GUIDE NO. AERB/NPP/SG/O-14

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GOVERNMENT OF INDIA

AERB SAFETY GUIDE

LIFE MANAGEMENT OF NUCLEAR POWER PLANTS

ATOMIC ENERGY REGULATORY BOARD

AERB SAFETY GUIDE NO. AERB/NPP/SG/O-14

LIFE MANAGEMENT OF NUCLEAR POWER PLANTS

Atomic Energy Regulatory Board
Mumbai - 400 094
India

March 2005

Price:

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FOREWORD

Activities concerning establishment and utilisation of nuclear facilities and use of radioactive sources are to be carried out in India in accordance with the provisions of the Atomic Energy Act 1962. In pursuance of the objective to ensure safety of members of the public and occupational workers as well as protection of environment, the Atomic Energy Regulatory Board has been entrusted with the responsibility of laying down safety standards and framing rules and regulations for such activities. The Board has, therefore, undertaken a programme of developing safety standards, codes of practice and related guides and manuals for the purpose. While some of theses documents cover aspects such as siting, design, construction, operation, quality assurance and decommissioning of nuclear and radiation facilities, other documents cover regulation aspects of these facilities.

Codes of practice and safety standards are formulated on the basis of internationally accepted safety criteria for design, construction and operation of specific equipment, structures, systems and components of nuclear and radiation facilities. Safety codes establish the objectives and set minimum requirements that shall be fulfilled to provide adequate assurance for safety. Safety guides elaborate various requirements and furnish approaches for their implementation. Safety manuals deal with specific topics and contain detailed scientific and technical information on the subject. These documents are prepared by experts in the relevant fields and are extensively reviewed by advisory committees of the Board before they are published. The documents are revised when necessary, in the light of experience and feed back from users as well as new developments in the field.

The Code of Practice on Safety in Nuclear Power Plant Operation (AERB Code No. SC/O, 1989) lays down the minimum requirements for safe operation of NPP. This safety guide is one of a series of guides, which have been issued or under preparation, to describe and elaborate on the specific parts of the code.

It is known that structures, systems and components of NPP undergo wear and degradation with age. The potential degradation mechanisms include corrosion, erosion, creep, irradiation and a combination of these factors. This guide addresses the issues related to such causative factors, timely detection of incipient failures and preventive/ mitigation measures. A programme for life management should be instituted for each NPP so as to ensure that, items important to safety of the NPP function without impairment of their reliability and intended safety margins. In drafting this guide, extensive use has been made of the information contained in the relevant documents of IAEA issued under its NUSS programme.

Consistent with the accepted practice, ‘shall’, ‘should’ and ‘may’ are used in the guide to distinguish between a firm requirement, a recommendation and a desirable option respectively. Appendices are an integral part of the document, whereas Annexures, footnotes, bibliography and list of participants are included to provide information that might be helpful to the user. Approaches for implementation different to those set out in the guide may be acceptable, if they provide comparable assurance against undue risk to the health and safety of the occupational workers and the general public, and protection of the environment.

For aspects not covered in this guide, applicable and acceptable national and international standards, codes and guides should be followed. Non-radiological aspects of industrial safety and environmental protection are not explicitly considered in this guide. Industrial safety is to be ensured through compliance with the applicable provisions of the Factories Act, 1948 and the Atomic Energy (Factories) Rules, 1996.

This guide has been prepared by specialists in the field drawn from consultants experienced in all aspects of NPP. It has been reviewed by the relevant AERB Advisory Committee on Codes and Guides and Advisory Committee on Nuclear Safety.

AERB wishes to thank all individuals and organizations who have prepared and reviewed the draft and helped in its finalisation. The list of persons, who have participated in this task, along with their affiliations is included for information.

(S. K. Sharma)
Chairman AERB

DEFINITIONS

Acceptable Limits

Limits acceptable to the regulatory body for accident conditions or potential radiation exposure.

Accident Conditions

Substantial deviations from operational states which could lead to release of unacceptable quantities of radioactive materials. They are more severe than anticipated operational occurrences and include design basis accidents as well as beyond design basis accidents.

Ageing

General process in which characteristics of structures, systems or component gradually change with time or use.

Ageing Management

The engineering, operations and maintenance actions to control ageing degradation and wearing out of structures, systems or components within acceptable limits.

Anticipated Operational Occurrences

An operational process deviating from normal operation which is expected to occur during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety nor lead to accident conditions.

Approval

A type of regulatory consent issued by the regulatory body to a proposal.

Atomic Energy Regulatory Board (AERB)

A national authority designated by the Government of India having the legal authority for issuing regulatory consent for various activities related to the nuclear and radiation facility and to perform safety and regulatory functions including enforcement for the protection of the site personnel, the public and the environment from undue radiation hazards.

Audit

A documented activity performed to determine by investigation, examination and evaluation of objective evidence, the adequacy of, and adherance to, applicable codes, standards, specifications, established procedures, instructions, administrative or operational programs and other applicable documents, and the effectiveness of their implementation.

Authorisation

A type of regulatory consent issued by the regulatory body for all sources, practices and uses involving radioactive materials and radiation generating equipment. (See also Consent)

Commencement of Operation of NPP

The specific activity/activities in the commissioning phase of a nuclear power plant towards first approach to criticality starting from fuel loading

Commissioning

The process during which structures, systems and components of a nuclear and radiation facility, on being constructed, are made functional and verified to be in accordance with design specifications and found to have met the performance criteria.

Common-Cause Failure

The failure of a number of devices or components to perform their functions, as a result of a single specific event or cause.

Consent

A written permission issued to the consentee by the regulatory body to perform the specified activities related to nuclear and radiation facilities. The types of consent are 'license', 'authorisation', 'registration', and 'approval' and will apply according to the category of the facility, the particular activity and radiation sources involved.

Construction

The process of manufacturing, testing and assembling the components of a nuclear or radiation facility, the erection of civil works and structures, the installation of components and equipment and the performance of associated tests.

Decommissioning

The process by which a nuclear or radiation facility is finally taken out of operation, in a manner that provides adequate protection to the health and safety of the workers, the public and of the environment.

Design Basis Accidents (DBAs)

A set of postulated accidents which are analysed to arrive at conservative limits on pressure, temperature and other parameters which are then used to set specifications to be met by plant structures, systems and components, and fission product barriers.

Design Basis Events (DBE)

The set of events, that serves as part of the basis for the establishment of design requirements for structures, systems or components within a facility. Design basis events (DBEs) include normal operations, operational transients and certain accident conditions under postulated initiating events (PIEs) considered in the design of the facility. (See also ‘Design Basis Accidents’)

Design Life

The period for which the item will perform satisfactorily meeting the criteria set forth in the design specification.

Deterministic Method

A method for which most of the parameters and their values are mathematically definable and may be explained by physical relationships and are not dependent on random statistical events.

Examination

An element of inspection consisting of investigation of materials, components, supplies or services to determine conformance with those specified requirements which can be determined by such investigation.

Full Power

It is the rated thermal power of the reactor, i.e. the gross fission power as established by station heat balance, using approved methodology.

In-service Inspection (ISI)

The inspections of structures, systems and components carried out at stipulated intervals during the service life of the plant.

Inspection

Quality control actions which by means of examination, observation or measurement determine the conformance of materials, parts, components, systems, structures as well as processes and procedures with predetermined quality requirements.

Items Important to Safety (IIS)

The items which comprise:

(1) those structures, systems, equipment and components whose malfunction or failure could lead to undue radiological consequences at plant site or off-site.

(2) those structure, systems, equipment and components, which prevent anticipated operational occurrences from leading to accident conditions.

(3) those features which are provided to mitigate the consequences of malfunction or failure of structures, systems, equipment or components.

Normal Operation

Operation of a plant or equipment within specified operational limits and conditions. In case of a nuclear power plant this includes, start-up, power operation, shutting down, shutdown state, maintenance, testing and refueling.

Nuclear Power Plant (NPP)

A nuclear reactor or a group of reactors together with all the associated structures, systems, equipment and components necessary for safe generation of electricity.

Operation

All activities following and prior to commissioning performed to achieve, in a safe manner, the purpose for which a nuclear/radiation facility is constructed, including maintenance.

Operating Organisation

The organisation so designated by responsible organisation and authorised by regulatory body to operate the facility.

Operating Personnel

Those members of site personnel who are involved in the operation of the nuclear/ radiation facility.

Operational Limits and Conditions (OLC)

Limits on plant parameters and a set of rules on the functional capability and the performance level of equipment and personnel, approved by Regulatory Body, for safe operation of the nuclear/radiation facility.
(See also Technical Specifications for Operations)

Operational States

The states defined under ‘Normal Operation’ and ‘Anticipated Operational Occurrences’.

Plant Management

The members of site personnel who have been delegated responsibility and authority by the operating organisation for directing the operation of the plant.

Prescribed Limits

Limits established or accepted by the regulatory body.

Pre-Service Inspection (PSI)

Inspection prior to or during commissioning of the plant to provide data on initial conditions supplementing manufacturing and construction data as a basis for comparison with subsequent examinations during service.

Quality Assurance (QA)

Planned and systematic actions necessary to provide adequate confidence that an item or service will satisfy given requirements for quality

Records

Documents which furnish objective evidence of the quality of items and activities affecting quality. They include logging of events and other measurements.

Regulatory Body

(See ‘Atomic Energy Regulatory Board’)

Regulatory Consent

(See ‘Consent’)

Reliability

The probability that a structure, system, component or facility will perform its intended (specified) function satisfactorily for specified period under stated conditions.

Responsible Organisation

The organisation having overall responsibility for siting, design, construction, commissioning, operation and decommissioning of a facility.

Safety Culture

The assembly of characteristics and attitudes in organisations and individuals which establishes that, as an overriding priority, the protection and safety issues receive the attention warranted by their significance.

Severe Accidents

Nuclear facility conditions beyond those of the design basis accidents causing significant core degradation.

Siting

The process of selecting a suitable site for a facility including appropriate assessment and definition of the related design bases.

Specification

A written statement of requirements to be satisfied by a product, a service, a material or process indicating the procedure by means of which it may be determined whether specified requirements are satisfied.

Surveillance

All planned activities namely monitoring, verifying, checking including in-service inspection, functional testing, calibration and performance testing carried out to ensure compliance with specifications established in a facility.

Technical Specifications for Operation

A document approved by the Regulatory Body, covering the operational limits and conditions, surveillance and administrative control requirements for safe operation of the nuclear or radiation facility. It is also called 'Operational Limits and Conditions'.

Testing (QA)

The determination or verification of the capability of an item to meet specified requirements by subjecting the item to a set of physical, chemical, environmental or operational conditions.

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APPENDICES

ANNEXURES

1. INTRODUCTION

1.1 General

1.1.1 The structures, systems and components (SSC) which include items important to safety (IIS) of a nuclear power plant (NPP) are required to function, without impairment of their safety margins and reliability, as per design specifications in all operational states during the service life of NPP.

1.1.2 The physical characteristics of SSC change with time and use, due to ageing process. If appropriate actions are not taken, the safety margins provided in the design are reduced due to ageing. Safety state (i. e. integrity and functional capability) of plant components, both passive and active, change with use and time resulting in reduction of safety margin. (Refer Appendix 1).

1.1.3 Careful assessment of the ageing characteristics of the SSC, factors influencing the ageing process and their consequences on safety margins and reliability, is essential for planning and implementing timely actions for assuring safe operation of an NPP during its life-time. Feasibility of upgrading of SSC, for complying with current safety standards will also require a planned approach for effecting necessary modifications.

1.1.4 A programme for all phases of management of NPP; pre-operational, operational and post-operational phases in respect of the SSC, should be instituted by the responsible organisation (RO)/operating organisation, (Op.O) for achieving the above objectives and those given below.

1.2 Objective

1.2.1 This guide details the essential factors that are required for a comprehensive assessment of the ability of the IIS for performing their intended functions reliably as per design specifications. Requirements for planning and implementing an effective life management (LM) programme for the IIS in NPP are addressed.

1.2.2 Factors that need consideration during the siting, design, construction and operating phases of an NPP for planning and implementing the LM programme are also explained in the guide.

1.3 Scope

1.3.1 The guide covers the following aspects of the life management of an NPP:

(i) Degradation of the SSC during pre-operational and operating phase

(ii) Analysis of factors influencing ageing and residual life of the SSC

(iii) Measures to mitigate ageing effects

(iv) Considerations and approach for license renewal at the end of design life

(v) Organisational aspects of life management.

Decommissioning and recovery of site are not covered in this guide.

1.3.2 The guide primarily addresses the requirements for life management of IIS. The recommendations given in this guide could also be gainfully used for other SSC for improving plant availability and economic generation of power.

2. LIFE MANAGEMENT PROGRAMME

2.1 Rationale for Life Management

2.1.1 Life management deals with the management of NPP starting from the stages of siting and continuing through the stages of design, construction, commissioning and operation. Life management also enables acceptable safety performance during operating phase and extended phase of operation. Special attention is required during the extended phase of operation, due to the ageing effect.

2.1.2 Existing programmes relating to various activities and management of NPP (operation, maintenance, in-service inspection, surveillance and radiological protection) for safe operation are detailed in other safety guides. Life management coordinates all these activities. It addresses issues like understanding, predicting and detecting effects of ageing and measures for mitigating actions. It brings to the notice of the designers, manufacturers, and operating organisation, the need for effective ageing management of SSC. It can be best accomplished under systematic umbrella type programme that coordinates existing activities relevant to management of ageing.

2.1.3 Considerations in formulating the life management programme should include:

(i) Degradation of plant SSC caused by a combination of ageing mechanism and premature degradation during various phases of plant life.

(ii) Understanding the role of service environment and various degradation mechanisms in causing premature ageing and implementing suitable O & M practices for minimizing degradation.

(iii) Up-gradation of safety levels to the extent feasible with increase in knowledge and improvement in technology.

2.2 License Renewal for Operation

2.2.1 License is issued to an NPP for design life, which is generally 30 to 40 years. During the process of licensing, aspects important to safety are assessed at stages, such as siting, design, construction, commissioning and operation. The regulatory body issues the license for operation of NPP after review of commissioning test results. Preliminary assessment of feasibility of decommissioning of plant at the end of design life is also considered. The Regulatory Body issues initial license authorising operation for a specified period. Renewal of authorisation for operation for further period is issued after assessment of safety performance of the NPP.

2.2.2 A periodic safety review (PSR) is carried out once in ten years as per safety guide AERB/SG/O-12 for authorization for operation during design lifetime.

2.2.3 Well before (atleast 5 years) the end of design life, a license renewal for further period of operation (beyond design life) is to be applied as per guidelines given in this safety guide. The basic approach and criteria for such license renewal are as follows:

(i) Original design safety levels should be met.

(ii) Upgradation in plant is carried out to achieve safety requirements as stipulated for the NPP built to earlier standards by Regulatory Body.

(iii) Revised environmental impact assessment should be prepared and reviewed.

(iv) All site parameters should be assessed and reviewed.

(v) Safety analysis should be reviewed and updated if the original safety analysis is less conservative as compared to current standards (Ref. Appendix-2 on 'Current Standards and Practices').

(vi) A comprehensive review of operational performance should be carried out.

(vii) In-depth review of life management including ageing msanagement programme (AMP) should be carried out.

2.3 Phases of Life Management

2.3.1 The life management of an NPP is carried out in two distinct phases: pre-operational and operational. (Refer Appendix- 3)

2.3.2 Pre-Operational Phase

This period includes siting, infrastructure development at site, design, manufacture of equipment and components, construction and commissioning. Pre-operational tests of all SSC including initial failures and corrective actions taken form base-line data for LM.

2.3.3 Operational Phase

This phase has a life span of 30 to 40 years and may be extended for further period of 20 to 30 years after appropriate life extension program, regulatory review and license renewal. Typical ageing management programme is given in Appendix-4.

2. 4 Organisational Aspects of Life Management

RO/Op. O should have appropriate organisational set up to develop and implement life management programme. This programme should be conceived and implemented from the stage of site selection, through the life of the NPP. Emphasis of the programme should be on proper site selection, design, construction, commissioning and operation so that the NPP is able to operate safely through the design life and beyond. The organisations involved in siting, design, construction, commissioning and operation should participate in the life management programme. Following aspects are considered:

(i) Organisation of life management

(ii) Human aspects and safety culture

(iii) Infrastructural development

(iv) Review of life management programme

Section 5 deals with the above aspects in detail.

2.5 Elements of Life Management

2.5.1 General

The objective of life management is successful operation of the NPP for its design life, in such a way that the SSC remain in satisfactory state so that the NPP can be re-licensed for operation for a further specified period. After such an extended life, it should be possible to decommission the NPP as per approved procedures. The main elements of life management are discussed in the following.

2.5.2 Ageing Considerations

2.5.2.1 General

2.5.2.1.1 Ageing management programme (AMP) has a three-step methodology.

(i) Selection of SSC for evaluation of ageing studies.

(ii) Ageing management studies for selected SSC.

(iii) Appropriate ageing management initiatives and related actions.

2.5.2.1.2 The Information/data collected for evaluating ageing characteristic of the selected SSC should include the following:

(i) Material properties, specifications and design life of components.

(ii) Operating conditions, stressors, degradation mechanism and condition monitors.

(iii) Effects of degradation and failure.

(iv) Effects of internal and external hazards.

(v) Generic ageing Issues.

(vi) Operational experience feedback.

(vii) Research and development on specific component.

(viii) Operation, surveillance and maintenance history.

(ix) Results of examination of components removed from service.

2.5.2.1.3 The ageing management programme (AMP) comprises of two strategies for SSSC i.e. managing active SSC and managing passive SSC.

(i) Active components are continuously monitored, or periodically functionally tested for performance and corrective measures taken.

(ii) Passive SSC are generally designed for plant life, e. g. pressure boundary components and civil structures. If premature degradation or failure is detected by ISI or surveillance, corrective action is taken.

2.5.2.1.4 The AMP focuses on co-ordination of existing programmes to avoid duplication of efforts. The existing programmes contributing to ageing management of SSC include:

(i) Preventive maintenance.

(ii) ISI, surveillance, testing and condition monitoring.

(iii) Control of plant parameters within operating limits and conditions.

(iv) Feed back of operational safety experience and analysis.

(v) Review and updating of operating procedures.

2.5.2.1.5 The data collected as above, provide information for evaluating degradation and ageing management of a SSC. An equipment master list is prepared and data is collected at periodic intervals specified for review. The qualified life of the SSC is the maximum period they are expected to perform its intended function, under normal operating conditions, anticipated operational occurrences, accident and post-accident conditions.

2.5.2.2 Selection of Structures, Systems and Components for Ageing Evaluation

2.5.2.2.1 The selection process of grouping of components for ageing management studies as given in Appendix-5 and 6, is used to progressively eliminate, after due consideration and justification, components, which can be managed by routine O & M practices.

2.5.2.2.2 Accordingly the following steps for selecting SSC should be instituted:

(i) SSC classified as per AERB safety guide on 'Safety Classification and Seismic Categorisation, AERB/SG/D-1'are selected.

(ii) Components in lower classification, but whose failure prevent other components in higher classification, to perform safety functions should be considered.

(iii) Assessment of SSC failure in terms of high risk (severity and frequency).

(iv) Failure of some components not acceptable from plant life consideration, as they are life-limiting components or their replacement could require unacceptable personnel radiation exposure.

(v) Assess potential for ageing degradation of above components.

2.5.2.2.3 After selecting the SSC as detailed above, ageing management issues need to be identified and mitigative measures taken. If necessary, pilot studies may be instituted. Selection process for SSC important to ageing management studies should be carried out as per the flow diagram given in Appendix-6.

2 5.2.3 Environmental and operational stressors

2.5.2.3.1 Following are the main environmental and operational stressors :

(i) Irradiation.

(ii) Temperature.

(iii) Humidity.

(iv) Process chemistry/chloride contamination (which induce inter granular stress corrosion and stress corrosion cracking).

(v) Hydriding of Zr material.

(vi) Erosion.

(vii) Impact of corrosion from outside corrosive environment.

(viii) Pressure and/or temperature transients and cycling.

(ix) Heating and cooling rate outside permissible limits.

(x) Stagnant fluid in crevices.

(xi) Frequent testing.

(xii) Frequent start and stop operations.

(xiii) Abnormal variation in voltage and frequency.

(xiv) Fatigue (thermal/vibration).

(xv) Mechanical wear/fretting/denting.

(xvi) Weld related cracking.

2.5.2.3.2 The list of degradation mechanisms, affected materials and equipment are enumerated in Annexure -IA and Annexure- IB for mechanical components and civil structures respectively. There may be more than one degrading mechanism affecting a component concurrently.

Following annexures give information regarding ageing mechanisms, which cause degradation of SSC.

Annexure-II	:	Ageing mechanisms causing degradation.
Annexure-III	:	Management of ageing of I & C equipment in NPP.
Annexure-IV	:	Management of ageing of cable.
Annexure-V	:	Degradation mechanisms in PHWR components.

2.5.2.4 Risk Informed Reliability Considerations in Ageing Assessment.

2.5.2.4.1 Defense in depth is achieved by providing multiple barriers against possible release of radioactivity into plant environment and/or public domain. These barriers are monitored, maintained and the status is verified to be in conformity with the design intent at specified intervals by administrative controls.

2.5.2.4.2 It is essential to monitor statistics of failure rates of IIS from ageing considerations. The mode of degradation of such IIS should be monitored on a planned basis to reduce the probability of their unanticipated failures.

Ageing impact should be evaluated based on:

(a) Modes of failures.

(b) Surveillance test intervals.

(c) Frequencies of observed failures.

2.5.2.4.3 A Level –1 probabilistic safety analysis (PSA), based on time dependent unavailability models for components, facilitating ageing evaluation and assessment of the risk importance of maintenance, surveillance and ageing management programme should be carried out for the NPP. Plant specific failure data should be used along with expert judgement for finding out ageing related unavailability values. Conclusions from such Level-1 PSA results in terms of increase in core damage frequency (CDF) due to ageing may be used for risk informed decision making.

2.5.2.5 Common Cause Failure Effect due to Ageing

2.5.2.5.1 Common cause failures (CCF) lower safety levels by affecting IIS sharing common environment. Environmental conditions in which such IIS operate should be monitored and controlled in compliance with design for minimising age related CCF. If such common cause failure is not controlled, defense in depth could be jeopardized. The age related common cause mechanisms could also affect redundant components. Staggered maintenance programmes should be followed for redundant equipment to minimise simultaneous failures.

2.5.2.6 Equipment Qualification for Items Important to Safety

2.5 2.6.1 The IIS are designed to cater to their functional requirements during and post design basis accident even when they are near the end of their service life. The design must provide for adequate margins for cumulative stresses for lifetime and anticipated mechanical loads for design basis accident (DBA) condition.

2.5.2.6.2 All IIS should be qualified for their service life, under expected service environment. The NPP should have a programme for monitoring and ensuring that environmental factors during service life are within accepted limits. The NPP should also have an inspection and testing programme to check their functional capability.

2.5.2.6.3 Procedure for Equipment Qualification

(i) During design phase, a qualification programme specifically tailored for the plant, should establish environmental conditions and safety functions for IIS. Equipment qualification specification are established based on:

(a) Duty/performance requirement during normal operation and DBA.

(b) Environmental and operational stressors.

(c) Degradation effect of ageing.

(ii) The equipment qualification is established by:

   (a) Type testing of actual equipment/component with simulated service conditions and/or accelerated ageing and service conditions as per EQ procedure.

   (b) Operating experience providing information on failure rates, modes and reliability analysis.

   (c) Analysis by building valid mathematical models of equipment/ components.

   (d) Combination of some or all of above.

(iii) If the qualified life of an equipment is less than the required installed life, an on-going qualification programme may be implemented by:

   (a) Testing identical equipment/component.

   (b) Installing additional equipment beside the required equipment (removed from service and left in location) before the end of qualified life period.

   (c) Original equipment can be considered for in-situ test or type test to determine residual life.

2.5.2.7 Ageing Management Database

2.5.2.7.1 Baseline Information

Baseline data should include equipment/component material history including physical and chemical properties, special treatments, stress analysis, concessions accorded during manufacture, acceptance criteria, pre-service inspection data, test results, functional capability, and service conditions during design life. Information dispersed in various reports should be stored equipment/system wise for retrieval.

2.5.2.7.2 Operating History Data

The data for a specific SSC, should include actual environmental conditions, process chemistry, system parameters, mode of operation, transients, and test results. Any significant deviations from design assumptions should be recorded. Equipment specific data on off-normal operations, failures and their method of detection, root cause analysis, corrective actions, and modifications/improvements should be part of operation history data. Operating experience feedback on similar equipment from other plants should be available for comparison. ISI data on failure due to IGSCC, SCC, pipe thinning, and results of condition monitoring should also form part of operating history data. All data pertaining to IIS should be compiled, equipment/component wise, in individual operating history documents.

2.5.2.7.3 Maintenance History Data

Data on maintenance of equipment including description of work, information 'as found' and modification history along with collective radiation exposure to personnel, should be available for initiating mitigating measures.

2.5.2.7.4 Comprehensive data is necessary for management of ageing effects. Data relevant to safety and long-term functional reliability of IIS should be identified, and reviewed at periodic intervals for detecting components and or system degradation.

The system for data collection and record keeping should include the following:

(i) Prediction of functional capability and residual life of IIS.

(ii) Identification and evaluation of degradation due to ageing effects.

(iii) Optimisation of operating conditions and practices to minimise ageing.

(iv) Decision on scheduling of predictive maintenance, repair/ replacement.

(v) Assessment concerning continued operation and renewal of license.

(vi) Programme for R & D for ageing studies.

(vii) Development of new materials, technologies and designs.

2.5.2.8 Ageing Management Studies

2.5.2.8.1 The following should be considered for ageing management studies of IIS.

(i) Understanding of ageing process.

(ii) Monitoring of ageing by detecting degradation before failure.

(iii) Adopting appropriate maintenance and repair practices.

2.5.2.8.2 Ageing management programme could be divided into two phases.

(i) Phase I ageing study (Refer Appendix-7) focuses on the following:

(a) Review and analysis of existing data of operating experience, condition monitoring, ISI, and failure analysis to identify and understand ageing process.

(b) Review of current techniques of monitoring and mitigation of ageing effects.

(c) Preparation of interim ageing assessment report for each SSC.

(d) Recommendations for Phase-II.

(ii) Phase-II in-depth ageing study (Appendix 8) is related with the following:

(a) Identified technology upgrade in Phase-I.

(b) In-depth review of operational experience.

(c) R & D on component and material ageing technology.

(d) Studies on understanding of ageing.

(e) Studies on monitoring of ageing to verify existing diagnostic and data evaluation techniques or to develop new technology.

(f) Studies on measures to extend life.

(g) Integrated assessment of ageing and recommendations.

The in-depth study would provide information for evaluating degradation and ageing management of SSC. Action plans are formulated for the currently operating NPP for operation, maintenance and design review.

2.5.3 Managing Technological Obsolescence

Ageing management involves ensuring continued availability of qualified spare parts or providing for timely replacement of obsolete equipment. Manufacturers discontinue supply of spares/equipment due to following.

(i) Upgradation of technology.

(ii) Change in specification by vendors.

(iii) Change in standards.

(iv) Discontinuation of product.

2.5.4 Generic Safety Issues

Operating experience from NPPs is compiled and analysed by various international organisations. Research and development aimed at understanding failures due to ageing of components and materials, lead to improvement of component design and advancement of material technology. First generation NPPs had brought forth many such generic issues. Improvements had been incorporated in subsequently designed NPPs to improve operational safety and reliability of plants.

2.5.5 Change in Regulatory Requirements

The regulatory requirements undergo modifications based on operational experience, international feedback, research findings, computational tools and methods and new developments in technology. Hence, life management programme should address changes in regulatory requirements during anticipated life of plant.

2.6 Planning of Life Management

For planning of life management, following technical aspects are considered:

(a) Screening and grouping

(i) Screening of SSC.

(ii) Grouping of NPP components for ageing management.

(b) Studies

(i) Study of ageing mechanisms and measures that can slow down degradation caused by them.

(ii) Study of safety margin of SSC and their assessment as per current practices and safety requirements.

(c) Internal inputs

(i) Material and equipment selection.

(ii) Operation and maintenance practices including modification/ replacement if needed for enhancing the life of IIS.

(iii)Periodic safety review of NPP.

(iv)Equipment qualification of IIS.

(v) Environmental considerations.

(d) External inputs

(i) Feedback from operating plants and pilot studies.

(ii) Research and development.

3. PRE-OPERATIONAL LIFE MANAGEMENT CONSIDERATIONS

3.1 General

Pre-operational activities such as site selection, compilation of baseline data on environmental conditions, design, manufacturing, storage, construction and commissioning have influence on safety and life span of an NPP. All site related parameters and other design inputs, which influence the life of the NPP should be evaluated and accounted for in design and construction. (Annexure-VI gives aspects of design, manufacture and storage).

3.2 Environmental Considerations during Site Selection

(a) Presence of aggressive chemicals e. g. , chlorides, sulfides and sulfur dioxide have deleterious effects on metals and concrete. Such sites should preferably be avoided.

(b) However, if such sites are selected for other reasons, mitigatory measures should be implemented including those for storage of materials, equipment and spares beginning with the construction phase and during the life span of the NPP.

(c) Cyclonic weather conditions, erosion of coastal areas, tidal waves and coastal flooding, will need considerations during site selection assessment.

(d) Apportionment of radiological limits to individual units shall be reconsidered while deciding multi facilities at existing sites.

3.3 Design

3.3.1 Taking into account the long life expected for the SSC, the following aspects should be considered at the design stage for ageing management:

(i) Reliable performance during service life.

(ii) Adequate provision for safety margin in operation and ease in maintenance and ISI.

(iii) Adequate residual life at the end of service period.

(iv) Effect of large off-normal variations in frequency and voltage for sustained periods should be taken into account during preparation of specifications.

3.1.2 The design engineering has to estimate the design life of the SSC, considering various service conditions and factors that would influence the service life.

This will form the basis for the ageing monitoring for assessing residual life. For example, the influence of the number of thermal cycles, low/high frequency operations and over-pressure incidents should be evaluated at specified intervals and corrective measures taken to enhance the service life. Additional design considerations are enumerated in Annexure-III.

3.4 Procurement, Manufacture and Storage of Equipment

3.4.1 The responsible organisation (RO) should ensure that the procured equipment/components meet the codes/standards. The design specifications should cover technical and QA requirements. The responsibility of the vendors should be defined and interface arrangements for approval and inspection should be specified. Selection of vendors should be based on experience feedback, existing manufacturing and QA facilities.

3.4.2 Manufacturer/vendor of IIS should establish and implement QA programme the level of which should be commensurate with the safety significance of the items manufactured. The RO will ensure compliance of QA requirements by vendor. Approved laboratories for physical and chemical properties should test materials used in manufacture. Random samples should be cross-checked by other reputed laboratories. The manufacturer will implement procedures for identifying material/components at receipt and all stages of manufacturing. Non-conformance should be documented and should be reviewed and approved by design engineering personnel. Manufactured equipment will be tested as per specifications and stored in clean environment.

3.4.3 Storage arrangements during manufacture, transportation and at site prior to commencement of erection/construction should be carried out to minimize those factors which influence ageing. Procedure for storage for IIS should be established to avoid deterioration prior to commencement of operation considering saline atmosphere or corrosive pollutants. Appropriate procedures and storage conditions should be established and maintained for spares and subassemblies to prevent degradation.

3.5 Construction and Erection

3.5.1 Construction and erection should be planned in detail with regard to sequencing of the different structures, systems and components. All equipment and components inside the reactor building should be handled under controlled atmosphere free from dust and humidity by preferably operating ventilation system. Clean room condition should be established during erection and assembly of critical components like coolant channels.

3.5.2 Equipment, pipes, valves and ducts openings should be covered by impermeable material such as plastic covers for preventing debris, dust and external contamination.

3.5.3 Construction should be carried out for nuclear equipment and components as per approved procedures, including requirements for safe handling and storage and use of approved consumables. Inventory records should be maintained for assuring clean system conditions on assembly and installation. A QA programme should be instituted to include the requirements for quality assurance checks, qualified personnel, procedures and documentation which should form part of records to be handed over to the commissioning group and for archival purposes.

3.6 Commissioning and Pre-operational Data Collection

During commissioning, base line data should be collected for comparison and trending during operation for detecting degradation due to ageing. Typical data are motor starting currents, acceleration timings, system fill up rates and timings, shut-off rod drop timings, mechanical clearances of wearing parts, setting of supports for equipment and piping, PSI data of coolant channels, feeders, elbows, steam generators, heavy water heat exchangers, strain deflection and leakage rates in containment structures, vibration of equipment, bearing temperatures, insulation resistance of electrical equipment and cables etc. Benchmark should be established for monitoring of foundation settlement for safety related structures and turbine-generator. Comparison over a period of time will provide useful information for understanding the progressive behavior and ageing of the SSC.

4. LIFE MANAGEMENT DURING OPERATION

4.1 General

4.1.1 The longest period in the life of NPP is the operating phase (30 to 60 years including extended life). During this phase, optimising of all activities based on ageing management studies is essential for achieving safety in operation.

4.2 Screening of Structures, Systems and Components Important to Safety

The SSC should be screened from ageing consideration as per procedure described in section 2.5.2 These SSC are further classified into the following categories. Typical categorisation for PHWR is given below (Refer Annexure-VII for general list of SSCs for BWRs).

4.2.1 Category-1 SSCs

Category-1 SSC, are normally not replaceable and therefore they limit the life of an NPP. They are designed to retain the required safety margin during lifetime, considering degradation caused by ageing, operational occurrences and design basis accident conditions, if any. Specific surveillance and periodic assessment of residual life is carried out considering operating experience, current knowledge on degradation mechanisms, international feedback, and pilot studies on degradation mechanisms and monitoring systems.

Examples: Calandria, end shields, primary coolant system inlet/outlet headers, moderator system piping and supports located in calandria vault, civil structure and liner of calandria vault.

4.2.2 Category-2 SSCs

Category-2 SSCs have limited accessibility and are difficult to replace due to radiation exposure and /or require long shutdown period. Condition monitoring, trending, preventive maintenance and ISI are possible to manage ageing effects.

Examples: Steam generators, coolant channels and primary coolant feeders.

4.2.3 Category-3 SSCs

For these SSCs preventive maintenance, ISI and condition monitoring is possible to manage ageing. They are replaced/repaired in a planned manner during operating phase.

Examples: Pressurising and fuelling machine supply pumps, fuelling machines, reactivity mechanisms, adjuster rods, shut-off rods, containment isolation dampers, steam discharge and steam safety relief valves, fire and emergency cooling water valves and pumps, diesel generator sets, class I power and control batteries.

4.3 Prioritisation of Safety Issues

In formulating the life management programme priorities have to be assigned based on operating experience on ageing and premature failures. An effective life management programme depends on the ability to detect degradation and initiate mitigating measures for maintaining functional capability without compromising safety margins as per design. Existing information on deterministic analysis/PSA and engineering judgement etc. will aid in prioritisation process. According to safety significance, safety issues are prioritised as high, medium and low.

4.3.1 High Priority Safety Issues

High priority safety issues have unacceptable impact on plant safety and on defense in depth. They need immediate corrective measures. The plant is required to be shutdown till interim or permanent corrective measures are implemented. The highest priority is given to those measures, which will achieve or restore defense in depth features that were applicable at the time of first licensing.

4.3.2 Medium Priority Safety Issues

Medium priority safety issues have significant impact on safety where defense in depth is degraded. After implementing interim corrective measures, plant operation may continue for limited time based on risk assessment. Permanent measures should be planned and implemented within specified time frame.

4.3.3 Low Priority Safety Issues

Low priority safety issues have low impact on plant safety. Plant operation can continue without any interim measures. Corrective measures may be implemented within a stipulated period.

4.4 Condition Monitoring and In-Service Inspection

4.4.1 Systematic data collection and analysis give basic information regarding degradation of plant equipment and structures. Off-normal operation, power transients, unit outages, failure/defect of equipment, human errors etc. may stress SSC more than in normal operation. Effects of various operating conditions and degradation mechanisms on SSC are studied. On the basis of such assessment, specified conditions of components are monitored to determine the degradation in safety margin of components. Systems for measurement of such conditions are condition monitors. Some examples of condition monitors are shut off rods drop timing, primary coolant temperature cycles measurement, vibration monitoring of rotating equipment, dissolved gas analysis of transformer oil etc. Annexure-VIII and Annexure-IX give examples of condition monitors for various SSC for PHWR and BWR respectively.

4.4.2 Surveillance of items important to safety is carried out as per Safety Guide on 'Surveillance of Items Important to Safety in Nuclear Power Plants (AERB/SG/O-8). All IIS are tested for their specified functional capability periodically.

4.4.3 In-service inspection of SSC is carried out as per safety guide on 'In-Service-Inspection of Nuclear Power Plants' (AERB/NPP/ SG /O-2). ISI is performed on equipment / components to monitor degradation or to locate failure by non destructive examination (NDE). In service inspection and review of test reports provide information regarding effects of ageing on SSC. NDE of passive SSC will yield directly applicable results for assessing residual service life. For SSC important to safety, provisions should be made for locating test samples at specific areas. Post irradiation examination (PIE) for radioactive component gives information for managing important SSC.

4.4.4 Provisions for the following condition monitoring techniques may be considered:

(i) Radiation embrittlement.

(ii) Thermal embrittlement.

(iii) Fatigue.

(iv) Loose part detection.

(v) Acoustic leak detection.

(vi) Crack growth.

(vii) Power/control cable condition.

(viii) Online diagnostic vibration analysis.

(ix) Strain.

(x) Settlement of containment.

4.5 Measures to Mitigate Ageing Effects

4.5.1 Control of Operating Parameters

4.5.1.1 Operating parameters are required to be maintained within operational limits and conditions as per technical specification and other approved O & M procedures to minimise avoidable degradation of plant equipment.

4.5.1.2 Reactor trips cause pressure and temperature transients and affect systems and equipment. It is therefore essential to reduce number of plant trips by improving system response to operational transients, use of proven design equipment and error-free operation. Lessons learnt from operating experience feedback and from external events are important sources of information for this purpose. Wide grid frequency variation has deleterious effect on plant equipment particularly turbine blades.

4.5.1.3 Fluid flow control: Coolant flow is restricted to avoid erosion at feeder elbows and vibration in heat exchanger tubes and pipes. High coolant flow rates in pressure tubes may induce fuel fretting and shifting of loose garter springs, which affect pressure tube life. High shell side flow in heat exchangers at entry damages heat exchanger tubes by fretting at baffles. Lack of service cooling water flow results in corrosion due to stagnant water for poised systems.

4.5.1.4 The maintenance of chemistry in various systems, within specified design limits during operation and shut down is necessary from LM considerations. Limits of chemical parameters are delineated in design and technical specifications of the plant. They have to be maintained within limits by proper and timely chemical control. Noncompliance may result in excursions such as high hydrogen in moderator cover gas, D_2/H_2 pick-up in pressure tubes, contamination of primary coolant system resulting in buildup of radiation fields, high conductivity in feed water system affecting steam generator tubes etc.

4.5.1.5 Containment environment control: Pump room and fuelling machine vaults may experience high temperature and high humidity. They directly degrade instrument systems, electrical equipment, and control and power cables. Leakage control by maintenance, effective operation of dryers and coolers and air flow balancing are a few of the mitigating measures.

4.5.1.6 Fuel cladding failure results in increase in radiation field in operational areas, contamination of system/components and increase in radiation exposure to O & M personnel. For mitigating such effects observance of quality assurance measures in fuel fabrication and handling, limiting channel power and reactor power ramps as per specified technical limits, are essential.

4.5.2 Maintenance Programme for Managing Ageing

Maintenance of an NPP is covered in Safety Guide on 'Maintenance of Nuclear Power Plants' (AERB/SG/O-7). For life management programme, planning of maintenance, equipment/component modification, failure analysis and determination of residual life for SSC need to be considered.

4.5.2.1 Maintenance Planning

Maintenance planning for SSC should include the following:

(i) Listing of SSC and categorisation as per section 4. 2 above.

(ii) Maintenance procedure with QA checks.

(iii) Procedure for post maintenance testing and acceptance criteria.

(iv) Training of staff for maintenance works.

(v) Identify causes of degradation (due to wear, stress, corrosion, thinning, fretting, irradiation and thermal cycling).

(vi) Identify requirement for replacement of SSC based on ageing degradation and modify preventive maintenance and ISI schedule.

(vii) Identification of common mode failure/degradation.

(viii) Identification of degradation of fire barriers.

(ix) Root cause analysis of failures.

4.5.2.2 Maintenance Issues related to Life Management

4.5.2.2.1A degraded SSC could fail simultaneously during transient/accident conditions. Such a failure may occur due to following:

(i) Sharing common environment (pressure, temperature, humidity, chemistry radiation and vibration).

(ii) Conceptual or engineering error.

(iii) Manufacturing/fabrication error.

(iv) Installation error (non-observance of approved procedures).

(v) Service ageing/degradation and lack of timely maintenance.

(vi) Procedural error in operation and maintenance.

4.5.2.2.2 Identification of critical component in equipment assembly is required. (Critical component is a part or subassembly, which is subjected to enhanced degradation and has shorter service life).

4.5.2.2.3 Elastomers need timely replacement and hence should be subjected to special surveillance. It is desirable not to use elastomers in equipment subjected to high temperature.

4.5 2.2.4 Material incompatible for the intended service conditions should be identified for early rectification.

4.5.2.2.5 Safety related SSC operating at elevated temperature and/or pressure and subjected to thermal and stress cycles would have limited life based on their fatigue life limits. Such SSC require monitoring and repair or replacement when necessary to ensure their operating margin. Maintenance and ISI schedule for such SSC should be reviewed at specified intervals for monitoring age and service related degradation for timely and appropriate action.

4.5.2 2 6 Equipment subjected to over-speed experience high stresses. Records of such events should be maintained and vendor's recommendations in this regard should be followed. Experience feedback on equipment failures, their causes and corrective actions taken should be shared within operating organisation and other power plants.

4.5.2 2 7 For age related degradation of concrete structures, Refer 'Maintenance of Civil Engineering Structures Important to Safety of NPP' AERB/SM/CSE-1, and Annexure-1B of this guide.

4.5.2.3 Failure Analysis

4.5.2.3.1 The cause of failure and statistical analysis of failure rate are reviewed. It is observed that time dependent failure rate is as indicated by well known Bathtub curve. High failure rate observed during initial operating phase, called "Burn-in period", could be due to teething problems in some equipment. Selecting proven equipment and proper pre-commissioning tests can minimise such failures during normal operation.

4.5.2.3.2 Systematic analysis of failure data including generic failure data is required for determining the severity, modes and root causes of failures. In addition to wear due to normal operation, fatigue cycles and cumulative stresses, following factors occurring frequently enhance degradation.

(a) Heating and cooling cycles.

(b) Transient / abnormal operating cycles.

(c) Frequency of operator / maintenance errors.

(d) Change in electrical power supply frequency.

(e) Over-speed test on turbine generator sets.

(f) Load throw off events causing flow transients across steam generator tube bundle.

(g) Total loss of off-site power incidents.

(h) Crash-cooling events.

(i) Cold pressurisation.

(j) Water hammering.

(k) Corrosive environment.

(l) Poor ventilation.

(m) High radiation field beyond design value.

(n) Excessive surveillance testing and maintenance.

4.5.2.3.3 Components exhibit increasing failure rates in later part of their life, characterised as the "wear-out" period. Components are replaced before wear out phase, as otherwise increase in failure rate results in:

(a) Decrease in reliability of component.

(b) Increase in initiating events causing system transients.

(c) Decrease in defense in depth due to possible concurrent failures.

4.5.2.4 Determination of Residual Life

4.5.2.4. Residual life and fitness for service can be assessed by fracture mechanic techniques and ISI inputs. Fracture mechanics assumes that all materials contain flaws. The inspection of materials and compononents aims to locate and characterise flaws and degradation in material properties that would lead to failure. The fracture mechanics technique enables to predict whether a crack of given size and characteristic would fail under a particular load if material properties like fracture toughness are known. Once the flaw is sized and load and environment are established, then the residual life is determined by fracture mechanics evaluation. Fracture mechanics thus integrates the material properties, design stresses and flaw size and if the detected crack size is smaller than critical size and grows slowly then it may be left in service and inspected at specific interval based on engineering review.

4.5.2.4.2 High pressure and /or high temperature equipment are designed as per ASME or other applicable codes. Stresses, fatigue and creep damage fractions are calculated taking into account the mechanical and thermal loading and transients. For the purpose of residual life determination, actual fatigue usage factor is to be calculated considering the service history. Systematic ISI involving measurement of thickness, as well as surface and volumetric examination at pre-decided intervals provide basis for life assessment.

4.5.2.4.3 Non-replaceable SSC are designed with conservative values of mechanical properties so that, they will have adequate residual life at the end of their design service life. Determination of residual life requires assessment of the following:

(i) Calculation of design life based on current codes and standards.

(ii) Changes in dimensions, ductility, fatigue and mechanical strength due to operational, environmental and unanticipated stresses (operation beyond design limit).

(iii) Size and orientation of flaw detected.

4.5.2.5 Modification

4.5.2.5.1 The design of an NPP undergoes improvement in safety system based on operational experience feedback. This results in modification in operating plants for upgrading the status of the plant for meeting current safety standards. (Refer Annexure-X).

4.5.2.5.2 Implementation stages for modification include:

(i) Identification of modification requirements.

(ii) Engineering of modification proposal and approval.

(iii) Prior approval from regulatory body for safety related SSC.

(iv) Schedule for supply of equipment/component and implementation of modification, including procedures for configuration management.

(v) Preparation of document on modification carried out on SSC.

(vi) Submission of documents to regulatory body.

(vii) Revision of technical specification.

(viii) Updating of drawings, procedures for operation, maintenance, surveillance and ISI requirements.

(ix) Training of staff on modification.

4.5.3 Chemical Cleaning

4.5.3.1 Chemical cleaning during pre-operational phase

(a) Hot conditioning is carried out for primary coolant system after light water flushing and installation of test coupons, as per the approved procedure to achieve 30 to 40 micron oxide layer as ascertained by test coupons.

(b) Flushing and chemical cleaning of condensate and feed water systems is carried out to clean up the system and remove loose scale during commissioning.

4.5.3.2 Chemical cleaning during operating phase

(i) Decontamination of nuclear systems equipment and components is carried out to gain access during planned maintenance. During this phase decontamination is carried out to remove contaminants due to fission and activation products from the inner surface.

(ii) Condenser and heat exchanger tube cleaning:

Thermal performance of various heat exchangers is monitored and trended. Condenser tube fouling is the major degrading factor requiring chemical cleaning. Online ball cleaning and chemical dosing of condenser cooling water is carried out for closed loop condenser cooling water system. During shut down, high pressure water lancing and chemical cleaning is also resorted to. The chemical waste is treated before discharge to prevent adverse environmental impact.

(iii) Steam generator shell side chemical cleaning/hot water soaking and flushing should be carried out to remove accumulated crud/ sedimentation from crevices and tube-sheets and corrosion product on outside surface of tubes and baffles to control degradation due to denting and corrosion; and to improve thermal performance. Wherever provisions exist, sludge lancing should be carried out periodically to remove accumulated sludge on tube sheet of steam-generators.

4.6 License Renewal

(Refer Annexure-I and Annexure-XII)

4.6.1 General

During the operating phase of an NPP, degradation of SSC occurs due to passage of time and service. The integrity and the functional capabilities of SSC should not degrade below the values considered during the design. New features may have been incorporated in the plant based on operational safety experience, research and development and advancement in knowledge. The design of the plant may not meet the current safety standards at the end of design life. Its continued acceptance for further operation has to be assessed consistent with the regulatory requirements. The SSC of NPP may have residual life at end of license period. Operating organisation may examine safety and economic feasibility of extending operating phase.

4.6.1 Consideration for License Renewal

(i) Safety margin initially provided in the design is affected during service life and passage of time. Safety standards are revised based on operating experience feedback and technology development.

(ii) During the license period of operation, safety level achieved, should be equal to the design target envisaged. During the life extension period it is expected that the safety level would be higher than that, which was envisaged during the original design due to continuous upgradation effected during the licensed life of the station. But under no circumstances the safety level should be lower than the original design intent.

(iii) Operating organisation should review residual life of SSC including balance of plant systems and equipment based on Periodic Safety Reviews (PSR) as per safety guide on "Renewal of Authorisation for Operation of Nuclear Power Plants" (AERB/SG/O-12) and mitigation measures instituted. It also takes into account operating history, conclusions of ISI and ageing studies carried out. Available data of similar plants of the same vintage should be compiled for comparison.

(iv) Adequacy and effectiveness of the technical management of the plant, safety culture, administrative controls and operating organisation's response to unusual and abnormal events are reviewed.

(v) Safety related civil engineering structures should be reassessed for ageing degradation/structural stability.

(vi) Power and control cables should be reassessed for ageing degradation and replaced conforming to current standards, if required.

(vii) Site characteristic, as prevalent at the time of site selection should be compared with current understanding. Change in population density, industrial and other activities within the emergency planning zone (EPZ), construction of public works such as water reservoirs, reclassification of seismic zone, change in course of rivers, cyclonic activities and rainfall and flood in area near the NPP, should be reassessed.

(viii) Environmental burden should be reassessed for continued radioactive discharges in the public domain and the ability of site to bear the additional waste management due to extended life of the NPP and other nuclear facilities.

(ix) Consideration for life extension could be based on Probabilistic Safety Assessment (PSA) complemented with deterministic judgment for both individual and collective issues for assuring compliance with safety requirements.

4.6.3 Method of Assessment

4.6.3.1 Periodic Review for Life Management (Refer Appendix-9)

4.6.3.1.1 Apart from continuous review, a detailed periodic review should be made at predetermined intervals (say ten years). All data gathered during previous years of operation should be carefully scrutinized to check capability to operate within technical specifications limits, adequacy of safety culture of plant, personnel and management response.

4.6.3.1.2 This periodic review should include the following.

(a) Corrective actions are taken for area of improvements in various systems, surfaced during PSR.

(b) The ageing management study results and mitigating measures for ageing are implemented.

(c) Appropriate actions are considered for any external threats and internal hazards, which were not considered at design stage.

(d) Corrective immediate actions have been taken when defense in depth is threatened.

(e) Plant control system and operator response were adequate to manage transients and abnormal occurrences.

(f) Operational experience feed back of similar plants considered.

(g) Major maintenance works and ISI carried out satisfactorily.

(h) Peer reviews were carried out and results implemented.

4.6.3.2 In-Depth Review for License Renewal (Refer Appendix-10).

4.6.3.2.1 For life extension beyond initial license period (30 to 40 years), it is imperative that a comprehensive in-depth review of design and operational aspects of the plant should be performed with reference to current safety standards to ascertain non-conformance to safety requirements. Safety issues should be identified based on acceptance criteria from the Regulatory Body. There are two important and complementary tools as given below that together can help in the exercise of judgment.

(i) Deterministic Safety Analysis Methods

Deterministic judgement process for safety issues apply internationally accepted safety rules and safety standards, derived from concept of defense in depth. As the complexity and the number of issues increase, this process relies on a systematic categorisation based on

(a) Frequency of occurrence of events.

(b) Potential consequences if the safety functions are impaired.

(c) Existing capabilities of safety function(s) affected by the safety issue.

This concept emphasises redundant means to ensure the performance of the primary safety functions controlling the power, cooling the fuel and confining the radioactive material. The effectiveness of primary safety function performance is considered to be an appropriate integrated approach for evaluating levels of defense affected by safety issues.

(ii) Probabilistic Safety Analysis Methods

This takes into account the initiating events that might lead to accidents considering probability of failure of safety systems/safety related systems intended to prevent/mitigate the consequencess of accidents. PSA relies on deterministic analysis to specify limiting conditions for adequate performance of a particular safety system. Probabilistic methods provide an insight into the relative importance of different features of systems. PSA needs data from operating experience. The comprehensiveness of the plant data collection programme and acceptability and accuracy of data are equally important to both probabilistic and deterministic methods.

4.6.3.2.2 The effect of external hazards on safety should be considered in both deterministic and probabilistic methods. One critical aspect of external hazards to be considered is their potential to induce common cause failures.

4.6.3.2.3 In the review path it is necessary to assess contribution of each deterministic deficiency which might have on plant damage and or external radioactive release.

4.6.3.2.4 Level-1 PSA identifies accident sequences that can lead to core damage and quantifies its frequency. It identifies dominant contributions to core damage and provides insight into the weakness/strength in plant design and operation and ways of preventing core damage. Level-2 PSA is carried out taking inputs from Level-1 PSA to quantify the magnitude and frequency of radioactive release to the environment, following core damage projection and containment response. In case this could not be carried out, due to lack of the state of the art methodology and software, atleast qualitative assessment of the containment based on design data and review of its performance for the severity of core damage should be made to evaluate magnitude and likelihood of radioactive release outside the containment.

4.6.3.2.5 During the review of defense in depth in a plant, balanced view should be taken between the earlier safety standards (based on which the original design was made) and the current safety standards. Many older plants may not conform to all the current criteria and standards for design. This does not necessarily make them unsafe in part owing to the earlier conservative margins, and in view of the measures taken to meet current safety requirements in spirit by upgrading/augmenting safety systems and operation and maintenance procedures.

4.6 3.2 6 In the seismic design area, current seismic standards which have evolved through the application of seismic hazard methods, seismic structural technology and better component fragility data can not be imposed retrospectively through a rigid design process, because such an approach would require many changes to the plant systems and structures which may be impractical to implement. However, insights from modern seismic technology and from a study of performance of equipment and structures in actual earthquakes can be applied to older plants and can provide substantial improvements in safety through 'seismic margins' assessment process.

4.6.3.2.7 The in-depth review of the results of the assessment will indicate the weakness of systems/subsystems and urgency of corrective measures.

4.6.4 Acceptance Criteria

4.6.4.1 General

If the safety level falls below the original design intent safety level or external hazard perception changes for the worse, the Regulatory Body may withdraw the license to operate and the plant shall be shut down. In the case of life extension, the plant reference safety is expected to be higher during the extended period of operation than that of original design. Acceptance criteria for life extension need to be evolved by the responsible organisation in consultation with the Regulatory Body. The issues are often plant specific and the key issues in connection with development of acceptance criteria are:

(a) Compliance with the current safety principles and practices.

(b) Safety goals, as in original design or higher due to various improvements implemented.

(c) Adequacy of residual life of SSC important to safety.

(d) Identification of systems requiring upgradation.

(e) Safety culture.

4.6.4.2 Safety Goals during Renewal of License

As mentioned in section 4.6.3.2 (on in depth review) measures for improvement of safety goals should be identified and their implementation for achieving higher safety goals should be undertaken. The plant reference safety level should improve after effecting modifications.

4.6.4.3 Ageing Management Programme (AMP)

Existence of an effective ageing management programme will be an important consideration for license renewal.

4.6.5 Achieving Acceptable Levels of Safety

4.6.5.1 The starting point of plant assessment is the in-house assessment followed by periodic reviews. It delineates the safety culture and the alacrity with which the responsible organisation, operating organisation and the plant management had addressed the safety issues and upgraded the safety goals over the years.

4.6.5.2 Atleast five years before the end of the licensed life, in-depth-life extension review should be done as mentioned in section 4.6.3.2. Based on the acceptance criteria and other safety considerations, deficiencies are identified and prioritised based on relative importance as given in the Annexure-XIII. Critical deficiencies, which could lead to significant likelihood of an accident, are given higher priority. Some of them are as follows:

(a) Any situation in which accidents considered in the original design basis would not be coped with, adequately.

(b) Deficiencies that could lead to failures which would not be coped with adequately, such as major deficiencies in the primary pressure boundary leading to an accident beyond the design basis.

(c) Inadequate shutdown capability or inadequate decay heat removal during normal operation and anticipated operational occurrences, including outages and abnormal events such as fire, flooding or complete loss of electrical power, taking into account single failure criteria.

(d) Inadequate containment or confinement capability such that credible failures or sequences of failures (includes single failure criteria) that can not reasonably be excluded on probabilistic or deterministic grounds could give rise to a large external release of radioactivity requiring significant emergency counter measures.

(e) Severe deficiencies in the conduct of operations.

4.6.5.3 All deficiencies in the high and medium priority safety issues (Annexure-XIII) should be rectified. In case of deficiencies, which could not be compensated by other means and are confirmed to be unacceptable by Regulatory Body, the plant shall be shut down. On the other hand, if the review does not identify critical deficiencies, the proposal for license renewal should be got approved by the regulating body.

4.6.6 Infrastructure Support

4.6.6.1 Sufficient staff with adequate competence and training should be provided. Retraining at periodic intervals as necessary for updating their knowledge base should be provided. Safety culture should be all-pervasive.

4.6.6.2 Simulator training and retraining shall be provided to all licensed/authorised operating personnel.

4.6.6.3 A catalogue of all failures and mistakes should be candidly recorded, read and understood by the operating personnel. Failures often repeat themselves. So knowledge about past failures and actions taken will go a long way to handle similar situations in the future.

5. ORGANISATIONAL ASPECTS OF LIFE MANAGEMENT

5.1 Organisation for Life Management

5.1.1 The life management programme has to be conceived and progressively implemented from site selection to the end of operating phase of the NPP. Concurrent with implementing a nuclear power programme, a systematic approach for accessing world wide experience in siting, design, construction and operation of NPPs, combined with its own experience, will provide the operating organisation, the means for achieving safe and economic operation, through its life, conforming to current safety and regulatory requirements. For achieving the above objectives, it is necessary to establish a proactive, diligent and cohesive management structure for the life management of an NPP.

5.1.2 The responsible organisation (RO) should promote and proactively participate in establishing a life management programme for the NPP. In this respect RO should promote and encourage safety culture amongst its staff. Knowledge and experience should be systematically disseminated through structured training programme for active participation of its personnel in the activities related to LM.

5.1.3 For effective implementation of the programme, it is essential to assign the responsibility to a dedicated group, and aided as required by inter-disciplinary experts groups of different disciplines. Accordingly RO should consider the following:

(i) Develop detailed procedures and methodology for life management activities.

(ii) Organise categorisation of SSC and of data for life management.

(iii) Arrange for systematic monitoring, processing, collection and collation of data from all NPPs.

(iv) Collate relevant international experience.

(v) Organise data evaluation and action plans.

(vi) Follow up measures to mitigate ageing effects.

(vii) Initiate research and development programme for type testing and development of special procedures.

(viii) Organise engineering consultancy services and manufacturers' participation.

5.1.4 Plant management is responsible for implementing action plans for life management at plant. Before implementing the action plans, the plant management should review and evaluate the plans for their feasibility and effectiveness for implementation. Modifications to action plans, should be carried in consultation with life management dedicated group. Plant management should provide feedback and data on action plans implemented.

5.2 Human Aspects and Safety Culture

5.2.1 The ultimate success of life management of an NPP depends on degree of understanding, acceptance and support of staff of the nuclear power plant. It is necessary to have awareness among the plant personnel about the usefulness and importance of life management. The plant personnel should feel a sense of ownership and belonging for the area of plant under their control. This approach greatly influences operation and safety of the plant.

Plant personnel need training in following aspects:

(a) Understanding of ageing and degradation mechanisms, environmental stressors and degradation of various equipment/components.

(ii) Importance of various limits and conditions of operation.

(iii) Understanding precursors as early warning of ageing phenomenon and sensitivity to its indication.

(iv) Prompt recognition and reporting of deviations that could indicate onset of degradation.

(v) Accurate and standardised data collection and documentation by plant personnel for subsequent analysis.

(vi) Developing competence of personnel to carry out assigned responsibilities.

5.2.2 The paramount importance of safety in all plant operations i. e. safety culture should be inculcated right from the inception of plant among all personnel. Plant personnel should be encouraged to have a positive work culture, and timely resolution of deficiencies. They must observe diligence in all operation for containing radioactivity within barriers and observe conservatism in decision making. Plant performance is compared with the performance of similar plants to strive for excellence.

5.2.3 Organisations should maintain sufficient number of skilled and qualified staff with adequate engineering and analytical background. It should also take care of organisational changes to ensure placement of competent people for respective assignments. All plant personnel should have medical check-up for physical and psychological fitness.

5. 3 Infrastructure Development

Infrastructure development should address the following:

(i) Most of the equipment and components of an NPP operate in high radiation environment, requiring development of remotely operated tools and equipment with high degree of reliability for their maintenance.

(ii) Test facilities are required to be set up for testing and qualifying equipment, components and materials used in the plant.

(iii) Research and development is required for:

(a) Material/component testing for understanding degradation mechanisms under mechanical loading, thermal stress and radiation fields.

(b) Post irradiation examination is required for microstructure evaluation and monitoring of oxide layer and hydride blisters by hydrogen estimation in samples.

(c) Development of ultrasonic, eddy current and acoustic emission testing techniques and tooling and associated computer software for coolant channel, calandria tubes and steam generator tube inspection. Development of laboratory techniques for H_2/D_2 measurement in sliver scrape samples.

(d) Tooling for repositioning of garter springs and replacement of coolant tubes.

5. 4 Review of Life Management Programme

5.4.1 Responsibilities

RO should establish suitable mechanisms, for overseeing the conduct of the programme and achieving its objectives. The mechanisms set out, should enable and ensure that the latest developments in the field get addressed and as appropriate, included in the various programmes. Provisions to detect, evaluate and mitigate the effects of unanticipated ageing mechanisms should be an integral part of the ageing management programmes

5.4.2 Review

It is necessary to review all the programmes and the results achieved, at periodic intervals for assessing and assuring their effectiveness. The review process may be categorised into three complementary types and will include self-assessment, peer review and periodic review. They differ in the independence of the review team, the degree of formality and the rigour. It is necessary to review all the programmes, and the results achieved, at periodic intervals. They provide, at one end of the spectrum, short-term managerial, operational and quality control aspects and at the other end, a comprehensive review of the effectiveness of the programmes for ensuring the material fitness of the plant for continued operation.

5.4.2.1 Self Assessment

Self assessment should be carried out annually by the plant management. Organisational arrangements and procedures set up for implementing the life management programme, should be assessed for its effectiveness in coordinating all relevant existing plant and external programmes and activities including:

(i) Promotion of LM programme.

(ii) Whether all SSC requiring AMP are covered.

(iii) Adequacy in the identification and delineation of SSC and or degradation mechanisms requiring ageing evaluations.

(iv) Adequacy of ageing evaluations and organisation for specialised expertise.

(v) Availability and usage of internal and external services for setting standards.

(vi) Implementing ageing management decisions.

(vii) Assessment of the effectiveness of the AMP and verifying whether actions /mechanisms set up meet the objectives of the AMP.

(viii) Availability of up-dated performance/ageing related technical information, and effective usage of such information.

(ix) Organisational lacunae, if any, in the functioning of LM/AMP unit.

(x) Technical competence and organisational ability of the LM/AMP unit, for co-ordinating and managing implementation of recommendations after assessment and review in close co-ordination with the plant management.

(xi) Adequacy of procedures/mechanisms for resolving any disagreements in implementing AMP team's recommendations.

(xii) Adequacy of R&D inputs.

5.4.2.2 Peer Review

Peer review of relevant existing programmes should be conducted at least once in five years, for identifying areas for improvement. The peer review team should consist of experts in technical support, operations, maintenance, chemistry, planning, training and administration for determining whether these programmes are in accordance with accepted standards. The review should

should include all the programmes and all aspects, important to ageing management. The review should cover documentation and the efficacy of document management systems for recording of plant performance including transients, event reports and performance data. An audit should be conducted through a series of direct observations for identifying both programme strengths and weaknesses. The review results should be formally reported to the plant management for appropriate action.

5.4.2.3 Periodic Review

5.4.2.3.1 The objective of a periodic review is to assess all programmes comprehensively once in ten years by the responsible organisation. It is carried out for determining whether the management of ageing in the NPP is effective and the required integrity and the functional capability of SSC are maintained and whether an adequate AMP is in place for future plant operation.

The following should be examined:

(i) General attributes of the AMP (e.g. programmatic aspects, such as policy, resources, procedures and records).

(ii) Scope of the AMP (e. g. screening methodology and SSC covered by the AMP).

(iii) Quality of programmes for ageing management of specific SSC (e.g. the degree of understanding of the ageing of SSC and the effectiveness of established detection and mitigation programmes).

(iv) Results achieved (e. g. the actual physical condition of SSC equipment qualification status).

5.4.2.3.2 The review should take into account the results of all existing programmes and activities, which are relevant to the AMP at the NPP, AMP self-assessments and peer reviews of relevant programmes. The review should be carried out in three major steps:

(i) Assessment of the current ageing management status:

Information on the AMP is reviewed. A list of deviations from the current requirements is then documented in a report. All programme strengths and deficiencies should be clearly identified.

(ii) Interim safety review:

In this step, each deficiency identified in the earlier assessment as above, is reviewed for determining its safety significance and for appropriate corrective action(s), using existing information and expert judgement, rather than detailed analysis, and is documented in a report. In cases where safety significance is high, immediate remedial actions should be implemented.

(iii) In-depth safety review:

In this step, the safety significance of individual deficiencies and the adequacy of corrective actions from the interim review should be verified. If the in-depth analysis shows that the actions resulting from the interim review are inadequate, the feasibility of other corrective actions should then be assessed. The acceptability of continued plant operation is then assessed, taking account of the remaining deficiencies.

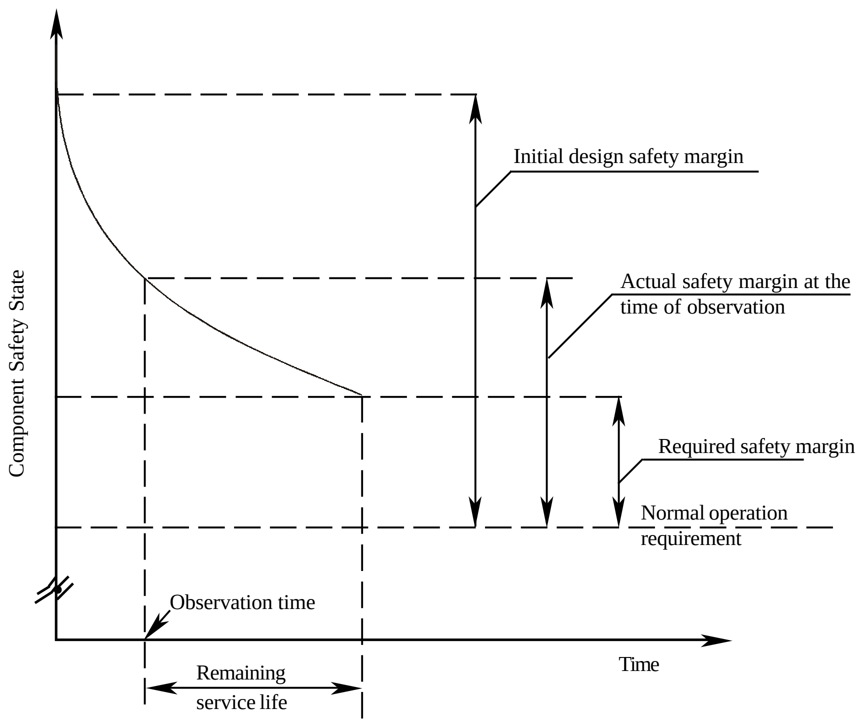
5.4.3 Regulatory Review

Results of the periodic reviews of the programmes instituted by the NPP management are evaluated by the regulatory body for verifying that plant/ equipment ageing is being efficiently managed and that effective programmes are in place for continued safety in operation of NPP. The programmes should also be assessed for their adequacy and effectiveness for timely detection and mitigation of ageing degradation in order to ensure that the required safety margins i.e., the integrity and functional capability of the SSC are maintained.

APPENDIX-1

(Refer Section 1.1.2)

SAFETY STATE OF COMPONENTS AND SAFETY MARGIN AS FUNCTION OF TIME



APPENDIX-2

(Refer Section 2.2.3)

CURRENT STANDARDS AND PRACTICES

Before a safety assessment is performed, the operating organisation and the regulatory body have to agree upon the safety standards, methods and practices to be applied. The safety assessment should be based on a comparison of the actual plant status, including all changes made so far, with current safety standards and operational practices. It needs to be pointed out, however, that non-compliance with these standards does not necessarily mean that the plant built to earlier standards is considered unsafe.

However, standards, methods and practices are evolving as a result of:

- research and technological development.
- operational experience.
- assessments of incidents and accidents.
- change in public opinion.
- extension of safety considerations and protective measures into the beyond design basis area.

The agreement between the regulatory body and the operating organisation on standards, methods and practices needs to be based on a clear understanding as to the reasons for the requirements, the rationale for assessing safety significance and the applicability to the specific plant built to earlier standards. The fact that current standards have usually not been developed to resolve safety issues for an existing plant built to earlier standards must be taken into account. Issues for plants built to earlier standards are usually corrected by a case by case treatment. Changes to codes and standards are usually based on optimised solutions for future design.

The revised Nuclear Safety Standards (NUSS) of IAEA covering governmental organisations, siting, design, operation and quality assurance and the set of NUSS supplementary safety guides constitute an internationally accepted frame of reference for the safety of nuclear power plants built to earlier standards. The Indian position on current standards is similar to above international position. For India, the current standards are defined as those described in the latest AERB safety documents, i. e. AERB codes, standards, guides and manuals, IAEA NUSS codes and guides which do not have corresponding AERB safety documents also need to be included while working out the current standards. Similarly latest documents from International Organisation for Standardisation (ISO), International Nuclear Safety Advisory Group (INSAG), American Society of Mechanical Engineers (ASME), Institute of Electrical and Electronics Engineers (IEEE) and from American Society of Civil Engineers (ASCE) not covered by AERB safety documents, may also be consulted.

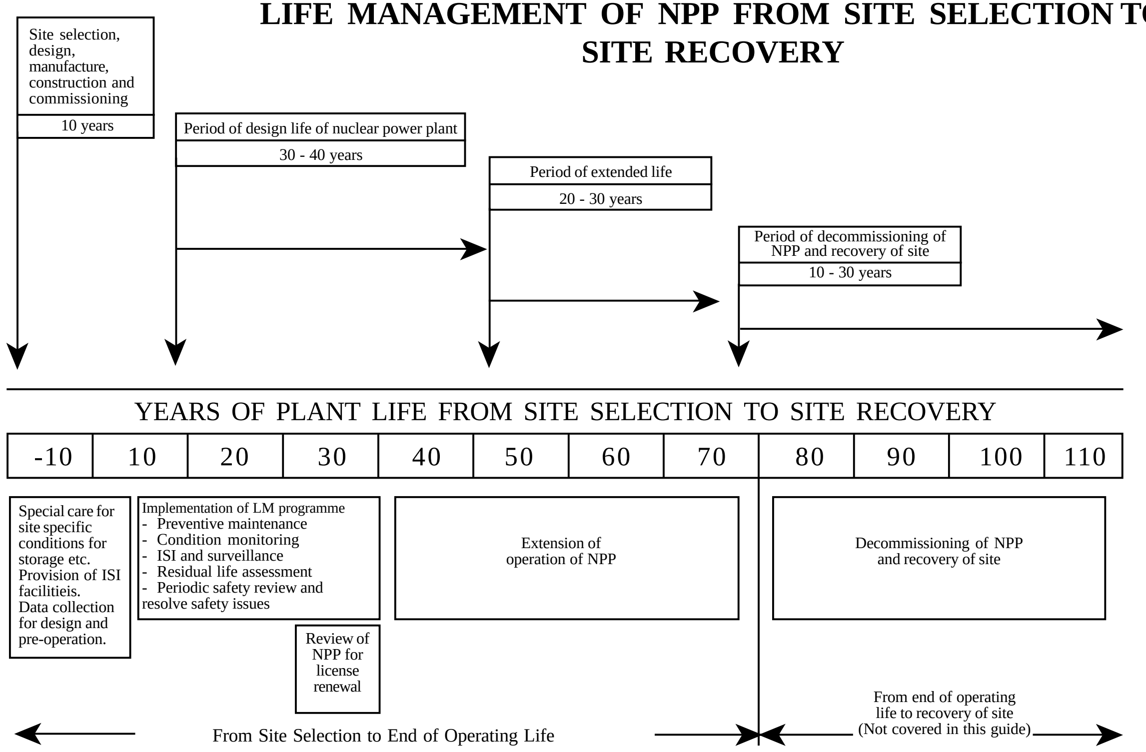
In the event that the national standards are upgraded over those used for the original licensing, the plant operator may agree with the regulator to use the upgraded standards in conjunction with analysis methods consistent with the original licensing, unless the upgraded standards are necessary for safety.

Where the plant design is based on imported technology and experience, it may be appropriate to consider applying the standards of the country of origin. The decision to adopt another country's national standards for the purpose of reactor safety needs to be based on an evaluation of their applicability and confined by ensuring that they are consistent with the IAEA NUSS codes and guides and the basic safety principles of INSAG-12 to the extent that those standards are applicable. One approach for comparison with current standards can be a comparison with the plant that has been built to current standards.

APPENDIX-3

(Refer Section 2.3.1)

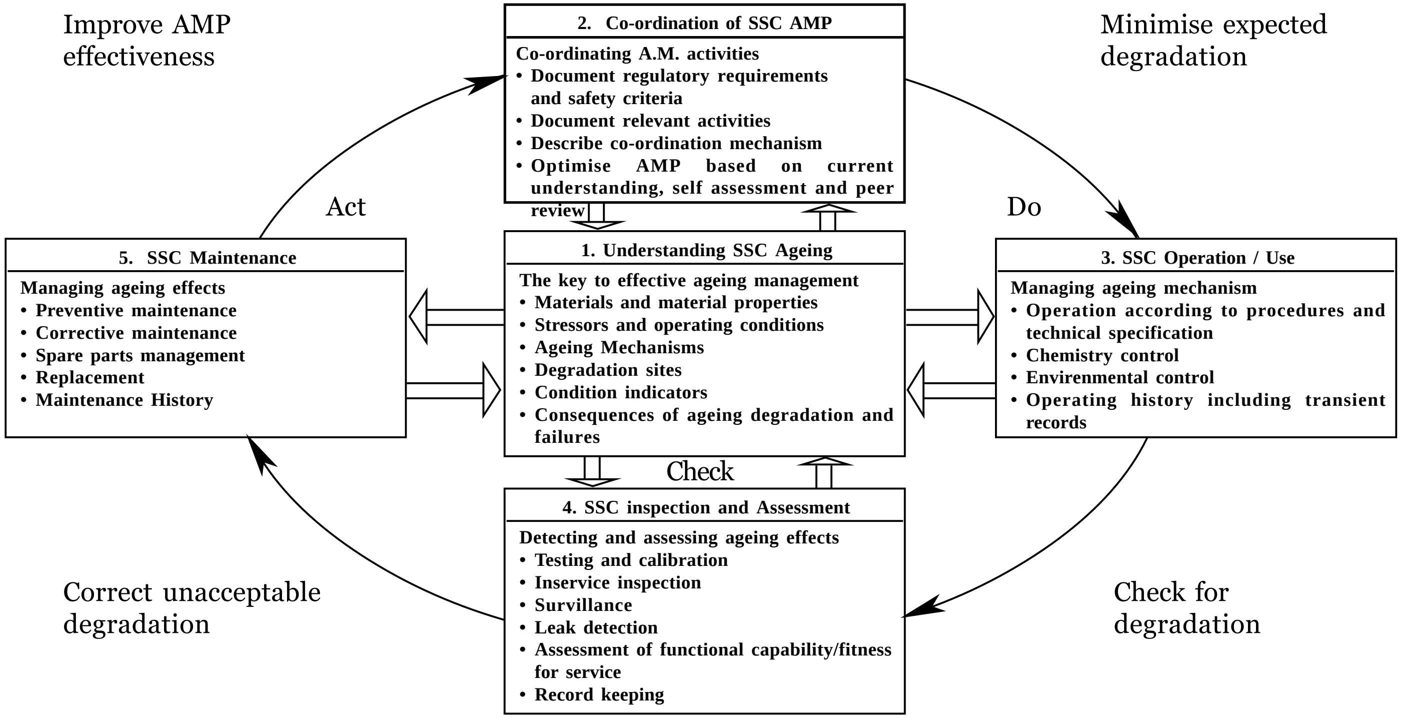
LIFE MANAGEMENT OF NPP FROM SITE SELECTION TO SITE RECOVERY



APPENDIX-4

(Refer Section 2.3)

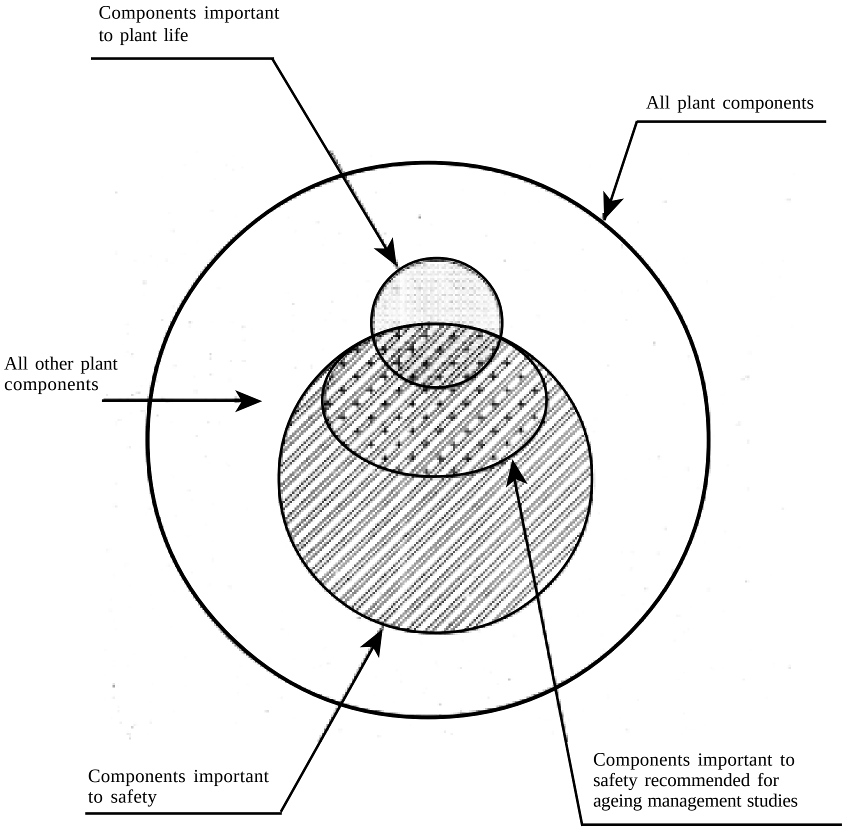
TYPICAL AGEING MANAGEMENT PROGRAMME PLAN FOR OPERATIONAL PHASE



APPENDIX-5

(Refer Section 2.5.2.2)

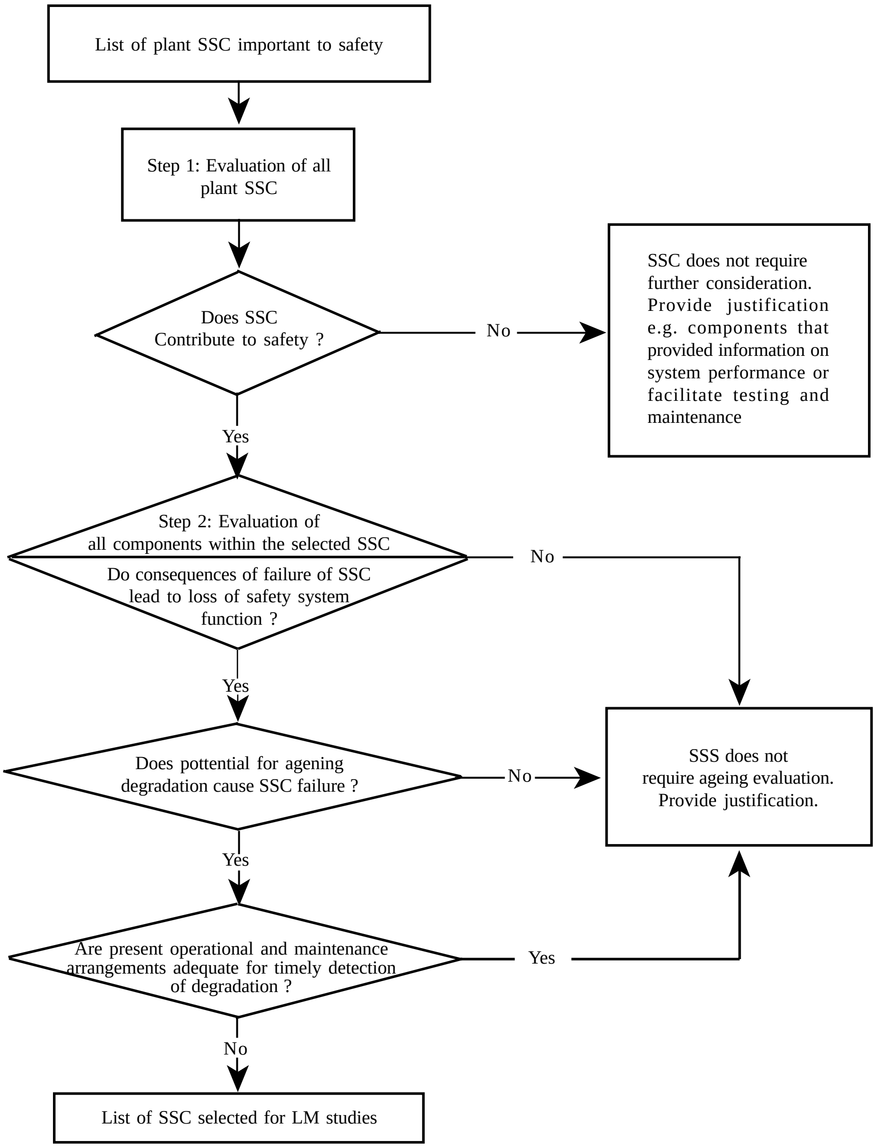
GROUPING AND SELECTION OF SSC OF NPP FOR AGEING MANAGEMENT STUDIES



APPENDIX-6

(Refer Section 2.5.2.2)

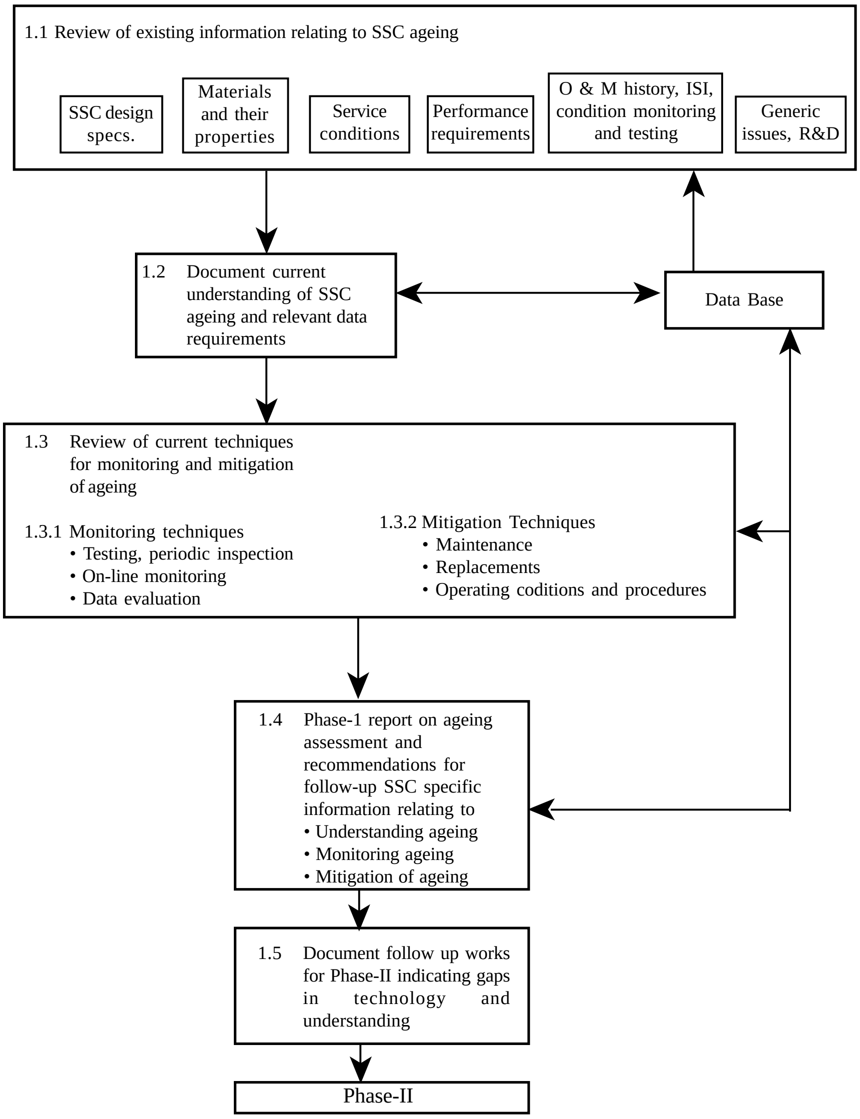
SELECTION PROCESS FOR SSC IMPORTANT TO SAFETY FOR PLANT LIFE MANAGEMENT STUDIES



APPENDIX-7

(Refer Section 2.5.2.8)

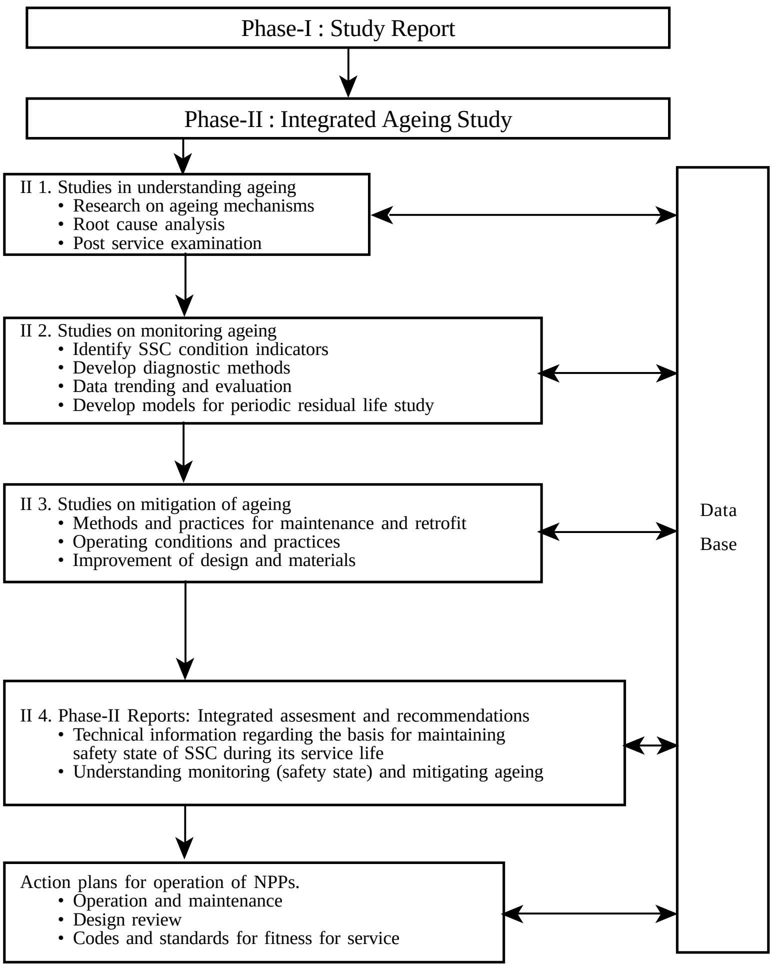
METHODOLOGY OF AGEING MANAGEMENT STUDIES FOR MITIGATING EFFECTS OF AGEING (PHASE-I: INTERIM AGEING STUDY)



APPENDIX-8

(Refer Section 2.5.2.8)

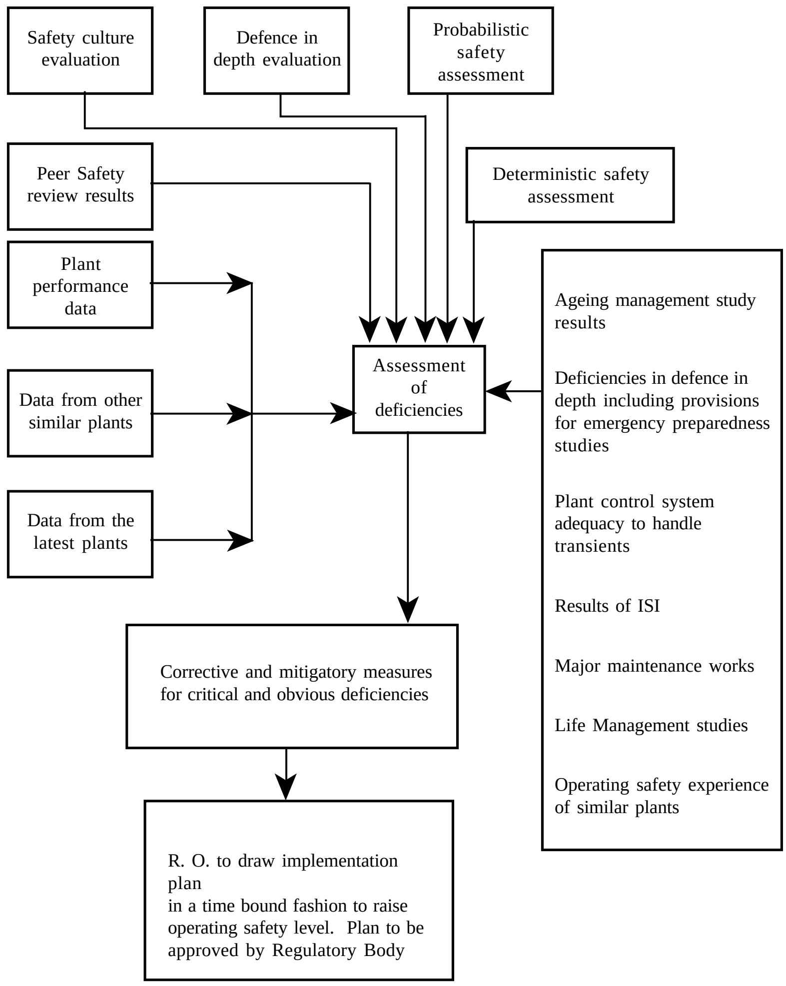
METHODOLOGY OF AGEING MANAGEMENT STUDIES FOR MITIGATING EFFECTS OF AGEING (PHASE-II : INDEPTH AGEING STUDY)



APPENDIX-9

(Refer Section 4.6.3.1)

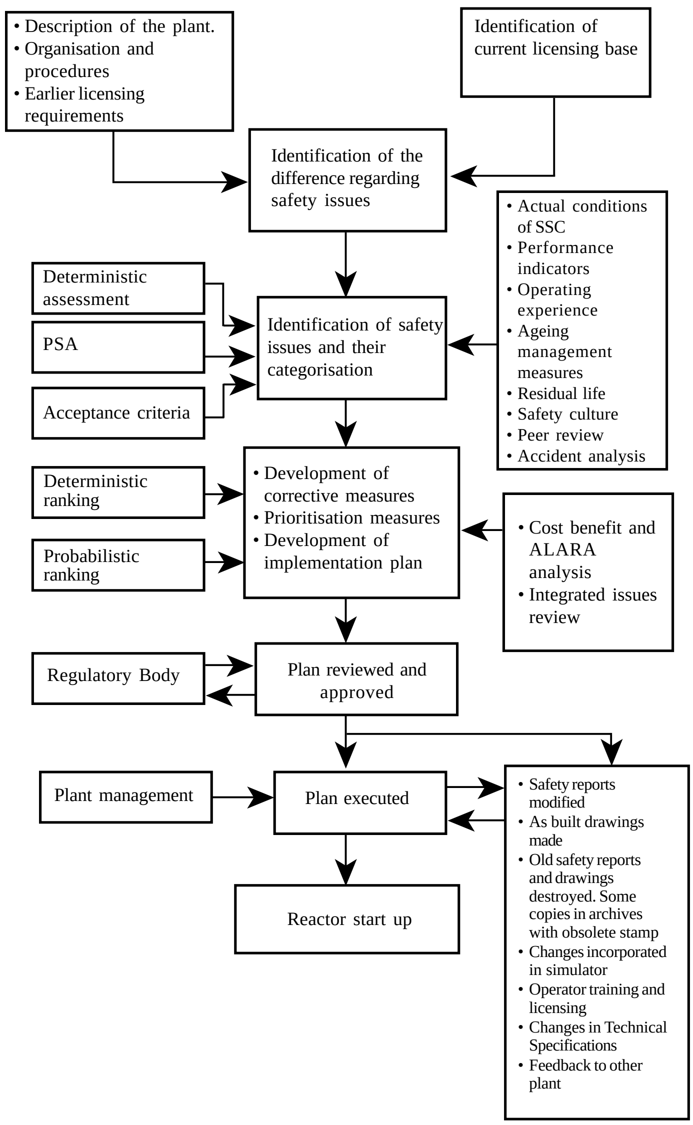
PERIODIC REVIEW FOR LIFE MANAGEMENT



APPENDIX-10

(Refer Section 4.6.3.2)

IN-DEPTH REVIEW FOR LICENSE RENEWAL



ANNEXURE-IA

(Refer Section 2.5.2.3.2)

ENVIRONMENTAL STRESSORS, DEGRADATION MECHANISMS AND AFFECTED / SUSCEPTIBLE MATERIALS AND COMPONENTS

S.No.	Stressors / Degradation Mechanism	Susceptible Materials and Components
1	General corrosion, pitting	Crevices and hideout regions, low to no flow component, ECCS, post incident systems, service water system, (pump, pipe, valves, HX) stand by process systems, fire water system
2	Stress corrosion cracking on internal/external surfaces low and high temperature).	Wet insulation with high chloride, off-normal chemistry parameters (Chloride, pH, sulfides, ammonia), stressed component due to welding, rolling, cold-working. SG , HX tubes, FW heaters and Condenser tubes, SS-304 / 316 pipe lines (primary piping of BWR)
3	Irradiation assisted stress corrosion.	Reactor pressure vessels and internals in BWR, coolant channel in PHWR.
4	Errosion corrosions (high temperature)	Steam and feed water piping, feed pump recirculation flow piping to deaeretor and recirculation valve trim, bleed steam non-return valve casing, extraction piping, HX, moisture-separator and reheater, HP and LP turbine blades
5	Crevices corrosion (low and high temperature)	Stagnant areas, proximity of welds, sleeved region of pipe, stub-shafts, welds with back-up rings, valve / pump drain line area in casing
6	Microbial influenced corrosion (low temperature)	Service / process water equipment (pumps, pipes, valves, HX) where hydro test is carried out with bacteria contaminated water, ingress of dirt /mud in equipment and laid up with water. Diesel generator with contaminated water in cooling circuit

ANNEXURE-IA (contd.)

S.No.	Stressors / Degradation Mechanism	Susceptible Materials and components
7	Corrosion fatigue (low and high temperature)	Thermal mixing regions in carbon and alloy steel components (FW nozzle of RPV and SG thermal sleeve area)
8	Fatigue (low and high temperature)	Rotating equipment support and piping attached to large component (piping and hanger support for MBFP, CEP, turbines and generator)
9	Weld related cracking	
9. 1	Lack of fusion, hot ductility, ferrite depletion, crevice formation (high and low temp.)	Similar materials welds, wrought materials to casting welds, low ferrite filler welds and seam welds.
9. 2	Dilution zone cracking (Heat Affected Zone weld cracks)	Dissimilar material welds. (CS to SS) Reactor vessel/SG channel cladding interface, RPV/SG nozzles to safe ends welds, valves and pumps casing to pipe line welds (CS to SS)
10	Low temperature sensitisation	Stainless steel components
11	Thermal embrittlement (high temperature)	Ferrite and cast SS components
12	Irradiation embrittlement	Reactor pressure vessel, internals and support structures (BWR) Coolant tubes, end-shields and calanderia support structure (PHWR)
13	Mechanical wear/fretting	Mechanical rotating equipment ball bearing fit-up area on the shaft, impeller vanes and turbine blades.
14	Oxidation due to environment	Relay and breaker contacts, lubricants, insulation materials associated with components (cables varnish etc.)
15	Creep and swelling (radiation assisted)	Reactor pressure vessel and internals (BWR), coolant channels (PHWR)
16	Hydrogen pick-up (high temperature and radiation)	Hydride formation in coolant channel (PHWR)
17	Abnormal rise in voltage and frequency	Cables, relays and windings

ANNEXURE-IB

(Refer Section 2.5.2.3.2)

DEGRADATION FACTORS AFFECTING THE PERFORMANCE OF SAFETY RELATED CONCRETE STRUCTURES

(a) Concrete

Ageing stressors/ service conditions	Ageing mechanism	Ageing effect	Potential degradation sites	Remarks
Percolation of fluid through concrete due to moisture gradient	Leaching and efflorescence	Increased porosity and permeability; lowers strength	Near cracks; areas of high moisture percolation	Makes concrete more vulnerable to hostile environments; may indicate other changes to cement paste;unlikely to be an issue for high quality, low-permeability concretes
Exposure to alkali and magnesium sulphates present in soils, seawater or groundwater	Sulphate attack	Expansion and irregular cracking	Subgrade structures and foundations	Sulphate-resistant cements or partial replacement of cements used to minimise potential occurrence.
Exposure to aggressive acids and bases	Conversion of hardened cement to soluble material that can be leached	Increased porosity and permeability	Local areas subject to chemical spills; adjacent to pipework carrying aggressive fluids	Areas prone to attacks are protected by suitable lining/coating

ANNEXURE-IB (Contd.)

(a) Concrete (Contd.)

Ageing stressors/ service conditions	Ageing mechanism	Ageing effect	Potential degradation sites	Remarks
Combination of reactive aggregate, high moisture levels, and alkalis	Alkali-aggregate reactions leading to swelling	Cracking; gel exudation; aggregate pop-out	Areas where moisture levels are high and improper materials utilised.	Eliminate potentially reactive materials; use low alkali-content cements or partial cement replacement
Cyclic loads/ vibrations	Fatigue	Cracking; strength loss	Equipment/ piping supports	Localised damage; fatigue failure of concrete structure unusual
Exposure to flowing gas or liquid carrying particulates and abrasive components	Abrasion; Erosion; Cavitation	Section loss	Cooling water intake and discharge structures	Appropriate hydraulic design and other protection measures
Exposure to thermal cycles at relatively low temperatures	Freeze/thaw	Cracking; spalling	External surfaces where geometry supports moisture accumulation	Air entrainment utilised to minimise potential occurrence
Thermal exposure/ Thermal cycling	Moisture content changes and material incompatibility due to different thermal expansion values	Cracking; spalling; strength loss; reduced modulus of elasticity	Near hot process and steam piping	Generally an issue for hot spot locations; can increase concrete creep that can increase prestressing force losses
Irradiation	Aggregate expansion; hydrolysis	Cracking; loss of mechanical properties	Structures proximate to reactor vessel	Containment irradiation levels likely to be below threshold levels to cause degradation

ANNEXURE-IB (Contd.)

(a) Concrete (Contd.)

Ageing stressors/ service conditions	Ageing mechanism	Ageing effect	Potential degradation sites	Remarks
Consolidation or movement of soil on which containment is founded	Differential settlement	Equipment alignment, cracking	Connected structures on independent foundations	Allowance is made in design; soil sites generally include settlement monitoring instrumentation
Exposure to water containing dissolved salts (e. g. seawater)	Salt crystallisation	Cracking	External surfaces subject to salt spray; intake structures	Minimised through use of low permeability concretes, sealers, and barriers

(b) Mild steel reinforcing

Ageing stressors/ service conditions	Ageing mechanism	Ageing effect	Potential degradation sites	Remarks
Depassivation of steel due to carbonation or presence of chloride ions	Composition or concentration cells leading to corrosion	Concrete cracking and spalling; loss of reinforcement cross-section	Outer layer of steel reinforcement in all structures where cracks or local defects (e. g. joints) are present	Prominent potential form of degradation; leads to reduction of load-carrying capacity
Elevated temperature	Microcrystalline changes	Reduction of yield strength and modulus of elasticity	Near hot process and steam piping	Of significance only where temperatures exceed~200^0C
Irradiation	Microstructural transformation	Increased yield strength; reduced ductility	Structures proximate to reactor vessel	Containment irradiation levels likely to be below threshold levels to cause degradation

ANNEXURE-IB (Contd.)

(b) Mild steel reinforcing (Contd.)

Ageing stressors/ service conditions	Ageing mechanism	Ageing effect	Potential degradation sites	Remarks
Cyclic loading	Fatigue	Loss of bond to concrete; failure of steel under extreme conditions	Equipment/ piping supports	Localised damage; fatigue failure of concrete structures unusual

(c) Prestressing

Ageing stressors/ service conditions	Ageing mechanism	Ageing effect	Potential degradation sites	Remarks
Localised pitting, general corrosion, stress corrosion, or hydrogen embrittlement	Corrosion due to specific environmental exposures (e.g. electrochemical, hydrogen, or microbiological)	Loss of cross-section and reduced ductility	Tendon and anchorage hardware of prestressed concrete containments	Potential degradation mechanism due to lower tolerance to corrosion than mild steel reinforcement
Elevated temperature	Microcrystalline changes	Reduction of strength; increased relaxation and creep	Near hot processes	Thermal exposure not likely to reach levels that can produce ageing effects in prestressing
Irradiation	Microstructural transformation	Increased strength; reduced ductility	Structures proximate to reactor vessel	Containment irradiation levels likely to be below threshold levels to cause degradation
Cyclic loading due to diurnal or operating effects	Fatigue	Failure of prestressing under extreme conditions	Tendon and anchorage hardware of prestressed concrete containments	Not likely as cyclic loadings are generally small in number and magnitude

ANNEXURE-IB (Contd.)

(c) Prestressing (Contd.)

Ageing stressors/ service conditions	Ageing mechanism	Ageing effect	Potential degradation sites	Remarks
Long term loading	Stress relaxation; creep and shrinkage of concrete	Loss of prestressing force	Prestressed concrete containments	Larger than anticipated loss of prestressing forces

(d) Containment liners

Ageing stressors/ service conditions	Ageing mechanism	Ageing effect	Potential degradation sites	Remarks (e. g. significance)
Electrochemical reaction with environment (metallic)	Composition or concentration cells leading to general or pitting corrosion	Loss of cross-section; reduced leaktightness	Areas of moisture storage/ accumulation, exposure to chemical spills, or borated water	Corrosion has been noted in several containments near the interface where the liner becomes embedded in the concrete
Elevated temperature (metallic)	Microcrystalline changes	Reduction of strength; increased ductility	Near hot processes and steam piping	Thermal exposure not likely to reach levels that can produce ageing effects in metallic liners
Irradiation (metallic and non-metallic)	Microstructural transformation (metallic); increased cross-linking (non-metallic)	Increased strength; reduced ductility	Structures proximate to reactor vessel	Containment irradiation levels likely to be below threshold levels to cause degradation
Cyclic loading due to diurnal or operating effects (metallic and non-metallic)	Fatigue	Cracking; reduced leaktightness	Inside surface of concrete containment building	Not likely as cyclic loadings are generally small in number and magnitude

ANNEXURE-IB (Contd.)

(d) Containment liners (contd.)

Ageing stressors/ service conditions	Ageing mechanism	Ageing effect	Potential degradation sites	Remarks (e. g. significance)
Localised effects (non-metallic liners)	Impact loadings; stress concentrations; physical and chemical changes of concrete	Cracking; reduced leaktightness	Inside surfaces of concrete containment building	Potential problem in high traffic areas

ANNEXURE - II

(Refer Section 2.5.2.3.2)

AGEING MECHANISMS CAUSING DEGRADATION

Major degradation mechanisms for various materials (metals, concrete and non-metals) are discussed below. This is only an indicative list .

II.1 Metal Degradation.

II.1.1 Irradiation

The effect of neutron irradiation on most of the metals is to increase the yield strength, ultimate tensile strength, and reduce toughness and elongation. For certain metals properties begin to change at about 10^{20} n/cm^2. The contribution of associated gamma radiation to the ageing phenomenon is only marginal. Ageing sensitivity of the materials will depend upon type of materials, heat treatment, initial mechanical properties and the presence of trace material such as copper, phosphorus and carbon. Zirconium alloys for fuel cladding, calandria tubes and low alloy steel end-shields are examples of components where effects of neutron irradiation have to be examined critically.

II.1.2 Thermal ageing

Thermal ageing in general is a degrading process causing changes in strength properties, hardness, ductility and toughness. Cast austenitic stainless steel which contains about 15% of ferrite phase has been known to experience thermal embrittlement when exposed to reactor operating temperature of 280^0-320^0 C. In such cases, the number of thermal cycles that critical equipment or components may be subjected to be specified and observed.

II.1.3 Creep

For certain materials at high temperatures various thermally activated processes in the microstructure result in degradation of the metal characteristics mainly their mechanical properties. The type of fracture whether ductile or brittle caused by creep depends on temperature strain rate. Components should be designed only in the primary and secondary regimes as the damage is still not irreversible.

II.1.4 Fatigue

Many structural components in an NPP are subjected to cyclic stresses due to fluctuations of operating parameters or process transients. Fatigue behaviour of a component is governed by a number of factors like stress range, mean stress, number of fatigue cycles, environmental conditions, metallurgy, and surface toughness of the material. Cracks can initiate at local geometric stress concentration points like notches, structural discontinuities and surface defects. Loading produced by fluctuating temperature can cause thermal fatigue. Vibrations of tubes in heat exchangers and steam generators due to design deficiencies or excessive shell/tube side flow can cause high cycle fatigue failure. NPP components subject to temperature transients and cyclic pressure loads are designed for a conservative frequency of transients. It becomes difficult to keep track of their transients. Advantage should be taken of on line fatigue monitoring programs available for such purposes.

Thermal fatigue can arise from thermal stresses due to a non-uniform temperature distribution in the wetted components/parts produced by cyclic changes in fluid temperature. Components like reactor vessel, reactor system piping, heat exchangers and pumps in some systems are subject to cyclic stresses. Some equipment in an NPP are kept in standby condition. When put into service, the temperature changes from ambient temperature to the operating temperature of the circulating fluid. This results in thermal stresses. To ensure that thermal fatigue does not become a significant ageing mechanism the rate of temperature change during normal start-up/shutdown should be kept within limits. The cumulative usage factor due to transients and accident conditions should be kept to the minimum. It is a good practice to keep cycle logs on important equipment. This allows the equipment replacement before fatigue failure can take place.

II.1.5 Corrosion

(i) In most cases corrosion is an electrochemical reaction characterised by material loss. Corrosion reduces the component wall thickness either locally for e. g. crevice corrosion, pitting, galvanic corrosion, micro biologically influenced corrosion, etc. or more uniform like rusting. Oxidations of fuel element cladding, reduction in wall thickness of steam generator tubes are cases of uniform corrosion. In general corrosion products occupy a larger volume than that of the metal itself. Thus in crevice corrosion build up of reaction products can fill up the available clearance with the consequence of build up of pressure.

(ii) Similarly in steam generator tube holder’s ferrite spacers, the build up reaction products can result in denting of the thin walled tubes. Pitting is an example of local corrosion attack. Crevice corrosion can be avoided by appropriate design features. Highest corrosion rates for metals can occur in system applications, where the fluid is low temperature air saturated stagnant water. Appropriate water chemistry control and operation procedure can help in avoiding de-alloying and keeping the corrosion rates within acceptable limits.

(iii) Galvanic corrosion can occur at dissimilar metal contact points, where surfaces are not adequately protected, and surfaces are exposed to moisture, for prolonged period of time. The effects of galvanic corrosion can be minimised by insulating the dissimilar metals from each other with non-conductive coatings such as painting or protecting the metals with a sacrificial anode.

II.1.6 Erosion and Erosion/Corrosion

(i) Erosion is removal of protective oxide layer on a metal and/or the base metal by mechanical action of a flowing fluid or particulate. Erosion/corrosion occurs when the fluid or particulate matter is also corrosive to metal. Changes in flow direction and changes in flow cross section can create eddies and high local turbulence. Consequences are thinning of wall until leakage or rupture occurs.

(ii) These processes are influenced by the material, water chemistry, operating temperature and fluid flow velocity. The material removal rate increases exponentially with increasing flow velocity. In feed water pipes, corrosion erosion effects have been reported at various plants. PHWR feeder pipes are also susceptible to erosion.

II.1.7 Stress Corrosion Cracking (SCC)

Certain alloys are susceptible to subcritical crack growth when subjected to stress (applied or residual) and a corrosive environment. Elimination or reduction in any one or more factors can reduce the possibility of SCC to occur. Depending on system material and corrosive environment the cracking can be intergrannular or transgrannular. Severe cases of intergrannular stress corrosion cracking (IGSCC) have occurred in piping of non stabilised SS under BWR conditions. Strain induced corrosion cracking can occur in piping and vessels of unalloyed or low alloyed ferrite steels in high purity water. Pipe welds are particularly prone to IGSCC.

II.1.8 Irradiation Assisted Stress Corrosion Cracking (IASCC)

Nonsensitised austenitic stainless steels become susceptible to intergranular failure after accumulation of a sufficient neutron fluence. Failures of BWR core shroud and other reactor internal components like control–blade sheaths and handles have been reported after the components have reached neutron fluence levels > 5 x 10^{20} $n \cdot cm^{-2}$, E > 1 MeV. High levels of neutron irradiation modify the mechanical properties of these components making them less flaw tolerant. Irradiation assisted stress corrosion cracking (IASCC) appears to occur with little or no stress. In addition to neutron fluence, the other contributing factors appear to be material composition, heat-treatment condition (sensitisation), and fabrication variables and residual stresses.

II.1.9 Corrossion Fatigue

In the case of fatigue loads, additional environmental effects can cause the number of cycles for crack initiation to be drastically reduced from that for an inert atmosphere. Examples of concern with regard to reduced lifetime in corrosive environments with high cycle fatigue or cyclic crack growths are:

(i) Parts in motion e.g. shaft of main circulating pump

(ii) Piping subjected to flow vibrations

II.1.10 Wear

(i) Wear may be erosive, adhesive or abrasive. In an NPP all wear mechanisms can occur. Some of the areas of concern are valves, electrical relays and contacts, parts of various machinery, pipes, turbine blades heat exchanger and steam generator tubes etc.

(ii) Erosive wear or erosion is a significant ageing mechanism for pump components like impellers, suction and discharge nozzles, casing, wear rings and internal radial bearings. For example, it can become a failure mechanism in service water pumps drawing water from lake, river or sea.

(iii) Adhesive wear, also called scouring, galling or seizing is characterised by transference of material from one surface to another during relative motion or sliding due to micro welding of asperities. It can be reduced by application of suitable lubricants. Parts that cannot be lubricated must be adapted to each other with respect to their adhesion tendency. Austenitic steels have high adhesion tendencies. Materials with high non-metallic phases like carbides, nitrides or ceramics are more suitable for such services.

(iv) Abrasive wear, commonly also known as scouring or gouging is characterised as displacement of material from a solid surface due to hard particles sliding along the surface. Pump shafts, plungers, internal valves, bearings, seals etc. are subjected to abrasive wear and should be designed for easy replacement.

II.1.11 Weld Related Cracking

(i) In austenitic stainless steel welds, lower thermal conductivity, higher coefficient of thermal expansion and viscous nature of the molten pool promotes distortion, high residual stress and lack of fusion defect. Other concern is solidification and liquation cracking. These type of cracks can be avoided by proper welding technique and selection of appropriate filler wire which will ensure 3% to 4% ferrite in the weld metal.

(i) Weld related cracking issues mainly pertain to high strength low alloy components and dissimilar metal welding. Special considerations must be given to dissimilar metal welding intended for high temperature service. Failures in austenitic/ ferritic steel joints may occur in ferritic steel heat affected zone adjacent to weld interface. These failures are attributed to high stress and creep at the interface due to difference in coefficient of thermal expansion of the weld and base metal.

(iii) Low alloy steel and high strength ferritic steels are susceptible to hydrogen induced embrittlement**.** Presence of hydrogen in the weld can lead to under bead cracking or it may result in reduced ductility of the welds. Use of low hydrogen electrodes is recommended for such applications.

II.1.12 Insulation Embrittlement and Degradation

(i) Electrical insulation systems are a vital part for most of the electrical rotating machinery and transformers.Electrical insulation gets aged by exposure to combination of thermal, electrical, mechanical, environmental and radiation stressors. This leads to deterioration of quality of insulation. The periodic measurements and diagnostic tests like IR, Polarisation Index, D.C. leakage current, AC charging test, Tangent Delta and tie up tests, dielectric loss energy test, partial discharge test, voltage surge test indicate the trend of integrity of insulation system and play a major role in life prediction.

(ii) Ageing tests indicate that for certain cable insulation materials, mechanical properties degrade prior to ageing degradation of electrical properties. The concern is that embrittled insulation can crack during an accident.

(iii) Mineral hydrocarbon oils are used to provide dielectric strength, insulation and as a cooling medium in transformers. During its service the oil undergoes oxidation leading to formation of peroxides, water, acids and sludge. This results in deterioration of insulating materials (oil and cellulose materials). Periodic oil tests and dissolved gas analysis provide information on deterioration and deciding about when to replace the oil.

II.2. Degradation of Concrete

II.2.1 The principal applications of concrete in nuclear safety related systems include its use in containment building, biological shield and auxiliary safety related buildings. Deterioration of concrete results primarily from cracking due to aggressive environment, corrosion of embodiments or extreme environmental exposure (temperature, pressure and irradiation, saline atmosphere, humidity etc.). For details refer Annexure-1B.

II.2.2 Cracking of concrete can occur when the concrete is either in plastic or hardened state. Cracks may not always affect the ability of the structure to bear load but they can expose the concrete to attack from hostile environments which can affect structural durability.

II.2.3 Cracking of hardened concrete results from shrinkage, salt crystallisation, thermal effects and chemical reaction including sulphate attack, carbonation, alkali- aggregate reaction and reinforcement corrosion.

II.2.4 Cracking and spalling of concrete can result from the corrosion of mild steel reinforcement bars due to build up of corrosion products.

II.2.5 Thermal gradients are deleterious to concrete structures. The normal practice is to limit the bulk temperature to less than 65^0C.

II.2.6 The potential causes of degradation of the reinforcing steel are corrosion, exposure to elevated temperatures and irradiation. Reinforcement corrosion problem occurs when the concrete pH is reduced to less than 11 destroying passive iron oxide layer as a result of improper construction techniques e. g. honeycombs formation or presence of cracks that permit access for hostile environments.

II.2.7 The need for the possibility of inspection, retensioning of prestress cable and replacement has made non-grouted post tension steel tendons, the dominant prestressing systems for containments. The prestressing systems are protected by filling the ducts containing the post tensional tendons either with microcrystalline waxes containing corrosion inhibitors for non-grouted tendons or with portland cement grout for grouted tendons. Potential degradation modes for the various prestressing systems include corrosion, exposure to elevated temperature and irradiation. Long term loss of prestressing is due to shrinkage of concrete, creep of concrete and relaxation of steel.

II.3 Non Metal Ageing

II.3.1 Electronics

Electronic components form the major part of instrumentation and control (I&C) systems of NPPs. Humidity, dust, switching cycles, temperature, vibration and radiation are the significant causative factors for ageing of these components. Most of the electronic components are expected to be replaced after a specified period during the life of the plant. Electronic equipment is generally located in low radiation fields.

II.3. 2 Polymers

Various polymers like elastomers, thermoplastics, and thermosettings find a variety of applications in an NPP. Elastomers as gaskets and O-rings are used for sealing function in pressure boundaries and as sealing material in containment. Polymers are used extensively as sheathing materials for cables. The stressors are temperature, irradiation, humidity, hydrocarbons, and ozone. Thermal effects result in polymers becoming hard and brittle, losing tensile strength and elastic qualities, which induces cracking as it ages. The oxidation reaction, which is dominant in thermal environments, is also operative in radiation environments resulting in reduction in elasticity and embrittlement.

ANNEXURE-III

(Refer Section 2.5.2.3.2)

MANAGEMENT OF AGEING OF INSTRUMENT AND CONTROL EQUIPMENT

III.1 General

Experience has shown that ageing and obsolescence have the potential to cause maintainability and operability of many I & C systems to deteriorate well before the end of the plant life. An I & C ageing management strategy is therefore required to control this threat.

III.1.1 Temperature and Pressure Sensors

Heat, humidity, temperature cycling, ionising radiation, mechanical vibrations and shock can cause the degradation in both the calibration and response time of the temperature and pressure sensors in NPPs. Ageing affects both the steady state (calibration) and dynamic (response time) performance of sensors. For example, RTD and thermocouple seals can fail (dry out, shrink, or crack) and allow moisture into the sensor causing a reduction in insulation resistance. Therefore, ageing management of temperature and pressure sensors is accomplished by periodic calibration and response time testing. Pressure transmitters should be calibrated periodically as per the requirement in technical specifications for operation.

III.1.2 Neutron Flux Detectors

Neutran flux detectors have a lifetime shorter than the reactor. They are consumable parts, which need to be changed often. Gas multiplication factor of proportional counters is excessively sensitive to presence of impurities. Oxygen traces or humidity may create negative heavy ions that stop multiplication and changes the characteristics of the sensor. The degradation of insulation resistance in self- powered neutron detector, has a similar effect in modifying the sensitivity. Ionic attack on BF_3 detectors can lead to sudden failure. The ageing effects typically change the response curve of the sensor.

Periodic calibration, verification and response time testing and trending of result can help to identify ageing effects and estimate the residual life of the detector.

III.1.3 Electronics

High temperature and temperature cycling are the dominant causes of ageing in electronic components and circuits. The dominant ageing mechanism for capacitors with liquid electrolyte is loss of electrolyte through the end cap seals. The increasing use of Teflon seals has reduced the extent of this problem. A variety of measures may be taken to guard against the consequences of loss of electrolytes. These include, periodic replacement, replacement of all similar components when the first failure is detected, periodic testing/monitoring of components and spare modules; leakage current, capacitance value, and power factor. Ageing management strategy should address over voltage, number of starts/power-ups and electrostatic discharge.

III.1.4 Fuses

Fuses are widely used as protective devices against over current in electronic or electrical circuits. As the lifetime of the component is related to the number of starts, the only effective management technique is preventive maintenance. This preventive maintenance may by conditional; replacement of all fuses on a set of equipment when the first fuse failure is encountered.

III.1.5 Relays

Sub-components of standard electromagnetic relays: the relay coil, the relay contacts, ancilliary components such as contact spacers and plugs and sockets are vulnerable to ageing. Ageing of relay coils is primarily a problem on relays which are continually energised. Excessive heat may be generated by the coil or associated components, which may cause the coil to burn out or adversely effect other components within the relay or nearby. In pneumatic time delay relays heat may cause embrittlement of the diaphragms causing set-point drift, or set point drift on under voltage relays.

Relay contact may age due to contact oxidation on normally open (NO) contacts, contact welding or pitting due to excessive current (possibly caused by switching of inductive loads) or due to chemical attack e.g. due to the use of high sulphur content rubber components within the relay. Internal ancillary component may deform due to temperature or chemical attack.

When in service, a periodic visual inspection may be helpful in identifying any chemical breakdown or degradation of components or contacts. Regular cleaning of relay contacts may also be beneficial in specific circumstances. Most relays are rated for a certain number of operations and life will depend on how the relay is used. Relays, which are repeatedly exercised, may need periodic replacement.

III.1.6 Connectors

The dominant ageing mechanisms for connectors are mechanical wear and oxidation of contacts. These lead to an increase in contact impedance or even to a complete open circuit. Connector should be left undisturbed wherever possible. Repeated breakinsg and making of connections may lead to mechanical wear. Heat drying of connectors by heating before installation can help eliminate failures due to moisture. Thermographic scanning of connector whilst in service can give an indication of high resistance points which may give early warning of failure.

ANNEXURE-IV

(Refer Section 2.5.2 3.2**)**

MANAGEMENT OF AGEING OF CABLES

IV.1 The intended service requirement for cables is to continue to be functional during LOCA occurring even at the end of NPP design life. The ageing degradation-stressors for cables are:

(i) Elevated ambient temperature and humidity.

(ii) Cyclic mechanical stress due to connected equipment vibrations.

(iii) Exposure to radiation.

(iv) Thermal oxidation of insulation and cable terminals.

IV.2 The age related degradation of cable material causes the following :

(i) Mechanical failure.

(ii) Loss of insulation resistance.

(iii) Degradation of cable conductor/insulation material.

IV.3 The following mitigating measures for life management may be considered :

IV. 3.1 Condition monitoring techniques for detecting degradation:

(i) Oxidation Indentor Time (OIT)- It is related to amount of anti-oxidation remaining in polymer and indicates residual life.

(ii) Insulation Resistance Spectroscopy- Cable deposit/cable sample assessment at high temperature/ radiation hotspot for degradation and residual life by OIT.

(iii) Rotary equipment vibration monitoring for assessing damage on cable and terminals.

(iv) Monitoring temperature and radiation field for determining hotspots on continuous/periodic basis using thermography or other techniques.

IV.3.2 Development of cable data-base indicating cable materials, location/routing, electrical loading, operating conditions, radiation field and temperature and their source, operating history of cables and replacement, mode of failure and trending and analysis of condition monitoring results.

(i) Maintain restriction on repeat start/stops and loading of electrical systems.

(ii) Avoid short radius bends for power cables and overloading of cable trays.

(iii) Control steam, hot water, oil leaks and impingement over cables.

(iv) Maintain cleanliness of cable surfaces for correct dissipation of heat.

(v) Monitor fire prevention and protection system to avoid fire.

(vi) Type/environment qualification including accelerated ageing, DBE and post DBE functional testing of cable, and evaluation of long term behavior.

IV.3. 3 Air and cooling water flow balancing for ventilation systems to reduce area temperature. Control thermal insulation breakage to reduce local area temperature rise.

ANNEXURE-V

(Refer Section 2.5.2.3.2)

DEGRADATION MECHANISMS IN PRESSURISED HEAVY WATER REACTOR COMPONENTS

V.1. Coolant Tubes

V.1.1 Under the operating conditions of stress, temperature, fast neutron flux and water chemistry, Zirconium alloy coolant tubes are susceptible to a number of ageing mechanisms. The primary ageing mechanisms are:

(i) Delayed Hydrogen Cracking (DHC).

(ii) Irradiation enhanced deformation.

(iii) Change of coolant tube material properties.

V.1.2 Hydrides are always present in coolant tubes at room temperature. At operating temperature hydrides form when Hydrogen concentration is greater than terminal solid solubility (TSS). Deuterium produced by radiolysis and corrosion action is absorbed by the coolant tube material. The pick up rates increase as the tubes' oxide thickness increases. The action is more pronounced in Zircalloy-2 as compared to Zirconium-2.5% Niobium (Zr 2. 5Nb) coolant tubes. Over the years this concentration will exceed the TSS at the operating temperature, leading to hydrogen embrittlement and limiting operating life of coolant tube.

V.1.3 Coolant tube irradiation enhances deformation. The crystal structure of Zirconium causes anisotropy deformation. Neutron irradiation causes growth and creep. Strains caused by thermal creep, irradiation creep and growth result in deformation in pressure tubes resulting in increase in length, sagging, increase in tube diameter and reduction in wall thickness. Appropriate allowances and monitoring procedures have to be provided at the design stage to take care of these deformations if these are not to become life-limiting issues.

V.1.4 Irradiation changes coolant tube material properties, increases tube hardness, yield and tensile strength and reduces ductility and fracture toughness. As the coolant tubes become more susceptible to fracture, the margins associated with leak before break (LBB) behavior decrease. Changes in tube material properties must be monitored over the lifetime to ensure the specified safety margins. The solubility limit of H/D in Zirconium is low. Delayed Hydrogen Cracking (DHC) is the main ageing issue for coolant tubes. For arresting the DHC the following causative factors that lead to local concentration of H/D should be avoided.

(i) Significant tensile stress concentrations and temperature gradients through the tube wall. These can be minimised by ensuring that significant flaws or defects beyond specified limits are not created during tube manufacture, commissioning or operation.

(ii) Large temperature gradient through the tube wall may occur due to displacement of garter spring from their intended locations of coolant tube and make contact with the cooler calandria tube creating the potential for formation of brittle hydride blisters. This may eventually lead to cracking of coolant tube. Therefore coolant tube and calandria tube contact must be prevented. If it does occur due to garter spring displacement it should not be allowed to exist long enough for a hydride blister to form and grow at the cold spot.

V.1.5 Ensuring low initial hydrogen in the pressure tube and low residual stresses in the rolled joints, selection of proper materials, maintaining proper coolant chemistry and oxidising atmosphere in the continuously flowing annulus gas, repositioning of garter springs if displaced help in enhancing the lifetime of the pressure tubes.

V.2 Steam Generators (SG)

V.2.1 The steam generator tubes act as a barrier between the radioactive primary side and the non-radioactive secondary side. Area wise, the thin walled tubes of SG constitute a major portion of the total primary coolant system pressure boundary. The primary coolant being at a higher pressure, any leakage from the primary to the secondary side of the SG can result in release of radioactivity to the secondary side and to the environment outside through the pressure relief devices.

V.2.2 There is a number of degrading mechanisms, which affect the performance of SGs. These include primary water side tube stress corrosion cracking (PWSCC), tube outside diameter stress corrosion cracking and trans-grannular SCC, fretting, wear, thinning, pitting, denting, high cycle fatigue, corrosion, and corrosion fatigue. PWSCC can occur if there is a combination of susceptible tubing material microstructure, high residual stresses and high temperature water. Present day materials like alloy 600, 690 and alloy 800M are not susceptible to PWSCC.

V.2.3 Appropriate fabrication and installation procedures can reduce residual stresses. Depending on concentration of corrosive impurities at dry out region, both mechanism IGSCC and intergrannular attack are active with ODSCC. The impurity levels in the secondary side systems are influenced by the type of cooling water, secondary degradation sites are tube to tube sheet and tube to support plate crevices in the sludge pile region. Flow induced vibration can lead to fretting and wear depending at support locations, stiffness of the supports and annular gap between tube and support. U-bend portion, if not properly supported may be subjected to high cycle fatigue due to flow induced vibrations. Combination of high vibration amplitude and low fatigue strength cause tube failure.

V.2.4 Condenser tube leaks and inadequate water treatment and impurities in treated make up water can introduce aggressive ions like chlorides and sulfides in the secondary side leading to local acidic conditions conducive to pitting. These may also manifest as under deposit corrosion in the tube sheet and tube support plate areas. Denting can occur if heavy build up of deposits occur in the tube support annular gaps.

V.2.5 Thermal fatigue and erosion/ corrosion can cause ageing degradation in feed water nozzles. Corrosion fatigue cracks can be caused by the thermal stratification of feed water and stress concentration in the nozzle area. Cyclical local stratification and thermal stripping due to turbulent mixing at the interface of hot and cold layer can initiate high cycle fatigue surface crack. Though unlikely to cause a failure; this type of degradation can weaken the SG against a pressure pulse or water hammer. These concerns are best addressed by appropriate design of the feed water nozzles.

V.2.6 Care should be exercised for avoiding operation of NPPs with low feed water temperature right from commissioning stage. Feed water nozzle temperature should be monitored for guarding against such degradation.

V.3 D_2O/H_2O Heat Exchangers

V.3. 1 Heat exchangers in the moderator or PHT systems are cooled by process water. The cooling water may come from a lake, pond, canal or sea. The outer surfaces of heat exchanger tubes are subjected to degradation mechanisms like corrosion, erosion, fatigue and wear. Vibrations of the tubes in a heat exchanger can lead to failure either due to fatigue of the tubes or to fretting corrosion where the vibrating tubes contact baffles. Tube vibrations can result from excessive shell side flow velocities across the tubes often in the window area where the shell side flow enters the tube bundle and can be attributed to improper design. Equipment of proven design should be used for critical services.

V.3.2 The tritium levels in both moderator and PHT system build up with time. With ageing the heat exchangers tubes become prone to leaks/failures. In NPPs where the process water is used to cool the heat exchangers directly, this becomes a problem. In case of a tube failure in any of the PHT or moderator system heat exchangers, the tritiated water gets mixed with the process water and can result in uncontrolled release of activity to the cooling water body, which may be in the public domain. The problem is less severe in plants with closed loop cooling system as the effluent activity discharge can be controlled. However this may require retention of the large quantity of tritiated water at the plant for long periods.

V.3. 3 ISI of the heat exchangers has to be carried out regularly. Acceptance criteria must be strictly followed. Good accessibility for ISI and testing, should be provided. Routine eddy current testing used for checking wall thinning has some limitations. Expertise and techniques to detect fine cracks needs to be developed. Tooling for pulling out a tube periodically to characterise the nature of degradations should be developed as this could provide basis for replacement of a heat exchanger reaching its end of life. Reliable tritium-in-water monitors can enable the plant to effectively handle and minimise release of activity in case of a tube leak.

ANNEXURE- VI

(Refer Section 3.1)

PREOPERATIONAL LIFE MANAGEMENT CONSIDERATIONS

VI.1 Design.

(i) Design considerations may have very important bearing on the life of a plant. Higher design margins will be conducive for long life e. g. heat exchangers have typically 10% extra tubes to cater for failures of tubes. It can be increased to say 20%. The design could be conservative with less number of tubes in window region.

(ii) The bends and elbows in primary coolant system should be of higher schedule to mitigate the effects of erosion.

(iii) Number of valves can be reduced, as there is improvement in the quality of equipment like pumps, heat exchangers etc. requiring less maintenance. Reduction in the number of valves will result in reduction in cost and maintenance efforts.

(iv) Under Indian conditions large voltage and frequency variations are endemic which are harmful to equipment like TG and pumps and motors. Manufacturers may be asked to supply turbo-generators, pumps, motors etc. with higher frequency and voltage variation - withstand capability.

(v) All equipment should be of proven and good design.

(vi) All design of critical components should be checked by a competent organisation/persons. All SSC have to meet stringent national / international codes and standards.

VI.2 Manufacturing and storage:

(i) Any defect during manufacture is best managed by avoiding it. Haste, less trained personnel on the job, inappropriate supervision both by the manufacturers and the client, (user), inadequate appreciation of the importance of the equipment and its inaccessibility, and the readiness to accord design and manufacturing concessions are some of the reasons of defects during manufacture.

(ii) As a general rule, for equipment located in inaccessible areas, no design concessions should be given. Where unavoidable, it should be done with proper considerations, record and long-term accountability and responsibility.

(iii) Life of many equipment is circumscribed due to inadequate storage either at manufacturer's place or at the site or central stores.

VI.3 Erection.

(i) Clean room condition should be maintained during erection. .

(ii) Training should be imparted to erection personnel to make them aware regarding design and operation requirements. Any design deficiency as well as construction difficulties should be brought to the notice of designers for corrective action.

ANNEXURE-VII

(Refer Section 4.2)

GENERAL LIST OF STRUCTURES, SYSTEMS AND COMPONENTS FOR BOILING WATER REACTORS

VII.1 As per classification and selection of components for life management as per item 5. 2, a list of SSC is prepared for BWR based on following criteria.

VII.2 Category 1: Not replaceable and only limited preventive maintenance and ISI is possible.

Examples:
(i) Reactor pressure vessel and internals and supports.
(ii) Civil structure, drywell (Containment), reactor building, ECCS building.

VII.3 Category 2: Preventive maintenance, ISI, and condition monitoring and trending is possible to mitigate ageing but difficult to replace due to radiation exposure and or long shut-down period.

Examples:
(i) Reactor recirculating pump motor assembly, shut-down heat exchanger and pumps, clean-up system pumps, heat exchangers piping and valves and feeders.
(ii) Secondary steam generator and supports.
(iii) Reactor recirculation pumps and piping and valves.
(iv) ECCS pumps, MOVs, Emergency condensers, Control rod drive mechanisms, steamline safety relief valves, main steam isolation valves, containment cooling fans and drywell sump pumps.
(v) Nuclear instrument systems (SRM, PRM, LPRM, TIP).
(vi) Turbine casing and generator stators and TG foundations.

VII.4 Category 3: Preventive maintenance, ISI, and condition monitoring possible to mitigate effects of ageing and replaceable on routine basis during operation phase.

Examples:
(i) CRD booster and feed pumps and hydraulic system components, instrumented relief valves for primary coolant, RBCW pumps and heat exchangers, fuel pool cooling pumps, fuelling machine, containment isolation dampers.

(ii) Main boiler feed pumps, secondary feed pumps and safety related PW pumps, fire system valves and pumps.

(iii) Start-up transformers, Emergency and black out diesel generator sets, class I, II, III power supplies and MG sets.

(iv) Condenser tubes, poison injection D.C pumps and post incident pumps.

ANNEXURE-VIII

(Refer Section 4.4.1)

AGEING MANAGEMENT PROGRAMME FOR STRUCTURES, SYSTEM AND COMPONENTS (PRESSURISED HEAVY WATER REACTORS)

S.No.	Component	Degradation mechanisms/ Degradation indicators	Monitoring systems	Mitigation actions
1	Calandria			
1. 1	Coolant tube (C.T.)	• Irradiation/ thermal cycling • Elongation of C. T. • C. T. sagging • Coolant and calandria tube contact • CT hydriding • Scratches on ID	• Creep measurement • Garter spring positioning by ECT • Sliver scrape sampling • PIE • H_2/D_2 assessment • CCTV inspection	• Creep adjustment • Relocate garter spring • Replacement of CT if H_2 pick-up limit exceed acceptable limit
1. 2	CT rolled Joint	• Thermal cycling and transients • Crash cooling/ Thermal fatigue • D_2O leak	• Annulus gas monitoring system • High humidity	• Gradual heating PHTS • Avoid crash cooling • Replacement of CT
2	End-shield	• Irradiation • Low temperature fatigue • Corrosion • Distortion of end-shield • Shift in NDTT	• Chemistry measurements • Temperature. gradient measurements between PHT and end shield • Monitor thermal cycles • Optical alignment check • Install sample coupon	• Gradual heating PHTS
3	Steam generator	• Thinning/ cracking due to SCC, IGSCC, corrosion • Denting fretting due to flow induced vibration and foreign materials	• Chemistry control of feed water and PHT • Estimation of leakage • Inspection for foreign material in shell and tube side • SG tube NDT	• Chemistry control • Lancing of tube sheet • Avoid low FW temperature • High flow blow-down • Condenser tube leak rectification

ANNEXURE-VIII (Contd.)

S. No.	Component	Degradation mechanisms/ Degradation indicators	Monitoring systems	Mitigation actions
3. 1	Feed water nozzle	•Thermal cycle fatigue	•Temperature. monitoring and control feed water temperature at the inlet nozzle	•Reduce period of low power operation
3. 2	SG tube sheet	•Corrosion product accumulation •Denting of tube O. D	•Visual inspection	•High flow blow-down •High pressure water jet lancing
4	PHT pump/motor			
4. 1	PHT pump mechanical seal	•Seal surface cracking •High seal leakage/ temperature. •Unbalance pressure at seal cavity	•Trending leak/ temperature, cavity pressure. •Leak rate testing of pressure reducing cell	•Replace/ refurbish seal faces
4. 2	PHT pump casing	•Erosion/ corrosion due to cavitation, vibration, fatigue	•UT for thickness measurement and radiography. •Visual inspection	•Repair eroded parts
4. 3	Horizontal joint bolts	•Thermal cycle fatigue, IGSSC due to chloride leaching from insulation	•Magnetic particle testing for bolts	•Change to metallic or mineral wool insulation
4. 4	Motor winding	•High winding and bearing temperature. •Insulation deterioration •Metal particles in bearing lubricating oil.	•Motor current •Bearing and winding temperature •Metal in bearing oil •Tan delta trending	•Winding and bearing maintenance •Reduce repeated starts •Avoid abnormal frequency operation.
5	Turbine			
5. 1	Turbine casing	•Distortion of casing •Corrosion at horizontal joint •Corrosion, erosion in casing	•Visual inspection of steam leaks •Casing top/ bottom temperature monitoring	•Low load/ abnormal frequency operation •Avoid load swings/ trip •Repair casing.

ANNEXURE-VIII (Contd.)

S.No.	Component	Degradation mechanisms/ Degradation indicators	Monitoring systems	Mitigation actions
5. 2	Rotor blades	• Thermal fatigue, corrosion/ erosion fatigue	• UT blade root • UT rotor discs • Vibration monitoring for unbalance and misalignment	• Inspect and repair lacing wire blade shroud
6	Condenser tubes	• Corrosion, erosion, SCC • Fretting • Scale deposit at I. D	• Condenser tube leak, • Ingress of CCW to condensate system • High temperature gradient across condenser	• Shell inspection and cleaning • Plugging of tubes • Chemical dosing • High pressure water jet cleaning
7	Containment	• Irradiation • High humidity and temperature • High strain levels during periodic testing • High electrical potential across pre-stressing cable	• Periodic leak test • Pipe and cable penetration leak test • MAL, EAL leak test • Ventilation inlet and exhaust damper leak test • Strain gauge • Half cell potentiometer	• Repair penetration leak and containment • Epoxy painting • MAL, EAL gasket replacement • Damper repairs

ANNEXURE - IX

(Refer Section 4.4.1)

AGEING MANAGEMENT PROGRAMME FOR STUCTURES, SYSTEMS AND COMPONENTS (BOILING WATER REACTORS)

S.No.	Component	Degradation mechanisms/ indicators	Monitoring Systems	Mitigation Actions
1. 1	Reactor Pressure Vessel (RPV)	• Irradiation due to neutron and gamma field resulting in shift in NDTT of RPV material	• Periodic testing of irradiation damage and evaluation of sample • Material installed in RPV inside the core • Radiography of vessel head weld • Periodic inspection of internals, core shroud and vessel internal surface with CCTV camera, for ultrasonic inspection of welds • Operating pressure hydro test during refuelling outage and visual inspection	• Avoid crash cool-down and scrams • Gradual raise in pressure and temperature as per NDTT requirement • Chemistry control of primary system
1. 2	BWR recirculation piping	• IGSCC • Thermal fatigue • Thermal embrittlement • Crevice corrosion	• UT of weld and heat affected zone • Radiography/UT • Acoustic emission • Core spray sparger flow test with open vessel condition • Inspection of FW sparger and core shroud for cracking • Inspection of weld as specified	• Chloride-free insulation for piping valves, pump casings to avoid IGSCC/ TGSCC • Avoid crash cool-down and scrams • Gradual raise in pressure and temperature as per NDTT requirement • Chemistry control of primary system • Normal and shock blow-down

ANNEXURE - IX (Contd.)

S. No.	Component	Degradation mechanisms/ indicators	Monitoring Systems	Mitigation Actions
1. 3	Dryer separator (D/S) assembly	• Thermal fatigue on hold down bolts • SSC/IGSCC • Corrosion • Disintegration of D/S elements	• Visual inspection under water during refueling outage by CCTV • Periodic steam dryness test	• Inspect/remove D/S as per approved procedure to avoid damage to hold down bolts • Primary/steam chemistry control • Avoid sudden load swing dumping incidents
2.1	SSG Tubing	• SSC (SS 304L) corrosion, fretting • Corrosion denting	• ECT of tubes primary/ secondary leak rate estimation • Monitor chemistry parameters • Visual inspection during water fill test	• Chemistry control of feed water system • Normal/shock blow down • Secondary cycle gradual loading to rated flow to avoid fretting • Tube sheet washing at high pressure water jet • Tube plugging as per fitness to service criteria • Remove tube based on leak/ ECT • Avoid CCW ingress and keep condensate demineraliser in service
2.2	Feed-water mozzles of SSG	• Thermal fatigue • Corrosion	• UT and radiography of welds • FW nozzle temperature gradient monitoring	• Avoid low feed water temperature for long period • Check SSG support and snubber • SSG nozzle replacement plan • Control water chemistry and chemistry treatment of FW.

ANNEXURE - IX (Contd.)

S. No.	Component	Degradation mechanisms/ indicators	Monitoring Systems	Mitigation Actions
3.1	Recirculation pump mechanical seal	• Seal leakage, cavity pressure/ temperature • Mating faces failure • Higher leakage through pressure reducing cell	• Surveillance of mechanical seal cavity pressure/ temperature and its trending • Leak rate monitoring and its trending • Visual inspection/repair during overhaul	• Crud control in primary system • Leak test of PRC for flow measurement • Replacement of mechanical seal
3.2	Pump casing	• Erosion of inner side of casing due to flow/ cavitation • IGSCC	• UT thickness	• Avoid 95^{0}C-100^{0}C temperature operation to control cavitation/ vibration • Chloride-free insulation
3.3	Casing bolts	• Pump vibration • Thermal fatigue • Visual leak check	• MPI of bolt for crack • Elongation / torqueing check while tightening	• Replacement of cracked bolts • Avoid thermal cycles • ISI of bolts as per PM
4	Main steam isolating valve	• Internal surface erosion due to stratified flow • Mating faces degradation due to closing impact • Stem scratches at OD • Gland packing deterioration/ leakage opening/closing time change due to elastomer degradation in pneumatic system elastomers	• Weekly closing/ opening timing test of MSIV • Containment leak rate daily monitoring and trending	• Replacement of elastomer component of pneumatic system • Valve inspection/ replacement of component • Repair casing eroded component • Trending MSIV close/open timing

ANNEXURE - IX (Contd.)

S.No.	Component	Degradation mechanisms/ indicators	Monitoring Systems	Mitigation Actions
5	Steam line from reactor to turbine	• Erosion at bends/ fitting in steam lines • Thermal cycling • Corrosion due to condensed steam • Distortion/ breakage of hangers/support	• UT thickness measurement at bends and fitting of steam lines • UT / RT of welds • Visual inspection of pipe hanger/ supports	• Adjustment of hangers/ supports • Draining of steam lines on start up after long outage • Trending thickness of steamline bends etc.
6.1	Primary containment (Dry well-D/W)	• Corrosion damage at internal/ outer surface of D/W plates and support structures. • High humidity and temperature	• UT thickness measurement of plate at embeded area • RT of weld joints on accessible area • Monitor D/W temperature of various locations identified as hot spot area	• Inspection of plate (external surface) by taking out sample along with concrete (core) • Painting of D/W internal surface and structure • Flow balancing of air and cooling water to D/W cooling fans and control water/ steam leakages
6.2	Dry well (D/W) (Integrated leak test)	• Unacceptable D/W leak rate conducted at specified period and test pressure	• Visual inspection of leakages at penetrations, isolation valves, equipment hatches, pipeline bellows, D/W head gaskets, and cavity bellows	• Replace elastomers • Carry out visual inspection of bellows and leak test of individual sub-components hatches, penetrations, seals, air locks etc.

ANNEXURE - IX (Contd.)

S.No.	Component	Degradation mechanisms/ indicators	Monitoring Systems	Mitigation Actions
7	Motor operated valves (MOVs)	• Shift in torque switches • Failure to close or open or hard to operate • Stem bending/ breaking • Gland packing leakage	• Signature analysis of MOV, MOTVAT system Manual operation to assess freeness of stem and visual inspection of gland packing	• Signature analysis at specified intervals to assess drift and initiate corrective action • Inspection of internals and lubrication of mechanisms as per PM. • Inspection of stem surface and replacement of gland packing as per PM
8	Nuclear instruments (SRM, IRM, PRM, LPRM)	• Degradation of detectors in presence of neutron flux • Cable and connector deterioration due to humidity, high temperature and radiation field	• Checking voltage v/s current calibration curve • IR resistance checks • Visual inspection of connectors.	• Replacement of detectors for SRM, IRM and LPRM as required based on detector/ sensor degradation. • Replacement of cables and connectors at scheduled intervals • Adjustment of locationand detectors or replacement for PRM • Upgradation of electronic system /technology for gain/signal amplification.

ANNEXURE - X

(Refer Section 4.5.2.5)

CASE STUDY OF RETROFIT/MODIFICATION

X.1 PHWR (Typical)

(i) Enmass coolant channel replacement was implemented based on material degradation (Zr-2 alloy replaced with Zr 2.5 Nb).

(ii) Retrofitting of ECCs system to mitigate LOCA in plants of earlier design.

(iii) Replacement of moderator heat exchangers due to tube material degradation.

(iv) Condenser re-tubing due to material degradation (admiralty brass replaced with 90 – 10 cupro-nickel material.)

(v) Replacement of obsolete control room monitoring / operator support system with computer-based system.

(vi) Upgradation of fire protection system (New FRLS type cables, cable routes provided with fire barriers and new fire detection and alarm system covering entire plant.)

X.2 BWR (Typical)

(i) Design modification of fuel support plugs-thermal sleeve holding device (reactor internals)

(ii) Deletion of two 6” dia. schedule 80 SS 316 bypass line and valve in reactor recirculation system due to IGSCC degradation and design modification.

(iii) Replacement of 8” dia. SS-316 schedule 80 pipe in reactor recirculation system due to IGSCC degradation with SS- 316 LN spool to reduce number of welds.

(iv) Modification of main steam line and turbine stop valve support due to crack initiated in support structure by excessive loading during turbine trip (design deficiency)

(v) Augmentation of online condensate demineraliser to improve water chemistry during condenser tube failure causing seawater ingress.

(vi) Modification of fuel pool cooling system to control chemistry parameter by introducing mixed bed demineraliser and filters.

(vii) Retrofitting of reactor vessel water level control system instrumentation with new design.

(viii) Replacement of containment multi-stage sump pump with improved single stage glandless pumps.

(ix) Augmentation and relocation of station lead acid power battery in air conditioned environment to avoid enhanced degradation.

ANNEXURE-XI

(Refer Section 4.6)

PLANT LICENCE RENEWAL CONSIDERATIONS (GENERAL)

XI.1 Reactors are generally designed for an operating life of about 40 years considering effects of irradiation, thermal cycling, pressure and pressure cycles, average temperature during operation, operational transients and various degradation mechanisms due to ageing and other deleterious effects. Even at the end of this period, SSC and IIS may have considerable potential to continue in service without jeopardising safety. Therefore it is necessary to examine in detail the economic and technical viability of extending the life of SSC by retrofitting, replacing, or using other techniques, as may be required and feasible. Economic issues are not addressed in this guide.

XI.2 Considerations for Licence Renewal.

XI.2.1 SSC and IIS are classified in section 5. 2 and Appendix-6 as category I, II and III for LM programme during operation. General list of SSC and IIS for BWR is given in Annexure-VIII. Operating organisation should initiate action for life extension, when an NPP has completed about three fourth of its operating life. This programme should be initiated in two phases: preliminary and comprehensive studies. During the preliminary phase, the 'fitness for service' of those SSC that primarily govern the feasibility of extending the life of NPP, should be identified and assessed. Regulatory requirements, for complying with current safety standards should be formalised.

XI.2.2 Having established the feasibility of extending the life of the NPP, in the second phase, detailed studies on the critical SSC identified in phase-I should be carried out. Balance SSC should then be evaluated for establishing the requirements for retrofitting or replacement. The schedules for detailed studies on critical SSC, repair and or retrofitting and submission of documents for regulatory consent should be finalised.

XI.3 Prioritisation of SSC for Evaluation in Phase-I

XI.3.1 Apart from national and international guidelines, the following attributes should be considered for prioritising SSC for detailed evaluation:

(i) Cost for replacement or retrofitting; impact on plant availability for replacement or retrofitting.

(ii) Radiation exposure to personnel involved.

(iii) Regulatory requirements.

(iv) Seismic qualification requirements.

(v) Replacement precedent including national and international experience and feedback.

(vi) Modes of failure related to service conditions.

(vii) Consequences of failure on plant and public safety.

(viii) Impact on plant operation.

(ix) Modifications required, to support the plant life management programme; for monitoring degradation and other surveillance systems.

(x) Generic issues with safety significance.

(xi) Changes in design and safety criteria or material, conceptual or engineering change, manufacturing errors, poor quality control and inspection practices during manufacture and installation errors.

(xii) SSCs due for replacement may be excluded from this exercise.

XI.3.2 Available reference materials for life extension study should be collated for establishing the data-base for stressors, degradation sites and failure mechanisms, along with documented in-house and international experience. Such data should include design specifications, manufacturing and construction history, safety analyses reports, operation and maintenance history, pre-service and in-service inspections, periodic safety reviews, peer and regulatory reviews.

XI.3.3 Evaluation should then be carried out on major prioritised SSC for assessing the feasibility for life extension. Prioritisation results should be compared with available international data on similar plants.

XI.3.4 Requirements for obtaining regulatory consent for life extension should be finalised with the Regulatory Body.

XI.4 Comprehensive Evaluation in Phase-II

XI.4.1 After establishing the feasibility for life extension of the NPP, comprehensive ageing evaluation should be carried out for all SSC, safety support systems, control, instrumentation and power cables and balance of plant.

The evaluations should include the following:

i) Overall requirements:

(a) Quality standards and records.

(b) Design bases for protection against natural phenomena.

(c) Fire protection.

(d) Environmental and dynamic effects design bases.

(e) Sharing of SSC.

(f) Plant layout, vis a vis fire protection, access control, segregation of sensitive equipment, and plant security.

(ii) Assessment of residual life.

(iii) Impact of various degradation mechanisms (refer section 4).

(iv) Failure analyses and modes of failure.

(v) Assessment for 'fitness for service' as per applicable codes and standards.

(vi) Correspondence with applicable current design criteria, codes and standards, including for the following systems:

(a) Reactor shut down system.

(b) Secondary shut down system.

(c) In-core and out of core nuclear instrumentation.

(d) Emergency core cooling and post incident cooling systems.

(e) Biological shield/ calandria vault cooling.

(f) Emergency power supply systems.

(g) Service water system.

(h) Spent fuel cooling system.

XI.4.2 Regulatory Consent for Plant Life Extension.

XI.4.2.1 The work on repairs, retrofits, replacements, and installation of modified or new systems to cater to current regulations, may be spread over a time period. After completion of the detailed evaluation and assessing the viability of NPP life extension, the RO should draw up a detailed implementation programme.

XI.4.2.2 The results of the detailed evaluations and the conclusions arrived at along with the implementation programme for life extension should then be submitted to Regulatory Body for their review and approval.

ANNEXURE-XII

(Refer Section 4.6)

RELICENSING OF NUCLEAR PLANTS BUILT TO EARLIER STANDARDS

XII.1. Assessment Phase

XII.1.1 Assessment of Level of Safety Culture.

- Assess if policy statement is up-to-date and the same is communicated to all concerned.
- Assess if the management structure is up to date and functional.
- Assess if allocation of resources (in training, qualification, retraining, etc.) is adequate.
- Assess the level of self-regulation of safety.
- Assess if delegation of responsibilities is adequate.
- Assess the training and qualification programme and its effectiveness.
- Assess the work of audit and review.

XII.1.2. Assessment of Status of Defence in Depth in NPP

- Assess the extent to which any actual abnormal operation could have been prevented.
- Assess the extent to which abnormal operations have been detected and controlled.
- Assess how design basis events have been detected and controlled.
- Assess if severe events (accidents) have occurred and actions taken thereon.
- Assess state of radiological control (personnel exposures, environmental releases, ALARA).

XII.1.3 Deterministic Assessment.

- Assess modification to plant layout, systems and structures and their adequacy.
- Assess in-service inspection reports on SSC.
- Assess state and adequacy of condition monitoring of items important to safety.

- Assess adequacy of performance tests of items important to safety.
- Assess other inspection and test results on items important to safety.

XII.1. 4 Probabilistic Assessment .

- Assess how any postulated initiating event (PIE) experienced in the NPP compare with design basis PIE.
- Assess how control and protection systems have responded to PIEs.
- Assess reliabilities of control and protection systems during operations and testing.
- Assess how core damage frequencies, based on plant performance, (i.e. on actual PIEs, actual reliabilities of control and protection systems, etc.) compare with analysis data.

XII.2. Corrective Measures

Based on the above assessments corrective measures for all deficiencies/ improvements will be drawn up with appropriate assessment of cost and time.

XII.3. Prioritisation

Corrective measures will be ranked high, medium or low depending upon their relative contributions to core damage frequency (CDF) and submitted to Regulatory Body for approval. Cost benefit and the effects of improvement on other systems and over- all plant perspective will be kept in mind during the decision making process.

XII.4. Action Plan

The RO, based on the above, will produce time bound implementation plan which should be acceptable to the Regulatory Body. Based on satisfactory completion of action plan Regulatory Body will consider re-licensing authorisation .

ANNEXURE-XIII

(Refer Section 4. 6. 4)

RELATIVE IMPORTANCE OF ISSUES AND IMPROVEMENT PRIORITIES

Fussell-Vesely Importance of Issue (Unit)	**Mean Core Damage Frequency***			
	10^{-5}	10^{-4}	10^{-3}	10^{-2}
$\geq$ 0.3	LOW	MEDIUM	MEDIUM	HIGH
0.1	LOW	LOW	MEDIUM	HIGH
0.01	DROP	LOW	LOW	MEDIUM
0.001	DROP	DROP	LOW	LOW
< 0.001	DROP	DROP	DROP	LOW

Notes: Once the classification is done, the RO shall recommend action plan to the Regulatory Body for approval. However high and medium priority issues shall be attended. This is for the purpose of illustration only. Appropriate goals for a particular plant should be established by the Regulatory Body.

Classification	**Range of Core Damage Frequency* (CDF)**
HIGH	CDF $\geq 10^{-3}$
MEDIUM	10^{-3} > CDF $\geq 10^{-4}$
LOW	10^{-4} > CDF $\geq 10^{-6}$
DROP	CDF < 10^{-6}

* Events per reactor year

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NOTES

AERB SAFETY GUIDE NO. AERB/NPP/SG/O-14

Published by **:** Atomic Energy Regulatory Board
Niyamak Bhavan, Anushaktinagar
Mumbai - 400 094.
INDIA

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