference are not discussed since the effects are usually transitory or catastrophic and their influence and effect on aged components are difficult to assess.

Instrumentation systems in nuclear power plants provide information pertinant to plant operation whereas plant control systems actuate the needed operational equipment. The interest here is in instrumentation systems which provide data on temperature, pressure, radiation and status of equipment such as valve positions. A wide variety of instruments are used to yield this information. Temperature measurements employ resistance temperature detectors and thermocouples; pressure measurements, from which level and flow information is also derived, employ bourdon tubes, bellows, cantilever mechanismus and piezoelectric detection devices; and radiation monitors employ a wide range of components such as Geiger-Müller tubes and ionization chambers. Neutron flux is generally measured with ionization chambers, fission chambers or rhodium detectors. Each of these measurement techniques requires some type of electrical circuit to provide the necessary data. The subject of discussion in this section is the electronics associated with these electrical types of detection and measurement circuits.

6.4.2.1. Components (devices)

Material degradation is discussed as it relates to components, but specifically the discussion concentrates solely on the material which make up electronic and electrical components. Common electrical components which are evaluated are resistors, thermistors, capacitors, diodes, transistors and operational amplifiers. These subcomponents are used in the electronic parts of pressure transmitters and acoustical accelerometers. Pressure transmitters and preamplifiers are also used in all areas of a nuclear power plant, including the containment. (Since the containment area also experiences relatively severe operating environment, one should expect the effects of any ageing mechanisms to appear first in equipment inside containment. But sensitive electronic parts are not used inside containment.)

Typical components of two types of transmitters and one preamplifier used in the USA are shown in Table 6.4.2.1.

Table 6.4.2.1. Pressure Transmitters and Acoustic Accelerometer Preamplifier Components

Pressure Transmitter (Barton Model 763)		Pressure Transmitter (Foxboro Models N-E11 and N-E13)		Acoustic Accelerometer Preamplifier (Technology for Energy Corporation)	
Component	Type	Component	Type	Component	Type
Resistor	Carbon composition	Resistors	-	Resistor	Carbon composition
	Film (glaze-metal)	Capacitors	Disk ceramic - Erie	Capacitor	Ceramic
	Wire-wound		Tantalum - Kemet		Silver mica
Potentiometer	10 turn	Diodes	Switching	Diodes	Switching
Thermistor	Disk		Rectifier		Zener
Capacitor	-	Transistors	Bipolar		
	Mica		Bipolar	Transistor	Bipolar
Diode	-			Printed circuit board	G-30 Polyimide
	Zener				
Transistor	Bipolar			Potting	Silicone type
Op-amp					
Other	Printed circuit board				
	Terminal blocks				

6.4.2.2 Service Environments

Electronic equipment has been designed to operate in a wide variety of environments. The environment in a nuclear power plant differs considerably from space or weapons environments. Radiation, temperature, humidity and oxygen are all important factors to be considered for nuclear power plants. Furthermore, the normal design life of a nuclear power plant is 40 years, though individual electronic components may only be designed for five, ten, or 20 years of use before replacement. Temperature and radiation are generally considered to be the greatest contributors to the ageing process, since most electronic devices are in sealed containers or enclosures. However, the long operational periods in power plants coupled with the other environmental factors (humidity, vibration, etc.) suggest that all possible factors affecting degradation must initially at the design stage be considered. For example, moisture leakage and gas penetration through some O-ring seals can be significant over long periods of time. This could result in corrosion problems in metal interfaces. Device lifetime can also be affected by oxygen and by products of the outgassing process for devices in sealed containers, as well as by ozone and ultra violet radiation. Cyclic stressor due to temperature and radiation changes also tend to affect the ageing process. Vibration associated with the integrity of bonds, seals, etc., may also be an important factor to the acceleration of ageing process. Thus it is important that the total environment be defined in power plants, and that moisture and gas leakage and penetration problems be understood. (Tables 6.4.2.2 and 6.4.2.3 give typical NPP containment radiation and other environmental parameters).

Table 6.4.2.2

Typical radiation environments

Radiation[a]	Nuclear plant containment Normal operation (40 year ageing)	 Accident[b]
Gamma Dose [rad]	3×10^4 (T) (10^{-3} - 10^8	2×10^7 (A) $< 10^6$ (A)
Dose rate [rad/h]	10^1 (T) (10^{-3} - 1^{-3})	$<10^6$ (A)
Neutron[c] Fluence (n/cm^2)	10^9 - 10^{14}	-
Flux (n/cm^2)	10^0 - 10^5	-
Electron & Proton Dose	-	2×10^8 (A)
Dose rate	-	

(a) Notation on dose and dose rates: T, Tissue ; A, air, Si, silicon

(b) Included for comparison with data on long term ageing

(c) Neutron energies: nuclear plant containment, 100 keV

Table 6.4.2.3

Comparison of other (non-radiation) environments

Environment	Shelf storage[a] (up to 40 yrs)	Nuclear plant containments Normal operation (40 years ageing)	 Accident[b]
Temperature (C°)	24-30	24 to 66	260
Humidity (% RH at 20°C)	10-100	10 - 100	100
Oxygen (volume %)	0-20	0 - 20	0 - 20
Voltage	off	on	on

(a) Inside sealed and desiccated volume

(b) Included for comparison with long term ageing data

The problem of defining the environment in a nuclear power plant is complicated because of large differences in plant design, age and location. It is difficult to define the exact levels, so only 'normal' variations can be bracketed. Compounding this problem are the problems in trying to distinguish what are 'normal' environments for those instruments in storage awaiting installation in a nuclear plant.

The cyclical nature of plant operations creates cyclical ageing environments which may accelerate ageing phenomena. Plants normally operate 11 months per year and the shutdown period can represent a 'cycle', as well as ageing at a lower temperature and at a lower radiation dose rate. Furthermore, the relatively low dose rates (compared with accident dose rates) for full power operations may lead to greater degradation for the same total dose than higher dose rates. There is a possibility that such effects may occur in electronic components, especially in semiconductor devices, since radiation-induced damage and annealing phenomena are not completely understood.

Temperature and humidity

In power plant control rooms and technical support centres the temperature levels are normally 20-25°C at 30-80% relative humidity. In non continuously occupied areas (e.g. pump rooms), temperatures and humidities are extremely variable. Humidity is generally not controlled in these areas. In the containment, temperatures and humidities also vary greatly depending on the location, the power level, the time of year and the performance of the air conditioning system. Humidity levels can vary from 10 to 100% relative humidity. The in-containment temperatures for one reactor for which data are available generally remained between 32°C-38°C with variations from 24°C to 66°C during the course of one operating cycle. The control rod drive motor area was consistently around 49-54°C, while the pressurizer shed had temperatures as high as 94°C. These types of variations make it difficult to define a 'normal' environment. In an ageing programme it is important to establish temperatures and other environmental factors at the locations of the equipment of interest.

Vibration

Vibration, may induce ageing mechanisms in instruments installed on piping, pumps, motors or any other structure that has some vibrational stressors. The problem is in determining the location of the equipment and the associated vibrational spectrum. Transmitters are normally mounted on bulkheads or other stable structures which dampen any vibrations such as those that emanate from motors or operating valves. For accelerometers, vibration is an input to the instrument. These instruments are typically mounted downstream from a valve, and the indication of whether that valve is open or closed depends on the sensing of pipe vibration. The ranges of frequency and the amplitude of such vibrations can be sizeable. Normal plant vibration of this kind is considerably different from vibration due to a seismic events, which is not an ageing problem but a design basis event (DBE) problem.

Radiation

The in-containment environment will invariably include radiation. However, the level varies considerably with location. The electronic equipment is generally located in areas of lower radiation levels.

In one study [31] of radiation inside an LWR containment but outside the secondary shield wall, where instruments and equipment are likely to be found, gamma dose rates varied between 4×10^{-3} and 7.4×10^{2} rad(tissue)/h, with the majority of the dose rates in the 10^{-2} to 10^{-1} rad(tissue)/h range.

Using a plant operating factor of 0.9, these dose rates translate into 40-year total gamma doses of 1 x 10^3 and 2 x 10^8 rad(tissue), respectively. However, the majority of measurements indicate that a 40-year integrated dose of 3 x 10^4 rad(tissue) may be more plausible. The neutron measurements show total neutron dose rates between 4.0 x 10^{-6} and 0.54 rad(tissue)/h. Neutron energy spectrum measurements showed no neutrons with energies above 500 keV. Most of the neutrons are below 300 keV, with the median energy somewhere between 40 and 100 keV. Thermal neutrons were estimated to account for about 20-30% of the flux (approximately 7-10% of the dose). If the dose is assumed to derive from 100 keV neutrons, these dose rate numbers for energy absorption in tissue convert roughly to 1.5 x 10^{-8} and 2.1 x 10^{-3} rad(Si)/h respectively. It should be recognized that these values represent nominal values rather than extremes. These numbers should be used only as an indication of the general reference for the in-containment radiation environment.

The current estimates of total gamma dose and neutron fluence over the 40 year life of the plant in some containment areas (3 x 10^4 rad(tissue) and 10^9 - 10^{14} n/cm^2) are in Table 6.4.2.2. These levels are near the radiation tolerance threshold for many (non-hardened) semiconductor devices and integrated circuits. Battisti et al. [32] showed that the operation of electronic components and circuits is affected by reactor neutron fluences of the order of 10^{13} n/cm^2, and some components completely failed at a fluence of 10^{14} n/cm^2. Estimates of the sensitivity of the components to Co-60 irradiation indicate that resistors are usable to 10^{11} rad(Si), capacitors to 10^{10} rad(Si) and semiconductors to 10^5 rad(Si). However, these numbers do not derive from the low dose rate, long term irradiations expected in power plants.

It should not be considered that operating conditions where neutron fluence is present are insignificant or that Co-60 gamma radiation is a good simulation of the radiation environment. Lambert, et al. [30] have compared the damaged caused by radiation to electronic components (resistors, capacitors, silicon diodes, integrated circuits, and operational amplifiers) irradiated in different radiation fields. They found that, in comparison with nuclear reactor irradiation, the effect of Co-60 gamma rays can be smaller by a factor of 100 or more. They conclude that care must be exercised in interpreting the results of radiation damage tests of electronic components in different radiation environments.

6.4.2.3 Electronic components and materials; characteristics and mechanisms relevant to ageing

A review of much of what is known about the ageing of materials and components of interest for nuclear power plant applications is given in a report by Carfagno and Gibson [4]. We summarize here some of the information contained in that work as well as information from other reviews and from individuals knowledgeable in this area [47]. Much of the emphasis here is on degradation and failure mechanisms and on radiation effects relevant to ageing.

Resistors

Resistors are among the most stable electronic components. With respect to steady state nuclear radiation damage, wire wound resistors are probably the most stable of all electronic components. Radiation exposure has never caused a permanent resistance change of more than 25% [34]. Resistors are less vulnerable to gamma radiation than to neutron exposure. Variable resistors used in calibration applications (potentiometers) under the same conditions and when operated within power ratings are not expected to degrade significantly with age [4]. Potentiometers used in control applications may experience a wear ageing mechanism if high use is required.

Carbon film resistors are, in general, less susceptible to radiation damage than carbon composition resistors [34]. Resistance variations due to construction, substrate and manufacturing processes are frequently of the same order of magnitude as changes due to radiation. For dose levels below 10^3 rad(C), radiation degradation appears to be less than or equal to that due to ambient condition of humidity, temperature or power level. Resistance changes of the order of 25% have been found for some of these resistors irradiated to 10^8-10^9 rad(C) gamma and 10^{13} fast-neutrons/cm^2. Degradation may be due to atomic displacements, change in coating resistance or formation of current leakage paths.

Carbon composition resistors are the least stable of the various resistor types. Radiation damage effects are often obscured by variations due to temperature, humidity and applied voltage. These resistors are generally used in non-critical service where $\pm$ 20% tolerance to allow for radiation damage would not be a problem.

Thermistors are thermally sensitive resistors and consist of mixtures of oxides and a binder. Metal oxide thermistors have stable characteristics in a reactor environment (to 10^{12} rad(C) gamma and 10^{12} fast-neutrons/cm^2). Silicon thermistors are considerably more sensitive to radiation [34]. The binder and type of oxide are the key factors in determining whether the thermistor can withstand large radiation doses, since the oxides have n-type or p-type characteristics. A shift in the ratio of n-type to p-type under irradiation is one of the mechanisms of failure.

Thin film resistors of tantalum nitride have been found to be stable without encapsulation. At 60°C their expected resistance change is less than 100 parts per million in 10 years, and at 125°C the change is about one part per thousand. These resistors do degrade under pulsed electrical operation. Depending upon the pulse conditions, ageing can be accelerated almost 1000 times over continuous operation. Care must be taken in pulsed ageing since the resistor can be destroyed by mechanisms related to temperature and voltage operating conditions which differ from (continuous operation) ageing mechanisms. Thin oxide resistors have experienced problems with incompatibility between the oxide and substrate.

Thick film resistors are produced from a metallic ink containing metals such as palladium and ruthenium by screen printing and then firing at an elevated temperature. The ageing characteristic appears to depend upon the resistor type (ink blend, etc.). Multiple failure mechanisms may be operative and these are difficult to quantify.

Capacitors

In capacitor ageing the two most important environmental variables are electrical stress and temperature. However, temperature cycling, humidity and radiation effects are also important. The principal cause of changes in capacitance during irradiation is dimensional change of the interelectrode spacing. This dimensional change is most pronounced with organic dielectrics.

Glass and porcelain capacitors are the most radiation resistant [4, 34]. They are stable to 10^{10} rad for gamma doses. The radiation sensitivity of ceramic type capacitors appears to vary depending on the manufacturer. Mica capacitors are reported as being usable to 10^8 rad. Generally, early failure can be eliminated with appropriate burn-in procedures (screening by testing under stress conditions). Main causes of failure of mica capacitors are attributed to corona degradation and of ceramic button capacitors to temperature cycling (producing dimensional changes). Although the failure mechanisms are not well understood, the application of the Inverse Power Model has proved, in many cases, to fit the test data well and to allow practical prediction of life.

Plastic and paper capacitors (thin films of paper, polystyrene, polyethylene or mylar with foil or deposited metal as electrodes) are much more sensitive to radiation than capacitors of inorganic materials such as glass, mica and ceramic. Plastic dielectric capacitors are more sensitive by a factor of ten than the inorganics, and paper capacitors are more sensitive by factors of 10^2 to 10^3. Oil-impregnated paper is more susceptible to failure than plain paper because of the formation of gaseous products which can increase pressure and produce mechanical failure as well as electrical failure. Polymer films also experience swelling and changes in morphology which affect ageing. In general, paper capacitors are reported to be unsuitable for use in the containment area (threshold failure levels are of the order of 10^6 rad gamma). Mylar capacitors experience total failure at 10^8 rad and are susceptible to moisture.

Electrolytic capacitors of aluminium and tantalum suffer significant ageing and failure for gamma doses of 10^4 to 10^8 rads [4]. The tantalum capacitors are generally more radiation resistant. In normal operation, the failure mechanism in aluminium electrolytic capacitors is loss of electrolyte, and this is a direct function of the core temperature. The maximum limit of temperature is 100°C. Aluminium capacitors generally have a short shelf life. The predominant failure mode in solid tantalum capacitors is electrical shorting caused by impurities. The ageing mechanisms for electrolytic capacitors can be accelerated by elevated voltages and temperatures.

Semiconductor devices

Thermal ageing mechanisms [4, 33, 35]

Experience with failure testing of semiconductor components (diodes, transistors) has shown that most semiconductor failures are log normally distributed and that an Arrhenius reaction rate model can be used to describe the relationship between temperature and median life. Apparent activation energies for good quality, burned-in transistors in integrated circuits are of the order of 1.0 eV. Several types of failure (or degradation) mechanisms have been identified for diodes and transistors. These include electromigration (0.5 eV), aluminium-silicon contact degradation (0.8 eV), surface degradation (1.0 eV) and aluminium-gold bond degradation (1.0 eV).

Procedures for estimating the median time to failure are well established when failure is a function of temperature only. Ageing procedures and models for other failure modes and for synergism due to different stresses (voltage, radiation, vibration, etc.) are less well understood.

A list of ageing mechanisms and degradation factors is given in Table 6.4.2.4. Elevated temperature can enhance chemical reactions in contact areas, diffusion of intermetallic materials at bonds and ageing of plastic materials which could release chemicals to react with other materials. Reduced temperature can produce higher failure rates owing to condensed moisture within a package and hot electron effects in n-channel metal oxide semiconductor (MOS) devices. Temperature cycling can induce fatigue stressing of materials, bonds, and seals in hermetic packages and at interfaces between the silicon device chip and the package.

Table 6.4.2.4

Failure Mechanisms in Silicon Semiconductor Devices [33, 35]

Material	Mechanism	Degradation factors
Oxide and oxide/semi-conductor interface	Surface charge buildup	Ionic conductivity, temperature, voltage, radiation
	Dielectric breakdown	Electric field, temperature
	Charge injection	Electric field, temperature
Metallization	Electromigration	Current density, grain size, geometry, temperature
	Corrosion	Humidity, contamination, temperature, voltage
	Contact degradation	Metals, impurities, temperature
Bonds	Intermetallic growth	Impurities, bond strength, temperature
	Fatigue	Temperature cycling, bond strength
Encapsulation	Diffusion and seal leaks	Atmosphere, pressure, humidity

Applied voltage and heat can accelerate effects associated with ionic movement. This can lead to surface inversion on transistor structures, surface charge movement on insulator surfaces, dielectric breakdown and diffusion of impurity ions (e.g. chloride ions) in plastic encapsulated devices. The accumulation of charge on the surface of the oxide or nitride layer can cause an inversion layer or channel to form in the base silicon, causing a phenomenon called 'parasitic MOS' which can lead to device malfunction.

Humidity in combination with applied voltage and raised temperature can enhance corrosion mechanisms. Corrosion of aluminium frequently produces a break in the circuit, whereas gold may produce dendritic growths which produce shorts. The corrosion of aluminium is also accelerated by the presence of chloride and sodium ions, and by phosphorus ions in the oxide near the edge of the protective oxide. The gold metallization corrosion mechanism is accelerated by trapped moisture and contaminants in cracks and pinholes in protective layers over the gold.

Current and higher temperatures can produce electromigration effects [36] which are dependent upon current density, metal grain size, metal alloy, metal line geometry, temperature and current gradients. Current induced metal atom transport can cause hillocks of metal at one end of the line and voids at the other, producing a break in the circuit.

Radiation ageing effects

In a radiation environment, semiconductor devices and integrated circuits in an electronic component are more senstive and vulnerable than resistors and capacitors. Typical tolerances for present electronics are 10^4 rad(Si) gamma and 10^{13} n/cm^2 for commercial hybrids and integrated circuits. For specially fabricated or selected radiation hardened devices tolerances of 10^6 rad(Si) gamma and 10^{14} n/cm^2 are typical [37]. These levels may be approached during normal operation of the station.

Most radiation testing to date has been related to nuclear weapons and space applications. Data on total dose radiation effects are available on many semiconductor device components and integrated circuits (see for example Refs. [38-40]). Some data are published on the devices used in reactor instrumentation. However, dose rate information in the range of interest for nuclear plant instrumentation is not readily available from laboratory testing. Furthermore, the interaction of operational temperature and long term irradiation has not been investigated.

In bipolar devices of the type used in some instrumentation in the containment, the surface states and charge buildup in the oxide passivation layer due to gamma radiation lead to increased leakage currents and current gain degradation [41]. The gain degradation is a function of the voltage bias condition during irradiation. Neutron irradiation damage in bipolar devices results in large reductions in transistor gain. The dominant mechanism for displacement damage is a reduction in the minority carrier lifetime.

To understand the effects of the radiation environment of the nuclear plant on semiconductor devices more work is needed on low dose rate, long term radiation effects. Most radiation effects tests use total dose test procedures and are accomplished over a short period of time compared with the actual time in practice.

A 'true' dose rate effect is defined as one that arises from actual physical processes occurring in the gate oxide. An 'apparent' dose rate effect involves circuit effects such as may arise from voltage bias or from annealing that can occur during the course of irradiation. (The annealing can be a function of bias applied both during and after irradiation). Lower dose rates appear to be less damaging simply because the longer time of irradiation allows some of the earlier damage to anneal. Although models have been developed which predict 'true' dose rate dependent effects (involving recombination lifetimes and space charge buildup), no 'true' dose rate dependence has been observed [42]. Apparent dose rate effects, however, have been observed. Although there are some general observations in the literature on such effects, some are occasionally contradictory.

6.4.3. Polymers [22]

Polymers are substances which consist of long macromolecules built up of small molecules (monomers) or groups of molecules as repeated units. They are divided into four groups: thermoplastics, elastomers, intermediate substances and thermosetting materials (Fig. 6.4.3).

POLYMERS

Thermoplastics	Intermediate Substances	Elastomers	Thermosetting Materials
Brittle or tough at normal, plastic and fusible at higher temperatures.	Materials showing characteristics of thermoplastics as well as of elastomers, used cross-linked or non-cross-linked.	Rubber or rubber-like substances hard-elastic at low, rubber-elastic and infusible at normal and higher temperatures, used cross-linked only.	Hard or brittle; infusible at all temperature up to the point of chemical decomposition. Curing during processing.

Fig.6.4.3. Classification of polymers.

Thermoplastic materials are brittle or tough at normal temperature, but become plastic at higher temperatures. They are fusiable and weldable and may be dissolved in certain organic liquids at higher temperatures.

Rubber and other materials having rubber-like characteristics are elastomers. They are hard elastic at low temperatures, becoming rubber-elastic at normal and higher temperatures, and are insoluble, but may swell and are not fusiable. These characteristics are due to the long molecular chains of cross-linked monomers. The cross-linking, usually referred to as vulcanization, is normally performed after shaping by heating the rubber compound, which contains special cross-linking agents. Elastomers are a popular choice for forming the insulation and cable sheath for electrical cable but are also extensively used in the formation of seals and gaskets. O-rings for both static and dynamic applications are chiefly made from elastomeric materials.

Intermediate substances form a group of materials commonly referred to as vinyls, which possess fewer rubber-like properties but are not brittle and tough.

Thermosetting materials are strongly cross-linked and stay hard or brittle at all temperatures (up to the point of chemical decomposition). They are unsuitable for insulation or sheaths of cables, but are used for insulating blocks, washers, casings and cable ties.

The addition of various other elements such as halogens (chlorine, fluorine, bromine, etc.) provides additional characteristics to the polymers to enhance their radiation absorbent, moisture absorbent and flame retardant properties.

Materials with a wide range of characteristics may be obtained by combined polymerization of different monomers. The group of rubber-like materials in which we find many synthetic rubbers and other substances with rubber-like characteristics, apart from caoutchouc, which is turned into rubber by vulcanization, may serve as an example.

Of all the polymers, polyethylene has the simplest chemical structure:

```
    H   H   H   H   H   H   H   H   H
    |   |   |   |   |   |   |   |   |
H - C - C - C - C - C - C - C - C - C - C - ...
    |   |   |   |   |   |   |   |   |
    H   H   H   H   H   H   H   H   H
```

The corresponding monomer is ethylene. Ethylene is important also as the basis of other polymers used with cables. The monomer of PVC for example, may be obtained by the chlorination of ethylene.

Ethylene is furthermore a basis for EPM, EPDM and EVA All these materials have in common that the long straight chains of carbon atoms that make up the macromolecules,have only single bonds and are saturated. It follows that they are particularly resistant to the influence of heat and oxygen, in contrast to unsaturated polymers having double bonds between carbon atoms, as for example natural rubber.

Most of the known elastomers contain a certain number of carbon double bonds (C==C). These are necessary for conventional vulcanization using sulphur or sulphur compounds. During this process only some of these double bonds are used up, however. The rest remain free within the molecular structure. At higher temperatures oxygen may be deposited on these double bonds, causing the material to deteriorate (ageing). A certain measure of protection against this ageing may be achieved by means of stabilizing additives.

The mechanical properties of the polymers, e.g., tensile strength, elongation, elasticity and resistance to cold, which depend on their differing chemical structures, and their resistance to external chemical influences, for example acids, bases or oils, together with their electrical and thermal characteristics, are the decisive factors determining the usefulness of cables insulated and sheathed with these materials.

6.4.3.1 Polymers and their application

The development and use of polymers over the last 50 years has been extraordinary. They have found application in almost every device and item of equipment in manufacture today.

Polymers are virtually everywhere and serve vital functions in electrical and mechanical equipment. Table 6.4.3.1 lists typical polymers and their applications. This list is by no means comprehensive but serves to demonstrate the wide variation in materials, properties, lists and applications [43, 22].

Text continued on p. 126.

Table 6.4.3.1.(a) Identification and Applications of Thermoplastic Resins

Material Identification	Fillers	Examples of Reference Codes	Applications: Material Used in Low Voltage Systems: Protection Envelopes	Supports of not alive parts	Supports of alive parts	Material Used in Medium Voltage Systems: Protection Envelopes	Support of not alive parts	Support of alive parts	Some Typical Applications
Polypropylene/PP	- glass fibres - asbestos fibres - talc	ASTM D 2146 UNI 7055 MIL-G-19978 MIL-L-39022 MIL-P-46109 MIL-B-52472	X	X		X	X		- boxes, components of lighting fixtures embedding of low voltage components
Polytetrafluoro-ethylene/PTFE, FEP (Copolymer TFE/hexafluorine propylene) PCTFE (polychlor-trifluorethylene)	- glass fibres	ASTM D (1430 (2116 ASTM D 1457	X	X	X	X	X	X	- parts of breakers, bushings, sheathing of electric componetns, insulating of high frequency coaxial cables, parts of kinematic motions
Cellulose Acetate/CA Cellulose Acetate Butyrate/CAB Thyl Cellulose/EC	- glass fibres - asbestos fibres fibres	ASTM D706-707 ASTM D1502	X	X		X			
		DIN 7742-7743 MIL-P-46074	X	X	X	X	X	X	
			X			X			
Nylon 66 Nylon 6 Nylon 11 Nylon 610 Nylon 612 Nylon 12	- glass fibres - glass micro-spheres	ASTM D (789 (2897 MIL-M-19887 MIL-M-22096 MIL-M-20693 ISO-R-1874	X	X	X	X	X	X	- Envelopes of batteries, connectors, terminal strips, parts of breakers & relays, insulators, plugs, insulating rods, parts of kinematic motions

Table 6.4.3.1.(a) (continued)

Material Identification	Fillers	Examples of Reference Codes	Applications: Material Used in Low Voltage Systems – Protection Envelopes	Support of not alive parts	Support of alive parts	Material Used in Medium Voltage Systems – Protection Envelopes	Support of not alive parts	Support of alive parts	Some Typical Applications
Polyurethane/PU			X	X		X			– envelopes, embedding, raceways
Polyethylene Texephthalate/PET Polybutyterephthalate/PBT	– glass fibres	MIL-P-46160 MIL-P-46161	X	X	X	X	X		– envelopes, parts of breakers, parts of kinematic motions
Polycarbonate/PC	– glass fibres	ASTM D (2473 (2474 DIN 7744	X	X	X	X	X		– envelopes, parts of breakers, cable terminations, insulators, parts of kinematic motions, connectors
Polyvinylchloride PVC, PVC/PP (Polypropylene PVCC (Chlorinated	– mineral fillers	ASTM D (1755 (1784	X	X	X	X	X	X	– insulating and jackets of cables, protective conduits, boxes
		ASTM D 2287 DIN 7748	X		X				
		UNI 7350	X		X				
Polypropylene Oxide/PPO	– glass fibres	MIL-P-41631	X	X	X	X	X	X	– parts of electronic components, parts of kinematic motions
Polysulfone/PPSU	– asbestos fibres	MIL-P-46120 MIL-P-46133	X	X	X	X	X	X	

Table 6.4.3.1.(a) (continued)

Material Identification	Fillers	Examples of Reference Codes	Applications						
			Material Used in Low Voltage Systems			Material Used in Medium Voltage Systems			Some Typical Applications
			Protection Envelopes	Support of not alive parts	Support of alive parts	Protection Envelopes	Support of not alive parts	Support of alive parts	
Acetal/POM	- glass fibres	ASTM D2133-2948 MIL-I-28964 MIL-P-46137	X	X	X	X	X		- parts of kinematic motions
Acrylonitrite-Buta diene-Styrene/ABS ABS/PVC (Poly-vinyl chloride) ABS/PC (Poly-carbonate)	- glass fibres	ASTM D1 788-3011 ISO DIS 2580 UNI 7041	X	X		X			- envelopes, boxes, protective pipes, diaphragms, canalizations
Polymethylmetha-crylate/PMMA PMMA/MS (Methyl-styrene)		ASTM D788-1431 ASTM D1547-3011 UNI 7067-7074 DIN 7745	X						- level gauges, special pipes, components of lighting fixtures
			X	X		X	X		
Polystyrene/PS PS/SBR (Copolymer Styrene-Styrene Butadiene)	- glass fibres	ASTM D (703 (1892 (3011 MIL-P (21347 (60312 UNI 7066-7073	X	X		X			- components of lighting fixtures, function boxes

Table 6.4.3.1.(a) (continued)

Material Identification	Fillers	Examples of Reference Codes	Applications: Material Used in Low Voltage Systems			Material Used in Medium Voltage Systems			Some Typical Applications
			Protection Envelopes	Support of not alive parts	Support of alive parts	Protection Envelopes	Support of not alive parts	Support of alive parts	
Polyethylene PE LD or HD (low or high density)	- glass fibres	DIN 7741 ASTM D (2103 (1248 DIN 7740 UNI 7054	X	X	X	X	X	X	- cable insulation, pipes, plates, embedding of electronic circuits

Table 6.4.3.1.(b) Identification and Applications of Thermosetting Resins

Material Identification	Fillers	Examples of Reference Codes	Applications						
			Material Used in Low Voltage Systems			Material Used in Medium Voltage Systems			Some Typical Applications
			Protection Envelopes	Support of not alive parts	Support of alive parts	Protection Envelopes	Support of not alive parts	Support of alive parts	
Phenolic/PF	- asbestos fibres - glass fibres - mica scales - wooden floor - paper - cotton - nylon	ISO R 800 UNI 4303 DIN 7708 ASTM D 700 MIL-M-14 BS-771	X	X	X	X	X		- supports, frame, envelopes of instruments, small site power and internment transformers; - parts and components of circuit breakers; - push buttons and parts of selector switches, etc.
	- glass fabric - asbestos sheets - cotton fabric - paper	CEI 15-5,15-10 DIN 7735 NEMA L11 ISO 1642	X	X	X	X	X		- insulating plates, diaphragms, supports of circuit breakers & panels; - parts of oil filled transformers
Melaminic/MF	- asbestos fibre - glass fibre - other mineral - fillers - wooden fillers - cellulose tufts - cotton	UNI 4302 DIN 7708 ASTM D 704 MIL-M-14	X	X	X	X	X		- components used in humid or polluted environment - parts of circuit breakers - terminals
	- glass fabric - asbestos sheets - cotton fabric	as PF	X	X	X	X	X		- insulating plates, supports, parts of breakers and switchgear - printed circuits

Table 6.4.3.1.(b) (continued)

Material Identification	Fillers	Examples of Reference Codes	Applications: Material Used in Low Voltage Systems – Protection Envelopes	Support of not alive parts	Support of alive parts	Material Used in Medium Voltage Systems – Protection Envelopes	Support of not alive parts	Support of alive parts	Some Typical Applications
Urea/UF	- cellulose fibres	UNI 4302 DIN 7708 ASTM C 705	X	X	X				- small breakers, push buttons, junction blocks, plates, boxes, plugs, sockets, etc.
Phenolic-Melaminic	- mineral fillers - wood - cellulose fibres	DIN 7708 MIL-M-14	X	X	X	X			- microswitches, small terminals, etc.
Epoxy/EP	- glass fibres - other mineral fillers	DIN 16912 ASTM D 3013 MIL-M-24325	X	X	X				- various insulating pieces
	- glass fabric - paper	AS PF	X	X	X	X	X		- printed circuits - insulating plates, diaphragms, etc.
	- mineral fillers	MIL-I-16923	X	X	X	X	X	X	- coils, cable terminals - sheathing of electric conductors, etc.
	- quartz powder - hydrated alumina				X	X	X	X	- stand-off and bushing insulators - parts of breakers, switchgear and connectors - instrument and power transformers - capacitors - parts of electric motors

Table 6.4.3.1.(b) (continued)

Material Identification	Fillers	Examples of Reference Codes	Applications: Material Used in Low Voltage Systems — Protection Envelopes	Support of not alive parts	Support of alive parts	Material Used in Medium Voltage Systems — Protection Envelopes	Support of not alive parts	Support of alive parts	Some Typical Applications
Polyurethane/PUR	- quartz powder - dolomite - hydrated alumina - glass micro-sphere	VDE 0291	X	X	X	X	X	X	- instrument and power transformers - coils - insulators and supports
Polyester/UP	- fibres - galss mat	DIN 16911 MIL-M-14 ASTM D 1201 DIN 16913	X	X	X	X	X	X	- parts of breakers & switchgear - parts of electric motors - insulators - diaphragms, envelopes, boxes, etc.
	- glass mat - cotton - paper	NEMA L1-1 DIN 7735 ISO 1642	X	X	X	X	X	X	- insulating plates, diaphragms, supports, - bars, parts of breakers and switchgear
	- glass micro-sphere - quartz powder, mica			X	X	X	X	X	- coils
Alkyd	- glass fibres - asbestos fibres - other mineral fillers	MIL-M-14	X	X	X	X			- parts of breakers, electric motors & various electrical components of high characteristics
Alkyl/DAP	- glass fibres - other mineral fillers	ASTM D 1636	X	X	X	X			- parts of breakers, connectors, electronic and electric components of high characteristics

Table 6.4.3.1.(c) Identification and Applications of Elastomers

Material Identification	Main Properties	Applications						
		Material Used in Pneumatic Systems		Electric cable	Material Used in Fluid Systems			Some Typical Applications
		Static Pressure Retaining	Dynamic Application		Static Pressure Retaining	Dynamic Application	Oils/Fuels	
Natural rubber (NR)	good good tensible good resistance to: impact, abrasion, water, alcohol, dilute acids, oxydation, weathering, high temperature, flammability, oils, hydrocarbons concentrated acids and bases	X	X		X	X	X	Pneumatic tires, tubes, transmission belts, gaskets, shock asborbing element sponges
Polyisoprene (IR)	similar to natural rubber better resistance to: oxydation and inorganic acids lower tensile and tear strength	X	X		X	X	X	Same as natural rubber
Polybutadienne (BR)	mechanical properties somewhat lower than natural rubber better resistance to agression and flex cracking limit resistance to ozone, weathering, oils, fuels, solvants	X	X		X	X	X	Pneumatic tires, soles, gaskets, seals, belting used in blends with other elastomers

Table 6.4.3.1.(c) (continued)

Material Identification	Main Properties	Applications: Material Used in Pneumatic Systems		Electric cable	Material Used in Fluid Systems			Some Typical Applications
		Static Pressure Retaining	Dynamic Application		Static Pressure Retaining	Dynamic Application	Oils/Fuels	
ISO butylene ISO prene (IIR)	Mechanical properties lower than natural rubber, polybutadiene and styrene but poor resistance to fuels and oil. Low gas permeability high electrical IR and dielectrical strength high resistance to ozone oxydation, weathering and water, good resistance to steam, acids, alcohols, esters and heat ageing	X	X	X	X	X		auto tires, inner tubes, steam hoses, diaphragms, electrical cable insulation, vibration absorption
Styrene-Butadiene	Excellent abrasion and impact resistance. Good resistance to flex cracking water, alcohol, poor resistance to ozone and hydrocarbons	X	X	X	X	X		Same as polybutadiene

Table 6.4.3.1.(c) (continued)

Material Identification	Main Properties	Applications: Material Used in Pneumatic Systems – Static Pressure Retaining	Material Used in Pneumatic Systems – Dynamic Application	Electric cable	Material Used in Fluid Systems – Static Pressure Retaining	Material Used in Fluid Systems – Dynamic Application	Material Used in Fluid Systems – Oils/Fuels	Some Typical Applications
CHLORINATED ELASTOMERS . Chlorinated isobutylene isoprene . Polychlororene (CR) (NEOPRENE) . Chlorinated polyethylene (CM) . Chlorosulfonated polyethylene (CSM) (HYPOLUN) . Epichloronydrids (CO, ECO)	Improves resistance to fuels and oils flammability. Good resistance to ozone oxidation, weathering, water, fuels, alcohols and heat ageing	X	X	X	X	X	X	Electrical wire|cable, betts, hoses, adhesives, gaskets, petroleum and chemical tank linings and hoses.
ACRYLIC RUBBERS (AR)	Excellent resistance sulfur bearing oils & fluids. Good resistance to ozone, oxidation, weathering, hydrocarbons Poor resistance to· aromatic & halogenated hydrocarbons, alcohols & acids, moderate tensile strength and abrasion, flex cracking, tear, impact.	X	X	X	X	X	X	Automatic ignition wire insulation, spark plug boots, power stearing hose, motor mounts, and transmission seals

Table 6.4.3.1.(c) (continued)

Material Identification	Main Properties	Applications: Material Used in Pneumatic Systems – Static Pressure Retaining	Applications: Material Used in Pneumatic Systems – Dynamic Application	Applications: Electric cable	Applications: Material Used in Fluid Systems – Static Pressure Retaining	Applications: Material Used in Fluid Systems – Dynamic Application	Applications: Material Used in Fluid Systems – Oils/Fuels	Some Typical Applications
Acrylunitrile Butodienne (Nitrile or Buna N) (NBR)	Good resistance to oils and fuels, non-oxidizine acids, mechanical properties better than (AR) Low permeability, High resistance to water Moderate resistance to weathering	X	X		X	X	X	Gasoline, chemical, oil hose gaskets, seals, "O" rings, conveyor belts
Ethylene Propylenediene (EPOM)	Good electrical & dielectric strength Good resistance to ozone, oxidation, weathering, water, steam, heat ageing, radiation, dilute acids, bases, alcohols, esters, abrasion, impact, flex cracking	X	X	X	X	X		Electrical insulations, hoses, conveyor belts, pipe and tank linings, vibration mounts, "O" rings, seals gaskets

Table 6.4.3.1.(c) (continued)

Material Identification	Main Properties	Applications						
		Material Used in Pneumatic Systems		Electric cable	Material Used in Fluid Systems			Some Typical Applications
		Static Pressure Retaining	Dynamic Application		Static Pressure Retaining	Dynamic Application	Oils/Fuels	
Fluorocarbons (FPM)	Good resistance to fuels oils, hydraulic fluids heat, ozone, oxidation, weathering, good tensile strength				X X	X X	X X	High temperature seals, "O" rings, lined valves, roll covering shaft seals
Polysulfide	Excellent resistance to fuels, oils, alcohols, ketones, esters Good resistance to ozone oxidation, bad weathering Low permeability Poor mechanical properties	X			X		X	Seals, gaskets, solvent hose valve seals, flexible mountings
Silicones . Methyl (MQ) . Vinyl Methyl (VMQ) . Phenyl Methyl (PVM) . Phenyl Vinyl Methyl (PVMQ) . Fluril Vinyl Methyl (FVMQ)	Excellent resistance to oxidation, weathering, heat ageing Good resistance to ozone corona Poor mechanical properties Phenylgroups (PMQ, PVMQ) show improved resistance to radiation	X		X	X			High/low temperature Electrical insulation, seals, gaskets, diaphragms, "O" rings

Table 6.4.3.1.(c) (continued)

Material Identification	Main Properties	Applications: Material Used in Pneumatic Systems		Applications: Electric cable	Applications: Material Used in Fluid Systems			Some Typical Applications
		Static Pressure Retaining	Dynamic Application		Static Pressure Retaining	Dynamic Application	Oils/Fuels	
Polyurethanes (AU, EU)	Excellent mechanical properties . tensile strength . tear . abrasion Excellent resistance to: ozone, oxidation Poor resistance to: halogenated hydrocarbons, acids Good low temperature performance	X	X		X	X		Tubing, bladders, seals, gaskets, wheels, casters, "O" rings, rollers

6.4.3.2. Service Environments and Ageing Mechanisms

Polymers used in conjunction with electrical and electronic equipment will experience the same environment as described in subsection 6.4.2.2. However, polymers are also used in mechanical applications such as for diaphragms in pressure switches, dynamic piston seals, electrical penetration modules, and even large equipment batch seals for containments. The environment for polymers includes not only the ambient conditions in the NPP buildings and containment but also direct exposure to internal process fluids, gases and lubricants.

It is fortunate that it is possible to produce such a wide variety of materials from polymers to function in these varied environments but this advantage also limits our ability to discuss generic environmental effects and degradation processes that are operative in polymeric materials. Given that one can only deal superficially with this area, the following environmental and electrical stressors will be discussed:

- temperature
- radiation
- humidity/electrical
- hydrocarbon (oils, fuels, solvents)
- ozone.

These stressors rarely act independently but usually in conjunction with one or more others, resulting in rather complex degradation mechanisms.

Traditional theory unfortunately deals with these stressors independently as ageing stresses and the resulting ageing mechanism. There are limits to the use of independent variable analytical techniques in understanding ageing and its effects; however, it does permit an understanding of the way a particular stressor affects the polymer.

Temperature

Temperature is traditionally believed to be the chief stress parameter promoting ageing during the material's normal service. A well known theory (of Arrhenius) is available to account for the effects of temperature on material over a time scale. Temperature index schemes have been developed to rate the material's retention of its original properties at rated temperatures.

One standard scheme [44] for moulded thermal setting plastics is to age material samples thermally at higher than usual temperatures until an endpoint of 50% reduction in tensile strength is reached. With several temperature level/50% endpoint time data points, an Arrhenius curve is plotted. This case of time versus temperature yields a temperature corresponding to a 100000 hour time point. This is then called the temperature index. One might be misled that the natural overall yield satisfactorily continues the performance and original characteristics at this temperature when in fact a 50% reduction in tensile strength has occurred.

Another scheme [44] for establishing the temperature index is one used for elastomers. The method is the same (Arrhenius). However, the endpoints are 70 hours and a hardness change $\pm$ 15 points, a tensile strength change $\pm$ 30% or an ultimate elongation change of $\pm$ 50%, whichever occurs first. The temperature at which the elastomer degrades to one of these levels within 70 hours is its temperature index.

These examples point out that the temperature index is not only a measure for rating material for service but also at this rating significant ageing of the material occurs.

The primary degradation process is a chemical reaction. Hence the applicability of the Arrhenius methodology. Oxidation is the primary chemical reaction operative in ageing at normal temperatures. At levels above its temperature index, secondary chemical reactions or other processes may dominate the degradation mode. These effects are often seen in accelerated thermal ageing tests.

Thermal stress usually results in polymers becoming harder (embrittlement), losing tensile strength and losing elastic qualities, and it induces cracking as it ages.

Most basic polymers are compounded with antioxidants which are effective in retarding the degradation but not in eliminating it.

Radiation

The oxidative reaction is dominant in thermal stress environments but is also operative in radiation stress environments. The process of chemical reaction is shown in Fig. 6.4.3.2.1. In the process a free radical is created through gamma interaction, then through chemical propagation additional radicals are established. These radicals in turn may cause scission (breaking of polymer chains), cross-linking (creation of additional/different polymer chains), disproportionation and non-radical products (through combination with antioxidants added to the polymer). In addition, non-radical products created during the propagation process can decompose and create free radicals which are available for chemical combination.

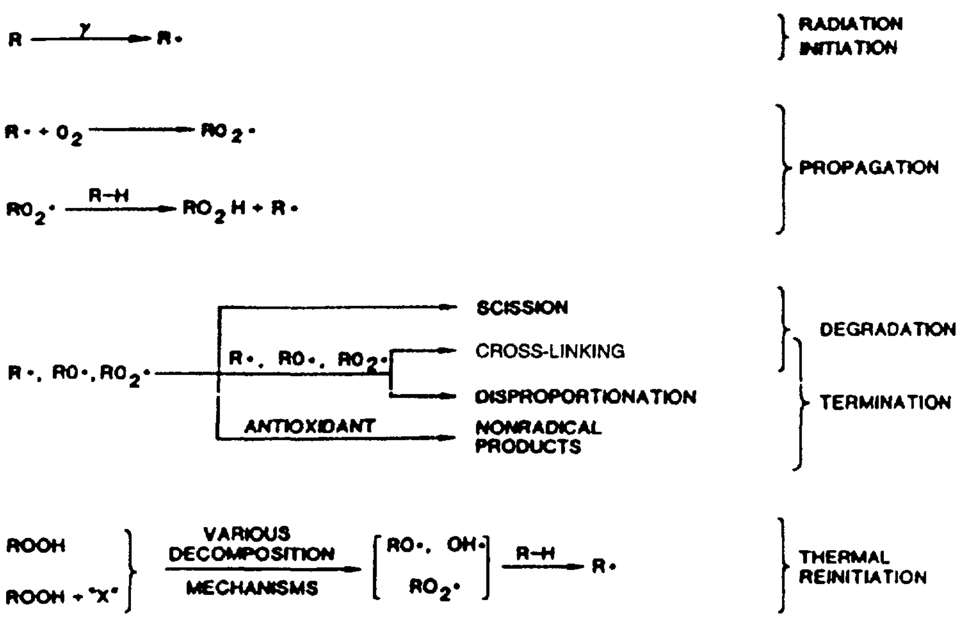

Fig 6.4.3.2.1. Chemical mechanism for radiation induced oxidation.

Scission and crosslinking are the two most commonly observed phenomena in polymer degradation under radiation stress environments. Some of the typical changes the material will see are given in Table 6.4.3.2.

Table 6.4.3.2

Primary Radiation Degradation in Polymers

SCISSION	CROSS-LINKING
INITIAL EFFECTS	INITIAL EFFECTS
Reduction of (- molecular weight, - Young's modulus	Increase of (molecular weight, Young's modulus, viscosity
- Reduction of ultimate tensile stress	- Increase of ultimate tensile stress
- Increase of elongation	- Reduction of elongation
- Reduction of hardness	- Increase of hardness
- Increase of solubility	- Reduction of solubility
	- Increase of softening temperature
- Reduction of elasticity	- Reduction of elasticity
	- Production of gases
- Embrittlement	- Embrittlement
- Formation of gases	- Formation of glassy solid
- Reduction of melting temperature	

In both thermal and radiation induced degradation the amount of oxygen present for reaction is significant. Fig. 6.4.3.2.2 is a dramatic demonstration of the results of exposure of a polyethelene material to the same conditions of temperature and radiation, but one set of samples was in air, the other in nitrogen. The tensile elongation (expressed as a ratio of original to exposed) changed very little over an exposure time of 100 days, whereas with oxygen the polyethelene essentially lost all elasticity.

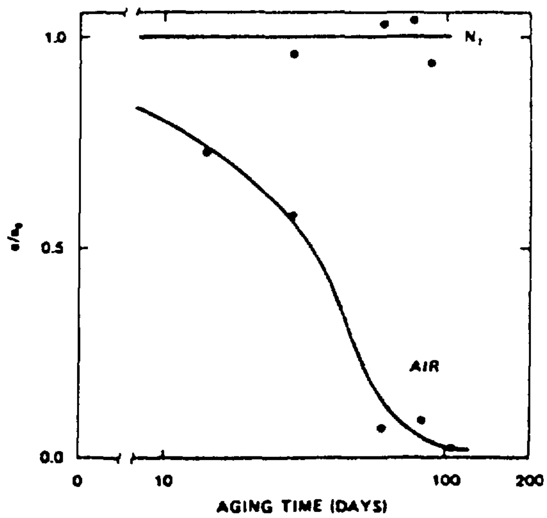

Fig. 6.4.3.2.2. Tensile elongation results for PE insulation as a function of ageing time at 5 krad/h, 80°C in atmospheres of air and N_2 [46].

The rate of material degradation may be sensitive to the radiation dose rate and lower dose rates may produce more damage for the same absorbed dose. These are commonly referred to as dose rate effects. Often material selection for equipment and electrical cable applications for radiation areas is based on tests that were performed at high dose rates (> 10^5 rad/h). At these rates the material's internal oxygen may quickly be absorbed and the chemical reaction significantly slowed by the depletion of the oxygen. Material permeability factors may then limit and control the rate of oxidation degradation. In actual power plant environments the low dose rates will facilitate renewal of the oxygen in the material. Thus actual material ageing may advance quicker than predicted if the polymer is sensitive to this effect. Thermoplastic, intermediate substance, and elastomeric polymers often display this sensitivity.

Figure 6.4.3.2.2 also describes a potential peroxide decomposition mechanism that produces free radials for further chemical reaction. This decomposition reaction was found to be significantin a PVC insulated cable material. The extent of this chemical decomposition process is not known for polymers in general.

Humidity, solid pollutants, electrical stress

The effect of humidity on polymers has been investigated in conjunction with radiation and thermal environments. These studies indicate that there are polymers where humidity may be a significant factor. However, if the polymer is resistant to the effects of humidity then the polymer's ageing behaviour under combined radiation and humidity and thermal and humidity progresses at a normal rate. The degradation phenomena are classified into surface effects and volume or internal effects.

Surface effects are only significant when humidity is combined with a sufficient voltage stress and solid pollutants such as dust. Superficial electrical arcing which initiates carbon tracking and erosion are the mechanisms that result.

Volume effects are related to the moisture absorption properties of the polymer. Swelling and hydrolysis mechanisms reduce the mechanical and electrical insulating properties of the polymer. Moisture absorption properties of polymers are readily available from manufacturers as standard engineering information and generally polymers are available with excellent moisture resistant properties.

Therefore the primary ageing mechanism related to humidity is surface tracking and erosion which can be controlled by maintaining the solid pollutants at low levels in electrical terminal blocks, circuit boards, etc.

Oils, fuels, and solvents

Polymers are also used in applications where there is contact with oils, fuels and solvents, such as diaphragms for pressure switches and transmitters made of elastomers, vacuum type pumps for fuel oil transfer as well as gaskets and seals. Solvents are often used to clean such equipment or systems.

The degradation process is by absorbtion and leaching. Once absorbed the polymer may swell, resulting generally in reductions in hardness, tensile strength and stiffness. Elongation properties can be augmented or impaired. Leaching can also occur, which can remove plasticizers and other components of a compounded polymer. If leaching is the predominant mode, general effects just the opposite to those of swelling are observed. In this case typical effects are hardness, stiffness and sometimes increase of tensile strength with a reduction of elongation.

Polymers are tested for their sensitivity to oils, fuels and solvents and cautions about limitations on use are available as standard engineering data. In addition, standard ratings have been established by organizations such as ASTM that characterize swelling and other factors.

Substituting oils or the use of solvents that come into contact with polymers may cause accelerated degradation owing to the aforementioned sensitivities.

Ozone

Although the concentration of ozone is normally very low, increased concentrations occur during lightning storms, around high voltage equipment where corona are is present or where electrical arcing occurs. For elastomers sensitive to ozone attack, a concentration as low as 0.0005% is sufficient to initiate cracking.

The ozone reacts with double bonds in the polymers. For unhelped material a thin sticky film or shield can form that actually limits further ozone attack. Where the polymer is elongated, such as in elastomer O-ring applications ozone causes the formation of surface cracks which then propagate, leading to reduction in tensile strength and elongation.

The rate of degradation is related to the type of polymer and its specific formulation (i.e. compound), mechanical stresses and the ozone concentration.

6.4.4. Lubricants - oils and greases

6.4.4.1. Applications of lubricants

Lubricating oils and greases are used in essential motors, pumps, gear boxes, slideways and pivoting devices [21]. The proper functioning of the machinery is dependent on proper lubrication.

Traditional methods of surveying ageing of greases and oils are through periodic replacement at selected intervals for small volume applications and sample testing for large volume applications. Large volume applications of lubricating oils often have filter systems to maintain contaminants at low concentrations.

6.4.4.2. Service environments and ageing mechanisms

The normal operating environment for lubrication is essentially the same as in any other industry in which the machine is in service, except for radiation environment. In addition, nearly all equipment dependent on a lubricant to remain functional receives a radiation dose well below any levels that could be damaging during its normal service, with a few exceptions. Table 6.4.4.2.1 lists these and gives typical levels depending on the reactor type.

Table 6.4.4.2.1

Estimated Radiation Levels for Different Nuclear Reactor System Rads/year

Component	Organic Moderated Reactor	Pressurized-Water Reactor	Liquid Metal Cooled Reactor	Boiling Water Reactor
Control Rod Mechanisms	10^8 max	10^5 - 10^8	10^4 - 10^5	10^6 max
Fuel Handling Devices	-	1 x 10^7 max	-	-
Primary Coolant Pumps	Negligible	10^6 - 10^7	10^8	10^6 - 10^7
Auxiliary Pumps	Negligible	10^4 - $3x10^6$	-	10^4 - $3x10^6$
Auxiliary Motors	-	10^5 - $3x10^7$	-	10^5 - 10^7
Turbines	25 max	10 max	Negligible	80 max

The primary causes of lubricant ageing are contamination of the lubricant and thermal ageing due to oxidation. These are well understood phenomena and will not be elaborated on in this section. Perhaps less well understood is the ageing due to radiation, on which this subsection will concentrate.

From these figures it is apparent that, in normal operation most components are subjected to relatively low doses of irradiation which will not call for specialized lubricants. During a five year cycle between overhauls only control rod mechanisms would be subjected to accumulated doses of radiation calling for specialized products.

The effect of radiation on fluid lubricants

There are four readily recognized ageing properties of an oil which change when it is subjected to radiation, which have been described in detail by Hollinghurst [48] with mechanisms explaining these occurrences.

(1) Viscosity changes occur that may be either increases caused by polymerization (cross-linking) or, in the case of certain synthetic products, chain breaking reactions (scission) causing decreases in viscosity.

(2) Chemical change accompanied by evolution of gas which will result in changes in fluid density.

(3) If air is present oxidation will occur which may be detected by infrared analysis or by changes in the acid value of the oil concerned.

(4) Readily detectable darkening in colour occurs.

Of the four property changes listed, it is the first, changes in base oil viscosity, and the third, oxidation of the base oil, which have the most significant effects on grease performance.

The changes in base oil viscosity can have the most significant effect when the order of change becomes double to four times the original value. At this level the lubricated bearing is being required to operate with an unsuitable fluid and this can often result in an increased operating temperature and hence accelerated deterioration of the grease. Conversely, if the result of radiation was to break down the lubricating fluid to a lower viscosity product, the bearing could operate with a fluid affording insufficient film thickness for correct lubrication and hence undergo rapid lubrication failure.

The effect of oxidation of the base oil is in general to increase the base oil viscosity. In this case, however, the viscosity increase is due to the products of oxidation (sludge), which in general accelerate further oxidation and, owing to their normally acidic nature, reduce the service life of the lubricated component.

Table 6.4.4.2.2 gives an example of some changes found by Vaile [49] to occur when a base oil is subjected to various levels of irradiation.

Table 6.4.4.2.2

Effect of Different Levels of Irradiation on a Solvent Extracted High VI Base Oil

Irradiation Megarads	0	10	100	1000
Specific gravity	0.863	0.864	0.365	0.881
Viscosity 37.8°C, cSt	32.6	32.8	35.8	154
Viscosity Index	107	109	117	118
ASTM Colour	1 1/2	1 1/2	2 1/2	5 -
Acid No. mg KOH/g.	0.05	0.05	0.05	0.15

The effect of radiation on thickeners and additives

The principal effect of radiation on thickeners, whether they are of the soap type or inorganic, is to reduce their efficiency and, as a result, to soften the consistency of the grease. The manifestation of thickener damage is often oil release which can, under some conditions, result in sufficient concentration of the thickener to compensate for lost thickener efficiency. Measurements of lost thickener efficiency have to be carried out by carefully selected techniques. This is because it has been found by Cox and Crawford that use of tests involving shearing of the grease to examine lost thickener efficiency results in partial 'healing' of the radiation damage, especially when this is a surface phenomenon, as in the case of organically modified clay thickeners.

It is not possible to make any generalization about the effect of radiation on additives except to point out that their efficiency in general is a function of their reactivity or ability to 'neutralize' the products of undesirable reactions. It has been found from infrared analysis of irradiated oils containing antioxidants that a large proportion of the normal doseage will be consumed in an oil sample irradiated to 500 megarads; thus, in a grease subject to irradiation, it can be presumed that the antioxidant will be consumed at a higher rate than would occur under the same operating conditions without irradiation. It has also been noted that oils containing certain EP additives develop high levels of sludge when irradiated. This sludge comes from the breakdown of the EP additives. It is therefore preferable that greases likely to be subject to irradiation should not contain high levels of EP/anti wear additives sensitive to radiation.

The effect of radiation on fully formulated greases of different types

Fig. 6.4.4.1 [50] shows the way in which the radiation resistance of each of the two principal components of a grease, the thickener and the fluid, can influence the overall stability of the grease under irradiation. Curves 1 and 2 represent the variation in consistency of greases having respectively a fluid and a thickener of limited stability irradiation. The third curve is that given by grease C, formulated to meet the requirements of a Schedule 1 grease for control rod lubrication for gas cooled reactors. The tendency for the grease made with the unstable fluid to thicken is at first masked by a softening caused by lost thickener efficiency; subsequently the polymerization of the fluid which occurs is so overriding in its effect that the product becomes rubbery and stiff, making it impossible to determine worked penetrations on the most highly irradiated samples.

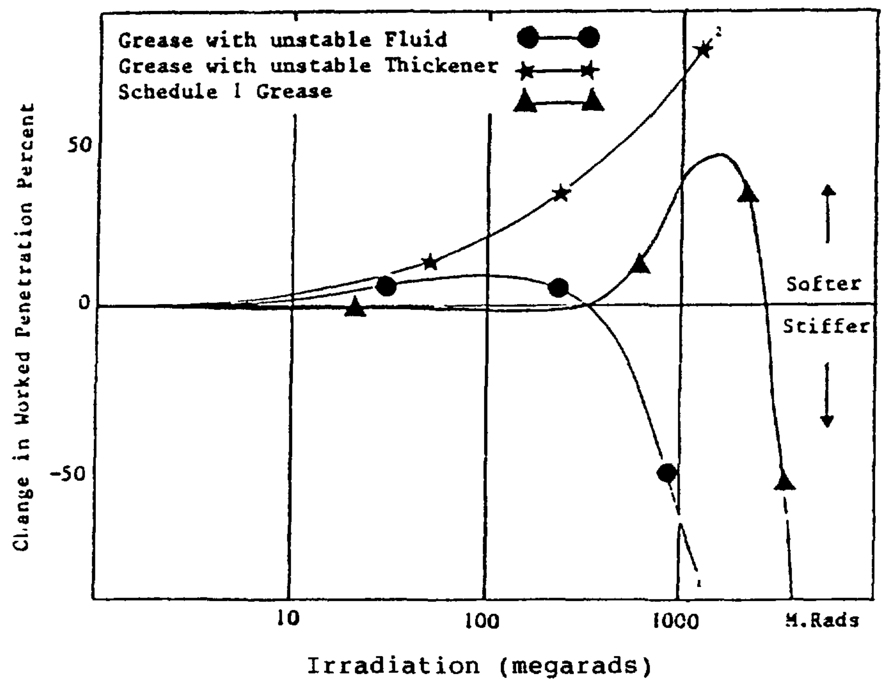

Fig. 6.4.4.1. Effect of radiation on greases of various stabilities.

6.4.5. Laminating resins [15]

Circuit boards are among the items whose mode of degradation and accelerated ageing are of considerable importance. While not pertaining directly to this subject, information on laminating resins and glass fibre structures may be of interest. Information is available on some 18 types of laminating resins used in glass fibre structures for microwave applications. The effect of thermal ageing on the laminates is illustrated in Table 6.4.5.1, Fig. 6.4.5.1 and Fig. 6.4.5.2.

Table 6.4.5.1

Effect of Ageing at 500°F on Flexural Strength of Style 120 Glass-Fabric Laminates

(Resin content approximately 40% weight)

Resin	Initial flexural strength ($lbf/in^2 \times 10^{-3}$)	Flexural strength after ageing for 100 h (percent)	Strength retention
DAIP	66.0	34.0	52
TAC Polyester	73.6	57.8	78
Diolefin	41.2	26.8	65
Epoxy	93.1	67.0	72
Novelac 1			
Polyester 1	73.0	34.1	47
Epoxy 1	99.2	62.5	63

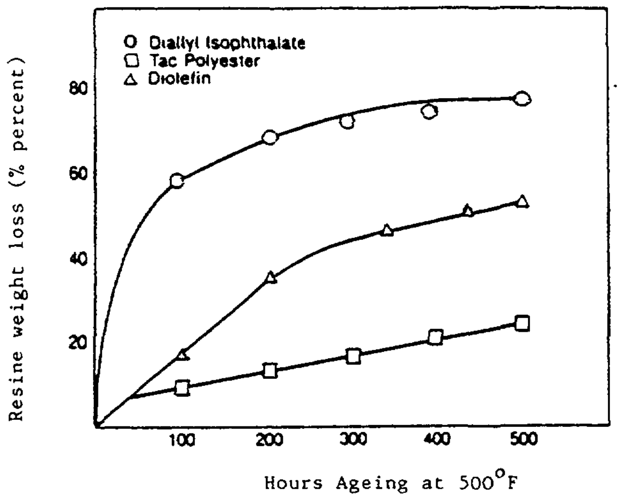

Fig.6.4.5.1. Percentage resin weight loss of style-120 glass fabric laminated during ageing at 500°F.

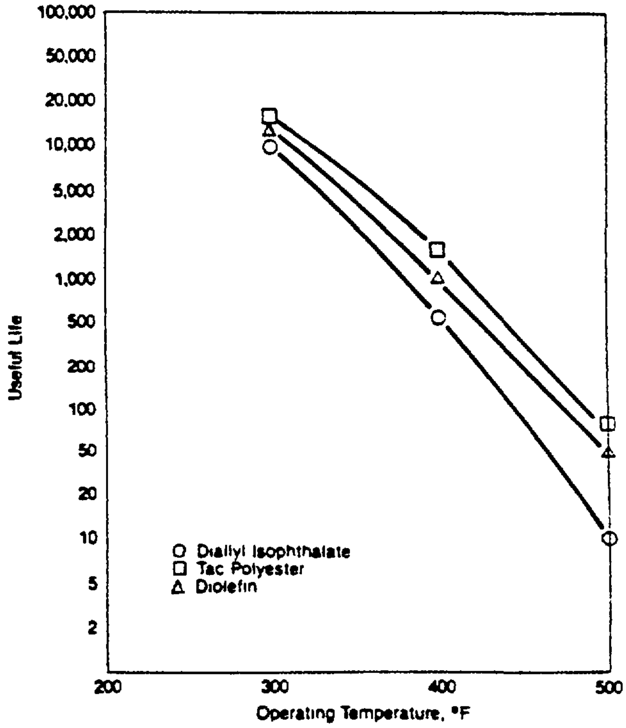

Fig.6.4.5.2. Thermal life curves of style-120 glass fabric laminated feeddomes, based on 10% maximum possible resin weight loss for a useful life.

The electrical properties of glass reinforced epoxy circuit laminates are satisfactory under normal conditions; however, under conditions of high humidity it is common to observe a loss of intercircuit impedance and fleshover strengths, especially where circuits involve plated through holes. A propriatory circuit laminate considered representative of the products commonly available has been examined. As received, such laminates have a volume resistance of 10^{15} ohms or better; however, electrical defects develop, with some variation in time, upon exposure to relative humidity of 95% or 100%, either at 20°C or upon temperature cycling. Volume resistance perpendicular to the face of the laminate, and therefore also to the glass reinforcement, is much higher than that parallel to the face, and is little affected by humidity exposure.

The resistivity parallel to the surface - and the glass - is not only considerably lower, but also more viable and directional in that plane. This anisotropic behaviour may be expressed by differences of some five orders of magnitude. Low resistance paths in the volume of circuit boards, as indicated by the first precipitous resistance drop, developed after an initial humidity exposure of the order of days. After removal from humid to room air conditions, high resistance was regained in minutes or hours; however, on recycling to high humidity, a sample whose resistance had previously fallen to a low value would often drop to the same low value in a period of hours rather than days, as in the first cycle.

A more detailed study of this failure behaviour suggests a physical model which ascribes the low volume resistance after humidity exposure to conduction in 'voids' accompanying filaments or threads in the glass cloth reinforcement. The existence of such voids is supported by independent work of others for different purposes. The voids may be structural defects, arising at the time of manufacture owing to lack of bonding between glass and resin interfaces, which tend to increase in number and extent with processing and thermal or humidity stress.

Since the glass reinforcement is definitely involved in the low resistance behaviour of the circuit laminates, substitution by organic reinforcement, such as Dacron or Nomex, or use of non-reinforced circuit boards might be suggested. However, these constructions tend to have other defects, such as poor heat resistance, lack of dimensional stability and mechanical impracticability for many purposes. In a multidielectric system (involving different dielectrics) the interface usually is the wearer of the either of the two dialectrics.

Commercial epoxy glass circuit laminates degrade in insulation resistance under high humidity service or test conditions, and leakage currents may increase to values unacceptable for high impedance circuits. Surface resistance may be controlled by cleaning and use of conformal coatings. The decrease in volume resistance has been ascribed to conductance in fine capillary voids paralleling individual glass filaments. These may be formed in the original process of manufacture and tend to grow in number and extent in processing and under thermal or humidity stress. Impedance loss may become critical with high impedance circuit elements.

6.4.6. Protective Coatings [15]

Protective coatings applied to the interior surfaces of reactor containment facilities are expected to be resistant to gamma radiation doses of the order of 10^9 rad and to withstand the high temperature/pressure, steam/chemical spray conditions associated with a design basis event (DBE). They must not flake or become detached from the surface as a result of a DBE, since the plugging of strainers, flow lines, pumps, nozzles or cooling channels by coating debris could make the safety systems inoperative. It is also desirable that the coatings can be easily decontaminated.

Coating tests were carried out to provide guidance for the writing of a coating qualification standard, ANSI N101.2-1971. The cooling systems were tested by exposure to two different borated solutions, either by submersion in the solution or by exposure to a spray; during these tests, the temperature in the vicinity of the specimens followed a profile that ranged from 300°F to 225°F.

Other tests included blowdown tests, steam/gamma radiation and air/gamma radiation tests, followed by autoclave testing. Such tests could serve as a suitable basis for evaluating protective coatings for service in pressurized water reactor or boiling water reactor containment facilities. Coatings which appeared to meet containment facility requirements included epoxy, modified epoxy, modified phenolic, inorganic zinc and polyurethane. Inorganic zinc primers, used as a prime coat, downgraded the performance of other coatings. Gamma irradiation of 1.0×10^9 rad do not appear to deteriorate or alter the performance in subsequent tests. The main cause of failure of most coating systems was blistering, especially of the top coat, caused by temperature intolerance rather than by chemical attack.

The change in physical properties of several exterior coatings as a function of accelerated ageing in the Atlas-Weather-Ometer has been studied. The coatings included a variety of vehicle and pigment combinations, either solvent based or in latex form, derived from linseed oil, alkyd, acrylic and vinyl tape vehicles. These materials were selected because of prior knowledge of their behaviour with respect to outdoor durability, and because they differed in chemical composition from those which have been described for use in nuclear facilities.

The methods and tests used for monitoring the properties of the plain films included thermal mechanical analysis, differential scanning calorimetry, scanning electron microscopy, moisture vapour transmission, the Sward Rocker hardness test, torsional shear modulus and logarithmic decrement tests, surface morphology, tensile strength and elongation, and glass transition temperature tests. The following changes were observed as the duration of exposure in the Weather-Ometer increased: elongation decreased rapidly; decrement decreased; shear modulus decreased; tensile strength increased, then decreased; Sward Rocker hardness numbers increased; moisture vapour transmission decreased; cracks and chalking increased; and embrittlement increased. These effects were attributed to increasing cross-linking and polymerization of the vehicle.

Fairly good correlation was observed between the results of different tests, but formal correlation with respect to ageing was not attempted. Acceleration factors with respect to real time ageing were not determined except for the approximation that, for a particular composition, 800 hours of Weather-Ometer exposure was equivalent to about one year of ageing under outdoor environmental conditions.

No information is available on the ageing of protective coatings used on containment facility surfaces, although new coatings in DBE tests fail principally by blistering and delamination. Testing at present is done on new coatings, and it remains to be determined whether ageing will increase susceptibility to this mode of failure or initiate or aggravate some other failure mode such as powdering, flaking or subfilm rusting. Several mechanisms may operate during ageing, such as deterioration of the coating adhesion leading to easier delamination. On the other hand, more complete cure and more complete solvent removal during ageing may lead to less blistering due to severe thermal stress. Ageing procedures cannot be recommended until DBE tests have been performed on naturally or artificially aged coatings.

6.4.7. Low-temperature solder glass [15]

Solder glass is used in the sealing of television tubes and other electronic display components. It is a low melting point, high load vitreous glass and is known to fail by stress corrosion, which is a function of temperature and humidity. Studies of this phenomenon under conditions simulating actual in-service conditions lead to the following equation:

$$\log t_f = \log K - 15.5 \times 0.0084RH + (0.37/kT)$$

where

t_f	=	time to failure of seal in seconds
RH	=	relative humidity in per cent
k	=	Boltzmann's constant, 0.8617×10^{-4} eV/K
T	=	absolute temperature in degrees, K
k	=	a constant which must be determined for the particular glass and geometry of the seal.

The value of 0.37 eV found for the activation energy of the failure of the high lead glass is low compared with that for sodalime silica, and no mechanism can be readily identified for it. The value of the constant k appears to be in the range of 65 to 70 for a typical high lead solder glass. A list of acceleration factors calculated for failure is given in Table 6.4.7.1.

Table 6.4.7.1

Calculation of Acceleration Factors from Button Data for Highlead Glass

Service condition		Lab. test condition		Acceleration factor
(°C)	(%RH)	(°C)	(%RH)	(X)
85	81	85	81	1
re	81	85	81	5
56	72	85	81	14
60	60	85	81	94
20	30-40	85	81	< 500

7. DETECTION AND MITIGATION OF AGEING EFFECTS

7.1. Introduction

Ageing degradation of NPP components, if not effectively monitored and controlled, may result in impairment of their performance characteristics leading to a reduction of the reliability of associated station systems (both active and standby systems may be affected).

Effective control of ageing degradation of NPP components requires timely detection and mitigation of the degradation. (The term components is used in a broad sense and includes plant structures and systems.) Nuclear utilities have been using different programmes or methods to prevent, detect, correct and mitigate failures of NPP components from any cause, including the effects of ageing degradation. These methods include testing, surveillance, preventive maintenance programmes, significant event reporting systems, periodic reviews of NPP performance, and operational, maintenance and design changes initiated on the basis of detected component failures or deficiencies. They have been under continuous development (based on lessons learned and new knowledge), and overall quite effective in maintaining acceptable operational reliability of NPPs.

In the past, ageing was considered just one of many possible causes of component failures, and ageing research as well as design, operation, monitoring and maintenance improvements were primarily in response to experienced operational problems or failures. However, in recent years the industry, regulatory and international organizations recognized that as the average age of NPPs increase, a more systematic and proactive approach to NPP ageing is needed to ensure continued plant safety. As a result, a number of programmes or projects have been initiated by different Member States, IAEA, Nuclear Energy Agency of OECD, etc., to understand ageing and to utilize existing or develop new methods for managing its effects. Studies of operating experience and specifically of age-related component failures have shown that a majority of ageing degradation or failures can be detected by good performance monitoring and preventive maintenance programmes. Although no drastic changes are needed, there is a need to review these programmes for their effectiveness in detecting and mitigating ageing effects, and where necessary to enhance them by new and improved methods. To be effective, the applied methods must be able to detect and mitigate ageing degradation of safety-related components before failures. (In this document, failure is considered to occur when a component is unable to meet its minimum performance requirements.)

This section describes the general principles of the methods for timely detection and mitigation of ageing effects.

7.2. Performance monitoring as a basis for preventive maintenance

If allowed, uncontrolled ageing degradation of NPP components would increase the corrective (i.e. breakdown) maintenance workload and slowly degrade plant performance in terms of both plant safety and production reliability. To maintain adequate plant performance, plant personnel must be able to monitor and predict component and system performance so that appropriate preventive actions can be done before projected failure. Three means of performance monitoring, namely condition monitoring, failures trending and system reliability monitoring, are discussed below.

7.2.1. Condition monitoring

Condition monitoring of components may be defined as a continuous or periodic observation and evalution of appropriate indicators to assess components' ability to continue to perform their specified functions during a period following the time of observation. The prescribed observations may take the form of measurements or periodic tests or inspections designed to produce consistent, repeatable results, in which current performance or condition is determined. Observed values are then compared with minimum acceptance criteria and with results of the same previous observation on the same component. A determination of present and projected conditions is made utilizing both absolute and relative comparisons of observed indicator values to define when and what maintenance is required.

Ideally, monitoring a single condition indicator would suffice to indicate the functional capability and exact criteria would exist to decide, based on the observed value of the indicator, what action to take at the time of the observation, i.e. to continue operating without change, to perform maintenance or calibration, to repair the component, or to replace the component. An acceptable indicator for condition monitoring must be a precursor indicator, that is an indicator providing a warning of impending functional degradation that may not yet be apparent. A precursor indicator must therefore have a change which is detectable before component failure.

An illustration of the ideal situation is given in Fig. 7.2.1 [90]. The bottom half of the figure shows the time variation of two indicators of a component condition; the top half of the figure displays the time variation of the component functional capability. At the observation time although some component degradation has occured, there is no decrease of the component's function capability. The 'unacceptable indicator', which correlates strongly with the component's functional capability, does not show any change. The component degradation is, however, indicated by the decrease in the value of the 'acceptable indicator' which may be some component characteristic such as abnormal voltage, response time or mechanical wear.

Criteria can be established that require maintenance to be performed when condition monitoring parameters deteriorate to a specific level or vary in a specified manner. These criteria should relate to component performance under both normal and accident conditions, and therefore should be established on the basis of operating and equipment qualification experience or laboratory research. In Fig. 7.2.1 the interval between the moment of observation and the time corresponding to the minimum allowable functional capability is an estimate of the remaining service life of the component if no corrective measures are employed to enhance the life of the component.

Conceptually, a criterion may take the form of a statement such as: When the level of the condition indicator reaches the value A, preventive action B should be taken. For the purpose of illustration, consider an instrument cable, with all its connections, between an alarm in the control room and a pressure transmitter in the containment of a nuclear power plant. Suppose it was established in the instrument cables qualification programme that the leakage current in the cable measured at 500 Vdc is an acceptable condition indicator, and if its exceeds 2.0 mA, the pressure transmitter signal may have excessive error should a loss of coolant accident (LOCA) occur one year after the observation. The condition monitoring criterion should take the following form: If the leakage current of cable measured during prescribed periodic inspection exceeds 2.0 mA at 500 Vdc, the corrective action must be taken within a year to identify and correct the cause of the leakage current (the cable must be replaced if no defects are found or the repair of the

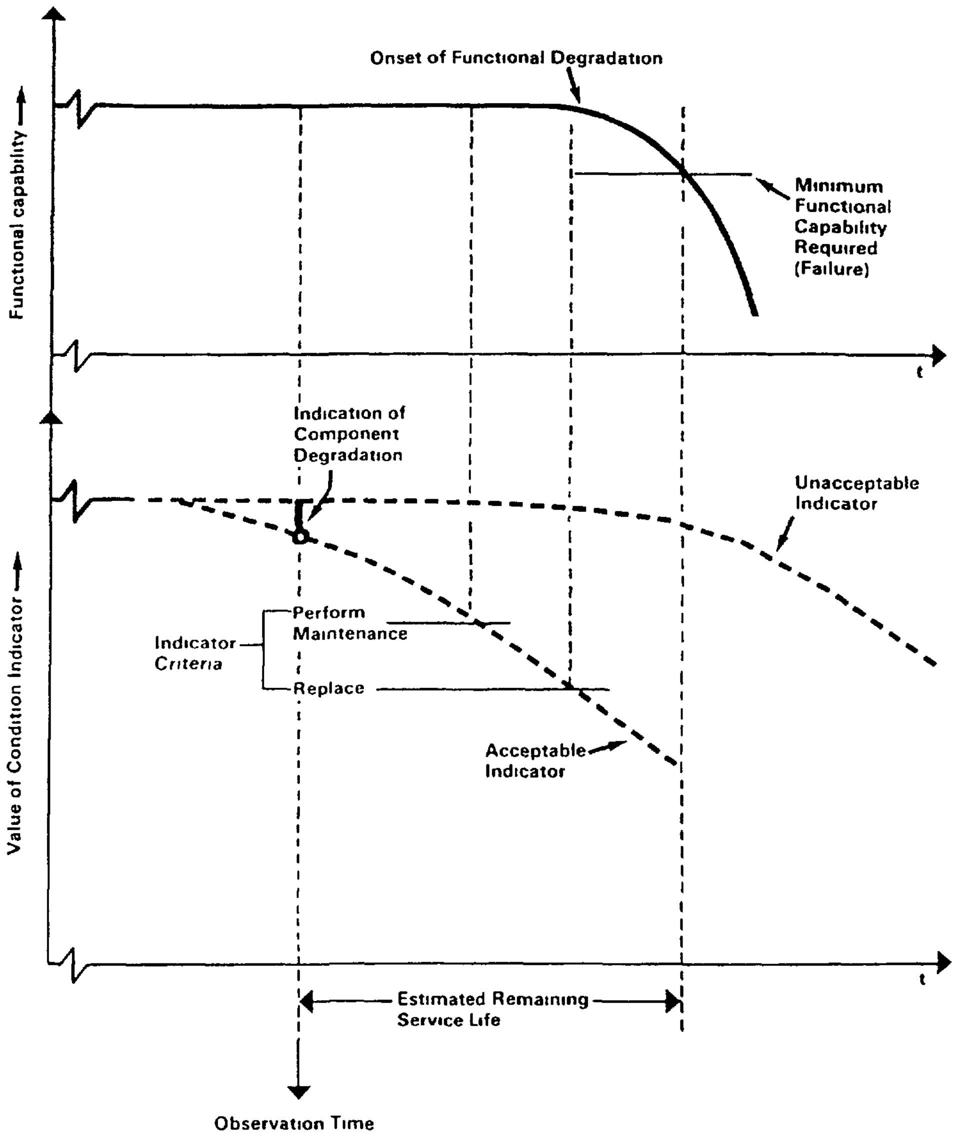

Fig. 7.2.1. Use of Condition Indicators for Condition Monitoring

identified defects does not remedy the situation). Since the rate of degradation is an important consideration for maintaining adequate functional capability, the timing of the corrective action should take into account the rate of change of the leakage current. The criterion may therefore include a requirement to increase the frequency of the leakage current measurements before the cable degradation is remedied.

The effect of maintenance on the functional capability of a component and its remaining estimated service life is shown in Fig. 7.2.2 [90]. With each maintenance, the component is normally upgraded and as a result the condition indicator value moves up, the functional capability curve moves to the right, and the residual service life is extended until a point is reached at which it becomes more cost-effective to replace the component. Of course, the replacement should be done sufficiently in advance of the expected component failure.

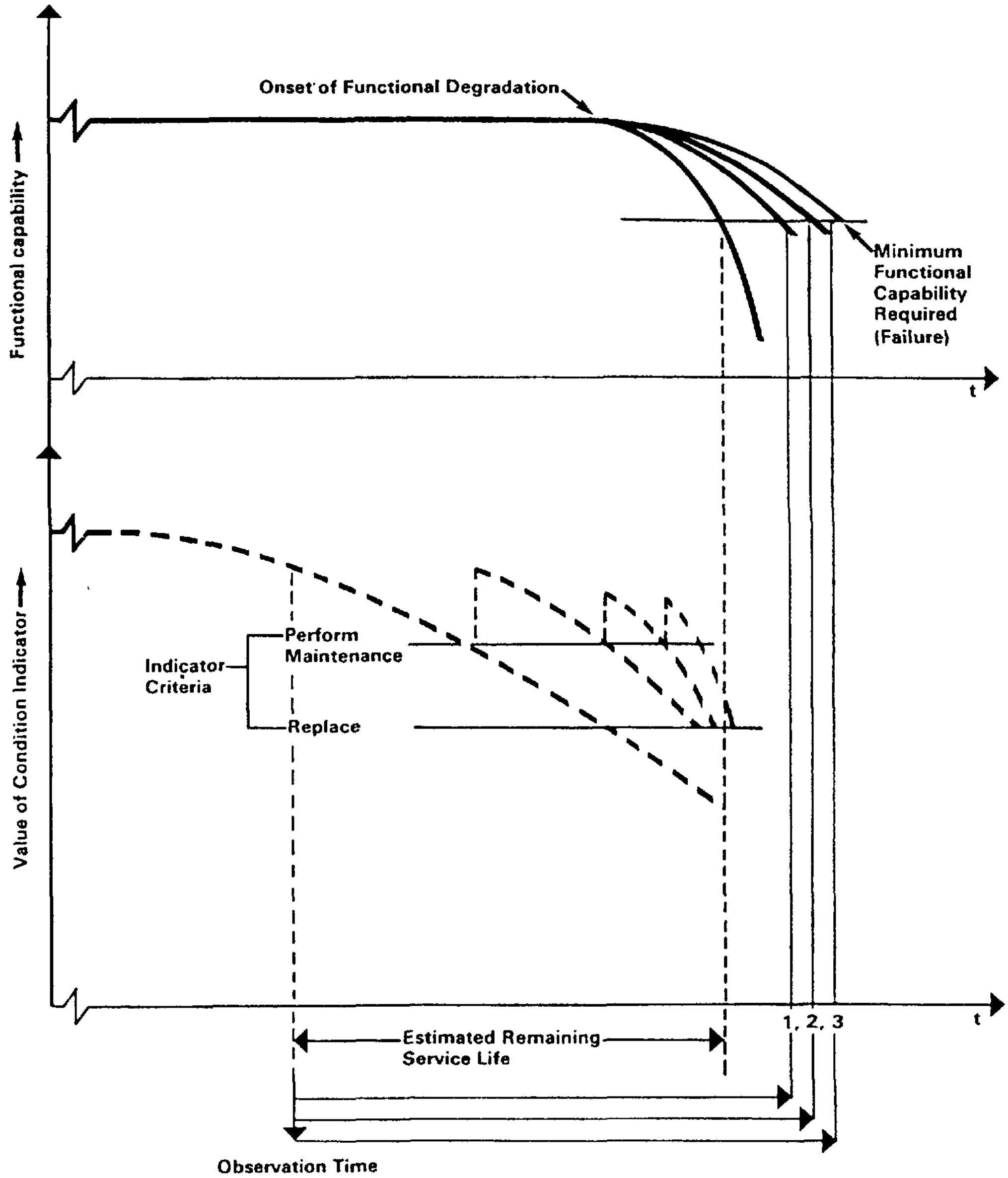

Fig. 7.2.2. Effects of Preventive Maintenance on Functional Capability and Remaining Estimated Service Life

In practice, the challenge is to identify indicators suitable for condition monitoring of different types of NPP components and the predictive criteria to decide what and when preventive action is required. A number of different condition monitoring techniques are being used and others are being developed on the basis of operating experience (including root-cause analysis of failures and correlation of observed indications with failure rate) and research and development (R&D). The techniques must be applied so as to ensure their repeatability which is prerequisite for condition and performance trending. This means that test conditions, such as process conditions, system valve lineup, test equipment accuracy and test personnel qualification, must be consistent from one test to another. This should be reflected in the condition monitoring programme procedures.

Examples of some of the common indicators of degradation include items such as voltages, currents, response times, setpoint drift and contact resistance change from routine tests and calibrations of instrumentation and control components. Also included are mechanical wear, corrosion, binding or bent parts from visual inspections or abnormal readings from comparisons of like process/operation parameters. Vibration monitoring of rotating equipment, motor current monitoring of motor operated valves and nondestructive examination techniques for pressure retaining components are three examples of condition monitoring methods which are briefly described below.

Examples of Condition Monitoring Methods

Vibration monitoring - Vibrations have long been known as indicators of rotating equipment condition. Long before the appearance of electronic vibration monitoring equipment, maintenance personnel were monitoring machine vibration by touching and listening to the machine. Modern piezoelectric accelerometers and velocity probes along with computer-based data acquisition and analysis equipment have brought a step change in the capabilities of vibration monitors. Detailed processing and analysis of the vibration spectra now allow automated diagnosis of machine problems. Since power plants have large numbers of rotating equipment, vibration monitoring has found widespread application [27] as a condition monitoring technique.

Monitoring of motor operated valves - The motor current during a valve stroke is a very useful diagnosis parameter for detecting and trending ageing degradation of motor operated valves (MOVs) [28]. From the standpoint of diagnostics, the motor is acting as a transducer, converting the drive train mechanical loads and their variations with time into variations in voltage across the windings. The resulting net current flow in the motor power leads then reflects the characteristics of the motor load. Changes over time reflect MOV degradation which appear as load changes.

Motor current can be measured remotely and non-obtrusively during plant operation, since no leads need to be lifted to attach a clamp-on current sensor. Reading and recording the motor current can be carried out rapidly, either manually or automatically. To utilize this technique for determining operational readiness, a data base is being developed, by Oak Ridge National Laboratory, to correlate diagnostic parameters with MOV degradation. The field tests were scheduled to begin in the summer of 1987.

It is anticipated that this technique will find application in the diagnostic monitoring of degradation in other motor driven components used in nuclear power plant safety systems, as well as in many non-nuclear applications.

A commercially available system for the assessment of motor-operator and valve functional performance is the MOVATS system [29]. The assessment is based on the measurements of stem thrust, motor-running current, bus voltage plus limit switch, torque switch and bypass switch activation positions within the valve stroke.

Non-destructive examination (NDE) techniques - NDE of pressure retaining components plays an important role in assuring their structural integrity. Procedures and detailed NDE techniques for periodic inspection are established by the utilities in accordance with various codes and regulatory requirements. NDE serves not only to detect component degradation, but also provides vital information about the materials, enabling appropriate corrective measures to be taken to prevent failure.

NDE techniques include eddy current methods, ultrasonic methods, radiography, magnetic particle testing, dye penetrant testing and visual inspection.

7.2.2. Failure trending

Another approach to component performance monitoring is to evaluate the statistical pattern of component failures over a period of time.

The failure rate of components as a function of time is given by the well-known bathtub curve, shown in Fig. 7.2.3. In their early life, which is called infant-mortality (or burn-in) period, components may experience a high failure rate. During their useful life, the failure rate of components is characterized by random failures, and the failure rate is approximately constant. Components experience an increasing failure rate in the late part of their life, which is characteristic of the wearout period. As shown in Fig. 7.2.3 electrical components generally experience a more constant failure rate in the useful life period than do mechanical components. The initial high failure rate of components can be eliminated in many cases before a plant is put into service by burn-in or commissioning testing. An effective performance monitoring programme can help to ensure that prompt corrective action is taken when the wearout region is reached.

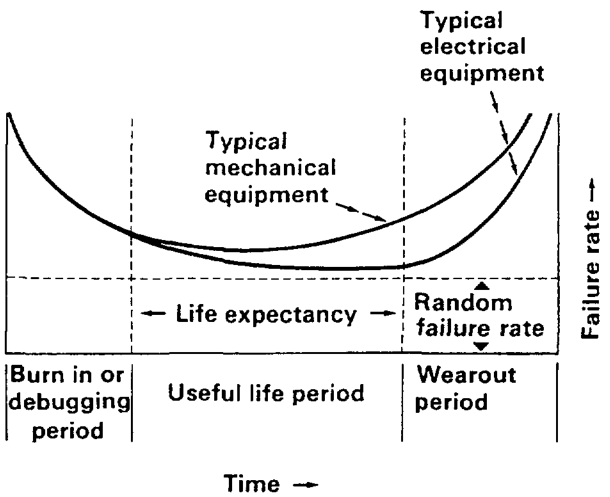

Figure 7.2.3. Time Dependence of Failure Rate

The failure pattern discussed here is statistical in nature, i.e., it indicates when the component will have higher probability of failure. If significant statistical data are available for such failure predictions, then surveillance, preventive maintenance, or replacement can be scheduled more effectively on the basis of failure trends. If the failure pattern of a device shows that the probability of failure increases significantly after a certain life service, scheduled replacement of the equipment, rather than replacement based on condition monitoring results, may be used.

Failure trending requires a systematic collection and analysis of failure data. Equipment deficiencies must be recorded in a specified, systematic manner to allow determination of the severity of failures, failure modes and the root cause/s of failures. Trends of failures and their causes may be then monitored. The component failure rate monitoring programme should periodically evaluate component failure rates and compare them against component lifetime experience and generic failure rate data. Because of the multiplicity of NPP components, this then may allow early recognition of increasing failure rate and timely elimination of developing weak spots before the failure of a system.

The failure rate pattern discussed above relates to normal service conditions. Establishment of frequency of inspection and maintenance must include consideration of the effect of postulated design basis event (DBE) service stresses. Figure 7.2.4 shows the effect of normal and accident service stresses on accumulated deterioration [90]. The DBE service conditions will cause stresses that are usually more severe that normal service stresses which include transients from process system upsets. Care must be taken in the development of inspection and maintenance intervals

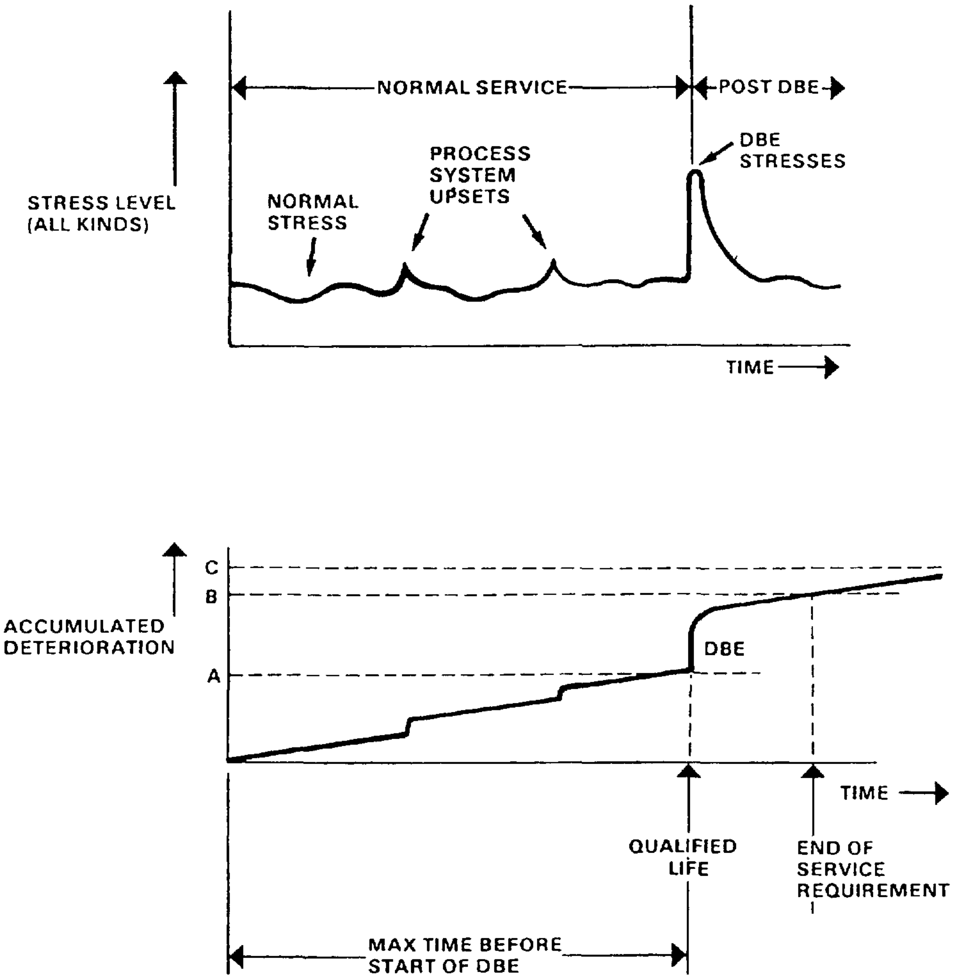

Figure 7.2.4. Accumulated Deterioration Related to Service and Design Basis Event Stresses (seismic, LOCA, MSLB)

A = maximum in-service deterioration allowed

B = maximum deterioration expected after DBE

C = end of life/component failure

B - C = performance/safety margin

MLSB = main steam line break

that sufficient capability to withstand DBE conditions remains throughout each interval. The combined effect of normal and possible DBE-related deterioration must be such that equipment is not allowed to significantly enter the wearout region where failure rates become exponential and the bulk of the equipment would be expected to fail. However, very few data exists relating the effect of DBE service conditions to changes in failure rate. Therefore, to use failure rate data as a basis of determining inspection and maintenance intervals, considerable conservatism must be incorporated to account for the effect of postulated DBE service conditions.

7.2.3. System reliability monitoring

System performance can be monitored using reliability methodology. This methodology permits a quantitative assessment of the system performance against performance standards or targets which can be established on the basis of experience, regulatory requirements, international recommendations and other considerations. If a performance target is exceeded, the cause (e.g. ageing degradation) as well as appropriate follow-up and corrective actions are identified to achieve and maintain a satisfactory level of performance. The corrective actions may include changes to testing, inspection and monitoring practices (e.g. frequency and parameters monitored), review and possible changes of design, operation and maintenance, and R&D work. Priorities for corrective actions should be set on the basis of the importance of the component in meeting the performance target of the system. The impact of the corrective actions on the system performance should result in an improvement of the reliability parameters followed by the reliability monitoring programme. Main elements of the process, outlined above, are depicted in Fig. 7.2.5 [55].

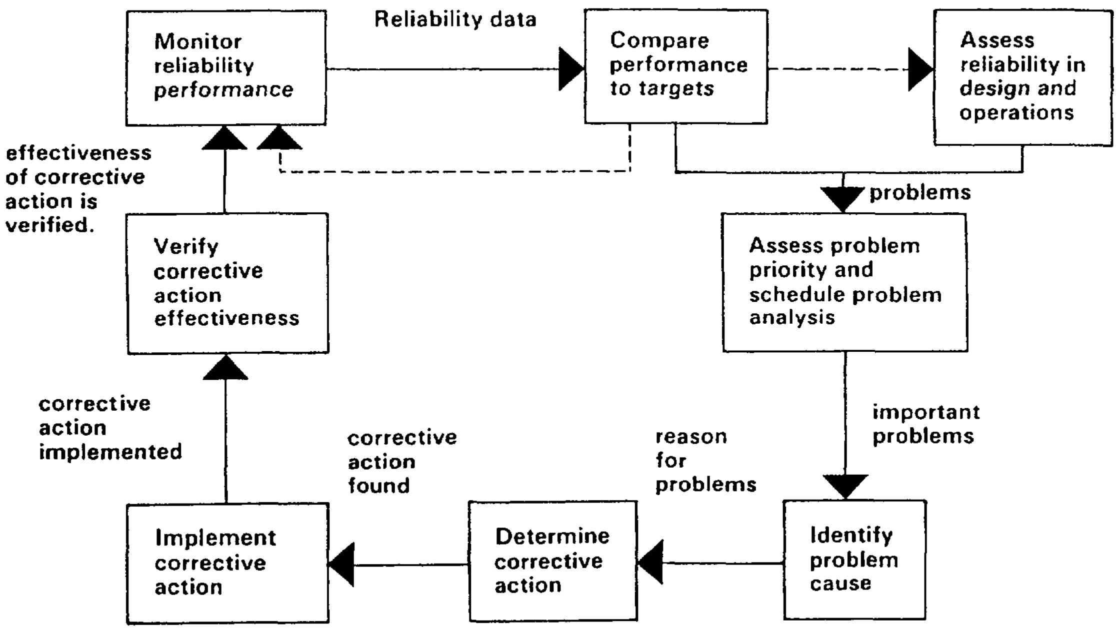

Figure 7.2.5. System Reliability Monitoring Process

A development of the system reliability monitoring programme involves the following steps:

- Setting of appropriate reliability targets

  (For examples, Canadian target for shutdown, emergency core cooling (ECC) and containment systems requires the unavailability of each system to be less than 10^{-3} a/a.)

- Deriving system reliability models

  (The level of detail of the models should reflect the expected quantity of failure data. For example, for the standby safety systems function block models may be used because of a limited amount of failure data collected due to the inherent reliability of each component.)

- Developing the test and inspection programme to provide confidence that standby systems would work when called upon

  (Components to be inspected, functions to be tested and the types and frequency of tests/inspections is determined using the system reliability target, reliability model, component failure rate data, the expected ageing processes and operating characteristics)

- Performing and recording of scheduled tests and inspections

- Analyzing test and inspection results to determine duration, severity and causes of observed failures

- Assessing periodically system operational reliability (i.e. performance) using system reliability models, the test and failure data

- Comparing past and predicted future system performance with reliability targets and where applicable identifying major contributors to system unavailability or unreliability

- Ensuring that appropriate follow-up and corrective actions are taken to remedy any system reliability problems

- Monitoring the expected impact of corrective actions on the system performance

An example of the system reliability monitoring programme is the programme used by Ontario Hydro, a Canadian utility with experience of over 170 reactor-years of operation [56]. The basic objective of the programme is to determine, or the basis of operating experience gathered on a particular system whether the system's performance over the past year was acceptable and will continue to be so in the future. The programme has been applied primarily to monitoring reliability of so called special safety systems, that is shutdown systems, ECC and containment systems. However, the methods are generally applicable to any plant system where additional insight into its performance is desired.

Because special safety systems are carefully designed and maintained, component failures are relatively infrequent. This fact, coupled with the built-in redundancy, makes total system failure highly improbable. Although such features contribute to high reliability, they also make meaningful performance assessments difficult because of the limited failure data.

Despite the data limitations, considerable insight can be gained by analyzing the operational data. The Ontario Hydro programme involves tracking of a number of reliability parameters because no single parameter can provide sufficient information to adequately judge reliability performance of a system in operation. The parameters tracked to assess performance of special safety systems are:

- system inoperability
- observed system unavailability
- derived system unavailability
- expected system unavailability

System inoperability is the time fraction of the past year when a system was fully incapable of providing protection for the events with which it was designed to cope. This parameter is determined directly form observation. It does not include marginal failures to meet the design intent of the system and is therefore a nonconservative measure of system performance.

Observed system unavailability is the time fraction of the past year when the system was unable to satisfy fully all design requirements. In other words, the system is deemed unavailable when it fails to provide required protection for one or more of the design basis events. (It follows, that the system's performance may be acceptable for all but one of the postulated events.) This parameter is again determined directly from observation, and includes all faults which resulted in a system inoperability and all other faults which reduced the system's functional capability below any of its design requirements. While the observed system unavailability provides a conservative measure of the actual system performance, it is susceptible to large statistical fluctuations from year to year. When used in conjunction with 'system inoperability', it is useful in distinguishing between major system faults which definitely affect public risk, and faults that may only marginally reduce the system's functional capability below the design requirements derived from the plant safety analysis.

Derived system unavailability is a parameter calculated using system reliability model and the system failure data from the past year. All unsafe faults that occurred over the past year are included and component unavailabilities are calculated by using an estimate of average future fault durations. This parameter provides more information than "observed system unavailability" on the contribution of individual component failures to overall system performance. It also provides an indication of the statistical significance of the value of the observed system unavailability in a particular year: for example, if an observed unavailability is much less than a derived unavailability it indicates a fortuitous situation that should not be expected to continue in the long term.

Expected system unavailability is calculated using the same reliability model as derived system unavailability and all relevant fault data collected over the life of the station. Where few or no component failures have been observed, a 50% confidence Chi-squared estimate of failure rate is used to provide an estimate of component performance. After a few years, this parameter provides a statistically valid upper limit estimate of long-term average system performance. Because it uses all relevant experience, it does eventually become relatively insensitive to sudden changes in the performance of equipment, but such changes are detected by other means (e.g. derived system unavailability).

A further check on the performance of special safety systems is carried out via the component failure rate trend monitoring programme. Each special safety system component is monitored on an annual basis. Its annual performance is checked against its own lifetime experience in addition to the standard generic failure rate data banks such as NPRDS. In this way, deteriorating component performance can be recognized before it impacts on overall safety system performance.

The reliability parameters used by Ontario Hydro have proven useful in interpreting data to better understand the actual reliability performance of special safety systems. Their use, since 1962, facilitated identification of the problems which caused systems not to meet their reliability targets. The programme also helped in understanding the significance of the problems and in setting priorities for corrective actions. Examples of ageing degradation problems identified and corrected with the aid of the programme include failures of ECC injection valves caused by the degradation of the grease used to lubricate limit switch assemblies of the valves, and failures of dousing system solenoid valves caused by the degradation of elastomeric internals in high temperature environment. Besides drawing attention to hardware problems, the system leliability monitoring programme has also helped to optimize testing programme and the operating and maintenance practices affecting safety system reliability performance.

7.3. Preventive maintenance as a means of managing of ageing effects

Effective preventive maintenance (PM) programmes are essential for ensuring acceptable operational reliability throughout the service life of nuclear power plants. This section does not attempt to describe a PM programme, but aims only to highlight those programme developments which are important to managing of ageing effects.

The role of PM to detect and mitigate component degradation before failure is accomplished through two kinds of preventive maintenance, scheduled maintenance and predictive maintenance.

Safety related components should be covered either by predictive or scheduled maintenance programmes, which means that their maintenance would be based either on some type of performance or condition monitoring, or on a call-up (time-directed) system.

To ensure effectiveness of PM programmes, they should be periodically reviewed and modified on the basis of operating experience. It has been shown [57] that the reliability-centered maintenance (RCM) methodology, developed in the aviation industry for a wide body jet aircraft, is an effective tool for the review and development of PM programmes for nuclear plant systems. RCM considers the contribution of each component to the functioning of the selected system, and using a logic tree analysis identifies practical and effective PM tasks.

The following subsections deal with the topics of predictive maintenance and scheduled maintenance, and also give further information on RCM application.

7.3.1. Predictive maintenance

Predictive maintenance, or condition-directed maintenance, is a maintenance initiated on the basis of observed present and projected future component conditions. The timing and the kind of maintenance work is determined utilizing trends of both absolute and relative comparisons of component conditions.

Maintenance decision methodology requires reliable techniques for failure or wear out predictions. The objective is to predict component performance so that maintenance or repairs can be planned and completed before the projected failure occurs. RCM approach can be used to develop appropriate plant specific PM procedures and plans.

RCM is a systematic method for developing or reviewing PM programmes. In brief, RCM is a systematic consideration of system/subsystem functions, the way functions can fail, and the safety and economic importance associated with these functional failures. Potential PM tasks are then considered and those tasks which can reduce failure probability and are economically viable are selected.

The process starts with the collection of all pertinent design, operating and maintenance information for a selected system. The system, its functions, boundaries and interfaces are described, and its functional failures (i.e. how it can fail) identified. The review of operating and maintenance experience is performed to determine actual failure modes, their root causes and effects (i.e. severities) for each functionally significant item. Condition indicators able to indicate component degradation before failure (i.e. precursor indicators) are then determined (where possible) for the most important failure modes identified in the review of experience. For the condition indicators to be monitored, standards or minimum acceptance criteria (based on the system functional requirements) must be determined, so that a performance monitoring programme which measures actual component conditions at appropriate time intervals can be designed. Finally, applicable and effective PM tasks to mitigate component degradation before failure are identified.

The outcome of the RCM based review of a PM programme for NPP safety related systems will be a mixture of predictive maintenance and scheduled maintenance or component replacement. For standby safety systems the principal strategy will be the predictive maintenance based on some type of condition monitoring.

To implement the predictive maintenance programme, it is necessary to store and analyze relevant data and report trends so that the extent of degradation can be determined and a decision on the kind and timing of preventive action made. The large amounts of relevant data to be processed (i.e. condition monitoring data, maintenance history data, operating data, and pertinent acceptance criteria), require use of computerized data processing. Experience will show which condition monitoring and data analysis techniques are appropriate for failure prediction. A number of iterations may be required to optimize a PM programme.

7.3.2. Scheduled maintenance

Although the trend in preventive maintenance is towards more predictive maintenance, some equipment is and will continue to be covered by scheduled maintenance. The reason may be a lack of practical condition monitoring method or the effectiveness of the scheduled maintenance for given equipment.

Initially, the scope and timing of scheduled maintenance is based on manufacturers' recommendations, warranty and regulatory requirements, and plant staff experience. Scheduled maintenance programmes should be periodically reviewed and optimized on the basis of operating experience to reduce corrective maintenance and thus improve component reliability. An optimization approach should incorporate analysis and engineering assessment of the plant operating and maintenance history to determine component failure modes and causes, the interval between failures, and when in plant life the failures occurred. Based on this information it is possible to determine which scheduled maintenance tasks are effective and when they should be performed. To repeat, the term components is used in a broad sense and includes plant structures and systems.

An example of the scheduled maintenance optimization strategy, developed by Advance Technology Engineering Systems Inc. [58], is outlined below, and the basic steps of the optimization process are shown diagramatically in Fig. 7.3.1.

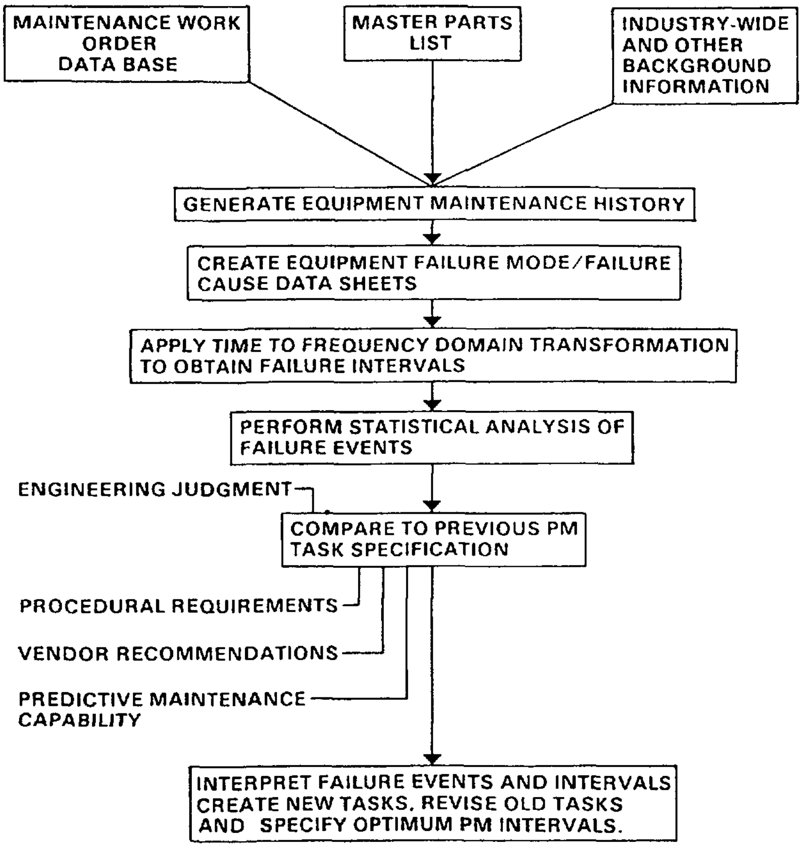

Figure 7.3.1. Scheduled Maintenance Optimization Process

The process starts with the generation of an equipment maintenance history package (including existing PM tasks, relevant drawings, etc.) for each equipment type on a system by system basis. The packages are reviewed to determine which equipment have significant failure histories. For each such equipment, significant failure modes and causes are determined and entered into annual data sheets (starting with year 1 of commercial operation). The data are then input to the time-series analysis (TSA) computer programme whose results show the frequency of equipment failures.

In addition to TSA, the data is statistically analyzed to identify if any relationships exist between apparently unrelated variables, and if there is any cause-effect relationship among system failures. The results of the TSA and statistical analyses are summarized and reviewed against the operating experience analyses results and the existing PM tasks to recommend changes to both PM tasks and PM intervals. In doing this, interviews with the plant staff are important to clarify non-cyclical events which are not necessarily preventable by maintenance.

Applications of the above described approach for optimizing scheduled maintenance have resulted in the reduction of equipment failures due to time-related degradation through effective PM as well as the correction of identified design and operational problems.

7.4. Developments needed in the areas of detection and mitigation of ageing effects

As discussed earlier in this section, managing the effects of ageing degradation requires effective methods to monitor and predict component and system performance so that appropriate preventive actions can be taken before projected failures of safety related components will occur. General principles and examples of methods for timely detection and mitigation of ageing effects have been given in the preceding sections. The question is: What needs to be done to improve our capability in these areas?

In the area of detection of ageing effects, most of the current testing and surveillance programmes adequately verify availability of NPP components at the time of observation, yet may not provide sufficient information for trending performance or failure prediction. Improvement of the effectiveness in mitigating ageing degradation of various plant components depends to a large extent, on better understanding of pertinent ageing mechanisms acting on various component materials under different service conditions. Thus, in general, there is a need to review existing performance monitoring and maintenance methods and practices for their effectiveness in detecting and mitigating ageing effects of safety significant NPP components, to identify specific needs for improvement, and resolve them by R&D and design or operation improvements, as appropriate. This general need is being addressed by a variety of studies and programmes, some of which are outlined in the following section.

The engineering studies are being performed on a component-type basis for the components which have been identified as having an ageing-related impact on the reliability of safety-related systems. These studies aim to identify and understand significant modes of component degradation, suitable condition indicators, preferred performance monitoring and maintenance methods and practices, and minimum accpetance criteria. Other research is currently being conducted to refine reliability approaches, risk prioritization schemes, statistical indicators to trend system performance and direct, risk-based predictive indicators of plant performance. Additional studies aim to develop technically acceptable methods for predicting remaining service life based on operating and maintenance history.

The area of condition monitoring development deserves further elaboration.

Typically, there are thousands of safety related components at a nuclear power plant. Although, in principle desirable, condition monitoring of all of these components may not be the necessary and most effective approach to managing of ageing effects. Therefore, guidelines for condition monitoring, as well as guidelines to develop realistic and quantitative criteria for surveillance and maintenance, should be established. The guidelines should provide, for specific components, the parameters most indicative of age related degradation; criteria for using condition monitoring data to make decisions concerning the type and timing of corrective maintenance or replacement; and practical methods for condition monitoring. The guidelines should also indicate the sample size of a population of similar components that should be monitored. Further, the guidelines should recommend the actions to be taken for other components within the population if a certain number of components being monitored degrade to an unacceptable level. For example, if a certain number of power cables are being monitored and show signs of unacceptable insulation degradation, then direction should be provided for testing and maintenance of the unmonitored cables that are operating under a similar environment.

Broader development of a condition monitoring programme is, in general, hampered by inadequate understanding of the degradation mechanisms. This, in turn, makes it difficult to identify or develop appropriate condition indicators and monitoring methods. In addition, at present, there is not an adequate correlation between the measurable condition indicators (ageing parameters) and the future functional capability during normal and DBE conditions for many types of components. Therefore, repair or replacement decisions are mostly based on experience and engineering judgement rather than predetermined standards or criteria. In this context, it follows that much can be done to enhance our capability for effective managing of ageing degradation.

8. MANAGING THE IMPACT OF AGEING ON NUCLEAR POWER PLANT SAFETY — EXAMPLES OF ACTIVITIES IN MEMBER STATES

Member States use similar but different approaches to managing the ageing of NPP components and its impact on plant safety and the reliability. These approaches involve, some or all of, the following major elements:

- Qualifying safety related equipment for all anticipated service conditions (including the postulated design basis accident and post-accident conditions) and taking into account all significant types of equipment degradation that may occur during its service life.

- Use of only 'well tried' equipment that has demonstrated high reliability under comparable conditions. Accelerated ageing methods may be used for 'well tried' equipment to address some environmental condition such as normal radiation that may not have been included in the comparable conditions.

- Preventive maintenance and surveillance programmes to prevent equipment failures through timely detection and mitigation of equipment degradation.

- Reviews of operating experience on the component, system and plant level against performance standards or targets to identify improvements needed to maintain adequate performance.

Much has been done to understand the ageing of different plant components and to deal with its effects on plant safety and availability. However, there is still a great deal to be learned about various material and service environment interactions causing time dependent degradation of plant components and about effective means of detecting and mitigating the ageing effects.

The following subsections give outlines of the approaches taken by some Member States to understand and manage ageing in NPPs and thus to maintain continued safe and reliable operation of the plants. These approaches include R&D work and other activities relevant to managing the ageing effects. They are based on documents available, mainly on the papers presented during the IAEA Symposium on Safety Aspects of NPP Ageing, held from 29 June to 3 July 1987 in Vienna, and on the documents submitted by the participants of the IAEA Technical Committee Meeting on the same subject held in September 1986.

8.1. United States of America

The USNRC's Nuclear Plant Ageing Research (NPAR) Programme is intended to resolve technical safety issues related to the ageing and service wear of equipment and systems at commercial reactor facilities and their possible impact on plant safety. Emphasis is placed on identification and characterization of the mechanisms of material and component degradation during service and evaluation of methods of inspection, surveillance, condition monitoring and maintenance as means of detecting and mitigating such effects. Specially, the goals of the programme are as follows:

- To identify and characterize ageing and service wear effects which, if unmitigated, could cause degradation of structures, components and systems and thereby impair plant safety.

- To identify methods of inspection, surveillance and monitoring, or of evaluating the residual life of structures, components and systems which will assure the timely detection of significant ageing effects prior to loss of safety function.

- To evaluate the effectiveness of storage, maintenance, repair and replacement practices in reducing the rate and extent of degradation caused by ageing and service wear.

Many of the issues concerning safe nuclear plant operation in the face of ageing processes have been, and are being, addressed by the nuclear industry through research, design, standards development and improved operation and maintenance. Nuclear utilities, EPRI, INPO, NSSS vendors, architect engineers, DOE, IEEE, ANS, ASME, ASTM, national laboratories and the USNRC currently participate in such efforts. However, because of the variety of components and structures and their applications, the complexity of the ageing processes and the limited experience with prolonged operation of commercial LWR, and the potential for adverse effects on safety, significant questions remain unanswered. These questions include:

(1) What structures, systems and components are susceptible to ageing effects that could adversely affect public health and safety? Which of these structures, systems and components are maintained and replaceable?

(2) What are the degradation processes of materials, components and structures which could, if unchecked (improperly maintained and/or not replaced) affect safety during the normal design lifetime and during extended life?

(3) How can the operational readiness of aged structures, systems and components be assured during the 40 year design lifetime and during extended life?

(4) Are currently available examinations and test methods adequate to identify all relevant ageing mechanisms before safety is affected? If not, what efforts are under way to improve them?

(5) What criteria are required to evaluate the residual lifetime of components and structures? What supporting evidence (data, record keeping, analyses, inspection, etc.) will be needed?

(6) How should structures, systems and components be selected for comprehensive ageing assessments and residual lifetime evaluations? Which structures, systems and components should be selected?

(7) How effective are current programme for mitigating ageing effects (e.g. maintenance, replacement and repair)?

(8) What kinds of reliability assurance and maintenance programme will be needed to ensure operational readiness of aged safety systems and components?

(9) What additional changes will be needed in codes and standards to address ageing beyond those changes already in progress? What schedule should be followed?

The NRC ageing research programme is carried out in discrete stages, as showed in Fig. 8.1.1.

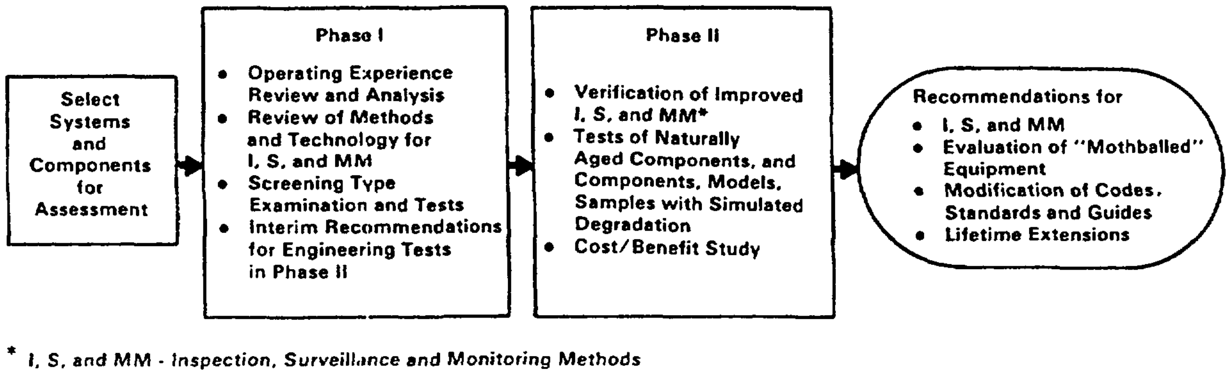

Fig.8.1.1. Research approach - NPAR programme.

A selection process is followed to establish priorities for detailed ageing assessments of specific systems and components. Selection criteria include the potential contribution to risk of the systems of components, nuclear plant experience and expert judgement regarding the susceptibility of the system or component to ageing degradation.

Risk and system oriented identification of ageing effects

The results of the completed scoping studies [2, 4, 6] and ongoing probabilistic risk studies and systems analyses are utilized to identify (a) the relative importance to risk of the ageing of various plant systems and components; (b) the stresses to which equipment will be subjected during operations and design basis events (including plant cycling and trips), and which could increase the potential for the common mode failure of aged redundant components; (c) levels of component functional degradation at which impairment of system performance would cause an increase in the severity or frequency of plant transients or an unacceptable degradation in the capability to mitigate the effects of design basis events. These evaluations are combined with the evaluations of ageing and service wear experience to establish general programme priorities and direction.

Review of nuclear plant experience

The LWR operating experience which relates to equipment ageing and service wear are evaluated to identify ageing effects which have occurred to date and which require further study to understand how such effects could impact safety over the expected 40 year life of the plant. Specific LWR oriented databases which are evaluated include Licensee Event Reports, the In-Plant Reliability Data System, Plant Preventive Maintenance Records, Nuclear Plant Experience Reports, In-service Inspection Reports, and the Nuclear Plant Reliability Data System. General trends from operating experience is used in conjunction with evaluations risk ageing systems.

US NRC's phased approach to ageing assessment

Phase I

Ageing assessments of selected systems or components involve two stages. Phase I includes evaluations of design specifications including: materials; operational environments; qualification tests; operating experience; systems interfaces; methods for inspection, surveillance, monitoring and maintenance; and screening type equipment examinations and tests. In a Phase I component or system analysis the product of the study is a preliminary identification of the significant modes of degradation, and evaluation of current inspection, surveillance and monitoring methods. Based on these evaluations, recommendations are developed to identify detailed engineering tests and analyses to be conducted in Phase II. Phase I evaluation of systems leads to recommendations for a systems level Phase II assessment or to recommendations for in-depth ageing assessments of additional components.

Phase II

Phase II assessments generally include some combination of: (a) tests of naturally aged equipment or of equipment with simulated degradation; (b) laboratory or in plant verification of methods for inspection, monitoring and surveillance; (c) development of recommendations for inspection or monitoring techniques in lieu of tests which cause excessive wear; (d) verification of methods of evaluation of residual service lifetime; (e) identification of effective maintenance practices; (f) in situ examination and data gathering for operating equipment; and (g) cost benefit analyses.

Component ageing assessment includes the examination and testing of equipment removed from service at operating LWRs and decommissioned reactors.* This equipment is used to identify failure modes related to ageing and service wear and indicators of such potential failure modes which may be monitored to detect the onset of failure in an incipient stage.

Tests of aged equipment simulate, where appropriate, stresses typical of the dynamic, seismic, electrical, mechanical, thermal, steam and radiation transients anticipated during operating service and representative of design base events. Selection of equipment, for which examinations and tests are performed, are considerations of risk ageing/systems evaluations and evaluation of previous experience.

In situ monitoring of operating equipment at LWRs is also performed to gain an understanding of the interaction between ageing and service wear defect characterization and inspection, surveillance and maintenance.

When available, ageing assessment is performed on equipment which has failed during operation as well as equipment which has survived extensive periods of operation. This is done in order to gain an understanding of those ageing effects which would only be precipitated during a trigger event accompanied by abnormal stresses.

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* A number of naturally aged components have been obtained during the decommissiong of the reactor at Shippingport in the USA and others are being requested from utilities and other decommissioned reactors.

Research strategy and program elements

Research program strategy

The NPAR program has been planned so that the major programme elements (the circled elements in Fig. 8.1.2) can be accomplished within a reasonable schedule. A number of supporting tasks (identified by rectangular blocks in Fig. 8.1.2) have been selected and scheduled so that research on the major elements can proceed with the full benefit of the completion of the necessary groundwork.

Where information is needed from manufacturers and utilities, avenues of communication are explored. Where related research is under way, liaison is accomplished. Insofar as practical, both government and industry supported programmes are incorporated. The overall programme plan includes the establishment of a central database of information (component and system specific) on: research relevant to plant ageing; standards and guides relevant to ageing; and citations and sources of published information on plant ageing. This central, integrated database of information benefits not only the NRC in its plant ageing research, but also all other organizations (laboratories, professional societies, utilities and manufacturers) concerned with plant ageing. Also, the programme will identify gaps in existing ageing related research that can be filled by government or industry sources, depending on which is most appropriate. To provide the maximum possible flow and exchange of information, one product of this programme includes the publication of reports on the research tasks. Table 8.1.1 provides specific equipment and systems of current interest and related reports which have been published. The research information is disseminated by the preparation of technical papers, presentations at technical conferences, sponsorship of workshops and symposia, and exchange visits and information and information with principal sources of related research.

A final product of the research programme is the recommendations for the revision of existing standards and regulatory guides and the development of new standards and guides, as necessary. The technical basis, developed from the research programme, accompany all of the recommendations.

The USNRC ageing research program, involving plant safety systems and components, has resulted in better understanding of ageing processes and will increase confidence in methods for detecting and mitigating ageing degradation. This will provide a basis for timely and sound regulatory decisions regarding continued operation of nuclear plants of all ages. In the programme it has been recommended that the detection and timely mitigation of ageing degradation at an early stage, before functional capability is impaired and before continued safe operation is jeopardized, will avoid unplanned and costly plant shutdowns. In addition, operating plant maintenance practices will become more effective. Wear from excessive testing can be minimized through the use of more effective surveillance techniques and equipment reliability will be improved.

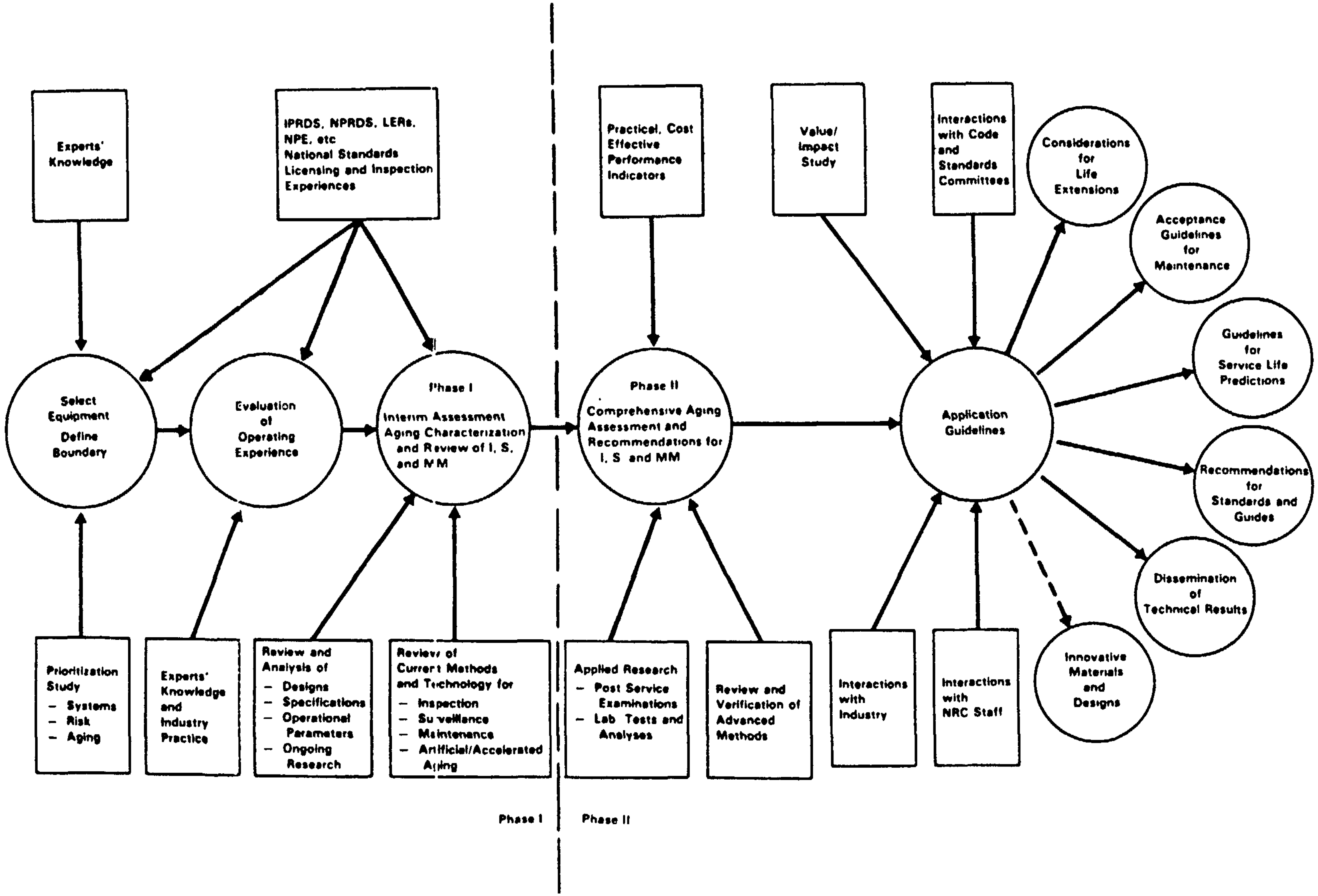


Fig.8.1.2. NPAR programme strategy.

Table 8.1.1

Categorization of Equipment and Systems of Current Interest*

Equipment	NUREG/CR
• Motor operated valves	4234, 4380
• Check valves	4302, 5159
• Auxiliary feed water pumps	4597
• Electric motors (large, inside containment)	4156, 4939
• Batteries	4457
• Chargers/inverters	4564, 5051, 5192
• Snubbers	4279
• Diesel generators	4590
• Circuit breakers and relays	4715
• Room coolers	PNL-5722
• Gang operated control switches	FRL-5384-13-4
• Solenoid valves	4819, 5008, 5141
• Power operated relief valves	4692
• Cables (power, control, instrument	4257-1
• Electrical penetrations	-
• Sensors/systems/transmitters -Temperature, pressure, level	4257-2
• Connectors, terminal blocks	
• Heat exchangers	
• Compressors	
• Transformers	
• Bistables/switches	
• Air operated valves	
• Surge arrestors	
• Purge and vent valves	-
• Safety relief valves	-
• Service water & component cooling water pumps	-
• Isolation consenders (BWR)	
• Pilot operated relief valves	4692
• Accumulators	
• Fan chillers	
• Main steam isolation valves	
Systems	
• High Pressure Emergency Core Cooling Systems	4967
• Low Pressure Emergency Core Cooling Systems	-
• Service Water System	-
• Component Cooling Water System	5052
• Reactor Protection System	4740
• Residual Heat Removal System/Heat Removal	-
• Class 1E Electric Distribution System	5181
• Engineered Safety Feature Actuation System	-
• Auxiliary Feedwater System	-
• Control Rod Drive Systems (PWR, BWR)	-
• Instrument and Control Air System	-
• Standby Liquid Control System (BWR)	-
• Reactor Core Isolation Cooling System	-
• Containment Isolation	-

* This listing includes the Primary System Pressure Boundary components (vessel, piping, steam generators, reactor internals, etc.) and containment structures. The ageing assessments of these components and structures are being addressed in other related programmes sponsored by the Division of Engineering of the US Office of Nuclear Regulatory Research.

US NRC's NPAR programme coordination with other programmes, institutions and organizations

It has been recognized in the NPAR programme that various institutions and the industry organizations have performed studies and instituted programmes relevant to ageing research. Results of the more important activies are reported in the Refs. [1, 3-5, 7 and 59-83]. Also, there are a number of ongoing programmes which are producing significant results which cannot and should not be duplicated. A major emphasis in the NPAR programme plan is that proper co-ordination and integration of plant ageing research activities is attained at various levels to achieve overall programme goals and objectives and to assure the efficient use of available resources.

Interfaces have been established are maintained with other related ongoing NRC programmes. External programmes involving both domestic and foreign organizations have similarly been reviewed, contacts and information exchanges have taken place and dialogues for cooperative research programmes have been initiated. Examples of the ongoing programmes and contacts are noted in the following:

NRC Programmes

- Licensee event reports (LERs)
- In-plant reliability data system
- Reliability assurance programme
- Procedures for evaluating technical Specifications (PETS)
- Maintenance and surveillance programme
- Plant performance indicator programme
- Licence renewal rulemaking
- Equipment qualification research - electrical, mechanical
- Materials research
- Mechanical components and structural research
- Equipment qualification - licensing activities

Domestic programmes outside NRC	Sponsor
• Ageing/seismic research	EPRI
• Equipment qualification research	EPRI
• Condition monitoring of electrical equipment	EPRI
• Decommissioning of Shippingport Atomic Power Station	DOE
• Nuclear plant reliability data system (NPRDS)	INPO
• Nuclear plant life extension	EPRI, DOE, NUMARC, Licensees

8.2 France

The problem of the lifetime of the 33 French units of 900 MW capacity and other units of 1300 MW capacity and the contingent appearance of premature ageing must, of course, be examined, especially with regard to its effect on the safety of the installations.

Research and development programmes of an ad hoc nature have been conducted in this field over the last ten years by Electricité de France (EdF) which is the operator, and also by CEA, the French Atomic Energy Commission. The approach adopted in this period has tended to result in the setting up of methods to characterize the ageing of those power plant components that are the most important in this respect.

EdF and CEA have decided to pool their resources to combine efforts in this field, with the agreement of the safety authorities.

Analysis of NPP experience

The lessons drawn from operation are used to detect the weak points not sufficiently taken into account by the design studies and to check that system availability remains at least equal to the level defined at the design stage. Specific tools, including ageing management, have been set up to help achieve these goals.

Recording and analysis of events

Data collected by the licensee (EDF)

Each plant has its own organization for detecting and reporting:

- incidents significant for safety.
- safety related events.
- unavailability of major safety related equipment.

Each plant records the significant incidents and the important events in a computer file kept at the disposal of the safety authorities.

Finally, each plant transmits to the safety authorities:

- at each refuelling shutdown, a file describing planned inspection and maintenance operations and modifications (2 months before) and then a report (4 months afterwards).
- a synthesis report of operating safety every calendar year.

Analysis of events by EdF

Each significant incident is analysed by the plant and the report is sent to the EdF Control Office and to the safety authority. It is analysed by the Control Office and the most significant incidents are the subject of a detailed study.

Analysis of events by the safety authorities

The "Institut de protection et de sûreté nucléaire" (IPSN), which is the technical support body of the safety authorities, analyses the documents transmitted by the licensee. The main purpose of these analyses is to provide the authorities with an evaluation of plant safety so that they can check whether the operating conditions proposed by the licensee are acceptable.

But the IPSN also performs a more in-depth investigation in order to detect as early as possible those problems likely to compromise safety.

Lessons drawn from the operation of the 900 MW(e) reactors

The first objective is to analyse the behaviour of the plant as a whole. In particular, all events and incidents are to be situated with respect to the design studies. This comparison may highlight favourable or unfavourable deviations. It may even show up shortcomings in the design studies, due more often than not to the difficulty, if not the impossibility, of foreseeing certain situations.

The second objective is to evaluate the actual behaviour of the equipment and its capability (and the maintenance of this capability) to perform the safety functions.

To summarize, the necessary corrective actions are based on a threefold approach:

- The discrepancies detected between the actual situations and the forecasts must be corrected if they are likely to compromise safety: modifications to the installation and/or to the operating rules should be examined and implemented, if necessary. A more in-depth analysis must be carried out in each case, not only to ensure that the modification provides a solution to the problem encountered, but also to check that there is no risk of harmful side-effects on other areas of plant safety. The study and application of the modifications require a strict organization to guarantee their quality. The time needed for these modifications could therefore prove long and require palliative measures.

- Surveillance of systems and components must be constantly adapted to the objectives. The experience acquired so far is widely used in re-examining the rules concerning periodic tests, non-destructive testing and preventive maintenance. Problems arising from the wear or ageing of the equipment generally result in an intensification of surveillance (this was illustrated with the steam generators). Conversely, the evaluation of the actual behaviour of certain components can lead to a relaxation of surveillance: as an example the protection system is now tested every two months whereas the frequency had initially been set at one month.

- The 'man-machine' interface must be improved in the light of operator experience. It was actually impossible to foresee entirely each aspect of human behaviour during the design studies. This is an ongoing task which involves changes in the control rooms, improvement of the procedures and operating instructions (which have been completely rewritten by Electricité de France), identification of rooms and equipment, personnel training and improvement of the quality organization.

To evaluate the overall safety level of nuclear plants, the risk they present must be quantified.

Use of a probabilistic model in evaluating the safety of reactors in operation

It is important to use rules defining the possibility of pursuing reactor operation and setting up the list of equipment required. These rules can only be based on an assessment of the risk increase in these situations given that, at the design stage, unavailabilities are not fully foreseen and that the steps taken during the construction of the plant are supposed to guarantee a high reliability of equipment.

The probabilistic model for the 900 MW(e) reactors was developed. Initially, the risk is assessed in terms of probability of core meltdown; secondly, it is assessed in terms of fission product releases.

The particulars of the model developed in France are as follows:

(1) The notion of success or failure of the control procedures likely to be used is introduced into the event trees so as to treat accident sequences as realistically as possible.

(2) All the reactor conditions are taken into account, in particular the cold shutdowns which, as worldwide experience shows, might prove to be a significant risk factor.

(3) A special effort is made to take account of the risk contribution of long term developments following an accident.

(4) The possibility of repairing faulty equipment or using additional resources in the case of such failures is also taken into account.

(5) The desire to create a safety analysis tool led to the setting up of a computerized system for recording the functional description of the plant and making rapid risk assessment computations.

The study initiated in 1983 is expected to provide by 1987 a risk assessment in terms of core meltdown and subsequently in terms of fission product releases. Beyond this, certain aspects will be studied thoroughly, notably the evaluation of the uncertainties and how to deal with human errors.

It should be noted that EdF undertook early in 1986 a study of the same type for the 1300 MW(e) series which will be conducted in close conjunction with the 900 MW(e) study previously described.

The feedback is integrated into the probabilistic model

This model must obviously depend on lessons drawn from operation of the reactors which, thanks to the standardization of the French reactors, constitutes an excellent information base. Feedback is used at two levels:

- On the one hand, feedback is used through a reliability database created from the Système de Recueil de Données de Fiabilité (SRDF) concerning 600 items of equipment of each 900 MW(e) reactor (about 30 units in France);

- On the other hand, feedback is used through analysis of incidents likely to highlight accident sequences not included in the probabilistic model, for example at system interaction level. We envisage that the model will be updated periodically as a result of feedback.

Generally, the probabilistic model allows evaluation of the variation in the risk obtained by implementing modifications to the plant design and operation. This is how a part of the model was used, for example, to evaluate that the gain factor obtained by the modifications proposed by EdF to counter the loss of the electric power supplies was of the order of 100.

The IPSN plans to use this model to evaluate in certain cases the gain resulting from the main modifications decided for these reactors, following the analysis of the operating incidents.

<u>Specific approach to ageing</u>

The tasks to be implemented are listed in Table 8.2.1 and they are as follows:

Task 1 - State of the art inventory

(a) Complete analysis of every possible evolution in equipment design used in a given plant, with particular emphasis on back-up equipment.

(b) Experimental or theoretical studies regarding behaviour justification (wear and tear, ageing of equipment, etc.) with a view to evaluating the life cycle of equipment (with identification of necessary data to permit this evaluation).

(c) Compiling, safe-keeping and use of data provided by the Inspectorate during construction and operation: besides up-keep and, more generally, feedback of experience.

(d) Research into functional criteria which would simply allow a guarantee that the installation should work according to the quality values chosen, throughout its lifetime.

(e) Development of new means of detection for equipment and its wear and tear during operation, with particular emphasis on studies mentioned in (b).

(f) Studies on means of enhancing equipment life extension (definition of new tests, supplementary hydraulic test trials).

(g) Studies on repair or replacement of equipment, including industrial programming costs and length of servicing shutdowns.

These established facts will lead to detailed reports. The conclusions should bring out the complementary measures of a similar nature that ought to be carried out, and their order of priority.

They will use to the extent possible EdF's own experience as well as that gleaned from the world's PWRs: especially the experience being gained in USA during the studies of "life extension".

Table 8.2.1.

TASKS	1986	1987	1988	1989
1. Reports : Analysis of both available data and procedures on hand	xxxxx	xxxx		
2. Study of the impact of operating methods	xxxxx			
3. Impact of changes in reglementation	xxxxx			
4. Expected economics	xxxxx	xxxxxxxx		
5. Definition, commitment and follow-up of complementary procedures (1)	xx	xxxxx		
6. Execution of complementary procedures (3)	x	xxxxxxxx	xxxxxxxxxx	xxxxxxxxxxxx
7. Elaboration of tools to aid judgement		xxxxx	xxxxxxxxx	
8. Summing-up and conclusions		xxxxx	(2) xx	xxxxxxx

Notes : (1) The period indicated is that of the initial definition and commitment to new complementary procedures. The follow-up of these engagements and those already committed, cover the whole of the duration.
(2) The summing-up of the end of phase 1 (milestone end 87) is marked by a presentation of the ensemble including the reports and the complementary engagement's program.
(3) Certain complementary engagements are set up before milestone, end 87 : on the other hand, certain procedures extend after the final summing-up.

In this capacity Electricité de France participated in 1985/86 in a pilot project regarding the life extension of the US plant Surry 1. This project, which was launched and partly financed by the DOE and EPRI, has been conducted by Virginia Power which owns the Surry plant.

Task 2 - Influence of operating procedures

This task is particularly important in France, for it concerns two vital functions - load following and frequency control. Any modifications concerning plant life extension must, therefore, be considered with this in mind. Likewise, changes in fuel management or control should be examined from the point of view of possible repercussions for vessel irradiation.

Task 3 - Significance of the development of regulatory rules

Safety regulations have tended to remain virtually unchanged in the last few years. Nevertheless, they could well change in the future. The risk is that any implementation of new rules on existing plants could economically be a real disaster.

Task 4 - Future economy

It is thought today that in all probability the competitiveness of new versus existing units has to be confirmed. On the other hand, PWR life extension is plausible, and it appears judicious to stress its optimization (programming of repairs, changes in performance, modifications, decommissioning decisions, etc.). In the short term, however, it is considered rational to concentrate on repair operations: especially the replacement of SGs in PWR 900s and the end-of-life optimization of the 'graphite gas' plants.

Task 5 - Definition, supplementary actions

The findings of Tasks 1, 2 and 3 are likely to confirm the need for supplementary actions. The latter would include further studies combined with R&D, with a follow-up in the framework of the life extension project. These actions could well also include new ISI or repair methods and new 'soft' operating procedures in order to decrease wear and tear and stress.

Task 6 - Execution of the complementary actions

The actions implemented as a result of Tasks 1-5 will need the active participation of a far greater number of organizations than do those tasks themselves. They will rely heavily on the Design and Research Division, on manufacturers, and on other organizations outside EdF. Collaboration with foreign organizations, within a framework of international co-operation, is also envisaged.

Task 7 - Development of tools designed to aid decisions

In the long term, end of life decisions have to be taken with respect to a coherent and full set of technical and economical data. The latter will be elaborated after Tasks 4 and 6 of the project have been carried out. EdF will doubtless need some kind of computer assistance to sort out the rules, in order to be convinced that the Company is pointed in the right direction. The aim of Task 7 is to implement such a programme.

As in shown in Table 8.2.1, the project will come to its conclusion at the end of the 1980s. (It is implemented by the operation and design divisions of EdF with assistance from the Company's R&D division, and the economics' team (EEF) of the Presidency.)

Major considerations retained by EdF today

Table 8.2.2 gives the primary list of components seen as critical, which are the subject of this project report. These major considerations are described below.

Table 8.2.2.

COMPONENTS	
1. R.C.S.	4. BALANCE OF PLANT
PRESSURE VESSEL PRIMARY LOOPS STEAM GENERATORS R C PUMP BODIES PRESSURISER	TURBINE GENERATOR
2. OTHER NSSS COMPONENTS	5. CONTROL ROOM AND I & C
CLASS 2 AND 3 PIPES ROD MECHANISMS PRESSURE VESSEL INTERNALS	6. ELECTRIC INSTALLATIONS CABLES
3. CONTAINMENT	7. COOLING TOWERS

Every piece of equipment in a power plant is subjected to very serious design studies - every possible or even probable incident, accident, fault or reason for damage is taken into consideration. Nevertheless, it is very difficult to appreciate the exact meaning of design life. There is one major exception to this discouraging fact: this is the reactor coolant system.

In France, the relevant regulatory authorities (Order dated 26.2.74) ask for a demonstration of the structure's resistance to a list of foreseeable types of mechanical damage, particularly due to fatigue. This analysis includes a precise definition of a stress loading, mainly transients, which could be encountered in normal, fault or accident conditions. The French code (RCCM) establishes criteria for any type of damage (deformation, cracking, fast fracture, etc.), taking into account the cumulative damage for a given number of instances of each type of situation.

In other words, one element of operational safety is to demonstrate that Electricité de France does not exceed the assumed figure, and is nottrapped in a situation which could lead to unanticipated stresses.

EdF keeps a record of all incidents from the outset of operation for each unit. This system is vital in the event of real and cumulative damage assessment. The first experience today shows that:

- Actual incidents are, in most cases, very similar to those anticipated.

- The actual rate of occurrence is lower than expected.

- A few cases have led to difficulties which turned out to require new 'soft' procedures or more precise analyses.

All these findings and operations will be very useful when establishing the NSSS safe condition in future years. Today they can already be used to improve adjustment of ISI and to reduce somewhat the need for ISI.

The pressure vessel

For the pressure vessel, the effect of irradiation is expected to be acceptable for 40 years for all units, with perhaps a few exceptions, where the copper content in the welds is rather high by French standards.

The studies of Electricité de France are far more conservative than those governed by NRC criteria. The latter would allow continuous operation over 40 years, without any special procedures, for all French units. The first results gleaned from pressure vessel behaviour confirm the calculated margins, which is very encouraging.

Consequently, EdF does not consider it worth financing a study to 'regenerate' or to replace the pressure vessel (which would doubtless be possible after some years).

Other NSSS components

The NSSS has no weak points, but some components are more difficult to make, or are submitted to more severe conditions than others. EDF believes that more supervision is necessary in the following areas:

- vessel pipe welds (bimetallic) where very small defects may be harmful;

- certain specific NSSS components, e.g. the pressurizer spray nozzles or the high pressure injection heads, where thermal fatigue is the major contributor to damage;

- stainless steel moulded elbows where the ferrite content can lead to a decrease in resilience with age.

However, all these points are submitted to a thorough in-service inspection; in any case, replacement is possible without major problems or costs.

Steam generators

After the reactor vessel, the steam generators (SGs) are the main recurrent expenditure for a PWR, mainly on current tube bundle condition and maintenance. Nevertheless, it must be remembered that the components that go to make up the SGs currently wear more rapidly than the vessel itself.

In any case, it is now known how to replace them at a cost which, although very high, is not unacceptable. If necessary, replacement is thus always justified, except possibly in the last years of the plant's life.

The objective of Electricité de France is to keep its SGs running for 40 years: however, not many experts believe that this is possible, particularly for the first models. A more realistic aim, is to ensure that they are replaced once during their 40 year life cycle. In this way, the last years will not hold any disagreeable surprises. As of today, it is necessary to make sure that the real life of the first SGs is as long as possible by using all available means of life extension, such as surface treatment (shot peening or rotopeening).

EdF is not satisfied with the existing repair techniques. Serious work is now being carried out to improve sleeving.

Containment

Containment ageing does not seem to limit a plant's lifetime. However, the long term behaviour of a very thin prestressed concrete structure might create a problem. Surveillance is very simple and efficient, and, if needed, 'fixes' should be available.

Turbogenerator

Nuclear units are creep free, and as such are likely at least to equal their 'fossil' partners in performance.

New damage such as stress corrosion or corrosion fatigue can appear with the arrival of new techniques which are needed for larger sizes: also because of the higher water content of the steam. Here again, efficient surveillance is possible, and replacement of an item, although costly, would not lead to unit replacement.

Auxiliaries

Most auxiliary equipment such as pumps or electric motors, switches, etc., if properly maintained, can reach a lifetime of 40 years in very good conditions.

Partial or total replacement is carried out under current maintenance practice and expense: just as with vintage aeroplanes, for instance, of which nothing original remains after a time except wing-spars and skins. Some of these pieces of auxiliary equipment, such as power or control cables, can pose a difficult replacement problem, but again, this does not constitute a real industrial or economic challenge if properly prepared for. EdF has experience of recabling in several fossil fired plants, for example following a large fire.

General

Technical obsolescence of any equipment in French PWRs is not likely to occur before another 40 years, or possibly longer. EdF's plants should reach this age with a safe reactor vessel, NSSS and containment building.

As regards steam generators and the turbogenerator, if properly maintained, they should be in a safe condition at that time. This fact should allow correct availability. Other critical components, after possibly one or several replacements, should be as good as new.

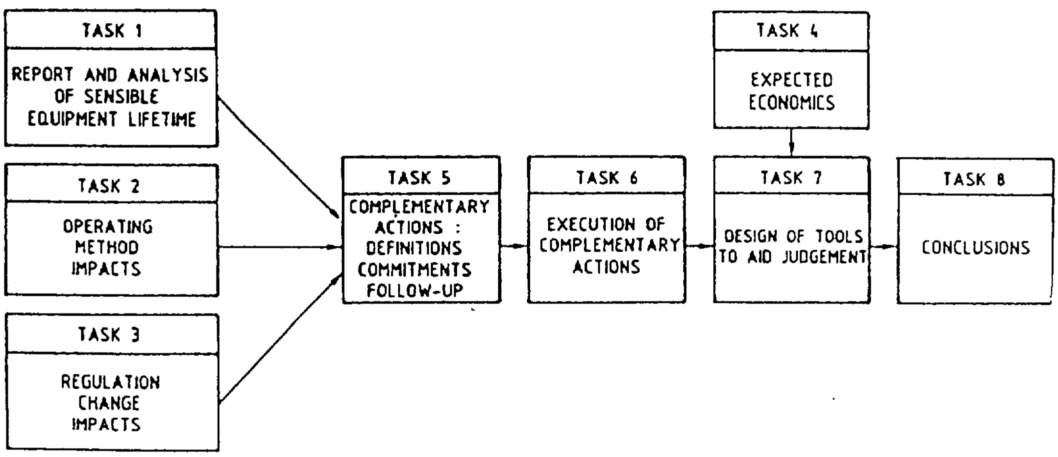
On the other hand, a very bad situation in say 25 or 30 years can be imagined. After a first replacement in year 15, let us suppose new steam generators could find themselves in a bad condition, due to a chemical accident or to any fabrication defects in the tubing. As for the turbo-generator, although well maintained, it could be destroyed due to an undetected crack in the shaft or in the main blade.

Back-up equipment, perhaps poorly maintained, could be in bad shape. Replacement would be possible, of course, but by a whole new set which would require state-of-the-art technology. This imaginary situation is not a nightmare only - it is a real predicament. Several units in the world have been "really" up against it.

Here is the real danger of obsolescence. A simultaneous end of life of numerous components which would entail a nightmare of maintenance, in order to keep the system running. This situation would involve rising costs, which in turn would still not necessarily guarantee continued reliability.

"A stitch in time save nine", - the proverb say. Electricité de France believes that there are two basic "musts". The first and most important, is the quality of maintenance, now, not tomorrow. Life extension begins today. The second maxim is - anticipation: Electricité de France, as one of the world's leading nuclear utilities, believes in this proverb. The conclusion is the "Lifetime Project" (see Table 8.2.3).

Table 8.2.3. Organization of the Life Time Project



Research and development programmes

For the sensitive items of equipments of which the list has thus been drawn up, subject to possible revision in the light of future developments, it is proposed to:

a) make a detailed inventory of items enabling life span assessment which are already available or easily accessible to enable evaluation of the periods during which it should be possible to use the equipment in safety, the means available for monitoring ageing and the possibilities of repair and replacement;

b) on the basis of these findings, define of the actions to be taken in the fields of design, research and development and the integration of experience feedback to supplement this information as necessary;

c) begin additional actions and pursue them to obtain the maximum benefits, especially as concerns knowledge of the state of ageing of the equipment, and its control by suitable operating, monitoring and maintenance measures;

d) combine in a coherent manner the data gathered by the project concerning the residual life of sensitive items of equipment at advanced stages of service, and the costs and durations of major repair or replacement. this information, together with that obtained by financial studies should supply the basis of a program of aid in decision making as concerns extension of life, contingent re-work or replacement of ageing units.

Following research programs are run or in preparation:

I. Research programs concerning ageing of electrical equipment and materials.

II. Programs concerning ageing of pipes and structures which includes:

1. Monitoring of irradiation effects
2. Ageing of cast austenoferritic steels
3. Corrosion

III. Additional action in reactor building containment surveying

With a view to improving monitoring of containments, it would appear necessary to start the following additional actions:

- historical site-by-site analysis of soil settling and ground water level,

- summarizing current behaviour of containments on the basis of Electricité de France survey reports

- inspection of cracking of containment concrete.

8.3 Italy

A limited programme intended to resolve issues related to the ageing and service wear of equipment at reactor facilities and their possible impact on plant safety has been started in Italy.

Operating experiences associated with research and licensing activities, induce to address the ageing effects on plant safety by applying an overall strategy, which includes the current approach at component level, but allows for assessment at systems level.

Such an approach could lead, among other things, through the analysis of the mutual functional interactions between aged components, to achieve more confidence about the maintaining of the safety functions throughout the plant life.

In view of the present limited nuclear programme in Italy, it is considered reasonable to pursue this objective through short and long term programmes.

In the context of the operating plants, the short term programme is addressed mainly to improve assessment at components level by:

- extending the use of sacrificial samples installed in the plant which are periodically taken and tested (e.g. specimens on the vessel, groups of short length of electrical cables, etc.).

  This activity allows, among other things, for improving the artificial ageing analysis and techniques and gives useful information to assess the residual life of the actual plant safety components;

- developing and applying condition monitoring techniques, to be used in the context of the usual preventive maintenance programmes, such as vibration monitoring, ultrasonic probing, infrared emission monitoring, as well as by implementing the water chemistry continuous monitoring techniques addressed to the control of pH, conducivity, impurities, oxygen content, etc., to prevent corrosion and erosion-corrosion phenomens.

These techniques, through the monitoring of proper parameters of selected components and the comparison with established acceptable level of such parameters, allows for taking adequate preventive actions to preclude failure of safety components.

Significant design improvement, at least at component level, can also be expected by the above said activities.

In this context, one of the activities started concern the examination and testing of equipments removed from Garigliano reactor, now awaiting decommissioning.

The main goal of the programme is to provide a basis for assessing the adequacy of industry methods for preconditioning prior to qualification testing. In fact, the experience has shown significant questions have arisen regarding acceptable methods for the artificial accelerated ageing of equipment called for in the standards as part of the qualification testing sequence.

The components up to now selected are:

- power and signal cables,
- thermocouples.

The activities for cables will be based on the execution of qualification test plans on thermal and low dose rates naturally aged cables and on the execution of qualification test plant on cable available at plant warehouse.

The long-term programme will be addressed mainly to perform assessment at system level.

In this regard a preliminary analysis will be performed to select fluid-mechanical systems and electrical systems, considered vital for plant safety, to be during plant operation.

Such an analysis will indicate also those process parameters suitable to estimate the system status. The purpose is to supervise, through consistent condition monitoring techniques, the possible fluctuation of the selected system process parameters and to compare it with established acceptable ranges, in order to take preventive actions before failures and to identify the possible age-related cause and in particular, the affected components.

Significant design improvement, among other things, at plant systems level can be expected by such an investigation and also useful elements to properly implement national NPP's licensing codes.

The large number of data and algorithms involved in such an analysis induce to connect the complete flexibility of such an approach at system level with the use in the NPP's of artificial intelligence machines.

8.4 Sweden

The ageing phenomena research is going on in Sweden as research into the longevity of nuclear plants. In the present review the following technical areas of importance have been identified.

- Work concerning inter granular stress corrosion cracking, IGSCC, and radiation induced stress corrosion cracking. These programmes include material research, chemistry, radiation chemistry, electrochemical measurements and irradiation of materials.

- Development of methods and tools for inspection and repair of control rod drive penetrations on the reactor pressure vessel (BWR).

- Recalculation with new methods of the thermal load resistance of the reactor pressure vessel and primary systems. Adjacent to this is reshaping of the documentation of plant thermal transients according to operating experience.

- Inventory of lifetime restrictions for electrical components with respect to room temperature, chemical pollution, etc.

- Qualification of safety related electrical equipment for accident conditions inside the containment for 40 years.

- Quality assurance rules including regulations for renewing components and systems in old plants. The work includes documentation of which rules and regulations have been applicable to the different plants and components during design and construction.

- Works to keep doses to personnel as low as possible.

Nuclear power plants are widely involved in state programme in co-operation with vendors and the Swedish Nuclear Power Inspectorate.

8.5 Canada

Background

The Canadian approach to managing ageing degradation of NPP components and maintaining adequate plant safety and reliability is an integral part of the overall Canadian approach to reactor safety. This approach does not depend on perfection of any safety measure but allows for design imperfections and occasional equipment failures and operator mistakes.

The effects of ageing are simply one factor in a comprehensive approach to ensuring the safety of nuclear power plants. For example, the design of CANDU NPPs strives to accommodate various ageing effects by using appropriate design features such as diversity, physical separation, redundancy and testability; and operating policies and procedures prescribe practices to minimize ageing effects (e.g., controlling operational cycling, proper chemistry and cleanliness of fluid systems).

Major components of the approach employed to manage NPP ageing are:

- operational safety management programme
- nuclear plant life assurance (NPLA) programme
- plant condition assessment and refurbishment
- equipment qualification
- research and development support.

Operational Safety Management Programme

The operational safety management programme [52] consists of; (1) a preventive maintenance programme, (2) the *Significant Event Reporting (SER)* system, and (3) a reliability based assessment of performance of safety related systems. These methods were put in place at the beginning of the Canadian nuclear power programme in the early 1960s with the objective of preventing, detecting, correcting, and mitigating failures of NPP systems and components from all possible causes. The effect of wear-out on equipment and system reliability was well recognized, and the intent has been to maintain plant systems and components in their "useful life" condition. All three methods mentioned above play an essential and complementary role in meeting this intent and achieving a high level of plant safety and production reliability.

(1) Preventive maintenance programmes include inspection, testing, surveillance, repair and replacement activities designed to preserve functional capability on NPP components for operational and emergency use. These activities are used to a greater extent in maintaining the required level of performance of safety equipment than process equipment. They are also the primary means of detection and mitigation of ageing and service wear effects in NPP components.

(2) The second major component of the operational safety management programme is the Significant Event Reporting system, whose purpose is to control and limit the risk associated with NPP operation.

This is accomplished through systematic detection, analysis and correction of deficiencies, including those caused by ageing degradation. SER system has the following important features.

a) Significant Event Reports are prepared for abnormal or unscheduled events that may have a detrimental effect on worker or public safety, environmental protection, production reliability and cost;

b) equipment and operation deficiencies are recorded in a specified, systematic manner to allow event review and analysis;

c) a multi level diverse screening process includes reviews by plant staff and management, different head office groups, a senior management review committee (when needed), and the regulatory staff;

d) trends of equipment and system deficiencies, identified by the SERs, and their causes are monitored; and

e) lessons learned from SERs are communicated to other CANDU stations, owners, designers and equipment manufacturers.

(3) Reliability based assessment of performance of safety related systems is an important part of the annual comprehensive and systematic review of nuclear power plant operation and maintenance. Performance of safety related systems is monitored on an ongoing basis and formally documented in the Quarterly Technical Reports.

The performance is measured against standards or targets which are established by the utilities on the bias of regulatory requirements, internal and external experience, international recommendations and other considerations.

For poised systems, that is, systems which are normally inactive and are triggered on demand, a test programme is developed and implemented to provide confidence that the system would work when needed. It is recognized that ageing degradation can be detected only if appropriate time-variant parameters are monitored. Components of functions to be tested and the test frequency are determine using reliability analysis and adjusted on the basis of in-service experience.

Reliability surveillance groups at NPPs monitor daily test records, control room logs, work reports and deficiency reports to ensure that all reactor safety deficiencies are properly recorded, classified as to their severity, and corrected. Component faults are classified in terms of their effect on the following two criteria:

a) active safety related systems should always operate in such a way that fuel temperatures remain within well defined limits; and

b) poised safety related systems should always be ready to operate according to their design intent.

To assess performance of poised safety related systems several reliability indicators are evaluated and monitored, including system actual inoperability and system actual and expected unavailabilities. The component failure rate monitoring programme annually evaluates component failure rates and compares them against component lifetime experience and generic failure rate data.

Monitoring and assessing performance of active safety related systems (i.e., process systems) involves recording and trending of both actual and near-miss serious process failures. Should either of these trends indicate a potential problem, a more detailed review of all component faults of a system is carried out to determine major contributors to undesirable performance and appropriate follow-up actions.

Nuclear Plant Life Assurance programme

A formal Nuclear Plant Life Assurance (NPLA) programme has been initiated by Canada largest nuclear utility, Ontario Hydro (OH), with the objectives of a) maintaining the long-term reliability, availability, and safety of OH nuclear plants during the normal service life of 40 years (life assurance), and b) preserving the option of extending the life of the plants beyond the currently assumed service life of 40 years (life extension) [85].

The proposed NPLA programme for OH focuses on relatively few components that are most critical to the long-term reliability and life of the entire plant since they cannot be easily and economically replaced. They include major components such as the calandria vessel, the calandria vault, nuclear piping, steam generators, reactor fuel channels and instrument, control and power cabling. Potential problems with these components can take the station out of service for a long time, and their repair or replacement is not covered by the normal operating and maintenance budget of the stations. (For example, technology for large scale replacement of fuel channels exists but large cost may be a factor.) NPLA programme does not address ageing of other plant components such as valves, switchgear and pumps, because existing maintenance programmes are considered to be adequate in the context of NPLA. However, it is recognized that current monitoring and maintenance methods and practices should be reviewed for their effectiveness in detecting and mitigating ageing degradation.

An NPLA methodology for an assessment of age-related degradation and for an engineering evaluation of the long-term reliability and life of critical components has been developed and is presented in Figure 8.5.

Proactive utility actions to manage age-related degradation of critical components include: initiatives to eliminate significant failure modes, early detection of age-related degradation, specialized preventive maintenance, some repair and upgrading, research and development work.

Currently the NPLA effort is directed at the lead plant - Pickering 'A' NPP. A preliminary review indicates that for many critical components, a number of proactive initiatives related to managing age-related degradation are already in place. For some other critical components, such as fuel channels, a technical committee has full scale programmes of study already underway. For other critical components, a full assessment of all possible proactive NPLA initiatives is required.

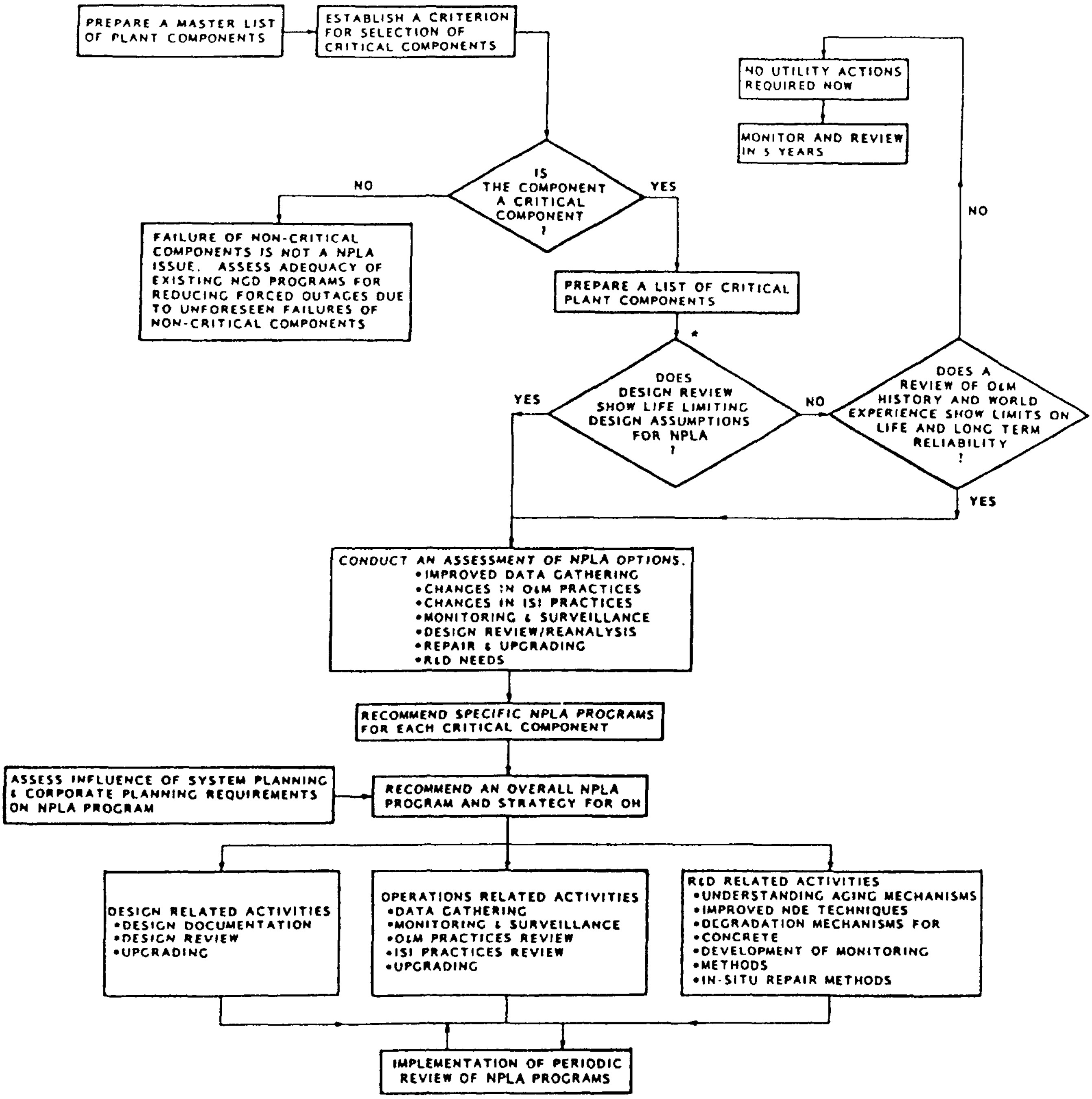


Fig.8.5. Methodology for a nuclear plant life assurance (NPLA) programme.

While it is recognized that OH nuclear plants are relatively young, 17 years and under, and that many ageing effects are small yet, the proposed NPLA programme must be implemented as soon as possible because after a long service, replacement in many instances is the only alternative at a much higher cost and outage penalty than prevention.

Plant condition assessment and refurbishment programme

During 1984-1987 Pickering NPP Units 1 and 2 have undergone a major condition assessment and refurbishment. Zircaloy-2 pressure tubes, some of which had degraded significantly, were replaced with Zr-Nb pressure tubes which are expected to be less susceptible to degradation due to a lower rate of deuterium uptake. At the same time, the Pick-Up (Pickering upgrade) project was carried out to determine the condition of seventy of major component types susceptible to aging degradation, to upgrade deteriorated components and/or to develop an optimum inspection and maintenance strategy for the future. The Pick-Up showed that most of the critical components are in good condition and identified desirable improvements in some of the monitoring and maintenance methods and practices. A similar plant condition assessment and refurbishment projects will be performed during future retubing outages of other units.

Equipment qualification

Equipment qualification (EQ) programme for new reactors (Darlington NPP) considers ageing degradation of safety-related components and accounts for it through the concept of qualified life. Earlier plants have a program to maintain the initial degree of qualification consisting of scheduled testing, inspection, maintenance, replacement and staff training activities. As well, status of EQ at these plants is being reviewed and upgraded on selected basis.

Research and development support

Numerous research and development (R&D) activities related to understanding and managing NPP ageing currently exist in Canada. R&D is being carried out or planned by the nuclear industry and the AECB on selected components in the following general areas:

- understanding of known and identification of as yet unknown age-related degradation processes
- development of inspection and monitoring techniques capable of detecting component degradation before failure
- validation of accelerated ageing techniques
- prediction of remaining service life
- development of effective maintenance methods.

Conclusion

The Canadian approach to managing ageing degradation of NPP components has originated and evolved from the overall Canadian reactor safety philosophy adopted at the start of the Canadian nuclear power programme thirty years ago. It has been under continuous development, incorporating lessons learned and new knowledge. The approach consists of mutually supportive and complementary programmes described above, namely: operational safety management programme, NPLA programme, plant condition assessment and refurbishment, equipment qualification and R&D. These programmes are expected to provide knowledge and technology needed to anticipate, detect and control the negative effects of ageing and thus to maintain adequate plant safety and reliability during the entire plant service life.

8.6. Finland [86]

The ageing and longevity problems in Finland are considered to be an integral part of general safety and availability of nuclear power plants.

The ageing aspects are considered within the normal inspection, control and licensing procedures of the safety authority, the Institute of Nuclear Safety and Radiation Protection (STUK). The utilities, the Imatra Power Company and the Industrial Power Co., Ltd., however, have taken special steps in their preventive maintenance and inservice inspection programmes to take into account the ageing of the operating plants. Several of the Government as well as utility funded research programmes which are carried out at the Technical Research Centre of Finland (VTT), are dealing with ageing and longevity of plant components.

Utility activities

Besides normal preventive maintenance, the utility activities specifically devoted to ageing and longevity problems include, e.g., the following:

- Identification of critical components, i.e. those which are very difficult to replace or repair.
- Strict recording of real transients and reporting to the safety authority.
- Extensive surveillance programmes for radiation damage monitoring including also cladding material and replacement of withdrawn surveillance capsules.
- Individual identification of materials chemistry, exact environments, loadings and repair procedures for pipings subject to stress corrosion cracking, including also consequence analysis of failures.
- Programme for replacement of certain in-core components.
- Extensive inspection and replacement programmes of pipings in turbine plant and in secondary circuit in PWR's for the prevention of damage due to erosion.
- Identification of parameters needed for the evaluation of certain components like valves, instruments and cables, for the determination of proper timing of component replacement.
- Planning for proper operation concerning components which are difficult to replace or repair.
- Optimization of preventive maintenance to decrease transients.
- Qualification of safety related instrumentation inside the containment.

Research activities

The ageing related activities concentrate on the following themes:

- Radiation damage of reactor pressure vessel materials including cladding as well as reactor internals.

- Stress corrosion cracking and corrosion fatigue of pressure boundary materials as well as in-core components.

- Water chemistry monitoring.

- Fracture mechanics including probabilistic aspects.

- Nondestructive evaluation.

- Remanent life time predictions through reliability theories.

Large part of the research programmes are carried out in wide international co-operation.

8.7. Federal Republic of Germany

The safety and availability problems in nuclear power plants in Federal Republic of Germany which may result from ageing and wear are counteracted by a system of inspection and maintenance measures [87]. Operating experience shows that most of the malfunctions were detected in the course of tests. The following is a description of the various measures leading to the detection of damage due to ageing and wear of components.

Inservice Inspections

Inservice inspections are carried out by the licensee of a plant. The supervision of these inspections with respect to the correct time and sequence is performed by a public authority. These inspections include the following:

- Visual inspection
- Leak tests
- Functional tests
- Nondestructive material examinations.

Planned Preventive Maintenance

Besides the testing and monitoring measures prescribed by the authorities, the licensees regularly apply a system of planned maintenance measures. This system covers the entire power plant including those sections which are not part of the safety system. Preventive maintenance includes testing, inspection, overhaul and preventive replacement. In most cases, preventive maintenance is carried out during plant outages.

Typical components subjected to periodic maintenance include the emergency power diesels, reactor coolant pumps, refrigeration systems and also valves which are not covered by the inservice inspections prescribed by the authorities.

Many of the maintenance instructions are suggested to the licensee of the plant by the manufacturer of the components concerned. This maintenance work refers in particular to the parts which are subject to ageing and wear.

In many cases, the manufacturers themselves perform the maintenance work. And as the manufacturers do so in quite a nubmer of plants, their often long years of experience provide them with good overview of the ageing and wear processes affecting "their components".

Preventive maintenance often detects and prevents possible failures due to ageing before the component in question actually fails. It is the general custom in FRG nuclear power plants, that in such a case all redundant components are also inspected in order to be on the safe side.

Other Maintenance Work and Inspections

Besides the "programmed maintenance" described above, which generally refers to larger components, there are a great number of measures used to supervise the state of other components.

An example of such measures is the impact pulse measurement on bearings of pumps and motors. During the operation of the plant, this measurement furnishes information on the state of the bearings.

Another examples is that of the chemical samples of internal lubricant and coolant circuits. The ageing condition of these fluids can be determined by means of these periodic inspections. If the fluids fall below their specified values a replacement will be made.

GRS (FRG Regulatory Body) has collected and evaluated operating data for 15 years. The major questions in this context related to the failure behaviour of components and systems.

A long-term and systematic collection of data makes it possible to recognize cumulations of defects on certain components. Comparisons of failure rates of the same components within certain intervals even permit an early recognition of slowly increasing cumulations of defects, such as they can frequently be observed in the case of ageing effects, i.e., before the failure of entire systems. As a result of the redundancy degree of German plants it is in particular necessary to prevent common mode failures. For this purpose, it is necessary that the values furnished by the data collection for single failures be interpreted correctly in order to eliminate the systematic weak spots which are developing, e.g., as a result of ageing or wear. Moreover, it is thus possible to optimize quality assurance during operation and, if necessary, develop new maintenance and testing strategies.

8.8. Japan

Ageing related actions are in Japan included into eight year LWR Plant Extension Program [88]. The program is divided into three phases and the phase I has been mostly completed by early 1987. The main subjects of the phase I were:

(1) Investigation on the present state-of-the arts in this field.

(2) Preliminary or trial estimation of the plant life on each one plant of both PWR and BWR.

(3) Listing-up of equipments or structures, which are considered to be critical for the extension of plant life.

(4) Recommendations on items of future study including experimental work.

Also, investigation has been done on the present situation of degradation diagnosis, its monitoring devices and replacement and repair technology.

To select critical equipment or structures it is necessary to consider:

(1) The effect of the loss of function of any reactor component or concrete structure on the power plant system.

(2) The amount of work involved in replacing or repairing the respective components, taking into account the radiation dose to personnel, and

(3) The current level of degradation diagnosis. Screening has been done on the basis of these considerations to select critical equipment and structures.

Several items for future investigation, including experimental work, were pointed out at the end of the second year. Among the items selected are: testing of equipment, materials characteristics, e.g., of low carbon steel and stainless steel under corrosive environment, and a method of reusing surveillance specimens of pressure vessel steel which have been Charpy tested.

Appendix

RISK OF COMMON CAUSE FAILURE: FUNCTION OF AGEING AND DESIGN BASIS EVENT STRESS

This appendix discusses the relationship of the risk of common cause failures, the ageing effects and the stress resulting from the occurrence of design basis events (DBE).

With reference to Fig. A.1, the ageing of NPP equipment consists of the degradation that equipment undergoes as a result of normal environmental and operational stresses. The risk associated with ageing processes is that as ageing takes place, equipment reliability may decrease. A precipitous drop in reliability increases the likelihood that multiple failures may occur before detection. This is a quasi-common failure mode effect; the 'common' factor is that redundant equipment simply becomes 'too old'. To represent the risk of common cause failure due to ageing, the risk function in Fig. A.1 is shown as an increasing function of age.

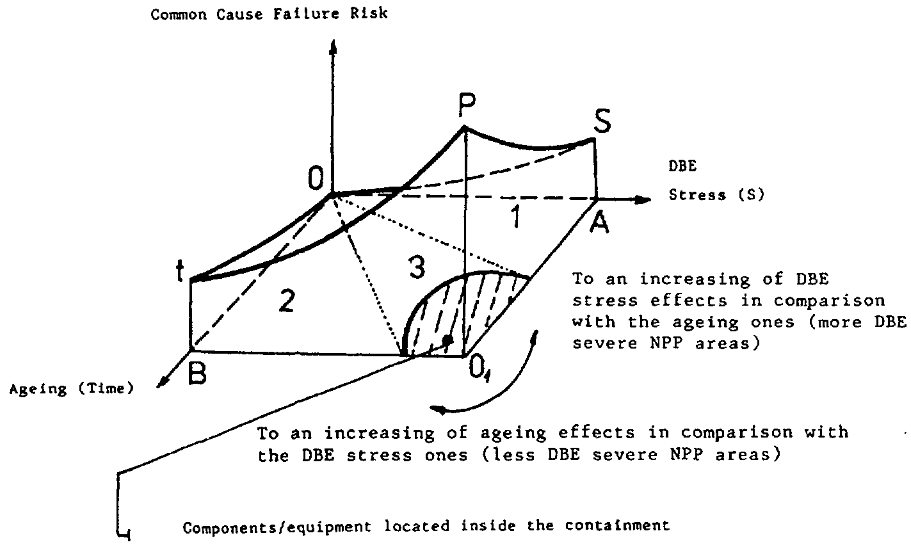


Fig.A.1. Risk of common cause failure: a function of DBE stress and ageing.

The mechanism of common cause failure is then described as a coupling between ageing and DBE stress and different approaches to the equipment qualification described below depend on the ageing and stress significance.

It is necessary to emphasize that the increase in the risk of common cause failure, as material degradation related to both DBE stress and age increases, depends on the real relationship connecting the material degradation to the degradation of the component/equipment function.

Referring to Fig. A.1, it is convenient to point out that the stresses affecting the equipment in the plant result not only from the elevated environmental conditions associated with DBE, but also from the rapid rate of change in these conditions. This rate of change effect is important, since one could otherwise argue that DBE stresses merely accelerate the ageing process that normally occurs.

If DBE stresses are great enough, and if the equipment is not designed adequately to cope with elevated stresses, the possibility of multiple failures during a design basis event can be appreciable. In Fig. A.1 this is represented as an increased risk of common cause failures along the DBE stress axis.

Beyond ageing alone and DBE stress alone manifestations, it is important to recognize a potential coupling relationship between the two effects. As normal ageing takes place, the attendant degradation may cause the equipment to be less and less capable of sustaining DBE stresses. Since all redundant equipment ages in a roughly similar manner, and since the equipment may be exposed to similar DBE stresses, we have the worst possible combination of effects. In Fig. A.1, the risk surface has a maximum (P) where the DBE stress and ageing are both at their highest values. The geometry of the risk surface illustrates a synergistic relationship between the two effects in the vicinity of the maximum probability of common cause failure.

Different areas of the ageing-stress plane represent NPP areas experiencing different levels of environmental severity. In the vicinity of the point O_1, it is possible to have substantial DBE stress and substantial ageing, which is the situation normally for equipment located inside the containment. P is the point of maximum probability of common cause failure, which may occur when both stress and service time are at a maximum.

From this area by means of an anticlockwise rotation one approaches the area of NPPs still with severe DBE environments but with minimal aging. This is characteristic of equipment that may be exposed to significant design basis stresses early in its installed life, or equipment for which degradation due to DBE stresses is large compared with ageing effects. Moving from point O_1 to point B, one approaches region of lesser severity, corresponding to equipment which is not subject to major design basis event stresses, but is subject to ageing.

In Fig. A.1, the mechanism of common cause failure is shown as a coupling between ageing and DBE stress, and different approaches to the equipment qualification described below depend on the ageing and DBE stress significance.

For each region of Fig. A.1, a somewhat different approach to qualification could be used. The simplest region to deal with is the first region (in the vicinity of point A). To test the hypothesis that the risk of common cause failure is not large, it is merely necessary to subject one (or more) equipment sample(s) to simulated DBE stress conditions. This is essentially the type test approach as described in Ref. [6]. No age conditioning (accelerated ageing) of equipment is required.

The second region of Fig. A.1, in the vicinity of point B can also be handled without accelerated ageing, but the analysis is more complicated. The principal approach here is reliability methodology. Areas of the NPP that do not experience high levels of stress from a DBE (mild environment) fall into this region.

In the third region of Fig. A.1, it is possible to have substantial DBE stress and substantial ageing. However, the risk of common cause failure is not uniform over the total range. In the vicinity of the origin O, reliability methods in combination with type testing should be adequate to account for the risk likely to pertain. There may, however, be a case where reliability and/or type testing will be insufficient to determine whether a credible risk is involved. This region is depicted as a cross-hatched circular sector in Figure A.1. In order to provide a definitive answer to the question of credible risk, accelerated ageing and testing under simulated DBE conditions may be necessary. To accomplish this, a considerable understanding of the underlying ageing and failure processes will be required.

For the second region in the vicinity of point B the reliability theory can be used. The equipment ageing problem is described from a reliability viewpoint, in Fig. A.2. For a population of items of equipment having the same design, the failure rate passes through three distinct phases. The early Phase I is characterized by a rapidly decreasing failure rate associated with "infant mortality". Preoperational testing and screening techniques are employed to minimize the number of faulty components placed into service. This is followed by Phase II, characterized by a roughly constant failure rate. This is the ideal area for highly reliable component service. A back-up measure may be inspection, surveillance or testing programmes to identify those failures which occasionally occur. Finally, a 'wear out' Phase III is reached in which the failure rate increases rapidly. On entering this phase, preventive maintenance, refurbishment or replacement of components is required. The three phases constitute the familiar bathtub shaped reliability curve so often used to describe life failure rate effects, in particular those of electronic components/equipment.

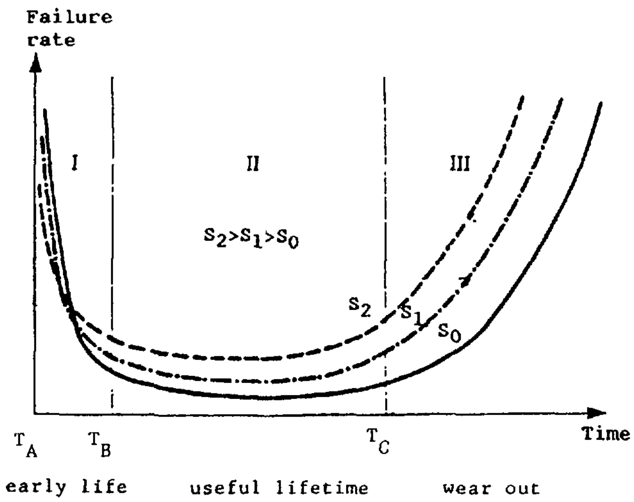


Fig.A.2. Bathtub shaped reliability curve.

The above rather general statements about failure rates and stress are merely hypothetical. Nevertheless, it is appropriate to examine the potential effect of these considerations within the context of nuclear power plant safety.

Firstly, for the simplest case, suppose a particular piece of equipment is to be exposed to a relatively steady environment throughout its installed life. The equipment is assumed to be 'protected', or will not otherwise be exposed to hypothetical abnormal or design basis event environments. The failure rate curve will usually resemble one of the curves in Fig. A.2.

In general, the failure rate curve will be a function of environmental and operational stresses acting on the equipment. The usual response to greater stress (in Regions II and III) will be a failure rate curve which lies above for nominal stress values. This is illustrated for stresses $S_2>S_1>S_0$.

The Phase I portion of the curve illustrates risk due to 'infant mortality'. The usual means of reducing this risk is the institution of pre-installation screening (including 'burn-in' techniques) and pre-operational testing programmes. These are commonly featured in high reliability applications. It is to be taken into consideration also that, because of improved technology, particularly in the electronic components field, the infant mortality area can be significantly reduced.

The Phase II portion represents a useful lifetime period with relatively steady failure rate. Failures that occur in this region are commonly defined as 'random failures'. They are thought to be the consequence of random variations in the properties of materials and components and in manufacturing processes; in particular, they are usually regarded as being independent of degradation due to ageing or wearing out. Generally, it is necessary to analyze failures carefully to determine their causes and then determine whether the cause is a result of a design deficiency or whether it can be classified as random. The risk can be minimized by using reliability analysis techniques to ensure that the likelihood of failures is minimal. If the lack of reliability data suggests an insufficient understanding of Phase II risks, the equipment should not be deployed in a safety application. Periodic in-service surveillance, inspection or testing programmes are used to identify those failures which occasionally occur.

The Phase III part of the failure rate curve represents a wear-out period. How does one assure that equipment has not aged to the extent that redundant components reach a failed condition simply on account of a rapidly increasing failure rate? It turns out that there are several safeguards against this possibility.

Reliability analysis techniques employed for the Phase II part of the failure rate curve can be extended to the Phase III part in a conservative manner. In addition to the basic conclusions that can be drawn from reliability analysis, the risk in Phase III can be reduced by surveillance, inspection, testing, and preventive maintenance; however, because of the increasing failure rate, the period of time allowed between these in-service activities becomes a crucial factor.

The 'dispersion' of in-service failures is important in determining whether multiple failures will occur in a surveillance interval before they can be detected. It can be shown that as time progresses, the dispersion effect tends to increase as a result of small differences in ageing rates. For some equipment, it may be possible to conclude that the likelihood of multiple failures in the Phase III region is acceptably small.

From the foregoing discussion, it can be concluded that equipment that is protected from abnormal or DBE stresses may be qualified strictly in terms of reliability analyses. For equipment subjected to modest DBE stresses, it may be possible to utilize accelerated stress or 'K' factors [4] within a reliability analysis structure to account for a more demanding environment. This approach, coupled with a type test on new equipment, may be sufficient to determine whether or not common mode failure presents a credible risk.

For the third region of Fig. A.1, the first step is to determine to what extent the DBE stresses are independent of the processes involved in ageing. If it can be established that they are independent, then no coupling relationship exists by which the risk is enhanced. In such cases, the risk surface does not achieve a maximum at a single point (maximum age, maximum stress); it should be sufficient to evaluate risk as before, using reliability in combination with a type test.

The second alternative is to determine to what extent ageing and DBE stresses result in essentially the same degradation processes. For example, thermal ageing may be a primary factor in determining the age degradation under normal conditions and over-temperature may be the dominant DBE stress. In this case, accelerated test data and reliability analysis serve to evaluate risk within the framework of K factor methods.

If the aforementioned alternatives are exhausted, there is no other option except that of accelerated ageing and DBE evaluation under simulated stress conditions.

The rater lengthy conceptual consideration of the ageing-DBE stress and common cause failure problem has been intended to illustrate the fact that accelerated ageing is not necessary in all qualification programmes. A brief description of some issues related to the ageing simulation are given in the following.

The stresses to be considered in the simulation of ageing are mainly heat, radiation, humidity, cycling during operation, electrical stress and background vibration. In special applications, other stresses may need to be applied.

For each of the applicable ageing stresses, the accelerated ageing test to be used, and the rationale for its selection, must be given. The reasons for not considering certain stresses must also be stated. The rationale for the sequence of ageing tests should be provided.

A failure modes and effects analysis (FMEA) and probability risk assessment (PRA) may be useful for identifying significant degradation mechanisms. Through this process, an effort should be made to identify the materials and components on which the safety function capability is most dependent and to exclude any that may be irrelevant or insignificant. The service stresses that are likely to produce significant degradation of the essential components and materials are the ones that must be considered in the simulation of ageing.

There is an acceleration ageing model for thermal stress, but there are no practical acceleration models for most other stresses. Therefore the selection of specimen conditioning procedures can only be accomplished within the limits of technology. This limits the investigator to modelling thermal ageing by the Arrhenius model; to simulating radiation ageing by exposing the specimen to a gamma radiation dose equivalent to that which would be received by the specimen during its installed lifetime, at as low a dose rate as is feasible considering the potential dose rate effect, the cost and the time; to determining the effects of operational cycling by estimating the number of cycles during the installed life; and to determining the effects of exposure to humidity, electrical stress and vibration on the basis of tests developed for military hardware.

In the discussion of Fig. A.1, reference has been made to areas of environmental severity. The severity level of an NPP area is commonly identified through the distinction of 'harsh' and 'mild' environmental areas.

Definitions of 'harsh' and 'mild' environments are given in Ref. [1] and are quoted below.

In this respect, and referring to Fig. A.1, moving from Point O_1 to Point A we cover harsh DBE environment area for NPPs, while moving from Point O_1 to Point B we gradually approach mild DBE environment areas for NPPs.

Definition of a harsh environment

An environment expected as a result of the postulated service conditions appropriate for the design basis and post-design basis accidents for the station. A harsh environment is defined as that resulting from a loss of cooling accident (LOCA)/high energy line break (HELB) inside the containment and post-LOCA or HELB outside the containment.

Definition of a mild environment

An environment expected as a result of normal service conditions and extremes (abnormal) in service conditions where seismic events are the only design basis event (DBE) of potential consequence.

Research is under way in some countries on possible coupling effects occurring between component ageing and seismic effects, which may result in common cause failures in an otherwise mild environment. However, other environmental service conditions may also be contributory factors. With regard to the seismic testing of equipment located in mild environments, a correct approach should foresee the need to include an evaluation of the existence of significant ageing mechanisms. If they are considered to be present, then they must be included in the qualification programme in the phase of equipment age conditioning prior to seismic testing.

It is useful to note that an ageing mechanism is considered significant if, in normal and abnormal service environment it causes degradation during the installed life of the equipment that progressively and appreciably renders the equipment vulnerable to failure to perform its safety function(s) under DBE conditions. Other terms used such as 'normal', 'abnormal' and 'DBE' service conditions are defined below; a reference can be made also to IEEE Std 323-1983 [1].

The specified service conditions define the applications of the equipment. Service conditions are of two types: environmental conditions, which define the surroundings of the equipment, and operating conditions, which include such parameters as input power and process variables. Both sets of service conditions must be defined for normal, abnormal and DBE conditions, including the entire period - following the start of each DBE - for which qualification must be demonstrated.

Environmental conditions include temperature, pressure, relative humidity, radiation, background vibration and any condition or conditions unique to a specific application, such as a corrosive atmosphere.

Operating conditions include input electrical supply parameters (such as voltage and frequency), process variables (such as fluid medium, pressure, temperature and flow rate of fluid), and load variables (such as current and force), as applicable.

For normal and abnormal environmental and operating conditions, it is not always sufficient to state the average and extreme values of the parameters. It is important to provide as much information as possible concerning the duration of each level of a service parameter, particularly in instances where the assumption of a conservative qualification can lead to an unnecessarily severe test. For example, the assumption that the temperature is always near its peak value because of the absence of information on the time dependent temperature distribution can lead to incorrect conclusions regarding thermal ageing characteristics of the specimen. As for another example, the assumption that a cable is always operating at its full rated load can lead to excessively severe testing.

It is essential also to account for the effects of equipment operation on the associated service conditions - for instance, the temperature rise and possible reduction of relative humidity caused by electrical loading.

The principal DBEs, and the parameters to be specified for each, are as follows:

(1) Loss of coolant accident (LOCA), main steam line break (MSLB) and high energy line break (HELB);

- Temperature, pressure, and relative humidity as functions of time

- Types of ionizing radiation and associated energy levels; for each type of radiation, the dose rate and the integrated dose as functions of time

- For chemical spray, the composition and pH (and any essential tolerance) and the spray onto equipment -- for example, $N(m^3/s)/m^2$ directed vertically downward onto the equipment expressing spray rate as m^3/s per unit volume of the reactor building.

(2) Submergence

- Composition of liquid (including potentially damaging contaminants), temperature of liquid and level of surface of liquid above equipment to be qualified.

(3) Seismic event

- Response spectra at equipment mounting points for horizontal and vertical components of motion

- Zero period amplitude

- Percentage of critical damping.

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