GUIDE NO. AERB/NPP/SG/O-2

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GOVERNMENT OF INDIA

A E R B S A F E T Y G U I D

IN-SERVICE INSPECTION
OF
NUCLEAR POWER PLANTS

ATOMIC ENERGY REGULATORY BOARD

AERB SAFETY GUIDE NO. AERB/NPP/SG/O-2

IN-SERVICE INSPECTION
OF
NUCLEAR POWER PLANTS

Atomic Energy Regulatory Board
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Orders for this guide should be addressed to:

Administrative Officer
Atomic Energy Regulatory Board
Niyamak Bhavan
Anushaktinagar
Mumbai-400 094
India

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 of ensuring 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. These documents cover aspects such as siting, design, construction, operation, quality assurance, decommissioning and regulation of nuclear and radiation 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, systems, structures 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 feedback from users as well as new developments in the field.

The 'Code of Practice on Safety in Nuclear Power Plant Operation' (AERB/SC/O) states the minimum requirements for ensuring adequate safety in plant operation. This safety guide is one of a series of guides, which have been issued or are under preparation, to describe and elaborate the specific parts of the code.

One of the prerequisites for operating a nuclear power plant is to establish and implement the in-service inspection programme to examine plant structures, systems and components for detecting and identifying possible deterioration and take remedial action. This guide provides necessary information to assist organisations/personnel participating in the development and implementation of the in-service inspection programme for nuclear power plants to meet the requirements specified in the 'Code of Practice on Safety in Nuclear Power Plant Operation' (AERB/SC/O). The guidelines given herein cover the mechanical components including the coolant channels of pressurised heavy water reactors. Some of the important aspects that are required to be met during design stage to facilitate in-service inspection have also been covered in this guide. However, a separate guide on 'Design for In-Service Inspection in Pressurised Heavy Water Reactor', AERB/SG/D-17 is currently under preparation, which will cover the design related aspects in more detail.

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, references/bibliography and lists 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. 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 Atomic Energy Regulatory Board, Bhabha Atomic Research Centre, Nuclear Power Corporation of India Limited and other consultants. It has been reviewed by the relevant AERB Advisory Committee on Codes and Guides and the Advisory Committee on Nuclear Safety.

AERB wishes to thank all individuals and organisations 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.

(Suhas P. Sukhatme)
Chairman, AERB

DEFINITIONS

Acceptable Limits

Limits acceptable to the regulatory body for accident condition or potential 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.

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.

Assessment

Systematic evaluation of the arrangements, processes, activities and related results for their adequacy and effectiveness in comparison with set criteria.

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 site personnel, the public and the environment against undue radiation hazards.

Audit

A documented activity performed to determine by investigation, examination and evaluation of objective evidence, the adequacy of, and adherence to applicable codes, standards, specifications, established procedures, instructions, administrative or operational programmes 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.

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 in accordance with design specifications and found to have met the performance criteria.

Competent Authority

Any official or authority appointed, approved or recognised by the Government of India for the purpose of the rules promulgated under the Atomic Energy Act, 1962.

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.

Documentation

Recorded or pictorial information describing, defining, specifying, reporting or certifying activities, requirements, procedures or results.

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.

Flaw

An imperfection, discontinuity, irregularity or fault in the material of a component such as a crack, inclusion, or porosity, lack of penetration, lack of fusion, etc.

Indication

The response or evidence from an examination that requires interpretation to determine relevance.

In-Service Inspection (ISI)

The inspection 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

The items which comprise:

- those structures, systems, equipment and components whose malfunction or failure could lead to undue radiological consequences at plant site or off-site;
- those structures, systems, equipment and components which prevent anticipated operational occurrences from leading to accident conditions;
- those features which are provided to mitigate the consequences of malfunction or failure of structures, systems, equipment or components.

Licence

A type of regulatory consent, granted by the regulatory body for all sources, practices and uses for nuclear facilities involving the nuclear fuel cycle and also certain categories of radiation facilities. It also means authority given by the regulatory body to a person to operate the above said facilities (see Licensed Person and Licensed Position).

Licensed Person

A person who has been licensed to hold certain licensed positions of a nuclear power plant after due compliance with authorised procedure of certification by the regulatory body.

Licensed Position

A position, which can be held only by persons certified by the regulatory body or a body, designated by it.

Normal Operation

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

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.

Nuclear Safety

The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of site personnel, the public and the environment from undue radiation hazards.

Objective Evidence

Term used in context of quality assurance, qualitative or quantitative information, record or statement of fact pertaining to quality of an item or service which is based on observation, measurement or test and which can be verified.

Operating Organisation

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

Operating Personnel

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

Operation

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

Operational Records

Documents such as instrument charts, certificates, logbooks, computer printouts and magnetic tapes, made to keep objective history of the operation of nuclear/radiation facility.

Operational States

The states defined under 'Normal Operation' and 'Anticipated Operational Occurrences'.

Plant Management

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

Potential

A possibility worthy of further consideration for safety.

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

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

Regulatory Body

(See 'Atomic Energy Regulatory Board').

Reliability

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

Responsible Organisation (RO)

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

Safety Limits

Limits upon process variables within which the operation of the facility has been shown to be safe.

Safety Report

A document provided by the applicant or licensee to the regulatory body, containing information concerning the facility, its design, accident analysis and provisions to minimise the risk to the public and to the site personnel.

Safety System

System important to safety, and provided to assure that under anticipated operational occurrences and accident conditions, the safe shutdown of the reactor followed by heat removal from the core and containment of any radioactivity, is satisfactorily achieved (Examples of such systems are: shutdown systems, emergency core cooling system and containment isolation system). It is also called the 'safety critical system'.

Severe Accident

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

Site

The area containing the facility defined by a boundary and under effective control of the facility management.

Site Personnel

All persons working on the site, either permanently or temporarily.

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, viz. monitoring, verifying, checking including in-service inspection, functional testing, calibration and performance testing performed to ensure compliance with specifications established in a facility.

Technical Specification 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 facilities. It is also called as "Operational Limits and Conditions".

CONTENTS

1. INTRODUCTION

1.1 General

1.1.1 During the operating life of a nuclear power plant (NPP), its components might be exposed to influences whose individual or combined effect cannot be fully predicted for the operating life of the plant with accuracy level desirable for nuclear safety. The most important influences are stress, temperature, irradiation, hydrogen absorption, corrosive attack, vibration and fretting, all of which depend upon time and operating history. These influences may result in changes in material properties such as embrittlement, fatigue, formation and/or growth of flaws and ageing.

1.1.2 The In-Service Inspection (ISI) involves periodic examination of components of NPP during its lifetime. The examinations required to determine the health of components form a part of ISI programme. The results of the Pre-service Inspection (PSI) of the components prior to the start of operation of the plant establish the baseline data required for comparison during subsequent ISI. PSI should therefore be carried out to collect baseline data before startup of the plant, to ensure that the components are of acceptable quality as per applicable standards/codes prior to the start of plant operation.

1.1.3 The extent of the periodic ISI is governed by criteria that ensure detection of unacceptable deterioration of the component(s) during the operating life of NPP. The components, whose unacceptable deterioration/failure could lead to or involve impairment of a safety system's functional capability to perform assigned safety functions and meet the design requirements or cause major damage of a process system, are required to be included within ISI programme. The ISI programme should include periodic inspection of fluid-retaining components and piping of systems related to heat transport from reactor, reactor shutdown, decay heat removal and all other systems and components whose failure could jeopardise the functioning and integrity of safety systems in the plant.

1.1.4 Both PSI and ISI should also be done of components when such components or materials are used beyond conditions1 of proven experience and do not otherwise fall under the purview of PSI/ISI as per the governing selection criteria. For example, the pressure tubes used in PHWR fall under this category and their dimensional, and volumetric examinations should be included in the periodic ISI programme.

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[1] Level A, B, C and D conditions: Service conditions as specfied in the relevant design documents lor which level A, B, C and D service limits are designated as defined in Section III of the ASME Boiler and Pressure Vessel code. The same has been reproduced in Annexure-I.

Similarly, volumetric examination of steam generator tubes, supplementary thickness checks for fuel channel feeder pipes and checks to record garter spring locations should be included in the ISI programme.

1.1.5 The interval between every ISI depends on the applicable code requirements, rate of change of operating conditions based on the operating history, evidence from earlier examinations and considerations of any known abnormality in operation.

1.1.6 ISI programme involves several methods of testing (including leakage testing of systems) at proper time intervals and administrative measures necessary thereof. The design of plant layout shall provide for accessibility to components to facilitate ISI. The shielding design should keep radiation exposure of ISI personnel as low as reasonably achievable (ALARA). The extent and stringency of ISI programme should be commensurate with the significance of the safety systems, safety-related systems and components following graded approach.

1.1.7 The ISI programme should be set on a firm and rational basis by considering not only the consequence of failure of the components but also the factors that determine the likelihood of such failures. Reliability of items and their importance measures obtained from generic and/plant specific Level-1 PSA results, when available, can be considered to form such a basis (Chapter on risk-based inspection in 'Probabilistic Safety Assessment Guidelines', AERB NF/SM/O-1 may be referred to for guidance on this aspect).

1.1.8 All inspection data should be analysed and necessary corrective actions commensurate with applicable codes and guides should be taken in a timely manner and reported to appropriate authorities.

1.2 Objectives

The main objectives of this safety guide are:

(a) to highlight the requirements for the responsible organisation to develop and establish an ISI manual incorporating the ISI programme from the design stage onwards to enable satisfactory implementation of the programme at the NPP;

(b) to focus on the importance of the PSI to be carried out prior to commencement of operation so as to provide baseline data for eventual comparison with the corresponding data of indications during the operating life of NPP to determine deteriorating trend, if any, and take timely remedial action;

(c) to emphasise on the operating organisation and the plant management the absolute necessity to conduct ISI at required intervals and extent during the entire operating life of the plant, including fresh PSI for structures, systems and components in the event of their replacement and modifications;

(d) to give guidelines on the formulations and implementation of PSI/ISI using approved techniques, qualified staff and maintenance of inspection records; and

(e) to indicate the positive contribution of in-service inspection in enhancing nuclear safety at the NPP by helping to prevent equipment/ component failures likely to be caused by deterioration during the operating life and to enhance the confidence in continued safe operation of the plant.

1.3 Scope

1.3.1 This safety guide outlines the provisions relevant to ISI of safety-related systems and pressure-retaining components including their supports. This guide also covers the classification of areas subject to inspection, responsibilities, provision for access, inspection techniques and procedures, personnel qualifications, frequency of inspection, documentation, records, evaluation of inspection results, disposition of non-conformances, and repair requirements.

1.3.2 This guide is applicable to all stationary land-based NPPs of PHWR and BWR types having thermal neutron reactors in India. The aspects specific to ISI of BWR have been outlined in Annexure-II. Though this guide is not intended for facilities other than Nucler Power Plants and to non safety-related items of NPPs, the principles stated herein can be employed as applicable to such items in NPPs and to other nuclear facilities, including research reactors.

1.3.3 This guide does not cover the ISI of equipment/components associated with electrical, control & instrumentation, and of civil structures since these are monitored by the surveillance programme. The functional testing aspects are separately covered in AERB safety guide on 'Surveillance of Items Important to Safety in Nuclear Power Plants' (AERB/SG/O-8).

1.3.4 The inspection requirements covered in this safety guide shall not be deemed as restricting and the inspection programme may include additional items found necessary by the state-of-the-art and operating experience.

2. GENERAL REQUIREMENTS FOR PSI AND ISI PROGRAMME

2.1 PSI/ISI Programme Manual and its Format

2.1.1 A comprehensive PSI/ISI manual, specific to each station, shall be prepared by the responsible organisation and issued for implementation to the station. The manual shall include the following:

(a) philosophy of PSI/ISI programme;

(b) responsibilities for implementation of the programme;

(c) list of examination areas;

(d) methods of examination;

(e) applicable codes and standards;

(f) extent of examination;

(g) examination interval;

(h) reporting of data;

(i) examination personnel including requirements of level of inspectors; and

(j) analysis of data.

2.1.2 The flow chart of activities to be carried out as part of PSI/ISI programme is given in Fig.1 of Appendix-I.

2.1.3 The manual should be available prior to start of PSI to enable appropriate collection of PSI data for the areas to be subjected to ISI later.

2.1.4 In view of continuing advancement in technology and the state of the art, the ISI programme manual should be reviewed and updated periodically at least once in 5 years.

2.2 Design Considerations for ISI

To effectively implement the PSI programme during construction and ISI programme during operating phase, the considerations that are needed to be taken into account at the design stage itself shall include the following:

(a) accessibility to areas and feasibility of the examination of components;

(b) shielding consideration for radiation levels to meet ALARA criteria;

(c) removal, storage and installation of structural members, shielding components, insulating materials and other equipment and components as necessary to perform the required examinations and tests;

(d) installation of supports, handling machinery, fixtures, platforms etc. to facilitate removal, disassembly, reassembly, placing and mounting of inspection equipment and/or probes;

(e) provision for conducting examinations by alternate methods in case of their need when indications are revealed;

(f) provision of facilities for decontamination of systems, equipment and working areas;

(g) provision to enable examinations remotely to reduce radiation exposure;

(h) aspects such as weld configuration, surface finish of components, crud or corrosion product build-up, and selection of materials;

(i) provision for ready detachability and temporary storage of heat insulation cover during inspection;

(j) provision for required repair or replacement of systems or components due to observed structural defects or flaw indications;

(k) provision of test coupons for assessing ageing effects of various operating conditions such as load, temperature and radiation on important material properties [e.g. Nil Ductility Transition Temperature (NDTT), strength and corrosion rate]. These ageing effects are creep, fatigue, radiation embrittlement, etc.;

(1) provision of power supply for PSI/ISI equipment and instruments that are required to be used inside reactor building for inspection; and

(m) provision of entry for power supply and instrumentation cables for inspection equipment and instruments to be used inside the reactor building, for which there may arise a need for drawing cables from outside reactor building, to avoid the possibility of containment bypass during inspection.

2.3 Considerations during Construction and Commissioning

Considerations shall include the following:

(a) geometry, configuration profile and surface finish of weld joints; and

(b) need for any change/deviation from existing design and design provisions for PSI/ISI.

2.4 Availability of all 'As Built' Data of Components

2.4.1 As built manufacturing and construction data shall be compiled for all the components which are required to be subjected to PSI/ISI. The data to be made available and maintained permanently at site along with the component shall include the following:

(a) as built drawings of the components and their manufacturing details, such as materials used, location of welds and deviations from original design;

(b) records of inspection, examination and test including deviations, if any. from the recommended/standard procedures during and after manufacture of components with approvals of deviations; and

(c) calibration standards for tests using eddy current, ultrasonic or any other examination techniques as applicable as per the ISI requirements of the component.

2.4.2 Scheduling of PSI

2.4.2.1 PSI shall be performed prior to commencement of operation either:

(a) after the component hydrostatic pressure test but before start up of the reactor; or

(b) before the component hydrostatic pressure test provided that a confirmatory examination as required by subsection 2.4.3 is performed after the component hydrostatic pressure test and the results indicate no significant change.

2.4.2.2 The PSI shall be carried out for all components (100%) which are subjected to ISI.

2.4.2.3 When a component is repaired or replaced, an initial examination of that component shall be carried out.

2.4.2.4 Shop and field examinations carried out during construction may be considered as part of the PSI provided confirmatory examination after the hydrostatic test is conducted as per subsection 2.4.3 below and method of ISI to be followed is comparable with the shop/field examinations.

2.4.3 Confirmatory Examination

2.4.3.1 Confirmatory examination, whenever required, by subsection 2.4.2.1 and 2.4.2.4 above, shall employ, as far as possible, the same methods, techniques and types of equipment and tools of comparable accuracy as those used for the PSI and preferably by same personnel, if possible.

2.4.3.2 Areas to be examined shall include:

(a) Piping

(i) At least 10% of the total area to be examined for ISI of the system.

(ii) All the areas for which recordable level (>20% of amplitude for reference standard) indications[2] were detected earlier.

(b) Vessels

(i) The full length of all major nozzle welds shall be inspected.

(ii) The most significant indications detected previously in the longitudinal and circumferential joints, such that at least 10 PSI/ ISI areas (or all, if fewer than 10 exist) per component or 10% of the indications (starting from the most severe indication in descending order), whichever is greater, shall be examined.

(c) Pumps

(i) At least 10% of all the pumps in each category for a reactor unit shall be examined.

(ii) The examination shall include all category A and B (see subsection 4.1.6.2) pressure-containing welded joints and internal surfaces of pump components designated as category A and B examination areas.

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[2]Recordable Level Indication is the level of indication for an individual NDT method at and above which the observed indications are required to be recorded and below which the indications are considered as absent and ignored. The level is decided by the utility and specified in relevant NDT procedures.

(d) Valves

(i) At least 10% of all the valves for a reactor unit shall be inspected.

(ii) The valve inspection shall include all category A & B pressure-containment weld joints and internal surfaces of valve components designated as category A & B inspection areas.

(e) Supports

At least 10% of all the supports of equipment, vessels, pipe, etc. (including snubbers) shall be examined.

(f) Steam Generator Tubes

(i) At least 10% of the tubes selected for ISI in each steam generator, including the tubes with the most significant indications detected previously, shall be examined.

(ii) Tubes to be inspected should be selected in accordance with the requirements mentioned in 13.1.2.

(g) Heat Exchanger Tubes

Inspection of 100% of the tubes is recommended in view of observed frequent failures of heat exchanger tubes.

(h) Coolant Tubes

The examination requirements for coolant tubes are given in subsection 13.2.

2.5 Examination Procedures

2.5.1 The PSI and ISI shall be carried out in accordance with documented and approved procedures.

2.5.2 The examination methods and techniques used shall comply with the requirements of the ASME Boiler and Pressure Vessel Code on ISI (Section-XI) and Non Destructive Examination (Section-V).

2.5.3 Procedures that deviate from the above requirements shall be submitted to the authority specified in the documented procedures as mentioned above in subsection 2.5.1 and approval obtained before the commencement of inspection.

2.5.4 For an examination method not included in the above mentioned requirement in subsection 2.5.2 the procedure shall employ the current practice that is consistent with the state of the art for that method and for the component or system to be examined.

2.5.5 For ISI of all critical components with complex geometry (such as coolant channel garter springs and the gap between PT & CT), mock up trials shall be carried out to qualify the ISI procedures, inspection personnel and equipment.

2.5.6 The specified calibration checks should be carried out using the actual system employed under site conditions, where calibration in laboratory conditions using reference/calibration standards differ from actual site conditions. These differences may arise due to use of different/modified systems suitable for remote operation and associated specific instrumentation, such as use of long bunched cables, intermediate stage amplifiers, and due to differences in surface condition, temperature and pressure of component, etc.

2.6 Staffing, Training and Qualification of Personnel

2.6.1 Adequate number of competent inspection personnel shall be deployed to carry out PSI /ISI. A certificate, wherever mentioned, means a written testimony of qualification of the inspection personnel.

2.6.2 Responsibilities of inspection personnel shall be as follows:

(a) witness or otherwise verify all the examinations and also make any additional examinations, if necessary, to ensure that all applicable requirements have been met,

(b) assure that the non destructive examination methods used adopt the techniques as applicable,

(c) assure that the examinations are performed in accordance with applicable documented qualified procedures and only by qualified personnel,

(d) certify the examination records only after satisfying himself that all the requirements have been met and that the records are correct, and

(e) keep himself adequately trained and qualified to an appropriate level and acceptable standards to be able to perform in a competent manner

2.6.3 Documented evidence shall be provided to ensure that personnel qualified as stated in 2.6.4 have performed the examinations.

2.6.4 Qualifications

This section specifies the minimum requirements that inspection personnel for the three levels of qualification should fulfil to perform inspections, examinations and tests. The requirements for each level are limiting with regard to functional activities but not with regard to organisational position or professional status.

(a) Capability Requirement (See Appendix-II)

(i) Capability of Level I Qualified Person

A Level I qualified person shall have capability to perform inspections, examinations and tests in accordance with documented procedures.

Capabilities with regard to associated inspection procedures, tools, measuring/test equipment, etc. shall include the following:

- familiarity with the tools and equipment to be employed and demonstrated proficiency in their use;
- capability to assure correct calibration status of measuring/ test equipment;
- capability to assure proper condition for use; and
- capability to assure approved status of the procedure for inspection, examination and tests.

(ii) Capability of Level II Qualified Person

A Level II qualified person shall have all the capabilities of a Level I qualified person for inspection, examination or test category or class in question.

In addition, a Level II qualified person shall have capability to:

- plan for inspections, examinations and tests;
- setup tests including preparation and setup of related equipment as appropriate;

- supervise or maintain surveillance over inspections, examinations and tests;
- supervise and certify lower level personnel; and
- report inspection, examination and test results.

(iii) Capability of Level III Qualified Person

A Level III person shall have all the capabilities of Level II person for inspection, examination or test category or class in question.

In addition a Level III qualified person shall have capability to:

- devise an inspection procedure and suggest alternate, complementary procedures;
- evaluate adequacy of specific programme for training and test personnel for inspection, examination and test; and
- write appropriate test procedures and suggest complementary techniques.

(b) Education and Experience Requirements

The following shall be the minimum requirements of education and experience of personnel for each qualification level. Other factors, which may demonstrate capability in a given job, are previous performance or satisfactory completion of testing.

(i) Level I

Individuals holding Level-I certificate in NDT from either a national or an international certification authority in accordance with IS-13805-1993 or Practice No. SNT-TC-1A of ASNT (USA).

(ii) Level II

Individuals holding Level-II certificate in NDT from either a national or an international certification authority in accordance with IS-13805-1993 or Practice No. SNT-TC-1A of ASNT (USA).

(iii) Level III

Individuals holding Level-III certificate in NDT from either a national or an international certification authority in accordance with IS-13805-1993 or Practice No. SNT-TC-1A of ASNT (USA).

(c) Performance Requirement

Persons who are assigned the responsibilities and authority to perform 'functions at Level I, II or III', shall have, as a minimum, the level of capabilities shown in Appendix-II.

(d) Requirement for Records

A filing system of records of personnel qualifications shall be established and maintained by employer.

(e) Requirement for Determination of Capability for Certification

The capabilities of a person for certification shall be determined by appropriate evaluation of his education, experience, training, test results or capability demonstration.

For this purpose the candidate shall be required to have passed both written and practical examinations conducted by a national body or, its authorised representative, with written certificate issued to this effect.

(f) Requirement for Written Certificate of Qualification

The qualification of personnel shall be certified in writing in an appropriate form including the following information:

(i) Identification of person being certified

(ii) Date of certification and date of certificate expiry

(iii) Level of capability

(iv) Activities certified to perform

(v) Basic criteria for certification

- records of education, experience and training
- test results
- results of capability demonstration

(vi) Results of periodic evaluation

(vii) Results of physical examination, when required

(viii) Signature and seal of body issuing the certificate.

(g) Requirement for Reevaluation of Performance

The job performance of inspection, examination and testing personnel shall be reevaluated at periodic intervals not exceeding 3 years. Reevaluation shall be by evidence of continued satisfactory performance or redetermination of capability. During this evaluation or, at any time, if it were determined by the responsible organisation3 that the capabilities of an individual are not in accordance with the qualifications specified for the job, that person shall be removed from the activity until such time as the required capability has been demonstrated.

Any person who has not performed inspection, examination or testing activities in his qualified area for a period of one year shall be reevaluated by redetermination of required capability.

[3] The responsible organisation shall identify any special physical characteristics needed in the performance of each activity. Personnel requiring these characteristics shall have themselves verified by examinations at intervals which do not exceed one year.

3. EQUIPMENT, METHODS AND TECHNIQUES

3.1 Equipment

3.1.1 All equipment used for examinations and tests shall be of acceptable quality, range, performance characteristics and accuracy in accordance with applicable standards.

3.1.2 All equipment together with the accessories shall be calibrated as stipulated in governing codes/standards. The equipment shall be properly identified with calibration records. Validity of the calibrations shall be verified regularly in accordance with Quality Assurance (QA) programme.

3.1.3 Reference blocks made to acceptable standards shall be used for calibration. If such standards for calibration are not established, these blocks shall be of identical material and surface finish and be subjected to the same fabrication (construction) conditions as the component being examined. The same reference blocks as used during manufacture and for PSI should be used for subsequent ISI wherever practical.

3.1.4 Reference specimen shall contain discontinuities and conditions that are comparable to existing or anticipated flaws or conditions. It shall be used to demonstrate the ability of the inspection system when the calibration specimen is not adequate for detection or evaluation of such flaws.

3.2 Methods and Techniques

3.2.1 General

Methods and techniques for the examinations shall be in accordance with requirements laid down in PSI/ ISI programme manual. The examinations are categorised as visual, dimensional, surface, volumetric and component integrity. Examples of the components and methods of examination normally considered for the ISI programme are given in Appendix-III.

3.2.2 Visual Examination

(a) a visual examination is used to provide information on general condition of a part, component or surface including such conditions as scratches, wear, cracks, corrosion or erosion on the surface: or on evidence of leaking. Optical aids such as television cameras, binoculars, boroscope, fibroscope and mirrors may be used.

(b) surface replication as a visual examination method is acceptable, provided that the surface resolution is at least equivalent to that obtainable by the visual observation.

(c) visual examination that requires clean surface or decontamination for valid interpretation of results shall be preceded by appropriate cleaning processes.

3.2.3 Dimensional Examination

The dimensional examination includes inspection for determining size, configuration, distortion, wear, alignment, corrosion and erosion by methods such as direct measurement (e.g. scale, micrometer, vernier callipers, gauges) and indirect measurement (e.g. theodolite, ultrasonic and other electronic methods).

3.2.4 Surface Examination

Surface examination is undertaken to delineate or verify the presence of surface or near-surface flaws or discontinuities. It is conducted by magnetic particle, liquid penetrant, eddy current or electrical contact method.

3.2.5 Volumetric Examination

(a) Volumetric examination is undertaken for the purpose of indicating the presence, depth or size of a subsurface flaw or discontinuity and involves radiographic, ultrasonic or, for tubing, the eddy current techniques.

(b) Radiographic techniques, employing penetrating radiation such as X-rays, gamma rays or thermal neutrons are used with appropriate image-recording devices to detect presence of flaws and to establish their dimensions and nature.

(c) Appropriate ultrasonic testing method is the most common method used to establish both the length and depth of flaws. Eddy current examination as well as ultrasonic testing are normally applied to tubing and tubular configurations to establish the existence and depth of flaws.

3.2.6 Integrative Examination

Integrative examinations are carried out for monitoring overall component integrity (e.g. leak detection acoustic emission and strain measurement).

3.2.7 Inspection of Component Integrity

The inspection of component integrity includes examinations for assessing design adequacy by evaluating strain, stress concentrations, discontinuities and distortion. Techniques such as holographic interferometry, X-ray diffractometry, infrared radiography, leak detection and strain gauging are used for the purpose.

3.2.8 Hydro-test for Pressure Retaining Components

Hydro-test is done for evaluation of the integrity of pipelines and pressure-retaining components. Hydrostatic leak testing is used to test components for leaks by pressurising them inside with a liquid. This testing method can be used on piping, tanks, valves and containers with welded or fitted sections.

3.2.9 Alternative and Complementary Examinations

Alternative examination methods or a combination of methods, or newly developed techniques may be substituted, provided the results yield demonstrated equivalence or superiority and results from both methods are comparable. If necessary, certain complementary examination may be performed to authenticate/verify the examination results.

3.3 Test Requirements

The test requirements for the pressure-retaining items are given in Annexure-III.

4. CRITERIA FOR SELECTION OF ITEMS AND EXTENT OF EXAMINATIONS

4.1 Criteria for Pressurised Heavy Water Reactors (PHWRs)

4.1.1 Typical Criteria for Selection

4.1.1.1 The systems and components, piping thereof (including supports), subject to inspection shall include the following or portions thereof:

(a) Pressure boundary of reactor coolant or any other systems whose failure may result in a significant release of radioactive substances.

(b) Systems essential for safe reactor shut down and/or safe cooling of nuclear fuel in the event of process system failure.

(c) Other systems and components whose dislodgement or failure may put in jeopardy the integrity of the system mentioned in (a) or (b) above, or both, such as garter springs, moderator inlet/outlet manifold, reactor coolant pump flywheel mounting on motor shaft.

4.1.1.2 Following system boundaries are required to be subjected to inspection:

(a) The inspection requirements shall apply to the fluid boundary portion of all components and piping and to their supports.

(b) The fluid boundaries referred above shall be considered as follows:

(i) For systems containing nuclear fuel or for systems connected to such systems, the fluid boundary subjected to inspection shall include all portion(s) which do(es) not have

- two additional barriers between fluid boundary and sheathing of nuclear fuel; or,
- two additional barriers between fluid boundary and the outside atmosphere; or,
- one barrier between fluid boundary and the sheathing of the nuclear fuel and the other barrier between it and outside atmosphere.

(ii) The number of barriers mentioned in item (i) above may be determined as follows:

- Metal boundary (e.g. vessel wall) - equal to one barrier.
- Valve - manual, automatic, or remotely controlled that remains closed during normal operating conditions equal to one barrier.
- Valve - self-closing, that may be open during level A operating conditions (see Annexure-I)- equal to half barrier.
- Valve - manual, that remains open during normal operating conditions should not be considered as barrier, unless shown that it's closure could be reasonably achieved within a time period consistent with permissible radiation releases where, it may be considered as equal to half barrier.
- Containment boundary - equal to one barrier.

(iii) For systems whose failure may result in significant release of radioactive substance, the fluid boundary subject to inspection shall include all portions whose failure may permit such a release.

(iv) For systems mentioned in item (a) of subsection 4.1.1.1, the examination requirements shall apply to all portions of the fluid pressure boundary whose failure may permit a significant radioactive release. The fluid boundary of vessels subject to inspection shall extend to and include all nozzle to vessel attachment welds. The component supports (e.g. for piping, vessels, pumps and valves) and rotating machinery whose structural integrity is relied upon to withstand the design loads and seismic-induced displacements should be inspected.

(v) For safety systems mentioned in item (b) of subsection 4.1.1.1, the examination requirements shall extend to and include all portions whose failure could prevent adequate safety function.

4.1.2 Extent of Examination for ISI Programme

4.1.2.1 The extent of inspection required shall be determined by inspection category as explained in subsections 4.1.5 and 4.1.6.

4.1.2.2 The ISI shall be extended to include a system, or component where operating conditions or operating behaviour differ significantly from that contemplated in the design. The inspection programme for such systems, or components shall be determined on special case basis.

4.1.2.3 The inspection of an item designated as spares replacement shall be the same as that required if it were not so designated.

Spares Replacement means an item that has all of the following characteristics:

(a) the item would normally be replaced rather than repaired;

(b) a replacement for the item is available; and

(c) the process of replacement does not involve welding or brazing.

4.1.3 Size of Failure, Fatigue Usage Factor, and Stress Intensity

To select the inspection category used for determining the extent of examination, the factors such as size of failure, fatigue usage factor and stress intensity are required to be evaluated (for details see Annexure-IV).

4.1.4 Stress Classifications of Inspection Areas

There are three stress classifications based on stress ratios (Rs): subsection AIV-6.1, Annexure-IV)

- Low stress intensity (L) : $Rs \leq 1/3$
- Medium stress intensity (M) : $1/3 < Rs < 2/3$
- High stress intensity (H) : $Rs \geq 2/3$

4.1.5 Determination of Inspection Category (see Appendix- IV)

4.1.5.1 Inspection categories shall be determined as follows:

(a) Determine the failure size classification (subsections AIV-1 to AIV-4, Annexure-IV).

(b) Determine the fatigue usage factor (subsection AIV-5, Annexure-IV).

(c) Determine the stress classification (subsection AIV-6.1, Annexure-IV).

(d) Designate category C2 for systems, components, and local areas classified as small failure size.

(e) Use Fig. 2 (A) of Appendix-IV to determine inspection category for systems, components, and local areas classified as medium failure size.

(f) Use Fig. 2 (B) of Appendix-IV to determine inspection category for systems, components, and local areas classified as large failure size.

(g) Locate the intersection of the stress intensity and fatigue usage factor on Fig. 2 (A) or Fig. 2 (B) of Appendix-IV, as appropriate, to determine the inspection category (A, B, C1 or C2).

4.1.5.2 For mechanical couplings, the inspection category shall be determined from Fig. 2 (A) or Fig. 2 (B) of Appendix-IV for the size of failure resulting from complete failure of the connection. The inspection categories for the various items in a coupling may be different.

4.1.5.3 For supports, the inspection category shall be determined as follows:

(a) The entire support shall be placed in category A where the supported component has a large failure size, and in Category B where the supported component has a medium failure size.

(b) For component attachment welds, the inspection category shall be determined from Fig. 2 (A) or Fig. 2 (B) of Appendix-IV for the size of failure resulting from complete failure of the weld. Where a support has two or more component attachment welds, the inspection category for the various welds may be different.

4.1.6 Sampling and Examination Required for Inspection Categories

4.1.6.1 Sampling for Examination

(a) Examination samples are selected from:

(i) the most significant acceptable flaws discovered during the pre-service inspection.

(ii) the areas most subject to corrosion or erosion.

(iii) the areas having the most severe conditions of service in terms of stress, particularly cyclic.

(iv) the areas most subjected to creep and irradiation.

(b) Successive examinations are performed on the same areas in each inspection interval, except where a change is indicated by examination results or special operating conditions. The inspection of an item designated as a spare shall be the same, as that required if it were not so designated.

(c) The weld joints between dissimilar materials (having different P-numbers as per Section IX, ASME Boiler and Pressure Vessel Code) shall be considered separately from joints of similar materials.

4.1.6.2 Examinations Required for Various Inspection Categories

The recommended degree of examination for individual components in each inspection category is as follows:

4.1.6.2.1 Category A - Inspection Requirements

Material that is in category A shall be selected for inspection as follows:

(a) Piping

(i) At least one joint in each pipe run4. The joint having the highest fatigue usage factor shall be selected.

(ii) Where fatigue usage factors are not calculated, the joint having the highest stress ratio shall be selected.

(iii) The weld joints for which recordable signals were obtained shall be included for subsequent inspection. (The weld and the base metal of minimum thickness value of half metal thickness or up to 0.5 inch, i.e. 0.013 m from the edge of the metal weld prepared, whichever is greater).

(iv) The joints from which no recordable signals were obtained shall be examined over 3 inspection intervals selecting one-third of such joint in each inspection interval (five years/ten years as applicable).

(v) The weld joints between dissimilar materials (having different P-numbers as per Section IX, ASME Boiler and Pressure Vessel Code) shall be considered separately from joints of similar materials.

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[4] Pipe run: A length of piping that has common specification and extends to but not beyond a large component (e.g. vessel, pump, anchor) or a piping intersection. The pipe run may extend beyond an intersection where the run pipe has an extruded outlet or a weld-on fitting for the branch connection.

(b) Vessels: All pressure-retaining welds during PSI and ISI. For identical welds the number of welds to be inspected may be reduced to F_A as given in Appendix-V.

(c) Heat Exchangers : Shell - 4.1.6.2.1 (b) is applicable.
Tubes - Complete length for all the tubes.

(d) Mechanical Couplings:

(i) All bolting

(ii) All ligaments between threaded stud holes

(iii) All other components.

(e) Pumps: All pressure-retaining welds.

(t) Valves: All pressure-retaining welds.

(g) Supports:

(i) All supports.

(ii) All component attachment welds.

(h) Rotating Machinery: All regions (e.g. flywheel).

4.1.6.2.2 Category B - Inspection Requirements

(a) Material that is in category B shall be inspected as defined in the appropriate item in subsection 4.1.6.2.2 (d) below, provided that a leak detection system is in use.

Leak detection system means a system that

(i) provides continuous monitoring of leakage; and

(ii) has a sensitivity that will readily detect and indicate incremental leakages that are in excess of 0.0083 kg/s (1.1lb/minute).

(b) When it can be demonstrated to the regulatory body that a material not complying with the conditions mentioned in subsection 4.1.6.2.2 (a) is acceptable, it shall be inspected as defined in the appropriate item in subsection 4.1.6.2.2 (d).

(c) Where the conditions of subsections 4.1.6.2.2 (a) and 4.1.6.2.2 (d) cannot be met, the material shall be placed in category A and inspected according to category A requirements (see subsection 4.1.6.2.1).

(d) The inspection for material complying with subsection 4.1.6.2.2 (a) or 4.1.6.2.2 (b) shall be in accordance with the following:

(i) Piping

- For each pipe run having one or more category B regions, the joint having the highest fatigue usage factor shall be inspected, except that if the pipe run has a category A region, no further inspection will be required.

- Where fatigue usage factors are not calculated, the joint having the highest stress ratio shall be selected for inspection.

- The weld joints for which recordable signals were obtained shall be included for subsequent inspection.

- The joints from which no recordable signals were obtained shall be examined over 3 inspection intervals selecting l/3rd of such joints in each inspection interval (5 years/10 years as applicable).

- The total quantum in Category B need not exceed 33 % of total number of joints.

- The weld joints between dissimilar materials (having different P-numbers as per Section IX, ASME Boiler and Pressure Vessel Code) shall be considered separately from joints of similar materials.

(ii) Vessels

- Pressure-retaining welds having the highest fatigue usage factors shall be inspected. The number of welds inspected shall not be less than one-third of the category B welds. For identical welds the number of welds to be inspected may be reduced to FB as given in Appendix-V.

- Where fatigue usage factors are not calculated, the joints having the highest stress ratio shall be selected.

(iii) Mechanical Couplings

- Bolting: 10% of the total number of fasteners in the joint, to the next higher integer, shall be inspected.

- Flange Ligaments: 10% of the flange ligaments between threaded stud holes, rounded to the next higher integer, shall be inspected.

- Other Components: The extent of the inspection shall be considered on a special case basis.

(iv) Pumps

- Pressure-retaining welds having the highest fatigue usage factors in this category shall be inspected. The number of welds inspected shall not be less than 1/3 of the welds in category B.

- Where fatigue usage factors are not calculated, the welds having the highest stress ratios shall be selected for inspection.

(v) Valves

- Pressure-retaining welds having the highest fatigue usage factors in this category shall be inspected. The number of welds inspected shall not be less than 1/3 of the welds in category B.

- Where fatigue usage factors are not calculated, the welds having the highest stress ratios shall be selected.

(vi) Supports

- All supports shall be inspected.

- Where a support has one or more component attachment welds, at least one weld (that having the highest fatigue usage factor) shall be inspected.

- Where fatigue usage factors are not calculated, the welds having the highest stress ratios shall be selected.

(vii) Rotating Machinery

The region that is defined as category B and has the highest stress in this category shall be inspected. However, if the component has a category A region also, only category A regions need to be inspected.

4.1.6.2.3 Category C1 - Inspection Requirements

(a) No ISI is required for material in this category, provided it is not composed of dissimilar metals.

(b) When it can be demonstrated to the regulatory body that a material not complying with the conditions mentioned in subsection 4.1.6.2.3 (a) above is acceptable, no ISI except that called for in subsection 4.1.6.2.5 are required.

(c) Where the conditions of subsections 4.1.6.2.3 (a) and 4.1.6.2.3 (b) above cannot be met, the material shall be placed in category B and inspected according to category B requirements (see subsection 4.1.6.2.2).

4.1.6.2.4 Category C2 - Inspection Requirements

No ISI is required for material that is in this category.

4.1.6.2.5 Additional Inspection Requirements - Corrosion and Erosion

(a) In addition to the foregoing ISI requirements, the following inspections shall be performed to determine the deterioration due to corrosion or erosion, or both.

(b) Category A Areas - Additional Inspection Requirements

(i) Where the system or component, except for pumps and valves, is known to operate under conditions that can be classified as non-corrosive5 and non-erosive, or inspection results on wall thickness are available to prove that non-corrosive and non-erosive conditions exist, no further inspection is required.

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[5] Non-corrosive condition:

a) where corrosion effects are known to be negligible; or,

b) where the corrosion effects arc reduced by chemistry or temperature control so that the reduction in material thickness over the intended service life does not exceed the smaller of 1.5 mm (0.6 inch) or 6% of the wall thickness.

(ii) For pumps and valves, a dimensional and/or visual inspection of the internal surface of the fluid boundary for loss of material shall be performed. For pumps and valves subject to corrosion related cracking, surface inspection of the internal fluid boundary shall be performed. The inspection area shall cover not less than 20% of the fluid boundary internal surface and shall include all areas having the greatest potential for corrosion-related cracking to occur.

(iii) For components and systems, except pumps and valves, those do not meet the requirements of subsection 4.1.6.2.5 (b) (i), the following dimensional inspections shall be performed:

- If non-corrosive conditions do not exist, the material thickness shall be determined at the location in the system considered to have the highest corrosion rate.

- If there is no area in the system that has a corrosion rate significantly greater than the average for the system, the material thickness shall be determined at the location having the highest calculated stress intensity.

- If non-erosive conditions do not exist, the material thickness shall be determined at the location in the system considered to have the highest erosion rate.

(c) Categories B and C1 Areas

The inspection areas in these categories shall be inspected according to the requirements for category A given by subsection 4.1.6.2.5 (b) except where no further inspection is required for systems that have an inspection area included in category A provided that the corrosion and/or erosion, of these areas is not greater than that of inspected category A area(s).

(d) Category C2 Areas

No additional inspection is required.

4.1.7 Extent of Examination for PSI

PSI should be performed on all components, supports, and portions of systems selected to undergo ISI and all replicas of such components, supports and portions of systems.

4.1.8 Criteria for Exemption

4.1.8.1 Areas of supports where the largest principal stresses are compressive need not be inspected.

4.1.8.2 Where it can be demonstrated to the satisfaction of the regulatory body that in the event of failure of a system or portion thereof covered by subsection 4.1.1.2 (b) (i) or (iii), without operation of the containment system, any resulting hazard to the public would be less than the dose limits prescribed by the regulatory body for a serious process failure (Category 2 event or above as defined in 'Design Basis Events for Pressurised Heavy Water Reactor', AERB SG/D-5), such system or portion shall be exempted from ISI, unless otherwise specifically included, based on any other considerations (e.g. Indian or International plant experience).

4.1.8.3 Components in systems, which are completely filled with liquid at temperatures below a value corresponding to a vapour pressure of 340 kPa (50 psi) and whose supports are shown to be adequate to withstand the forces resulting from failure of the fluid boundary can be exempted from inspection.

4.2 Criteria for Boiling Water Reactors (BWRs)

Annexure-II describes the requirements for the Boiling Water Reactors.

5. FREQUENCY AND SCHEDULING OF PSI/ISI

5.1 Pre-Service Inspection (PSI)

5.1.1 A PSI shall be performed before the commencement of operation to provide data on initial conditions supplementing manufacturing and construction data as a basis for comparison with the results of subsequent examinations. This examination shall therefore make use of the same methods, techniques and types of equipment as those planned to be used for ISI.

5.1.2 The PSI shall be extended to all components that are subjected to ISI. Where this includes a sample of welds, the full length and width of the welded zone and the specified portion of the adjoining base material shall be examined.

5.1.3 When a component is repaired or replaced, an initial examination shall be performed on that component.

5.1.4 Shop and field inspections performed during construction may form part of the PSI, where inspection after final installation and testing is not practical, provided :

(a) such inspections are conducted under similar conditions and with equipment and techniques equivalent to those that are planned to be employed during subsequent in-service examinations;

(b) inspections conducted before a hydrostatic (or pneumatic) pressure test are followed by a confirmatory inspection (subsection 5.3) after the test on a sample of inspection areas to demonstrate that no significant change has occurred;

(c) for components classified as pressure vessels only, inspections are performed after the hydrostatic (or pneumatic) test; and

(d) the shop and field inspection records are documented and identified in a form consistent with the recommendations of this guide.

5.1.5 PSI data should be collected as mentioned in subsection 2.4.2.

5.1.6 A PSI can also be performed before the component hydrostatic or other pressure test, provided a confirmatory inspection, as mentioned in subsection 5.3, is performed after the component hydrostatic pressure test and the results indicate no significant change.

5.2 In-Service Inspection (ISI)

5.2.1 All the recommended examinations of the ISI programme of a particular NPP shall be completed during each inspection interval, the length of which shall be based on conservative assumptions to ensure that deterioration, if any, of the most exposed component is detected before it can lead to failure. The inspection schedule provides for repetition of the inspection programme during the operating life of the NPP. The inspection schedule may involve evenly distributed inspection intervals (see example in Table-3, Appendix-VI and Table AII-1, Annexure-II) or, alternatively, the variably distributed inspection intervals, during the operating life of the plant (see example in Table AII-2 of Annexure-II) to improve the correlation between inspection intervals and the probabilities and characteristics of component failures. The inspection interval for the evenly distributed schedule may be chosen to be from a few years to about ten years; in the variably distributed schedule these intervals may be shorter in the early years of the plant life and then lengthened as experience permits. No matter which programme is adopted, information on flaw severity may require a shortening of the interval towards the end of plant life.

5.2.2 The inspection interval is subdivided into inspection periods during which a required number of examinations must be completed, depending upon the component, the type of examination, or the accessibility allowed by the normal plant operations or by scheduled outages. These examinations may be considered as a part of the total inspection required for the whole interval.

5.2.3 Examinations, which require the disassembly of components (such as disassembly of pumps or valves to examine large bolting volumetrically) or the removal of fuel or core support structures in reactor vessels to examine welds or nozzle radius sections, may be deferred until the end of each inspection interval or can be timed to coincide with dismantling for maintenance or other purposes except where, on the basis of results of examination conducted on analogous components, an earlier inspection is necessary. Dismantling of equipment shall not be considered to include the removal of insulation or access covers.

5.2.4 For a system or component that is subjected to conditions that differ significantly from those contemplated in the design specification, the periodic inspection interval shall be determined on a special case basis.

5.3 Confirmatory Inspection

5.3.1 Confirmatory inspection, when required by subsections 5.1.4 (b) and 5.1.6, shall employ the method used for the PSI.

5.3.2 Areas to be inspected shall include the following:

(a) Piping

The most significant indications detected previously such that the number of inspection areas equal at least 10% of the periodic inspection sample for the system.

(b) Vessels

(i) The full length of all major nozzle welds.

(ii) The most significant indications detected previously in the longitudinal and circumferential joints such that at least 10 periodic inspection areas (or all if fewer than 10 exist) per component or 10% of the indications, whichever is greater, are inspected.

(c) Pumps

At least 10% of all the pumps for a station shall be inspected, including all pressure-containing welded joints and internal surfaces of pump components.

(d) Valves

At least 10% of all the valves for a station shall be inspected, including all pressure-containing welded joints and internal surfaces of valve components.

(e) Steam Generator Tubes

Same as that mentioned in 2.4.3.2 (f)

(f) Heat Exchanger Tubes

Same as that mentioned in 2.4.3.2 (g)

5.4 Dormant Systems

Components in systems or portions of systems that are dormant (i.e. systems required to function in a passive way) should be subjected to ISI and the inspection interval for such systems should be less than or equal to half of the interval corresponding to inspection interval of the non-passive systems. Examples of dormant systems are suppression pool (PHWR), gravity addition of boron (GRAB) system (PHWR) and safety grade decay heat removal system (PFBR).

6. ACCEPTANCE CRITERIA FOR PSI AND ISI

6.1 General

6.1.1 Acceptance standards for visual, surface and volumetric examinations shall be established before the start of the programme. As-manufactured specific standards may be taken as the basis for arriving at acceptable standards and shall be submitted to the regulatory body for review, when required (e.g. before licensing for operation or first criticality). Reporting standards should be established such that when they have been reached, margins still exist between them and the acceptance standards. Indications that show no detectable change has occurred since the previous inspection should be acceptable.

6.1.2 For cases where the acceptance standards are not in existence or are not relevant to the situation, acceptance standards shall be established in consultation with the regulatory body.

6.2 Additional Examinations

6.2.1 When a flaw exceeding the acceptance standards is found in a sample, additional examinations shall be performed to include the specific problem area in an additional number of analogous components (or areas) approximately equal to the number of components (or areas) examined in the sample.

6.2.2 In the event that the additional examinations indicate further flaws exceeding the acceptance standards, all of remaining analogous components (or areas) shall be examined to the extent specified for the component or item in the initial sample except as modified by requirements mentioned in subsections 6.2.3 and 6.2.4.

6.2.3 Where the required piping examination in the sampling programme is limited to one loop or branch run of an essentially symmetric piping configuration, and examinations indicate flaws exceeding the acceptance standards, the additional examinations, mentioned in subsection 6.2.1, shall include an examination of a second loop or branch run.

6.2.4 In the event that the examinations of the second loop or branch run indicate further flaws exceeding the acceptance standards, the remaining loops or branch runs that perform similar functions shall be examined.

6.2.5 For multi-unit stations with identically designed plants, the area corresponding to that having the indication shall be inspected on the identical component in each reactor unit, immediately following the next reactor cool down. The period of time permitted to complete this additional inspection shall be based on the significance of the indication evaluated as unacceptable and in no case shall it exceed 20% of the examination interval specified in section 5.

6.3 Repetitive Examinations in Successive Inspection Intervals

6.3.1 The sequence of component examinations established during the first inspection interval shall for repeated during each successive inspection interval, to the extent practical.

6.3.2 Where examination of a component results in the evaluation of flaw indications in accordance with the provisions of section 7 and qualifies the component as acceptable for continued operation, that portion of the component containing such flaws shall be re-examined during each of the next three inspection periods, as an extra requirement above the original programme schedule.

6.3.3 In the event that the re-examinations required by subsection 6.3.2 indicate that the flaws remain essentially unchanged for three successive inspection periods, the component examination schedule may revert to the original schedule of successive inspections.

6.3.4 For multi-unit stations, in case of any indication seen during repetitive examinations in any unit, similar inspection shall be carried out in the relevant area in the other units as stated in subsection 6.2.5.

7. EVALUATIONS AND ANALYSIS OF RESULTS

7.1 The operating organisation shall ensure that the results of any examination are evaluated to determine compliance with acceptance standards.

7.2 All indications shall be interpreted as relevant or otherwise. An unacceptable indication shall be treated as defect.

7.3 All relevant indications shall be further investigated as to the nature, size and location of defects. The doubtful indications should be confirmed by alternate NDT techniques.

7.4 Interpretation of results shall be done by qualified Non-Destructive Examination (NDE) personnel in the respective technique.

7.5 The results of the examination shall be analysed by the Responsible Organisation (RO) for continued safe operation. When the analyses show that continued operation of the component is unacceptable, it shall be repaired or replaced as per subsection 10.1.

7.6 When fracture mechanics is employed for analyses required as per subsections 7.5 and 7.8, the following should be ensured:

(a) The stresses in the area of the flaw shall be analysed for all conditions of operation, including postulated accident conditions and actual as well as predicted normal operating conditions.

(b) The worst stress case shall then be selected, and the flaw should be circumscribed in elliptical or circular shapes and projected into a plane perpendicular to the worst stress case.

(c) The values of the material properties used in the analysis should be those actually measured, but where such data do not exist, it should be assumed that the properties conform to the most conservatively accepted values for the particular type and grade of the material containing the flaw.

(d) For the case where the properties of the material may be altered by its environment, such as by irradiation, samples should be used to establish the actual change in material properties and where samples are not in existence, it may be assumed that the changes due to irradiation follow the curves published for the most sensitive heats for the type and grade of the material in the irradiation zone. Care should be taken to consider all aspects of the problem so that assumptions always involve the worst case in the analysis. The calculation methods should be in accordance with accepted standards.

7.7 Where the result of an examination does not comply with the acceptance standards, evaluation shall be extended to include the following:

(a) Notifying the regulatory body of the indication.

(b) Further examination by other non-destructive methods where practical, to assist in the determination of the nature of the indication (location. size, shape, orientation, etc.). Care should be taken, when choosing any supplementary technique, to ensure that the conditions affecting the component are thoroughly investigated.

(c) For additional examinations as specified by subsection 6.2.

(d) Appraisal of the examination results to determine disposition i.e. acceptance of the indications in accordance with subsection 6.1, revised examination programme, repair or replacement.

(e) Submission of the disposition proposal to the regulatory body for acceptance.

7.8 Indications that do not comply with the acceptance standards may be considered as acceptable till the next ISI, provided it can be demonstrated to the regulatory body by suitable analyses that

(a) the integrity of the component is still adequate, and

(b) the predicted deterioration will not seriously reduce the integrity of the component prior to the next ISI.

8. NON-CONFORMANCE CONTROL

8.1 Procedures are required to be established for control of materials, parts, components, systems or processes that do not conform to specified requirements. The procedures are required to provide for prompt recording and reporting of non-conforming items. These procedures shall also provide clear identification and segregation of non-conforming items (physical segregation, tagging, etc.) for preventing inadvertent use.

8.2 A non-conformance report (NCR) for each of the non-conforming items should be prepared by the operating organisation and submitted to RO.

8.3 The NCR should include the following:

(a) Statement of non-conformity with drawings and sketches as required.

(b) Reasons for occurrence of non-conformity.

(c) Corrective actions to avoid recurrence.

(d) Disposition proposal along with technical analysis and justification.

8.4 Disposition of non-conforming item(s) should be documented. RO should review and approve the document mentioning the selected/approved disposition for all the non-conforming items.

9. VERIFICATION

9.1 Arrangements should be made for independent verification to ensure that the examinations and tests comply with the requirements. These verifications shall be carried out by inspectors from either the operating organisation, the regulatory body or from an organisation recognised by the competent authority.

9.2 The final results of PSI and ISI shall be confirmed by the operating organisation6 as satisfactory for continued operation and submitted to the regulatory body when required.

[6] Responcible organisation (RO) or plant management as designated by RO, if no operating organisation exists.

10. REPAIRS, REPLACEMENTS AND MODIFICATIONS

10.1 Repairs and Replacements

10.1.1 Indications revealed by pre-service or in-service examinations that are considered to make a component unacceptable by the provisions of section 6 shall require corrective action in the form of repair or replacement.

10.1.2 Components should be repaired in accordance with the codes and standards applied at the time the component was constructed and in accordance with the QA programme in effect at the time of repair.

10.1.3 Replacements should meet the provisions and requirements of the codes, standards and other special instructions that were applied to the construction of the component or the part of the component to be replaced. Alternatively, replacements may meet the requirements of later editions of codes or new codes and standards, or portions thereof, provided:

(a) The requirements affecting the design, fabrication and examination are reviewed and it is determined that the original safety requirements are not diluted.

(b) For mechanical interfaces, fit and tolerances affecting performance are not changed by the later editions of, or the new codes or standards.

(c) The materials are compatible and suitable for installation and operating requirements of the system.

10.1.4 Components that are repaired or replaced for any reason shall be re-examined in accordance with the provisions of this guide, and before the pressure retaining components are returned to service they shall be tested in accordance with subsection 3.3. This re-examination shall include the technique that was used to detect the deterioration, and the re-examination shall form the new basis for subsequent ISI.

10.1.5 When systems or components require modification, alteration or addition, the provisions in this guide for repair and replacement shall be used.

10.2 Modifications

10.2.1 Requirements

10.2.1.1 Systems and components that are extended, reduced, or otherwise altered after the initial startup shall be reviewed to establish any change in:

(a) the hazard to health and safety;

(b) potential failure consequences;

(c) the extent of systems subject to inspection; and

(d) the inspection category.

10.2.1.2 A change in any of the above classifications shall call for amendment to the periodic inspection programme to ensure that the inspection of the revised system complies with the requirements of this guide.

11. DOCUMENTATION

11.1 The documents necessary for adequate implementation of the ISI programme shall be developed, maintained and readily available to the operating organisation.

11.2 The documents shall be clearly identifiable by the date, name of the plant and operating organisation, the competent authority, and be readily retrievable.

11.3 The documents shall be developed and maintained from the design stage onward and shall include:

(a) the selection of components to be examined,

(b) specifications, as-built drawings, data of the component including material specifications, heat treatment records, records of manufacturing process, fabrication and installation specifications, drawings and records of acceptance of deviations from specifications,

(c) samples of materials used wherever applicable,

(d) determination of the type of examination,

(e) selection, location and extent of areas to be examined and examination frequency,

(f) PSI data and reports,

(g) examination procedures,

(h) a copy of ISI programme document and all subsequent amendments,

(i) all ISI records and reports,

(j) calibration records and charts,

(k) acceptance standards,

(l) record of all defects discovered by both PSI and ISI, their disposition, the corrective action taken, details of any repairs or replacement and subsequent examinations, and

(m) record of radiation doses received and also current estimation of expected radiation doses that may be received during the examination or tests for the planning purpose.

11.4 The relevant information such as identification of component, location and size of area examined, examination technique, type of equipment used, type of sensor calibration equipment, sensitivity standards, testing agency, name and level of inspector and date and time of inspection, etc. shall be documented so that examination could be repeated and similar authentic results obtained.

11.5 All indications that are in excess of the minimum recording level and all pertinent information concerning the indications (e.g. location, magnitude, length) shall be documented.

11.6 Comparisons with previous examination results and evaluations shall be documented.

11.7 Records, which are directly applicable to an individual component, should be maintained for the life of the component. Other records should be maintained for the life of the plant.

11.8 The detailed examination procedure should be clearly identified in the IS1 programme to include the following:

(a) scope of the examination,

(b) applicable codes and standards,

(c) supporting documents,

(d) requirements related to personnel qualifications,

(e) methods and equipment to be used,

(f) preparation of components to be examined,

(g) requirements for calibration and re-calibration,

(h) examination procedure,

(i) minimum recording level indication if applicable, and

(j) data to be recorded.

11.9 Recording of the results of the examination and test shall be in a form that will show that the examination and test have been properly completed.

11.10 Detailed examination and test procedures shall be prepared, reviewed and approved before the examination and test are carried out. This is to allow sufficient time for the personnel to be trained and equipment to be setup and tested.

11.11 The ISI programme and procedures shall be reviewed at regular intervals and the revised version shall be documented.

12. AUDITING

12.1 General

All activities related to ISI shall be periodically audited to determine the adequacy of and adherence to established procedures, instructions, specifications, codes, standards and other applicable documents and effectiveness of implementation.

12.2 Audit Personnel

12.2.1 ISI programme audit shall be carried out by appropriately trained personnel who are not directly responsible for the areas being audited. They shall have sufficient authority and organisational freedom to make the audit meaningful and effective.

12.2.2 Selection of audit personnel for ISI programme shall be on the following basis:

(a) Specialised knowledge and experience in the field of 1S1 programme, various NDT methods and tests.

(b) Knowledge and experience of auditing techniques.

(c) Knowledge of applicable codes and standards.

12.2.3 Certification of Audit Personnel

Auditors will be appointed by RO for external audit and operating organisation for internal audit, based on qualification, knowledge and experience of ISI programme. Audit personnel should possess certification by either a national or an international certification agency. If certified auditors are not available, approval shall be obtained from RO for audit personnel based on oral/written/practical test(s), conducted by RO or a certification agency recognised by RO, to evaluate individual qualifications. Only certified/RO approved audit personnel shall be engaged for the audit function.

12.3 Frequency of Audit

Frequency of audit shall be at least once in three years.

12.4 Audit Plan

Audit shall be planned in advance. The plan shall include the scope of audit, composition of audit team, the activities to be audited, the sections/agencies to be notified and the applicable documents.

12.5 Audit Notification

The organisation to be audited shall be notified well in advance before the audit takes place. The notification should be in writing and should include information such as the scope and schedule of the audit, names of the auditor and the documents to be kept ready for reviewing.

12.6 Pre-Audit Meeting

A pre-audit meeting should be conducted with the sections concerned to confirm the scope and schedule of audit and to establish the channels of communications.

12.7 Audit Performance

Assessment shall be performed for the following activities:

(a) PSI/ISI programme for completeness and adequacy,

(b) completeness and adequacy of procedures and instructions,

(c) implementation of the procedures and instructions in the work areas being assessed,

(d) records of personnel training and level of qualifications,

(e) availability and updating of records like inspection, reports, history cards, and

(f) compliance with applicable codes.

12.8 Post-Audit Meeting

At the conclusion of the audit, a post-audit conference shall be held by the audit team with the audited organisation to present findings and clarify any misunderstandings. It is desirable that agreement be reached on audit findings at this conference.

12.9 Audit Report

A report shall be prepared immediately after the auditing and the same shall be signed by the lead auditor and representative of the audited section. The report shall include the following information:

(a) purpose of audit,

(b) list of standards, procedures, or other documents used as bases for audit,

(c) list of audit team members,

(d) pre-audit conference details,

(e) summary of audit findings, and

(f) suggestions for correcting non-conformance or deficiencies in the 1SI programme.

12.10 Follow-up Activity

12.10.1 By the Audited Section: The audited section shall review and report to the auditing organisation the progress achieved in completing corrective actions.

12.10.2 By the Audit Organisation: Audit organisation shall evaluate the response and confirm that corrective actions are accomplished as scheduled.

13. SUPPLEMENTARY INSPECTION

13.1 Steam Generator Tubes Inspection Requirements

13.1.1 Scope

Subsection 13.1 establishes the requirements for the supplementary PSI/ISI of tubes in steam generators (SGs).

The volumetric examination of all the tubes should be carried out in one SG during PSI and ISI. For remaining SGs the following should apply.

13.1.2 Sample Inspection

Tubes to be inspected for PSI and ISI should be selected from the two sample categories; specific (13.1.2.1) and random (13.1.2.2) as follows:

13.1.2.1 Specific Sample

The following should apply for specific sample:

(a) Tubes from regions considered being at higher risk of in-service degradation should be included. .

(b) Full length of each tube should be inspected, to the extent practicable, for degradation mechanisms reliably detected by standard bobbin coil eddy current inspection.

(c) For degradation mechanisms not reliably detected by standard bobbin coil eddy current inspection:

   (i) specialised NDE probes and techniques should be validated and used to provide information for comparison; and

   (ii) where a postulated degradation mechanism is only applicable to specific section(s) of the tube, such section(s) should be subjected to inspection.

13.1.2.2 Random Sample

For the random sample, the following should apply:

(a) The number of tubes in the random sample should be at least 50% of the required total;

(b) Tubes to be inspected should be chosen to cover all areas of the steam generator, with the exception of those covered by full length inspection under subsection 13.1.2.1.

(c) To the extent practicable, the full length of each selected tube should be inspected.

13.1.3 PSI

For PSI carried out prior to or during plant commissioning the following should apply:

(a) Subject a minimum of 25% of the tubes in each steam generator in each reactor unit.

(b) Select tubes for inspection in accordance with subsections 13.1.2.1 and 13.1.2.2.

(c) Perform PSI as follows:

(i) either after the primary side hydrostatic pressure test but prior to startup of the reactor; or,

(ii) prior to PHT system hydrostatic test provided that a confirmatory inspection of 10% of the tubes (as required by (d) below) after the PHT system hydrostatic pressure test is performed and the results indicate no significant change.

(d) The confirmatory inspection, mentioned in subsection 13.1.3 (c) (ii) above, should employ the method(s) used for the PSI and the confirmatory sample should include tubes with the most significant indications detected previously.

13.1.4 ISI

The following should apply for ISI:

(a) A minimum of 10% of the total number of tubes in one steam generator, chosen from those in the pre-service sample, should be subjected to ISI.

(b) Select tubes for inspection in accordance with subsections 13.1.2.1 and 13.1.2.2.

(c) Visually inspect the tubes and support structure on secondary side of one steam generator. If this is not possible, all the tubes in one SG should be examined by multi-frequency eddy current to check the condition of baffles/supports as far as possible.

(d) Remove a section of one tube in a deposit region from one steam generator for metallurgical examination, to check tube degradation by deposition on tube sheet. If the tube removal is not practicable, a condition should be maintained to ensure that metallurgical degradation of tubes does not take place by accumulated deposition on tube sheet surface and they remain in place. This may be ensured by frequent removal of deposits by using methods like sludge lancing. In addition, visual examination of tube-to-tube sheet area (shell side) should be carried out.

13.1.5 Inspection Interval

(a) ISI should be planned for the time intervals, which do not exceed five years or one-fifth of the component design life, whichever is less. These inspections should be performed over the last half of the inspection interval.

(b) Altered Intervals

If steam generator tubes are subjected to conditions that differ significantly from those contemplated in the design specification, the periodic inspection sample size and inspection interval should be reviewed and, if necessary, the periodic inspection programme should be revised.

13.2 Coolant Channel Inspection Requirements

13.2.1 PSI

All the channels should be subjected to PSI after hot conditioning and prior to criticality. The information obtained should be compiled with the fabrication and installation details. As a minimum the following requirements should be met:

(a) Manufacturer's mechanical, chemical, volumetric and dimensional inspections, and corrosion tests data should be compiled and made available at plant site. All history dockets should be prepared for individual channels.

(b) Data obtained from the post installation inspections/dimensional checks including thickness measurements, detection of garter spring location and tilt, if any, and minimum front and back gaps (kept for thermal expansion and creep) measurement will be recorded.

13.2.2 Base Line Inspection (BLI) as first ISI

(a) Sample Size and Time of Inspection

BLI should be performed within a two effective full power years (EFPY) period, commencing after 7000 effective full power hours (EFPH) of operation. Appropriate sample size should be chosen based on channel selection criteria given in 13.2.2 (c).

(b) Extent of Examination

The BLI should include

(i) The full volume of the tube including the areas adjacent to rolled joints.

(ii) Measurements to determine the pressure tube to calandria tube gap may be:

- either determination of garter spring location, and tube deflection (sag); or

- pressure tube to calandria tube (PT-CT) gap measurements.

The sag or gap measurements should be separated by no more than 250 mm or half fuel bundle length, whichever is less.

(iii) Internal diameter and tube wall thickness measurements at not less than three equally spaced circumferential position, with axial spacing separated by no more than 250 mm or half fuel bundle length whichever is less.

(iv) Measurement(s) of channel length from 'E' face to 'E' face and determination of the fuel coolant channel position on its bearings.

The selected channels should be same in all the reactors of same type (e.g.PHWRs of same Group, see Annexure-V).

(c) Channel Selection Criteria

The channels vulnerable from various considerations should be selected for BLI based on the following considerations:

(i) minimum two channels from low flux zone,

(ii) minimum three channels from high flux zone,

(iii) one channel from regions close to adjuster rods other than those selected in (i),

(iv) one channel each from regions close to moderator inlet and outlet lines (applicable to RAPS 1& 2 and MAPS),

(v) the channels that have seen high fuel failures,

(vi) all the channels with expected life shorter than the design life based on creep contact time and blister growth,

(vii) any other channels identified during pre-commissioning stage, and

(viii) any abnormality observed during operation that may affect service life of coolant channel.

13.2.3 ISI

(a) Sample Size and Time of Inspection

The minimum sample size for ISI should be based on the selection criteria mentioned in subsection 13.2.2 (c)7. These selected coolant channels should be subjected to the complete inspection. The first such inspection covering at least half of the selected channels should be carried out within five EFPYs unless the operational history dictates otherwise. Subsequent to the first periodic inspection these selected coolant channels should be inspected within a period of next five to fifteen EFPYs and again within twenty five EFPYs respectively by distributing them evenly over the specified period. After each IS1 the results should be reviewed and analysed for changing the frequency of coolant channel inspection if required.

[7] For the present design of PHWRs, a minimum number of 12 channels are subjected to complete examination during ISI.

(b) Extent of Examination

The requirements of subsection 13.2.2 (b) should apply. The monitoring of additional characteristics and inclusion of additional number of channels on the basis of PSI/ISI to determine the extent of Deuterium ingress by insitu scrape sampling or other alternate techniques should be done. This should be supplemented by examination of pressure tube removed from reactor. The removed pressure tube should also be analysed for degradation of mechanical properties such as

(i) fracture toughness, by direct or indirect methods;

(ii) critical crack length at different temperatures;

(iii) ultimate tensile strength, yield strength and ductility; and

(iv) delayed hydride cracking velocity.

(c) Additional Inspection

(i) Where indications found during volumetric inspection are greater than the reference specimen, the area corresponding to that containing the indication should be inspected on additional coolant channels in the same reactor. The extent of additional inspections should be determined as part of the disposition as referred in subsection 13.2.7;

(ii) In the event of an indication being found in the coolant channels of one reactor that exceeds the volumetric acceptance standards, the need to extend inspection to other reactor will be taken into consideration by RO; and

(iii) When there is a reason to suspect the channel during the course of inspection, all the channels which are likely to come under the same suspicion should also be included for such inspection after due evaluation of the channel which was found suspect in the first instance.

(d) Volumetric Inspection Reference Specimen

A reference specimen should be made from a pressure tube of the same material and geometry as the pressure tube that is to be inspected, and should contain the following discontinuities:

(i) Ultrasonic Shear Wave Calibration Slots

The dimension of each discontinuity should be 6.0 ± 0.5 mm long, 0.15 ± 0.05 mm deep and 0.15 ± 0.05 mm wide. The depth of discontinuity should be decided based on the minimum wall thickness of material used (e.g. 3% for Zr tube and 2% for Zr-Nb tube). The discontinuities in the reference specimen will be decided by the RO for respective NDT techniques for PSI/ISI based on material design specifications and standard practices adopted.

Reference discontinuities should represent all possible anticipated flaws in longitudinal, circumferential and randomly oriented directions as follows:

- internal longitudinal slot,
- internal circumferential slot,
- external longitudinal slot, and
- external circumferential slot.

(ii) Ultrasonic Longitudinal Wave Calibration Slots

Three flat-bottomed slots machined from the outside surface. The slots should be 0.75 ± 0.05 mm wide, 1.5 ± 0.1 mm long, and should be machined to depths of 15%, 50% and 85% of the nominal tube wall thickness.

(e) Mock-up Facility for Dimensional Inspection

The RO should provide mock-up facility reference specimens, which is appropriate to the dimensional measurement techniques employed to satisfy dimensional inspection requirements. Items (i), (ii) and (iii) given below specify only the minimum requirements for this mock-up facility. The dimensional mock-up facility should be used to verify the performance and accuracy of the equipment and techniques used to perform the required measurements.

(i) Sag and Gap Determination

A mock-up facility should be made from a pressure tube and calandria tube (if applicable) of the same material and geometry as the channel to be inspected.

The mock-up facility should be assembled so as to contain dimensions covering the full range of measurements expected during inspection/ measurements.

(ii) Internal Diameter and Wall Thickness

A reference specimen should be made from a pressure tube of the same material and geometry as the pressure tubes to be inspected. The reference specimen should contain both the minimum and maximum internal diameters, and the minimum and maximum wall thickness specified in the design documentation for the pressure tube to be inspected. This is to verify the linearity in required range. Alternate method can be used to satisfy the same purpose.

(iii) Fuel Channel Bearing Position Determination

A reference specimen of the same or equivalent material as the fuel channel component(s) to be measured should be made. The reference specimen should be manufactured so as to contain dimensions covering the full range of measurements expected during inspection. This is to verify the linearity in required range. Alternate method can be used to satisfy the same purpose.

13.2.4 Station Specific ISI

13.2.4.1 General

This part of the programme envisages study of specific characteristics of the coolant channels over their operating life.

13.2.4.2 Channel Removal

To determine the characteristics controlling the generic degradation of coolant tube material, removal of channel from high flux region for post irradiation examination (PIE) is essential. From a reactor with highest full power operation in each group of reactors (Group 1, 2, 3 or 4; see Annexure- V) one channel in every two EFPYs should be removed and analysed. The first removal should be after the five EFPYs. For other reactors, one channel in every five EFPYs is to be removed for analysis and comparison with the initial data that is available or that can be generated out of off-cut archive samples. The off-cut samples of coolant channels should be archived and preserved such that the same may be used for comparison in future, as and when required. Channels with evidence of any blister formation should be included for removal.

13.2.4.3 Determination of Deuterium/Hydrogen Ingress Rate

Whenever additional channels are taken up for inspection, monitoring should be done for all the parameters envisaged in the reaotor/station-specific ISI programme to generate the engineering data on Deuterium ingress and assessment of ingress rate by direct or indirect method (on channel removed as per the requirement mentioned in subsection 13.2.4.2). Sufficient number of coolant channels should be selected for measurement of Deuterium ingress (equivalent Hydrogen concentration) and assessment of ingress rate by direct or indirect methods to generate sufficient data for each type of reactor.

13.2.4.4 Blister Formation and it's Growth Rate

Channels selected as per the requirements mentioned in subsection 13.2.2 (c) (vi) should be thoroughly inspected for any blister formation and growth rate. In absence of a method for direct measurement, analytical tool may be used.

13.2.5 Evaluation

The station should be responsible for evaluation of the results of each stage of the inspection programme. Wherever necessary, validated computer codes and programmes designed for the particular inspection activity should be used.

The results from each inspection should be evaluated for acceptance in accordance with the following:

(a) indications that show that no detectable change has occurred since the previous inspection should be acceptable;

(b) indications that show a detectable change should be evaluated to determine compliance with the acceptance criteria specified in subsection 13.2.6; and

(c) indications that do not comply with the acceptance criteria should be acceptable provided it can be justified to the regulatory body that the requirements mentioned in subsections 13.2.6 (d) and 13.2.7 are met.

13.2.6 Acceptance Criteria

(a) General

Relevant indications that do not exceed the limits mentioned in subsection 13.2.6 (b) below and show no detectable change since the previous inspection should be acceptable. When an indication is found, it may be worthwhile to revisit the signal location at a latter date to see if any growth of defect has taken place. In such cases, the signal should be clearly distinguished from a noise signal or a defect site to warrant revisit.

(b) Volumetric Inspection

The following conditions should be acceptable:

(i) For ultrasonic shear wave inspection, indications which

- are not crack-like, and
- have a response less than that from similarly oriented slots in the reference specimen specified in subsection 13.2.3 (d).

(ii) For ultrasonic longitudinal wave inspection, indications which

- are not crack-like; and
- have laminations parallel to the inside tube surface.

(c) Dimensional Inspection

Pressure tubes should be acceptable provided the following conditions are predicted to exist at the next inspection:

(i) The internal diameter is within the maximum specified in design documentation.

(ii) The wall thickness is not less than the minimum specified in design documentation.

(iii) The fuel channel remains on its bearings.

(iv) No pressure tube to calandria tube contact. For contacted Zircaloy-2 pressure tube the size of blister must not be more than 0.2 mm.

(v) Adjustments for free expansion of channel are made whenever the creep exceeds the design margin. Differential axial creep between two adjacent channels should not be greater than the respective designed feeder gap. In such cases, the actual feeder gap shall be measured and monitored during subsequent annual shutdown.

(d) Fitness for Service Assessment

Indications or dimensional conditions that do not comply with the acceptance criteria should be acceptable, provided it has been demonstrated to the regulatory body that

(i) the integrity of the component is still adequate; and

(ii) the predicted deterioration will not seriously reduce the integrity of the component before the next scheduled ISI.

13.2.7 Disposition

When the result of an inspection does not comply with the acceptance criteria mentioned in subsection 13.2.6, evaluation and further actions should include:

(a) notifying the regulatory body;

(b) further inspection by other non-destructive methods, where necessary, to assist in the determination of the characteristics (size, shape, location and orientation) of the indication or dimensional condition;

(c) appraisal of the inspection results, as specified by subsection 13.2.6 to determine disposition, i.e. acceptance, repair or replacement;

(d) submission of the proposed disposition to the regulatory body for acceptance. The disposition should also include consideration of changes to the extent and frequency of inspection. For power plants, consideration should be given to the inspection of other units in the same reactor group; and

(e) obtaining acceptance of the disposition from the regulatory body prior to returning the reactor to operation.

13.2.8 Material Surveillance

13.2.8.1 Initial Material Surveillance

RO should establish initial fracture toughness, delayed hydride-cracking velocity and critical crack length of the tubes selected for material surveillance.

13.2.8.2 Inspection data of components/items obtained from a similar unit (in terms of effective full power operation and fast neutron fluence) during specific inspection period of operation may be taken advantage of with proper justification in ISI programme. This unit may be called as lead unit for material surveillance.

13.2.8.3 (a) First material surveillance should cover the following activities:

(i) Initial material surveillance in accordance with subsection 13.2.8.1.

(ii) Baseline measurements of Hydrogen isotope concentration of six pressure tubes within a two EFPYs period, commencing nine EFPYs after generation of first net power. Should the unit be designated (as lead unit) for material surveillance, then a baseline measurement should be performed on the newly designated unit. The pressure tubes selected for baseline measurements should be those subjected to material surveillance over the remaining life of that unit.

(b) (i) The following periodic material surveillance measure-ments should be performed:

- Hydrogen isotope concentration.
- Fracture toughness.
- Delayed hydride-cracking velocity.

(ii) If a pressure tube is removed to meet the above requirements, the following should also be performed:

- Visual examination of surfaces and determination of garter spring impression.
- Volumetric examination.

(c) The surveillance intervals should be as follows:

(i) One pressure tube should be removed and subjected to the material surveillance requirements of subsection 13.2.8.3 (b) after every two EFPYs, commencing 12 EFPYs after the generation of first net power.

(ii) Subsequently (i.e. after 14 EFPYs), one pressure tube should be subjected to the requirements of subsection 13.2.8.3 (b) at intervals not exceeding three EFPYs; and

(iii) In addition, each pressure tube subjected to the surveillance requirements of item (ii) above, should be monitored for Hydrogen isotope concentration 3 + 1 EFPYs prior to its surveillance.

13.2.9 Acceptance Criteria and Evaluation Procedures for Material Surveillance

Acceptance criteria and evaluation procedure for the following measurements should be submitted to the regulatory body for approval:

(a) Hydrogen isotope concentration.

(b) Fracture toughness.

(c) Delayed hydride-cracking velocity.

Acceptance criteria should include both absolute and rate of change values.

13.2.10 Recording Criteria

(a) Volumetric Inspection

The following should apply:

(i) Specific information as defined in subsection 13.2.10 (a) (ii) below shall be recorded for ultrasonic indications, which exhibit any of the following characteristics:

- crack-like indications; or

- relevant indications equal to or greater than 50% of the corresponding indications from the discontinuities in the reference specimen defined in subsection 13.2.3 (d).

(ii) For ultrasonic indications meeting the criteria of subsection 13.2.10 (a) (i), the following information shall be recorded:

- Signal characterisation.

- Dimensions of the indication.

- Axial and circumferential location of the indication.

- Signal amplitude with respect to discontinuities in the reference specimen.

(iii) When alternative volumetric inspection methods (i.e. other than ultrasonic inspection) are used, the recording criteria should be submitted to the regulatory body and approval obtained prior to commencing inspection.

(b) Dimensional Inspection

All dimensional data collected should be recorded to permit comparison with previous and future inspections.

(c) Material Surveillance

All data collected should be recorded to permit comparison with previous and future inspections (see subsection 13.2.8).

13.2.11 Reporting Criteria

(a) Volumetric Inspection

Data recorded in accordance with subsection 13.2.10 (a) should be reported.

(b) Dimensional Inspection

The following should be reported:

(i) The maximum/minimum internal diameter, its location, maximum change in internal diameter since previous inspection, and the rate of change in diameter.

(ii) The minimum wall thickness, its location, and the maximum change in wall thickness since previous inspection and rate of change in wall thickness.

(iii) The minimum PT-CT gap, the location of the minimum gap, maximum change in gap since previous inspection and the rate of change in gap.

(iv) The most extreme bearing position and the predicted rate of change.

(v) Sag of pressure tube, curvature and slope data.

(c) Material Surveillance

The following should apply:

(i) The RO should submit to the Regulatory Body a report describing the results of the surveillance performed and the subsequent evaluation(s) to demonstrate compliance with the acceptance criteria mentioned in subsection 13.2.9.

(ii) The results of Hydrogen isotope concentration measurements should be reported to the regulatory body within 120 days after completion of that stage of surveillance.

(iii) All other surveillance results should be reported to the regulatory body within one year of commencement of surveillance activities.

13.3 PHT System Feeder Pipes Inspection Requirements

13.3.1 PSI

The following should apply:

(a) A visual PSI of all feeder pipes and supports should be performed on pre-operational reactor units and a record should be made of any observations arising from the inspection relating to the pressure retaining integrity of these components.

(b) Wall thickness measurements on a minimum of 20 feeder pipes should be performed on all pre-operational reactor units. The inspection areas should be chosen from those which are accessible and likely to experience the greatest reduction in wall thickness (e.g. bends).

13.3.2 ISI

13.3.2.1 General

The following should apply:

(a) A direct or remote general visual inspection of readily accessible feeder piping and its supports should be carried out. Disassembly of feeder pipes supports is not required. The use of remote inspection equipment is permitted.

(b) Direct or remote visual inspection of indications recorded during the PSI of items mentioned in subsection 13.3.1 (a) should be carried out.

(c) Wall thickness measurements should be performed on feeder pipes as required below.

13.3.2.2 Sampling

A minimum of 10 feeder pipes, especially at bend locations, selected from the PSI sample should be subjected to ISI.

13.3.2.3 Inspection Interval

The scheduling of inspections should comply with the requirements given in subsection 5.2.

13.3.2.4 Acceptance and Recording Criteria

The following should apply:

(a) Indications of reduction from the initial feeder pipe wall thickness of 20% or greater should be recorded and reported.

(b) Indications of reduction from the initial feeder pipe wall thickness of 40% or greater should be submitted to the regulatory body along with the result of analysis required for disposition (see section 7).

(c) For visual inspection, the requirements mentioned in section 6 should apply.

APPENDIX-I

FLOW CHART FOR ISI OF NUCLEAR POWER PLANTS (PRESSURISED HEAVY WATER REACTOR)[8]

[see subsection 2.1.2]

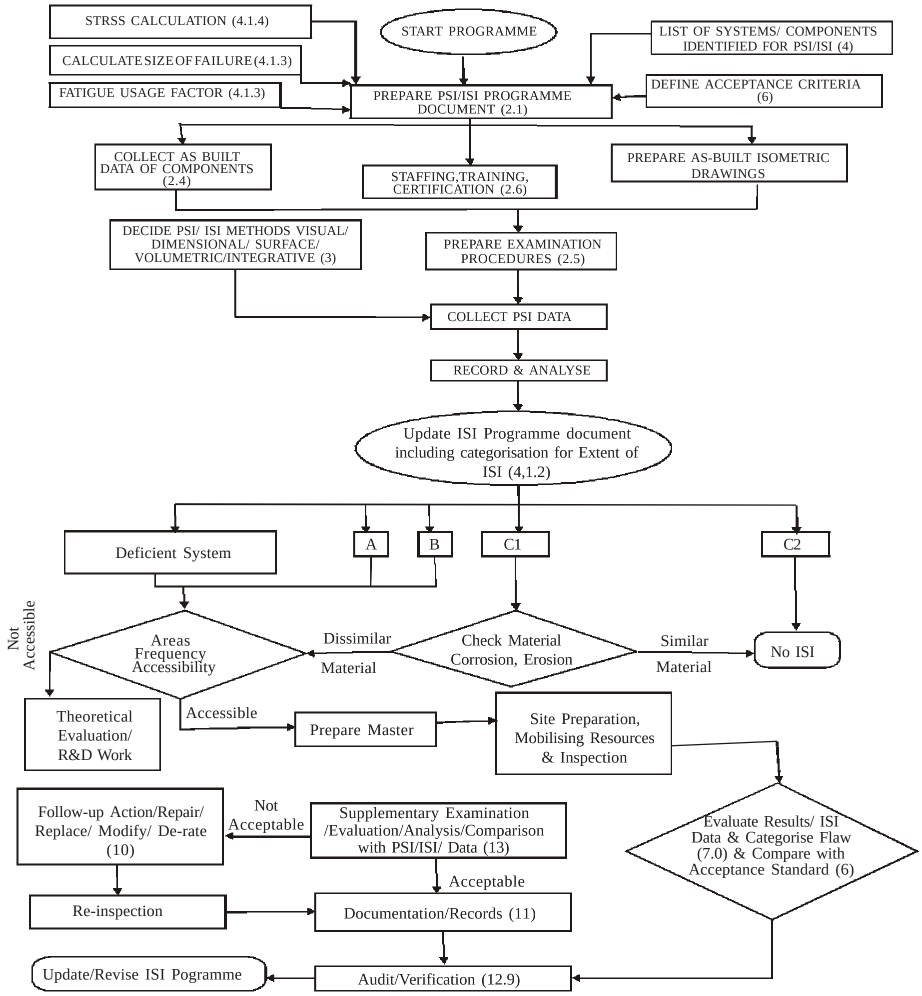

FIGURE 1: FLOW CHART FOR ISI OF NUCLEAR POWER PLANTS (PRESSURISED HEAVY WATER REACTOR)

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[8] Numbers mentioned in parenthesis correspond to relevant section of this guide.

APPENDIX-II

REQUIRED CAPABILITIES OF INSPECTION PERSONNEL

[see subsections 2.6.4 (a) and (c)]

TABLE-1: REQUIRED LEVELS OF CAPABILITIES OF PERSONNEL FOR INSPECTION FUNCTIONS

No.	Project Functions	Levels		
		Level I	Level II	Level III
1	Recording inspection, examination and testing data	X	X	X
2	Implementing inspection, examination and testing procedures	X	X	X
3	Planning inspections, evaluation and tests, setting up tests including related equipment	X	X	X
4	Evaluating the validity and acceptability of inspection, examination and testing results		X	X
5	Reporting inspection, examination and testing results		X	X
6	Supervising equivalent or lower level personnel		X	X
7	Qualifying lower level performance		X	X
8	Evaluating the adequacy of specific performance used to train and test inspection, examination and testing personnel			X
9	Qualifying same level personnel			X

Note: X means applicable.

APPENDIX-III

METHODS OF EXAMINATION

TABLE-2: COMPONENTS AND CORRESPONDING METHODS OF EXAMINATION
[see subsection 3.2]

Items to be Examined	Methods	
Vessel and piping	Visual, volumetric and integrative	
Pump and valve - Weld - Interior surface	 Visual, volumetric and integrative Visual and surface[9]	
Support - Weld - Others	 Visual, surface and volumetric Visual	
Rotating Machinery	Visual, surface and volumetric	
Mechanical Couplings	Bolting ≤25mm	Bolting ≥25mm
- Bolt	Visual	Visual, surface, volumetric and integrative
- Stud	Visual	Visual, surface, volumetric and integrative
- Nut	Visual	Visual, surface, volumetric and integrative
- Ligament between threaded stud hole	Visual	Visual, surface, volumetric and integrative
- Bushing	Visual	Visual, surface, volumetric and integrative
- Other components	Visual, surface, volumetric and integrative	
All components - Corrosion - Erosion	 Dimensional Dimensional	
Heat Exchangers/ Steam Generator - Shell - Tube	 Visual, surface and volumetric Volumetric	
Coolant Tube	PT-CT gap, garter spring location, volumetric, dimensional and hydrogen pickup	
Feeder Pipe	Visual, feeder gap, volumetric and thickness	

[9] If material is subject to stress corrosion or erosion.

APPENDIX-IV
DETERMINATION OF INSPECTION CATEGORIES
[see subsections 4.1.5.1 (e) and (f)][10]

FIGURE 2 (A): DETERMINATION OF INSPECTION CATEGORIES (A, B, C1, C2) FOR MEDIUM FAILURE SIZE
[see subsection 4.1.5.1 (e)][10]

Stress Intensity Ratio, R_{SA} or R_{SM} (see Section 4.1.3)	Low Fatigue	Medium Fatigue	High Fatigue	
1 – 2/3	C1	B	A	**High Stress Intensity**
2/3 – 1/3	C1	B	B	**Medium Stress Intensity**
1/3 – 0	C2	C1	C1	**Low Stress Intensity**
Fatigue Usage Factor (see section 4.1.3)	0 – 0.01	0.01 – 0.1	0.1 – 1.0	

Fatigue Usage Factor (see section 4.1.3)

FIGURE 2 (B): DETERMINATION OF INSPECTION CATEGORIES (A, B, C1, C2) FOR LARGE FAILURE SIZE
[see subsection 4.1.5.1 (f)][10]

Stress Intensity Ratio, R_{SA} or R_{SM} (see Section 4.1.3)	Low Fatigue	Medium Fatigue	High Fatigue	
1 – 2/3	B	A	A	**High Stress Intensity**
2/3 – 1/3	C1	B	A	**Medium Stress Intensity**
1/3 – 0	C2	C1	C1	**Low Stress Intensity**
Fatigue Usage Factor (see section 4.1.3)	0 – 0.01	0.01 – 0.1	0.1 – 1.0	

Fatigue Usage Factor (see section 4.1.3)

[10] Inspection requirements for various categories are given in 4.1.6.2.

APPENDIX-V

DETERMINATION OF MAGNITUDE OF ISI FOR IDENTICAL COMPONENTS

[see subsections 4.1.6.2.1 (b) and 4.1.6.2.2 (d) (ii)]

A-V-1 For identical components[11] that are operated under similar conditions, the following sampling requirements may be acceptable:

(a) For identical components in inspection category A areas, the number of identical components to be inspected should not be less than F_A. The value of F_A can be determined using the curve F_A in Fig. 3.

(b) For identical components in inspection category B areas, the number of identical components to be inspected should not be less than F_B. The value of F_B can be determined using the curve F_B in Fig. 3.

(c) Where the number determined by item (a) or (b) above includes a fraction greater than one-third, it shall be rounded up to the next integer.

A-V-2 The components to be inspected shall be chosen to include those components that are considered to be subjected to the most severe conditions.

A-V-3 In order to include the most significant indications in the inspection programme, as required by criteria given in section 4.1.6.1, it may be necessary to select more areas than required by criteria given in subsections A-V-1 and A-V-2.

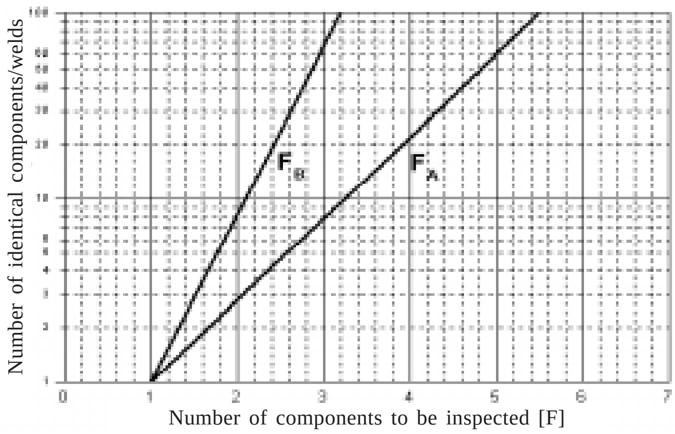

FIGURE 3: INSPECTION SAMPLE FOR IDENTICAL COMPONENTS/WELDS

[11]Identical coponents are components that have same specifications and attributes with respect to design, manufacture, service conditions and operating history.

APPENDIX-VI

INSPECTION SCHEDULE FOR PRESSURISED HEAVY WATER REACTOR BASED NUCLEAR POWER PLANTS

[see subsection 5.2.1]

TABLE-3: DESIRED EVENLY DISTRIBUTED INSPECTION SCHEDULE FOR PRESSURISED HEAVY WATER REACTOR BASED NUCLEAR POWER PLANTS

Inspection interval	Inspection period indicated as calendar year of plant service from commencement of operation	Minimum % examinations required to be completed	Maximum % examinations credited
1st (5 years)	0-2 2-5	16 100	34 100
2nd (10 years)[12]	5-8 8-12 12-15	16 50 100	34 67 100
3rd (10 years)	15-18 18-22 22-25	16 50 100	34 67 100
4th (10 years)	25-28 28-32 32-35	16 50 100	34 100 100

[12] Or one-third of design operational life of the plant, whichever is shorter.

ANNEXURE-I

DESIGN, SERVICE AND TEST LIMITS

(Excerpts from NCA-2142.4 ASME BPVC, Section III, Division 1, 1998)

[see subsections 1.1.4, 4.1.1.2 (b) (ii) (3), AIV-1 and AIV-6.1]

AI-1 Level A Service Limits

Level A service limits are those sets of limits which must be satisfied for all Level A service loadings identified in the design specifications to which the component or support may be subjected in the performance of its specified service function.

AI-2 Level B Service Limits

Level B service limits are those sets of limits which must be satisfied for all Level B service loadings identified in the design specifications for which these service limits are designated. The component or support must withstand these loadings without damage requiring repair.

AI-3 Level C Service Limits

Level C service limits are those sets of limits which must be satisfied for all Level C service loadings identified in the design specifications for which these service limits are designated. These sets of limits permit large deformations in areas of structural discontinuity which may necessitate the removal of the component or support from service for inspection or repair of damage to the component or support. Therefore, the selection of this limit shall be reviewed by the owner for compatibility with established system safety criteria (NCA-2141).

AI-4 Level D Service Limits

Level D service limits are those sets of limits which must be satisfied for all Level D service loadings identified in the design specifications for which these service limits are designated. These sets of limits permit gross general deformations with some consequent loss of dimensional stability and damage repair, which may require removal of the component or support from service. Therefore the selection of this limit shall be reviewed by the owner for compatibility with established system safety criteria (NCA-2141).

AI-5 Alternate Service Limits

Components or supports may be alternatively designed using more restrictive service limits than specified in the design specification. For example, Level B service limits may be used where Level C service limits have been specified.

AI-6 Test Limits

(i) The limits for test loadings shall meet the requirements of the appropriate subsection of this section; and

(ii) The selection of limits for other tests defined by the owner [NCA 2142.3 (b)] shall be included in the design specification.

ANNEXURE-II

ISI OF BOILING WATER REACTOR BASED NUCLEAR POWER PLANTS

[see subsection 4.2]

AII-1 Introduction

This annexure outlines principles of PSI/ISI programme applied to plants with light water cooled reactors of BWR type. The governing requirements of ISI should be as per ASME Section XI.

AII-2 PSI

All examinations are required to be performed completely on all components listed in the ISI programme as a pre-service examination before initial plant startup, including essentially 100% of the pressure-retaining welds. For the irradiation region of the reactor vessel, these examinations include the material (base metal) of any weld repair areas where the repair depth exceeds 10% of the nominal wall thickness. This region surrounds, and extends to the length of the fuel element assemblies.

AII-3 ISI

AII-3.1 Components may be exempted from the surface and volumetric examinations where the loss of coolant after a failure of those components would not exceed the make-up capacity, using only on-site power required for the normal shut-down, cool down and maintenance of the cool down condition. This exemption extends to components whose connections have a nominal pipe size of 25 mm or less, and to piping systems with a nominal pipe size of 25 mm or less.

AII-3.2 Components exempted from surface and volumetric examinations should be visually examined during periodic system hydrostatic tests.

AII-3.3 Welds of the reactor pressure vessel and of the steam generators, where it is specified in the ISI programme, should be examined 100% over the inspection interval. Examination of the pressure-retaining welds in other components and in components installed in other systems (e.g. safety systems and auxiliary systems) need not meet the 100% requirement, and the extent of the examination may be reduced by a sampling approach. Weld examinations should extend over the heat-affected zone on either side of the welds, wherever possible.

AII-3.4 A sampling programme should be developed for the examination of components with the same constructional design, manufacturing method and manufacturer, and which performs functions in the system such as those performed by pumps, piping and valves.

AII-3.5 Select sufficient sample of representative areas of cladding for examination to provide assurance of the integrity of all remaining areas.

AII-4 Inspection Schedules

AII-4.1 Either an evenly distributed inspection schedule or a variably distributed inspection schedule may be selected, and one may be substituted for the other during the first three years of the operating life of NPP.

AII-4.2 The preferable schedule is the variably distributed inspection schedule, which will provide a greater confidence of safety and eliminate potential deficiencies in the early years of plant life.

AII-4.3 The inspection interval specified in subsection AII-4.4, in which all examinations in the programme are to be completed, may be decreased or extended by as much as one year to enable an inspection to coincide with a plant outage during this period.

AII-4.4 The two different types of schedules based on inspection periods of three to four years are given in Tables AII-1 and AII-2 as possible implementation of the requirements in subsection 5.2 of the guide.

In these schedules assuming a 40-year plant life, the minimum percentage of the examinations required to be completed during each inspection period are shown. There is a specified maximum percentage of examinations that is accepted as having been completed, even if the percentage of examinations actually performed is in excess of this figure. These maxima are shown in column 4 of Tables AII-1 and AII-2 under the heading 'Maximum % Examinations Credited'. The schedule for other periods of operating life assumed in the design may be adjusted accordingly.

AII-5 Test requirements

AII-5.1 The system leakage tests and in-service system hydrostatic pressure tests should follow the provisions set out in ASME Section XI.

AII-5.2 In establishing the requirement for in-service hydrostatic pressure test, a number of competing factors are considered. Among the more important of these factors are a demonstration of overall pressure tightness; the stresses and strains under test conditions compared with those under operating conditions; the changes in material properties over the plant life; the extent to which assurance is increased that large flaws may not exist; the significance of fatigue cycling and crack propagation. The test requirements need continual review since on-going research and development in this field might provide new information to be taken into account in future.

AII-6 Typical Criteria for Selection of Items

(a) Pressure-retaining parts of components in the reactor coolant system.

(b) Components of and components connected to the primary reactor coolant system essential for ensuring the shutdown of the reactor and the cooling of the nuclear fuel in relevant operational states and in postulated accident conditions.

(c) Other components, the dislodgement or failure of which might put in jeopardy the systems mentioned in items (a) and (b) above.

(d) In general, the selection of items for ISI should follow provisions of ASME Boiler and Pressure Vessel Code, Section XI.

AII-7 Extent of ISI Programme

AII-7.1 In establishing the extent of the ISI programme consideration shall be given to the following systems and components in accordance with their importance to safety as mentioned above in section AII-6.

AII-7.2 Components subjected to ISI in accordance with subsection AII-7.1 shall be examined by visual, surface and volumetric methods as a general rule. In addition, the integrity of pressure-retaining components shall be checked by system leakage test and the hydro-test.

AII-7.3 Consistent with their importance to safety some components may be exempted from surface and volumetric examinations, either because of the size of their connections, or of the number of barriers between the component and the fuel or the outside environment. In such cases, however, these components are not exempted from the examinations for evidence of leakage as part of the system leakage tests.

AII-7.4 The number, frequency and extent of ISI of identical systems and components may be reduced by a sampling programme that will vary according to the design, the number of identical components or systems involved, operational requirements, or the existence of identical units in a multiple unit plant. The sampling criteria should be consistent with the importance to safety of the component and the rate of degradation.

AII-8 Extent of PSI

PSI requirements are same as described in subsection 4.1.7.

AII-9 Inspection Schedule

Either of the two kinds of examination schedules may be followed. The 'Evenly Distributed Inspection Schedule' and the 'Variably Distributed Inspection Schedule' for BWR are given in Table AII-1 and Table AII-2 respectively[13].

TABLE AII-1: EVENLY DISTRIBUTED INSPECTION SCHEDULE FOR BOILING WATER REACTOR BASED NUCLEAR POWER PLANTS
[**see subsection AII-4**]

Inspection interval	Inspection period indicated as calendar year of plant service from commencement of operation (years)	Minimum % examinations required to be completed	Maximum % examinations credited
1st (10 years)	0-3 3-7 7-10	16 50 100	34 67 100
2nd (10 years)	10-13 13-17 17-20	16 50 100	34 67 100
3rd (10 years)	20-23 23-27 27-30	16 50 100	34 67 100
4th (10 years)	30-33 33-37 37-40	16 50 100	34 100 100

[13] Note: 1. Plant life is assumed as 40 years for the tables.
2. These may not be applicatble to inspection of steam generators, heat exchangers or feeder pipes.

TABLE AII-2: VARIABLY DISTRIBUTED INSPECTION SCHEDULE FOR BOILING WATER REACTOR BASED NUCLEAR POWER PLANTS

[see subsection AII-4]

Inspection interval	Inspection period indicated as calendar year of plant service from commencement of operation (years)	Minimum % examinations required to be completed	Maximum % examinations credited
1st (3 years)	0-3	100	100
2nd (7 years)	3-7 7-10	33 100	67 100
3rd (13 years)	10-13 13-17 17-20 20-23	16 40 66 100	34 50 75 100
4th (17 years)	23-27 27-30 30-33 33-37 37-40	8 25 50 75 100	16 34 67 100 100

ANNEXURE-III

TEST REQUIREMENTS FOR PRESSURE RETAINING ITEMS

[see subsection 3.3]

AIII-1 Pressure-retaining items shall be subjected to the following :

(a) A system leakage test as a part of ISI.

(b) A system leakage test, undertaken before resuming operation, following each reactor outage where the integrity of the reactor coolant pressure boundary may have been affected.

(c) A system hydrostatic pressure test at or near the end of each inspection interval, if required.

(d) The pressure retaining components shall be visually examined to the extent practicable while the system is under the test pressure and temperature. The test pressure and temperature shall be maintained for a sufficient period of time before the examinations, to permit leakage to be identified.

(e) If leakages (other than normal/controlled leakage) are detected during the above test, the source of leakage shall be located, and the area examined to the extent necessary to establish if any corrective action is required.

(f) The system leakage test shall be performed at the test pressure that is not less than the specified system operating pressure.

(g) The duration of a test performed at a pressure higher than system design pressure shall be limited to prevent excessive stressing of the systems and components.

ANNEXURE-IV

SIZE OF FAILURE, FATIGUE USAGE FACTOR AND STRESS INTENSITY

[see Section No. 4.1.3]

AIV-1 Size of failure for components of systems located inside the containment boundary, considering Level A and B conditions (Annexure-I), can be determined as follows:

AIV-1.1 Piping, Pumps and Valves

Size of failure is expressed as the ratio of the maximum energy release rate from the failure being considered to the maximum energy release rate from most severe failure considered during the design of the systems that directly transport heat from the nuclear fuel.

Where the maximum energy release ratio (R_E) is not determined by detailed analysis, it shall be determined as follows:

$$R_E = \frac{A_F}{A_D} \times \frac{h_F}{h_D} \times \frac{\sqrt{P_F}}{\sqrt{P_D}} \times \frac{\sqrt{r_F}}{\sqrt{r_D}}$$

where

A = Flow cross-sectional area (guillotine failure).

h = Enthalpy at operating temperature and pressure minus the enthalpy of saturated liquid at atmospheric pressure.

P = Operating pressure (gauge).

r = Density of fluid.

Subscript F = Conditions for location being considered.

Subscript D = Conditions for design maximum energy release rate.

Failure size classification:

Small failure	: R_E £ 0.1
Medium failure	: 0.1 > R_E < 0.3
Large failure	: R_E ³ 0.3

AIV-1.2 Vessels

A size of failure based on energy release rate should be determined as mentioned above, using flow cross-sectional areas of the lines connected to the vessel that would continue to supply fluid to the failure. A failure size based on containment pressure rise shall also be determined. It shall be expressed as the ratio of the immediate containment pressure rise caused by the instantaneous release of fluid in the vessel, assuming a sealed containment, to the lowest pressure rise required to cause closure of the containment ventilation system.

Where the containment pressure rise ratio (R_v) is not determined by detailed analysis, it shall be determined as follows, assuming uniform vapour distribution and no fluid heat loss:

R_v = Containment pressure rise ratio, P_C/P_S

where

P_C = Containment pressure rise = $P \dfrac{V_T}{V_C}$

P_S = Lowest containment pressure rise to cause the closure of containment ventilation system

V_C = Containment volume

V_T = Vapour volume at pressure, $P = Wg \times Vg$

Vg = Specific volume of saturated vapour at pressure P

P = Operating pressure (gauge)

Wg = Weight of vapour produced, $\dfrac{W_l\,(h-h_f)}{h_{fg}}$

W_l = Maximum weight of liquid in vessel (operating condition)

h = Enthalpy of liquid at operating condition

h_f = Enthalpy of saturated liquid at normal containment pressure, P

h_{fg} = Enthalpy of evaporation at pressure, P

Failure size classification based on containment pressure rise shall be as follows:

Small failure : $R_v < 1$

Medium failure : $1 \leq R_v \leq 3$

Large failure : $R_v > 3$

If the two failure sizes determined from R_E and R_v differ, the larger should be taken as the size of failure.

AIV-1.3 Associated Pipelines Connecting to Failed Pipe or Component

Flow from a pipeline connected to a failed pipe or component need not be included in computing size of failure if the line incorporates two remotely operated or self-closing valves in series, provided that such valves and their power supplies are

(a) not connected to the same power source; and

(b) sufficiently remote from one another that both would not be subjected to damage or malfunction from the same failure.

AIV-2 Components in Systems Located Outside or Forming Part of the Containment Boundary

For components in systems classified under subsection 4.1.1.1 (a) located outside or forming a part of containment boundary, all failures capable of causing a radiation hazard equal to or exceeding the AERB dose limits for a serious process failure (category 2 event or above as defined in 'Design Basis Events for PHWR', AERB/SG/D-5) shall be classified as large.

AIV-3 Size of Failure for Components of the Systems Classified in Subsection 4.1.1.1 (b)

(a) Failure that may prevent functioning of the systems as mentioned in safety report shall be considered as large.

(b) Failures that do not prevent functioning of the system as mentioned in safety report shall be considered as small.

AIV-4 Size of Failure for Components of the Systems Classified in Subsection 4.1.1.1 (c)

(a) The size of failure shall be classified according to the total effect of the initiating and resulting failures.

(b) For dislodgement, that may lead to failure of a system classified in sub-section 4.1.1.1 (a), the relevant rules in section AIV-1 or AIV-2 above shall be used to determine the size of failure.

(c) For dislodgement, that may lead to failure of a system classified in sub-section 4.1.1.1 (b), the rules in subsection AIV-3 above shall be used to determine the size of failure.

AIV-5 Fatigue Usage Factor

AIV-5.1 Fatigue usage factor used in establishing inspection categories shall be determined by the rules given in Section III, ASME Boiler and Pressure Vessel Code.

AIV-5.2 Fatigue usage factor classification shall be as follows:

Low fatigue - Fatigue usage factor of 0.01 or less;

Medium fatigue - Fatigue usage factor greater than 0.01 and less than 0.1; and

High fatigue - Fatigue usage factor of 0.1 or greater.

AIV-5.3 Where the fatigue usage factor is not calculated, a value of 0.05 should be used.

AIV-6 Stress Intensity

AIV-6.1 Stresses or stress intensities may be used to calculate stress ratios as follows:

For Level A and B conditions (Annexure-I) the calculated stress or stress intensity shall be compared to the allowable stress or stress intensity such that

$$\text{Stress ratio} = R_s = \frac{\text{Calculated stress or stress intensity}}{\text{Allowable stress or stress intensity}}$$

This should be determined for all stresses and stress combinations that are calculated to meet design requirements for normal, upset and emergency conditions. These stress ratios are determined for each potential inspection area, and the highest ratio then determines the stress classification.

AIV-6.2 The use of more than one method of stress calculation may produce different calculated stress or stress intensities. The calculated stress intensity to be used in subsection AIV-6.1 should be taken from the most precise method employed. The following methods are listed in order of precision:

(1) Methods of analysis more refined than those given in Section III, ASME Boiler and Pressure Vessel Code, e.g. finite element analysis.

(2) Methods using the rules given in Section III, ASME Boiler and Pressure Vessel Code.

(3) Methods for determining tentative metal thickness and similar formulae given in Section III, ASME Boiler and Pressure Vessel Code.

AIV-6.3 Where the stress analysis considers more than one condition several values of R_s may be obtained. The maximum value of R_s shall be equal to R_{SM} .

AIV-6.4 For material where the value of S_a corresponding to 10^6 cycles is greater than S_m the ratio R_{SM} may be reduced as follows:

R_{SA} = stress ratio adjusted = R_{SM} ´ S_m / S_a

where

S_m is defined in NB-3200, Section III, ASME Boiler and Pressure Vessel Code.
S_a is defined in NB -3222, Section III, ASME Boiler and Pressure Vessel code.

ANNEXURE-V

GROUPS OF EXISTING PRESSURISED HEAVY WATER REACTOR BASED NUCLEAR POWER PLANTS

[see subsections 13.2.2 (b) and 13.2.4.2]

The existing NPPs are grouped into four categories based on the pressure tube material and garter spring types as follows:

GROUP 1: Early generation reactor with open annulus and zircaloy-2 pressure tube having two loosely fitted garter springs (RAPS-1 and MAPS-1).

GROUP 2: Reactors with closed annulus and zircaloy pressure tube having loosely fitted four garter springs (NAPS-1 & 2, KAPS-1).

GROUP 3: Early generation core replaced reactor with open annulus and Zr-Nb pressure tube having four tightly fitted garter springs (RAPS-2 and MAPS-2).

GROUP 4: New reactors with closed annulus and Zr-Nb pressure tubes having four tightly fitted garter springs (KAPS-2 onwards).

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LIST OF PARTICIPANTS

WORKING GROUP

Dates of meeting:	February 1, 17 & 25, 1993	June 5 & 6, 1997
	March 12, 1993	June 16 & 17, 1997
	August 26, 1993	December 17, 1998
	September 27 & 28, 1993	January 15, 1999
	November 4 & 5, 1993	August 27, 2001
	September 7 & 8, 1994	September 5, 2001
	April 30, 1997	November 12, 2002

Members of the working group:

Dr. S.K. Shrivastava (Chairman)	:	NPCIL
Shri D.N. Gaur	:	NPCIL
Shri C.P. Mungikar	:	NPCIL
Shri B.B. Rupani	:	BARC
Shri S. Vijay Kumar	:	NPCIL
Shri U.K. Paul (Member-Secretary)	:	AERB

ADVISORY COMMITTEE FOR CODES, GUIDES, AND MANUALS FOR SAFETY IN OPERATION OF NUCLEAR POWER PLANTS (ACCGASO)

Dates of meeting :	September 27 & 28, 1996 April 3 & 4, 1998 March 27, 1999 August 27, 1999

Members of ACCGASO:

Shri G.V. Nadkarni (Chairman)	:	NPCIL (Former)
Shri V.S. Srinivasan	:	NPCIL
Shri V.V. Sanath Kumar	:	NPCIL
Shri Y.K. Joshi	:	NPCIL
Shri K.M. Sinha	:	NPCIL (Former)
Shri Ravindranath	:	NPCIL
Shri Ram Sarup	:	AERB (Former)
Shri R.S. Singh	:	AERB (Former)
Shri S.T. Swamy (Co-opted)	:	AERB
Shri S.K. Warrier (Member-Secretary)	:	AERB

ADVISORY COMMITTEE ON NUCLEAR SAFETY (ACNS)

Date of meeting: July 12, 2002

Members of ACNS:

Shri Ch. Surendar (Chairman)	:	NPCIL (Former)
Shri S. K. Sharma (Vice-Chairman)	:	BARC
Dr. V. Venkatraj	:	BARC
Shri R.K. Sinha	:	BARC
Shri S. P. Singh	:	AERB (Former)
Shri S.S. Bajaj	:	NPCIL
Shri Ramesh D. Marathe	:	L&T, Mumbai
Shri S.K. Agarwal	:	AERB
Shri K. Srivasista (Member-Secretary)	:	AERB
Shri B.B. Rupani (Invitee)	:	BARC
Shri S.K. Mishra (Invitee)	:	BARC
Shri S. Vijaykumar (Invitee)	:	NPCIL

ADVISORY COMMITTEE ON NUCLEAR SAFETY (ACNS) (contd.)

Date of meeting:	November 27, 1999

Members of ACNS:

Shri S. K. Mehta (Chairman)	:	BARC (Former)
Shri S. M. C. Pillai	:	Nagarjuna Group, Hyderabad
Prof. U. N. Gaitonde	:	IIT, Bombay
Shri S. K. Goyal	:	BHEL, Hyderabad
Shri Ch. Surendar	:	NPCIL (Former)
Dr. U.C. Mishra	:	BARC (Former)
Shri S. K. Sharma	:	BARC
Dr. V. Venkatraj	:	BARC
Shri S. P. Singh	:	AERB (Former)
Shri G.K. De	:	AERB (Former)
Shri K. Srivasista (Member-Secretary)	:	AERB

PROVISIONAL LIST OF SAFETY CODES, GUIDES AND MANUAL ON OPERATION OF NUCLEAR POWER PLANTS

Safety Series No.	Provisional Title
AERB/SC/O	Code of Practice on Safety in Nuclear Power Plant Operation
AERB/SG/O-1	Staffing, Recruitment, Training, Qualification and Certification of Operating Personnel of Nuclear Power Plants
AERB/NPP/SG/O-2	In-Service Inspection of Nuclear Power Plants
AERB/SG/O-3	Operational Limits and Conditions for Nuclear Power Plants
AERB/SG/O-4	Commissioning Procedures for Pressurised Heavy Water Reactor Based Nuclear Power Plants
AERB/SG/O-5	Radiation Protection during Operation of Nuclear Power Plants
AERB/SG/O-6	Preparedness of the Operating Organisation for Handling Emergencies at Nuclear Power Plants
AERB/SG/O-7	Maintenance of Nuclear Power Plants
AERB/SG/O-8	Surveillance of Items Important to Safety in Nuclear Power Plants
AERB/SG/O-9	Management of Nuclear Power Plants for Safe Operation
AERB/SG/O-10A	Core Management and Fuel Handling in Operation of Pressurised Heavy Water Reactors
AERB/SG/O-10B	Core Management and Fuel Handling in Operation of Boiling Water Reactors
AERB/NPP/SG/O-11	Management of Radioactive Wastes Arising from Operation of Nuclear Power Plants
AERB/SG/O-12	Renewal of Authorisation for Operation of Nuclear Power Plants
AERB/NPP/SG/O-13	Operational Safety Experience Feedback on Nuclear Power Plants
AERB/NPP/SG/O-14	Life Management of Nuclear Power Plants
AERB/NPP/SG/O-15	Proof and Leakage Rate Testing of Reactor Containments
AERB/NF/SM/O-1	Probabilistic Safety Assessment Guidelines

AERB SAFETY GUIDELINES NO. AERB/NPP/SG/O-2

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