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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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TECHNICAL BASIS FOR COMMENDABLE PRACTICES ON AGEING MANAGEMENT

SCC and Cable Ageing Project (SCAP)

Final Report

The complete version is only available in PDF format.

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The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full member. NEA membership today consists of 28 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, Norway, Portugal, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

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Committee on the Safety of Nuclear Installations

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of senior scientists and engineers, with broad responsibilities for safety technology and research programmes, and representatives from regulatory authorities. It was set up in 1973 to develop and co-ordinate the activities of the NEA concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations.

The committee's purpose is to foster international co-operation in nuclear safety amongst the OECD member countries. The CSNI's main tasks are to exchange technical information and to promote collaboration between research, development, engineering and regulatory organisations; to review operating experience and the state of knowledge on selected topics of nuclear safety technology and safety assessment; to initiate and conduct programmes to overcome discrepancies, develop improvements and research consensus on technical issues; to promote the co-ordination of work that serves to maintain competence in the nuclear safety matters, including the establishment of joint undertakings.

The committee shall focus primarily on existing power reactors and other nuclear installations; it shall also consider the safety implications of scientific and technical developments of new reactor designs.

In implementing its programme, the CSNI establishes co-operative mechanisms with the NEA's Committee on Nuclear Regulatory Activities (CNRA) responsible for the programme of the NEA concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with the following NEA committees on matters of common interest: Committee on Radiation Protection and Public Health (CRPPH), Radioactive Waste Management Committee (RWMC) and the Nuclear Science Committee (NSC).

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Executive summary

The number of ageing nuclear power plants is increasing in OECD/NEA member countries. Accordingly, maintenance programmes, in-service inspection and testing of structures, systems and components important to safety have been implemented to ensure that levels of reliability and effectiveness remain in accordance with the design assumptions. This is often being done using an integrated ageing management strategy based on state-of-the-art technologies.

Ageing effects, especially material degradation, have progressively been experienced world wide and since the start of nuclear power plant operation. Material degradation is expected to continue as plants age and operating licenses are extended.

Two subjects – stress corrosion cracking (SCC) and degradation of cable insulation – were selected as the focus of the SCC and Cable Ageing Project (SCAP) due to their relevance for plant ageing assessments and their implications on nuclear safety. In order to achieve that goal, 14 NEA member countries joined the project in 2006 to pool their knowledge, and 3 additional countries joined during the course of the project. The International Atomic Energy Agency (IAEA) and the European Commission participated as observers.

When establishing the project it was recognised that in the limited time available ageing management could not be addressed in detail over a large range of topics. SCC has been and continues to be a serious problem and in recent years cable ageing has been identified as an area requiring more attention from both regulators and industry. Incidents in these areas often occur and draw attention. These two topics were therefore chosen for specific study in the project as examples of areas in which ageing management has been applied for many years and in which ageing management still needs to be developed through an internationally co-ordinated study which could yield greater insights into the management of these topics.

The objective of this internationally co-ordinated project is to share the corporate knowledge and operating experience to understand the failure mechanisms and identify effective techniques and technologies to effectively manage and mitigate active degradation in nuclear power plants.

The specific objectives of the project are to: *i)* establish a complete database with regard to major ageing phenomena for SCC and degradation of cable insulation through collective efforts by NEA members; *ii)* establish a knowledge base in these areas by compiling and evaluating the collected data and information systematically; *iii)* perform an assessment of the data and identify the basis for commendable practices which will help regulators and operators to enhance ageing management.

The scope of this project involves the development of a knowledge base and commendable practices that address common elements in the management of ageing and mitigation of failures for components and cables: study of ageing effects, investigation of failure mechanisms, mitigation of influencing factors, prediction of conditions for replacement, safety assessment of components, qualification testing (environmental qualification for cables) and condition monitoring.

According to IAEA Safety Guide NS-G-2.12 “Ageing Management for Nuclear Power Plants”, effective ageing management throughout the service life of SSC requires the use of a systematic approach to managing ageing that provides a framework for co-ordinating all programmes and activities relating to the understanding, controlling, monitoring and mitigation of ageing effects of the plant component or structure.

The SCAP SCC knowledge base was established by analysing and evaluating the data from the viewpoint of the implementation of appropriate ageing management beneficial both to the regulators and licensees. The practices of particular importance (*e.g.* key to the maintenance of SCC) were identified as commendable practices for the purpose of appropriate ageing management of SCC, the establishing or improvement of ageing management programmes under the technical information basis by analysing and evaluating the data from the viewpoint of the implementation of appropriate ageing management beneficial both to the regulators and operators.

The SCAP SCC event database, knowledge base as well as commendable practices will be of great help for less experienced staff to access past and present experiences and knowledge for the evaluation of new events. It will support a balanced assessment of the new events in the context of all applicable historic knowledge, aiding all stakeholders in terms of informed decision making. Four years is an extremely short time to establish such a complex event database and knowledge base in the field of stress corrosion cracking. Without the input from the OECD Piping Failure Data Exchange Project (OPDE), the event database could not have been developed in the time frame available. Members of the SCC Working Group have expressed a firm commitment to continue to populate and develop the event database and the knowledge base. It is envisaged that co-operation with the OPDE project will provide a suitable platform for this continuation.

For cable ageing, the crucial point is the knowledge about the qualification procedure for harsh environment and the predictive capability to estimate the remaining qualified life. The cable condition-monitoring techniques shared by the participants have become an up-to-date encyclopaedic source to monitor and predict the performance of every unique application of cables. The experience from events is rather limited and failures are in part strongly related to past manufacturing and installation practices. The insights collected in this project have offered greater levels of knowledge in ageing mechanisms and technically sound bases to address life extension and continued qualification of cables. In addition to the technical data and operating experience, the SCAP cable database and knowledge base provides up-to-date information on environmental qualification of cables that are to remain functional during and following a design basis event. The database has incorporated the publications on recent research results on ageing mechanisms and continued efforts in enhancing condition monitoring capability.

The Cable Working Group has created an encyclopaedia on cables useful for both novice and experienced NPP regulators and operators. This will be a living reference book on the web and its value will be inestimable if it is kept up-to-date.

The two working groups have brought together representatives of regulators, licensees, vendors and academics working in the field, and this combination has been found to be invaluable for the successful execution of the project. It is expected that the expert network will facilitate the sharing of knowledge as well as increase co-operation among experts outside the project.

The products of each working group are a database, a knowledge base and a report describing these and the commendable practices that the groups consider should be used in the ageing management of cable and stress corrosion cracking.

The commendable practices identified in this report are intended to strengthen technical approaches to optimise ageing management in the areas of SCC and cable ageing. The SCAP SCC and Cable data- and knowledge bases provide extensive information to benefit all stakeholders in designing, constructing, operating and regulating nuclear power plants and also provide commendable practices applicable to new reactors.

The working process of SCAP has also provided an important example to demonstrate how such a challenging task can be effectively addressed and therefore could be used as a basis for other topics in ageing management. Vital elements of the working process have been the identification of priority items of common interest, the assignment of a dedicated project co-ordinator, chairperson and clearing house with expert knowledge and lead organisations providing input to start the discussion and provide orientation.

This report summarises the product of SCAP work resulting from four years of technical interactions and shared knowledge from all participants from June 2006 to June 2010. The project was financed through a Japanese voluntary contribution to the NEA.

Chapter 1: Introduction

1.1 Background

The number of ageing nuclear power plants is increasing in OECD/NEA member countries. Accordingly, maintenance programmes, in-service inspection and testing of structures, systems and components important to safety have been implemented to ensure that levels of reliability and effectiveness remain in accordance with the design assumptions. This is often being done using an integrated ageing management strategy based on state-of-the-art technology.

Ageing effects, especially material degradation, have progressively been experienced world wide and since the start of nuclear power plant operation. Material degradation is expected to continue as plants age and operating licenses are extended. It is clear that unanticipated and unmanaged structural degradation could result in significant loss of safety margins, undermining public confidence and straining the resources of both regulatory authorities and the operators.

For regulatory authorities, it is also important to verify the adequacy of the ageing management methods applied by the licensees, based on reliable technical evidence. Two subjects – stress corrosion cracking (SCC) and degradation of cable insulation – were selected as the focus of the SCC and Cable Ageing Project (SCAP) due to their implications for nuclear safety and their relevance for plant ageing assessment.

In order to achieve that goal, 14 NEA member countries initially joined the project in 2006 to pool their knowledge. A further 3 countries joined while the project was being conducted, for a total of 17 participating countries. The International Atomic Energy Agency (IAEA) and the European Commission also participate as observers. The project was financed through a Japanese voluntary contribution to the NEA. Japanese technical institutions are also actively co-operating in the project under the co-ordination of the Nuclear and Industrial Safety Agency (NISA) of Japan.

When establishing the SCAP project it was realised that in the limited time available ageing management could not be addressed in detail over a large range of topics. Stress corrosion cracking has and continues to be a serious problem and in recent years cable ageing has been identified as an area requiring more attention from both regulators and industry. Incidents in both areas continue to provide periodic surprises. These two topics were therefore chosen for specific study in the SCAP project, being examples of areas in which ageing management has been applied for many years (SCC) and one in which ageing management still needs to be developed (cable ageing), in an internationally co-ordinated study which was anticipated could yield greater insights into the management of these failures.

This report summarises the project results of SCAP after four years of operation, from June 2006 to June 2010.

1.2 Objective

The objective of this internationally co-ordinated project is to share the corporate knowledge and operating experience so as to better understand the failure mechanisms and identify effective techniques and technologies to manage and mitigate active degradation in nuclear power plants.

The specific objectives of the project are to:

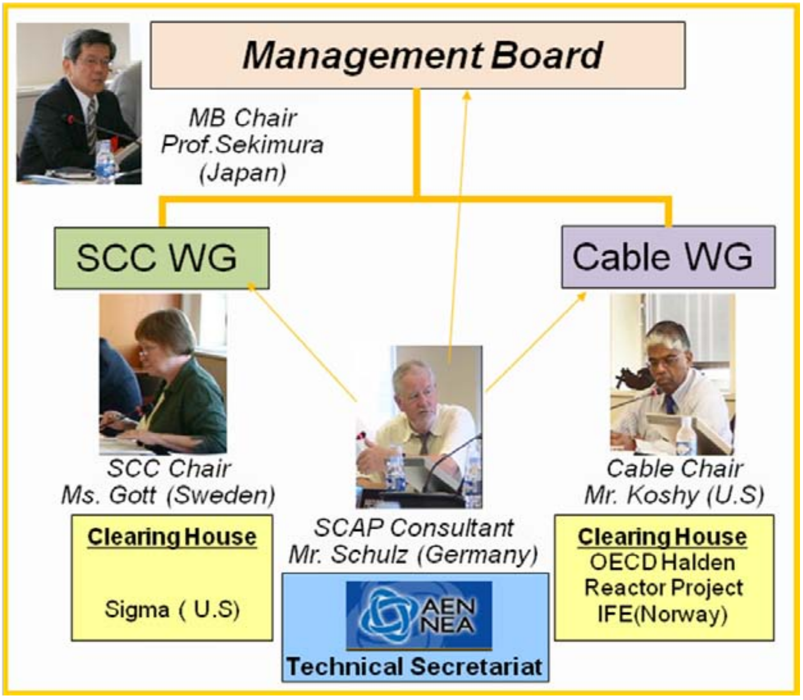
- establish a complete database with regard to major ageing phenomena for SCC and degradation of cable insulation through collective efforts by OECD/NEA members;
- establish a knowledge base in these areas by compiling and evaluating the collected data and information systematically;
- perform an assessment of the data and identify the basis for commendable practices which will help regulators and operators to enhance ageing management.

1.3 Project organisation

The project participants are experts in the fields of SCC and cable ageing and come from regulators, industry, research institutions and academia. They provide the relevant information and perform the assessments needed for the proper execution of the programme.

The Project Management Board (MB) runs the project with assistance from the NEA Project Secretariat (Figure 1.3-1). The MB responsibilities include, but are not limited to: approving the programme of work to be carried out by the working groups on SCC and cable; monitoring the project's progress in terms of results and time lines; and supervising reporting within and outside the project.

Figure 1.3-1: SCAP organisation structure



There are two working groups, one dealing with SCC and the other with cable insulation degradation. The working groups are responsible for carrying out the programme of work and ensuring the quality and timeliness of the reporting within and outside the project. Each working group is supported by a clearing house. The clearing houses work to ensure the consistency of the data contributed by the participating countries. They verify whether the information provided complies with the SCAP Coding Guidelines. They also verify the completeness and accuracy of the data, and maintain and distribute copies of the databases.

Seventeen (17) member countries are participating in one or both of the SCAP working groups: Argentina, Belgium, Canada, the Czech Republic, Finland, France, Germany, Japan, Mexico, Norway, the Republic of Korea, the Slovak Republic, Spain, Sweden, Switzerland, Ukraine and the United States.

Participation in the project is open to the government of any country, whether or not they are a member of the OECD, or to any national agency, public or private organisation designated by such government provided the government agrees with the Terms of Reference which were specified during the first Management Board meeting. Each member country nominates a national representative for both the SCC Working Group and/or Cable Working Group respectively who is responsible for the administration of the project within his/her respective country.

Each participating country has submitted data through its national representative. The data has been entered according to a coding format which was specifically developed for the databases and which is explained in the Coding Guidelines and the Quality Assurance Programme for the two working groups. Each participant has been exclusively responsible for its use of information generated under the project. Both the SCC and the cable databases are password protected, and the contents and data analysis results are distributed only among active working group members under the project Terms of Reference.

1.4 Scope

Based on differences in the fundamental knowledge concerning the SCC and cable insulation degradation mechanisms, as well as the operating experience associated with SCC and cable insulation degradation events, it was expected that the scope and focus of the databases for these two topics would differ from each other. The SCC event database is based on event occurrences, including piping and component failures. On the other hand, since cable failure or event occurrences are rare under normal operating conditions and because of the highly magnified cable stresses in an accident mitigation environment, the cable database is focused on cable material qualifications and condition monitoring methodology and its validation.

The scope of this project involves the development of a knowledge base and commendable practices that address common elements in the management of ageing and mitigation of failures for components and cables: study of ageing effects, investigation of failure mechanisms, mitigation of influencing factors, prediction of conditions for replacement, safety assessment of components, qualification testing (environmental qualification for cables) and condition monitoring.

1.5 Ageing management

The NEA has initiated various activities in the last decade to collect information and share experience regarding ageing mechanisms, developing databases, technical reports, position and guidance documents in support of assessment and management of long-term operation (the more recent NEA reports are Refs. [1-5]).

The IAEA has worked on ageing problems and their relevance for the safety of nuclear power plants since the mid 1980s. The first documentation was published in 1990 as an overall description of the problem [6]. Shortly after, in 1991 and 1992, IAEA published two documents describing methods to approach ageing management [7,8]. At the end of 1990 a document was published concerning the introduction and supervision of ageing management programmes [9]. Since 1992 a number of specific advisory documents have been published concerning different components [10,11], including their relevance for safety and why they should be included in an ageing management programme, which degradation mechanisms can be expected to occur and how to detect and manage them. In 2009 the collective experience of these documents was summarised with the publication of an IAEA Safety Guide [12-14].

Ageing problems are best tackled through a systematic programme in which existing relevant activities for ageing management are co-ordinated. Therefore ageing management includes the documentation of relevant programmes and activities and a description of how these different programmes are co-ordinated in a systematic manner that guarantees continuous improvement by incorporating operational experience and relevant research results. The documentation should also address which maintenance, control, inspection and monitoring should be covered is necessary as well as the frequency and the scope of these activities. To maintain degradation to an acceptable level, it is necessary to understand possible degradation mechanisms; suitable operational conditions that are designed to minimise degradation; control, inspection and monitoring techniques that need to be used to detect degradation in time; evaluation criteria to determine if sufficient safety margins remain if degradation is detected, and methods to manage, repair or replace components. In order to work in a systematic manner, it is necessary to understand the underlying design principles, including relevant regulations, codes and standards, operational and maintenance histories, results from inspection programmes, safety evaluation criteria and procedures, generic operational experience and research results.

According to the IAEA Safety Guide, effective ageing management throughout the service life of structures, systems and components (SSC) requires the use of a systematic approach to managing ageing that provides a framework for co-ordinating and harmonising all the programmes and activities related to the understanding, controlling, monitoring and mitigation of ageing effects of the plant component or structure. This approach is illustrated in Figure 1.5-1, which is an adaptation of Deming's "PLAN – DO – CHECK – ACT" cycle to the ageing management of SSC based on understanding the ageing of a structure/component; the closed loop of Figure 1.5-1 indicates the continuous improvement of the ageing management programme for a particular structure or component, on the basis of feedback of relevant operating experience and results from research and development, and results of self-assessment and peer reviews, to help ensure that emerging ageing issues will be addressed adequately. Every ageing management programme should have nine generic attributes described in the IAEA Safety Guide:

- scope of the ageing management programme;
- preventative actions to minimise control and ageing degradation;
- detection of ageing effects;
- monitoring and trending of ageing effects;
- mitigation of ageing effects;
- acceptance criteria;
- corrective actions;
- operating experience feedback and feedback of research and development;
- quality management.

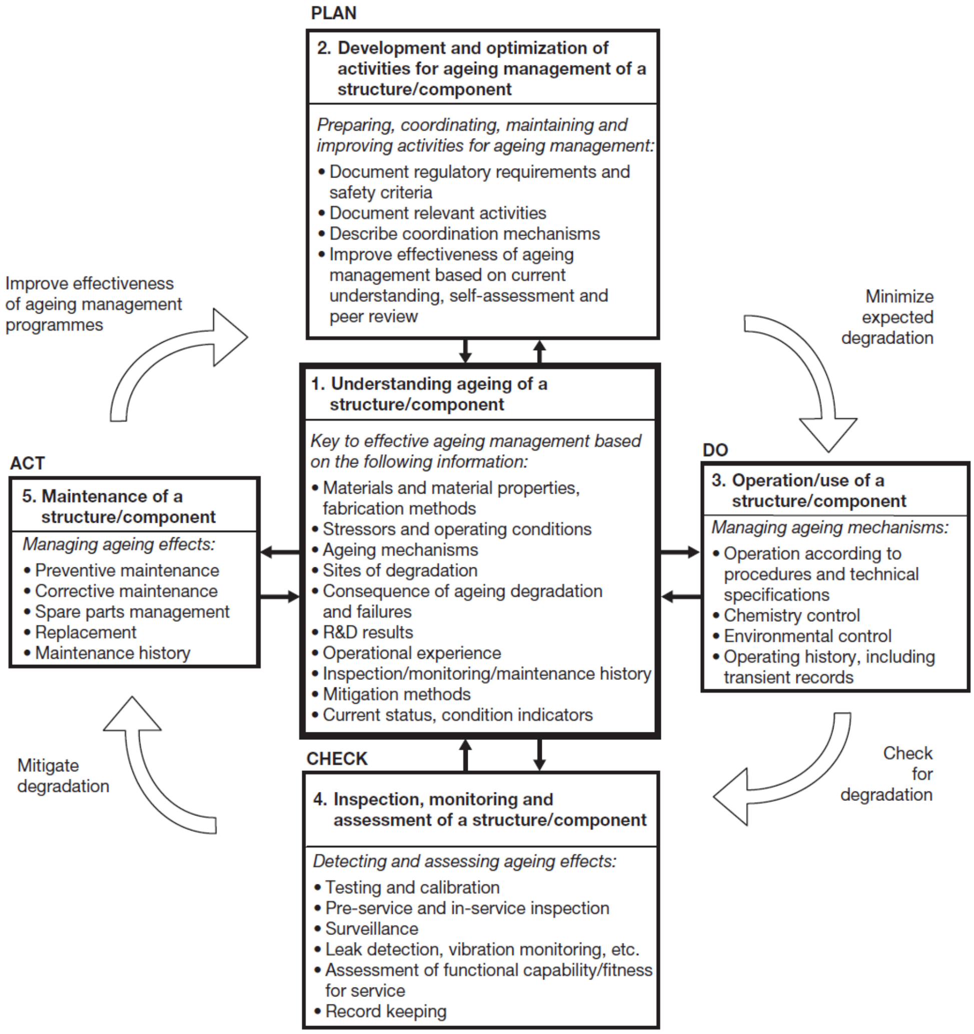
For the safe long-term operation of nuclear power plants it is necessary to develop and implement effective ageing management programmes. It is generally recognised that a proactive approach is preferred since this allows the nine attributes of a successful programme in accordance with the IAEA Safety Guide to be implemented in a timely manner.

However, effective ageing management can be jeopardised by several factors that might lead either to unexpected or premature ageing (*i.e.* ageing degradation that occurs earlier than expected). These factors or weaknesses need to be identified and addressed if the proactive approach to ageing management for safe long-term operation is to be successful. The most frequently encountered weaknesses of ageing management are:

- insufficient understanding or predictability of ageing;
- lack of data for ageing management;
- inadequate communication and co-ordination;
- inadequate safety culture;
- error-induced ageing;
- inappropriate use of reactive ageing management;
- insufficient capability for dealing with unforeseen ageing phenomena.

Throughout the plant operating period, it is important to implement a comprehensive ageing management programme starting early in the plant life to ensure safe long-term operation. In many countries comprehensive ageing management activities are now included from the early stages of plant operation and are reviewed regularly both through routine supervision and as part of the

Figure 1.5-1: Systematic approach to managing ageing of a structure or component



Periodic Safety Review (normally every ten years). Some countries perform a first extensive review in connection with the Periodic Safety Review associated with 30 years' operation and others as part of the license renewal process.

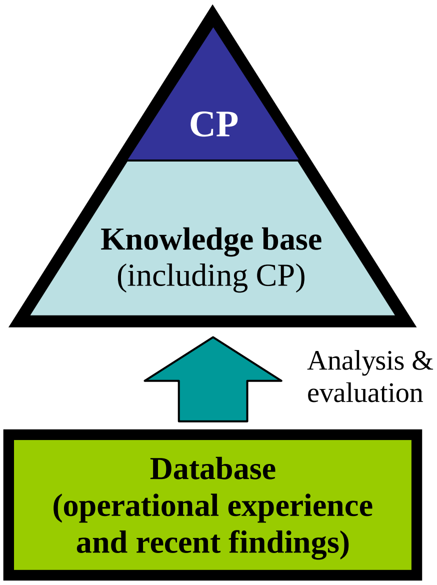
Therefore, knowledge should be extracted from the database (operational experiences and recent findings) by analysing and evaluating the data from the viewpoint of the implementation of appropriate ageing management and maintenance activities beneficial both to regulatory authorities and operators. The knowledge which can contribute to the smooth "PLAN – DO – CHECK – ACT" cycle for ageing management of SSC based on understanding the ageing of structures/components, and the closed loop of Figure 1.5-1.

Under the "PLAN – DO – CHECK – ACT" cycle, knowledge has been extracted from the database of each member country (*i.e.* analysis and evaluation of database) and transferred to the knowledge base through discussion with experts. Ageing management and maintenance activities are to a large extent based on international events and technical knowledge. As the regulator, it is very important to review the adequacy of ageing management conducted by utilities/operators, based on a reliable technical information basis (TIB). The TIB should be updated regularly to ensure timely, adequate ageing management and maintenance activities to improve ageing management.

The commendable practices are extracted from the knowledge base collected from project members, for the appropriate management of ageing phenomena (*e.g.* those common to member countries or key to the maintenance of individual ageing phenomena) and the improvement of ageing management programmes as described in the IAEA Safety Guide. It should be noted that commendable practices are included in the knowledge base. The definition and process for determining commendable practices is illustrated in Figure 1.5-2.

The database, knowledge base and commendable practices form a living technical information base and are updated continuously to reflect the latest findings.

Figure 1.5-2: Definition and determining process of commendable practices



1.6 Definitions

In this report, the definitions of ageing management and maintenance activities from the IAEA Safety Glossary [15] are used as follows:

Ageing management

Engineering, operations and maintenance actions to control within acceptable limits the ageing mechanism of structures, systems and components.

- Examples of engineering actions include design, qualification and failure analysis. Examples of operations actions include surveillance, carrying out operating procedures within specified limits and performing environmental measurements.
- Life management (or lifetime management) is the integration of ageing management with economic planning: *i)* to optimise the operation, maintenance and service life of structures, systems and components; *ii)* to maintain an acceptable level of performance and safety; *iii)* to maximise the return on investment over the service life of the facility.

Maintenance

The organised activity, both administrative and technical, of keeping structures, systems and components in good operating condition, including both preventive and corrective (or repair) aspects.

Corrective maintenance

Actions that restore, by repair, overhaul or replacement, the capability of a failed structure, system or component to function within acceptance criteria.

- Contrasted with preventive maintenance.

Periodic maintenance

Form of preventive maintenance consisting of servicing, parts replacement, surveillance or testing at pre-determined intervals of calendar time, operating time or number of cycles. Also termed “time-based maintenance”.

Planned maintenance

Form of preventive maintenance consisting of refurbishment or replacement that is scheduled and performed prior to unacceptable degradation of a structure, system or component.

Predictive maintenance

Form of preventive maintenance performed continuously or at intervals governed by observed condition to monitor, diagnose or trend a structure, system or component’s condition indicators; results indicate present and future functional ability or the nature of and schedule for planned maintenance. Also termed “condition-based maintenance”.

Preventive maintenance

Actions that detect, preclude or mitigate degradation of a functional structure, system or component to sustain or extend its useful life by controlling degradation and failures to an acceptable level.

- Preventive maintenance may be periodic maintenance, planned maintenance or predictive maintenance.
- Contrasted with corrective maintenance.

Chapter 2: SCC Working Group

2.1 Introduction

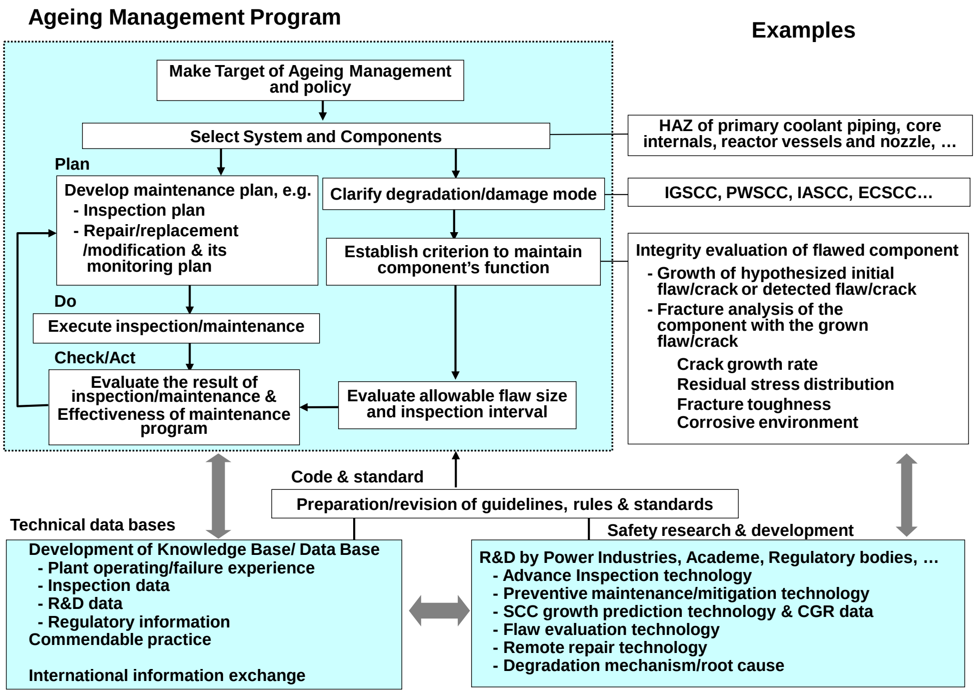
Stress corrosion cracking (SCC) is an ageing mechanism for which many events have occurred in different structures, systems and components (SSC) of nuclear power plants since the 1970s, continuing through to the present time. The causes of these events have been identified as sensitisation of material, local high residual stress, surface finishing and hardness associated with certain environments such as high-temperature water. For these reasons, it is necessary to carry out maintenance activities, inspections with the appropriate interval, monitoring, preventive maintenance/mitigation repair/ replacement and safety assessments to minimise the occurrence of future events.

The counter measures for SCC should be included in the ageing management programme (AMP). To achieve the targets of an ageing management policy for safe long-term operation, it is necessary to select the systems and components, identify the specific SCC degradation mechanisms/phenomena, and determine the criteria to ensure that the intended function of the system or component is maintained. It is necessary to assume an operating period and include the evaluation result in a preventive maintenance plan (or AMP covering inspection, repair, replacement, monitoring or surveillance).

To achieve this, a well-grounded Technical Information Basis (TIB) is required. To maintain the TIB, it is necessary to take into account the feedback from the results of research and development, codes and standards, trends of the specific degradation mechanism, and consolidation of appropriate maintenance programme.

The establishment of an AMP for SCC is illustrated schematically in Figure 2.1-1.

Figure 2.1-1: Schematic ageing management programme for stress corrosion cracking



The relationship between the nine generic attributes of the IAEA Safety Guide [12] and the activities of the SCC Working Group are shown in Table 2.1-1.

Table 2.1-1: Comparison of the generic attributes of an effective ageing management programme in IAEA Safety Guide No. NS-G-2.12 and the items of SCAP SCC knowledge base

IAEA Safety Guide No. NS-G-2.12		Items of SCAP SCC knowledge base
Attribute	**Description**	
1. Scope of the ageing management programme based on understanding ageing	• Structures (including structural elements) and components subject to ageing management. • Understanding of ageing phenomena (significant ageing mechanisms, susceptible sites): structure/component materials, service conditions, stressors, degradation sites, ageing mechanisms and effects.	Introduction of this report
2. Preventive actions to minimise and control ageing mechanism	• Identification of preventive actions. • Identification of parameters to be monitored or inspected. • Service conditions (*i.e.* environmental conditions and operating conditions) to be maintained and operating practices aimed at slowing down potential degradation of the structure or component.	(3) Preventive maintenance
3. Detection of ageing effects	• Effective technology (inspection, testing and monitoring methods) for detecting ageing effects before failure of the structure or component.	(2) Inspection/ monitoring/ qualification
4. Monitoring and trending of ageing effects	• Condition indicators and parameters monitored. • Data to be collected to facilitate assessment of structure or component ageing. • Assessment methods (including data analysis and trending).	(2) Inspection/ monitoring/ qualification
5. Mitigating ageing effects	• Operations, maintenance, repair and replacement actions to mitigate detected ageing effects and/or degradation of the structure or component.	(3) Preventive maintenance/ mitigation
6. Acceptance criteria	• Acceptance criteria against which the need for corrective action is evaluated.	(5) Safety assessment (including flaw evaluation)
7. Corrective actions	• Corrective actions if a component fails to meet the acceptance criteria.	(4) Repair/ replacement
8. Operating experience feedback and feedback of research and development results	• Mechanism that ensures timely feedback of operating experience and research and development results (if applicable), and provides objective evidence that they are taken into account in the ageing management programme.	(0) Event data (6) Research and development
9. Quality management	• Administrative controls that document the implementation of the ageing management programme and actions taken. • Indicators to facilitate evaluation and improvement of the ageing management programme.	Out of the scope of SCAP

Items of (1) Regulation/code and standards and (6) Research and development of SCAP SCC knowledge base are related to all attributes of IAEA.

The general aim of the knowledge base and commendable practices is to reflect basic international technical information (event data, R&D, etc.) for AMP and to provide a state-of-the-art description of the SCC phenomena treated and the main factors influencing their occurrence, the locations affected, as well as the strategies available for mitigation and repair. This information is useful for both regulators and utilities to establish and review AMP.

The working group was to identify commendable practices for ageing management of the different SCC degradation mechanisms for safe long-term operation. In order to achieve this, the working group agreed to establish an event database using similar principles to those already established within the OPDE project. The event database also covers experience such as field data from light water reactor events and field experience from inspections. In addition it was recognised that a knowledge base must also be established in order to identify commendable practices. The two together have been evaluated and provide the basis for the SCC portions of this report.

The scope of the event database, knowledge base and commendable practices covers Class 1 and 2 pressure boundary components, reactor pressure vessel internals and other components with significant operational impact, excluding steam generator tubing.

The ageing mechanisms included in the event database and materials affected (base metal, weld metal and cladding) are:

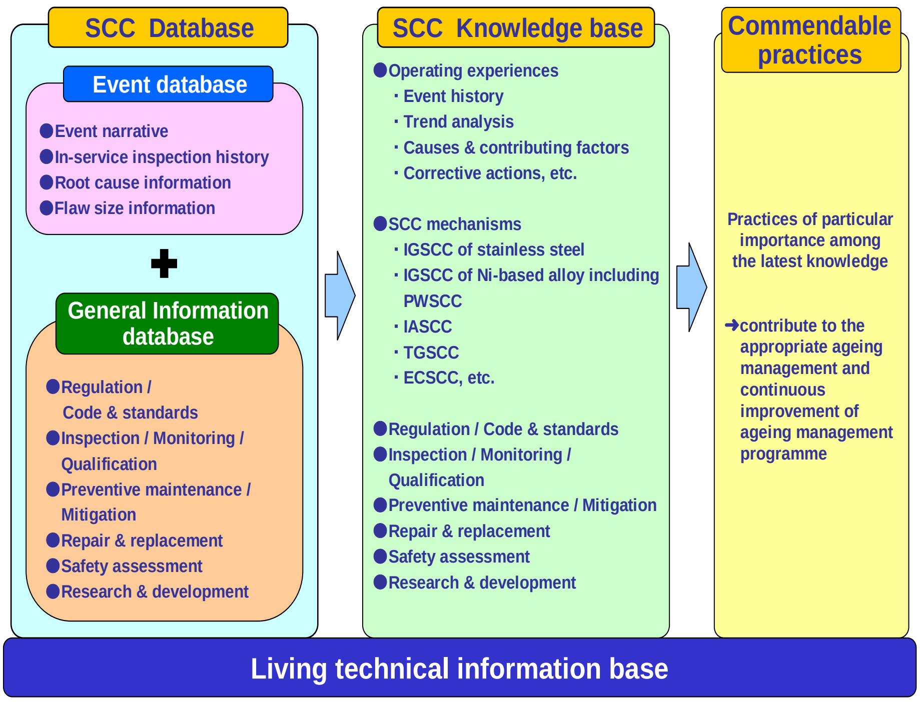
- intergranular SCC (IGSCC) of stainless steel;
- IGSCC of Ni-based alloy including primary water SCC (PWSCC);
- irradiation-assisted SCC (IASCC);
- trans-granular SCC (TGSCC);
- external chloride SCC (ECSCC);
- strain-induced corrosion cracking (SICC);
- corrosion fatigue/environmental fatigue.

The entire SCC database consists of an event database and general information. The general information consists of regulations/codes and standards, inspection/monitoring/qualification, preventive maintenance/mitigation, repair/replacement, safety assessment and R&D. Together these comprise the knowledge base.

The knowledge base is the identified knowledge, as an outcome of analysis and evaluation of the entire SCC database (event, operating experience and recent findings) for appropriate ageing management and maintenance activities. The knowledge base has also been identified from the key data and information contained in the database to establish an AMP for a specific ageing mechanism from the database of member countries, as well as discussion among the experts. The knowledge base can be used for establishing an AMP and the technical information basis (TIB).

The database, knowledge base and commendable practices are the living technical information base and are updated continuously to reflect the latest findings as illustrated in Figure 2.1-2.

Figure 2.1-2: Relationship between the SCC event database, general information, knowledge base and commendable practices



The SCC Working Group has been supported by the project consultant, a clearing house and two expert consultants, one for BWR and one for PWR aspects. The working group has held ten formal meetings as well as a number of informal meetings with the Japanese SCC Working Group, the consultants and the working group chairperson. Thirteen countries (Canada, the Czech Republic, Finland, France, Germany, Japan, Mexico, the Republic of Korea, the Slovak Republic, Spain, Sweden, Switzerland and the United States) have actively participated in the activities of the group, contributing to the event database and the knowledge base. The clearing house has together with the NEA IT-group established the event database which is a web-based relational database using Microsoft® Access. The project consultant has together with the NEA IT-group established the knowledge base which is available on the project website to all members. The two SCC experts have written comprehensive reports of SCC in BWR and PWR based both on their own knowledge and experience correlated with an evaluation of the information in the event database.

Most members of the working group are also members of the OECD/NEA Pipe Failure Data Exchange (OPDE) project. This database contains a considerable number of piping events caused by stress corrosion cracking. It was agreed at a very early stage that the SCAP-SCC event database would be based on the OPDE database following a review of the fields and identification of additional fields. Events could then be extracted from OPDE and comprise a major part of the SCAP-SCC event database. More information on the contents of the event database is given in Section 2.2.

Based on previous experience from populating the OPDE database a decision was taken to concentrate on so-called “representative events”. A representative event is typical for several identical or very similar events and is intended to contain all the relevant information and references for such events. This simplifies the data input since similar events can be entered in a simplified format with only the basic information and a note that they are coupled to the reference event.

National reporting requirements and practices vary considerably in the member countries. This means that for some of the member countries confidential information has been entered to make the entries comprehensive and to improve utilisation and interpretation of the database contents. This has led to the need to restrict access to the database to those countries that have contributed to the database. This is also consistent with the terms of the OPDE database, which is only available to members supplying data to the project. The event database can only be accessed with individual user names and passwords.

2.2 SCC event database

2.2.1 Scope

The SCAP SCC event database addresses passive components degradation or failure attributed to SCC occurring at nuclear power plants in participating countries. The opening screen is shown in Figure 2.2.1-1. The scope of the event database includes Class 1 and 2 pressure boundary components[1], reactor pressure vessel internals and other components with significant operational impact, excluding steam generator tubing.

The following mechanisms are considered in the event database: intergranular SCC in austenitic stainless steel and nickel-based material, irradiated-assisted SCC, primary water SCC, external chloride SCC and trans-granular SCC.

2.2.2 Structure

The SCAP SCC event database is a web-based relational database (Figure 2.2.2-1). The data entry is managed via input forms, tables, roll-down menus and database relationships. Database searches and applications are performed through user-defined queries that utilise the tables and built-in data relationships. The data entry forms are organised to capture essential passive component failure information together with supporting information. The four data entry forms are described below, and an example is shown in Figure 2.2.2-2.

1. Class 1 and 2 pressure boundary components are defined by the American Society of Mechanical Engineers (ASME) as follows: Class 1 includes all reactor coolant pressure boundary (RCPB) components; Class 2 generally includes systems or portions of systems important to safety that is designed for post-accident containment and removal of heat and fission products.

Figure 2.2.1-1: SCAP SCC event database opening screen

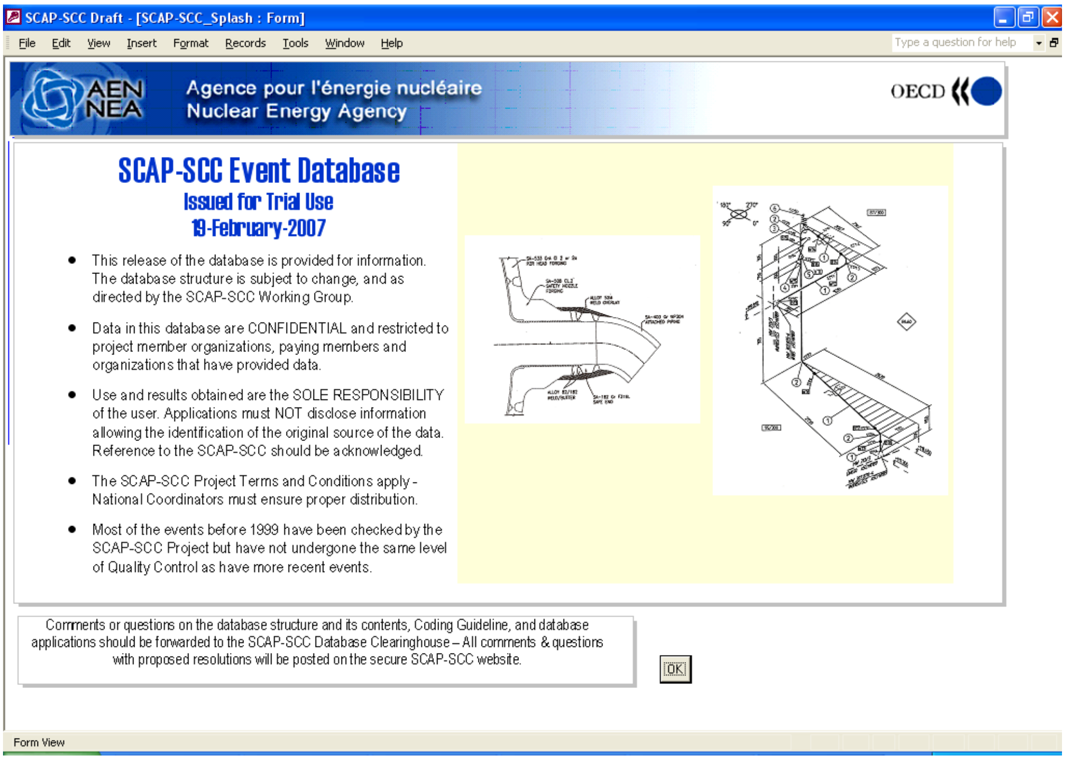


Figure 2.2.2-1: SCAP SCC event database relationships

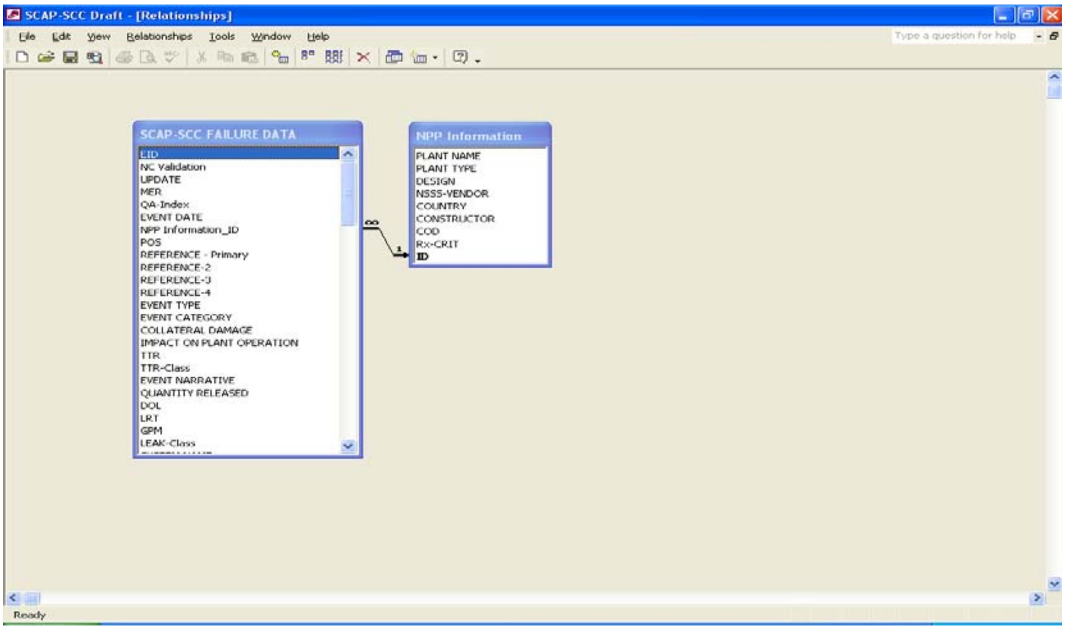
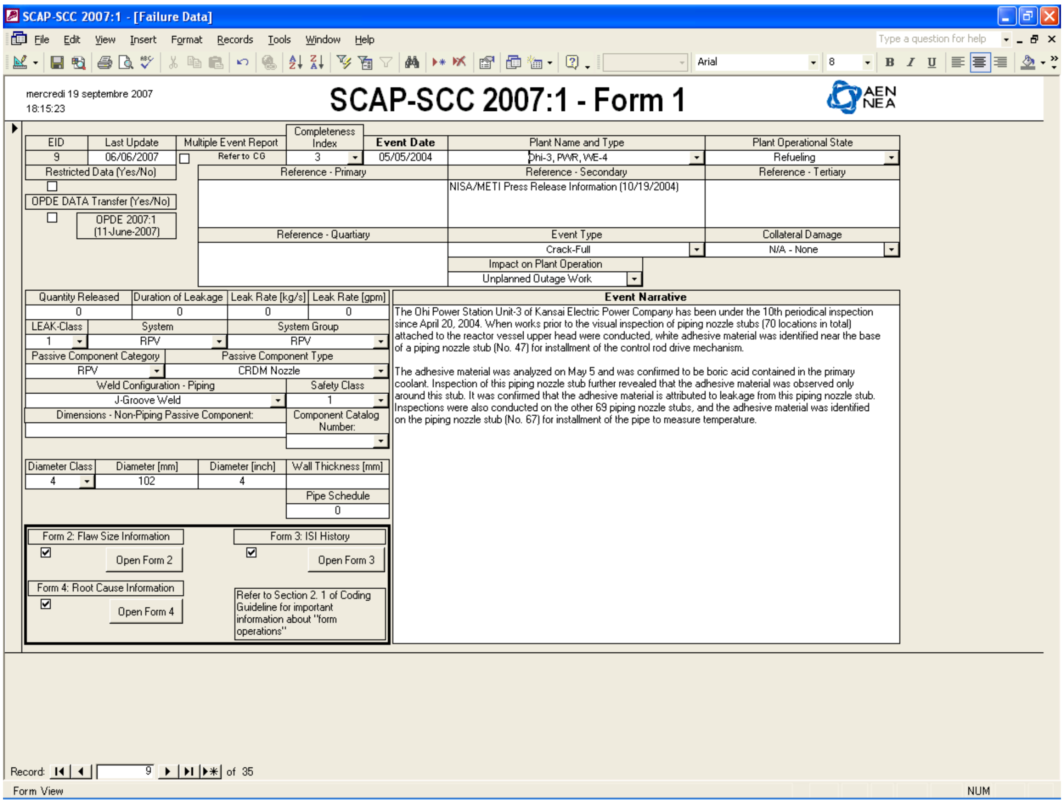


Figure 2.2.2-2: SCAP SCC event database – example of input format



Failure data input

This form defines the minimum data requirements. All data entry starts from here. It contains 35 fields, including the plant's name and operational state at the time of discovery of the event.

This allows differentiation between events with an operational impact, *e.g.* forced shutdown, and those events discovered through scheduled or augmented inspections. It also contains information regarding the event type, with a roll-down menu offering options such as a through-wall crack without active leakage, a partial through-wall crack and different types of leaks.

Information regarding collateral damage related to operational events involving active through-wall leakage is included. A menu defines the different corrective actions taken at the plant. A detailed description of plant conditions prior to the event and plant response during the event, the method of detection, and the corrective action plan are included in the event narrative field.

All the relevant information that characterises the degraded component is also included such as code class, dimensions, base metal and weld metal material designation, mechanical properties, for example yield strength and hardness, and the type of process medium at the time of detection.

Flaw characterisation

This form contains 11 fields with information that characterises the flaw (description, information about size and further details according to the type of flaw).

ISI history

This form consists of four fields. While primarily intended for recording in-service inspection (ISI) programme weaknesses, the free-format field may be used to document any information pertaining to the ISI of the affected component, or ISI history such as the time of most recent inspection.

Root-cause information

This form consists of 52 fields and includes information regarding the estimated age of the component, *i.e.* the in-service life at the time of failure. If the affected component has a known repair or replacement history this is to be taken into consideration.

A free-format field is provided to describe the location of failure, *i.e.* line or weld number or using a piping and instrumentation (P&I) reference. Roll-down menus present different options for choosing the method of detection, the apparent cause and contributing factors. Finally, a free-format field is included to provide information relevant to the root-cause analysis and cause-consequence relationship.

2.2.3 Search capability

The SCAP SCC event database is a Structured Query Language (SQL) database designed for managing relational information on stress corrosion cracking events. All fields in the database are searchable. In its present form, the database structure supports two types of searches. First, the web-based database allows searches for specific types of event reports based on plant type, plant name, event date and event characteristics (*e.g.* flaw characteristics, SCC mechanism, component type, material). Second, the SCAP SCC event database can be exported to a database platform such as Microsoft® Access. Within the Access database platform queries may be defined to support amongst other things statistical evaluations of the database content.

2.2.4 Limitations of the event database

It is a complicated task to establish a forum for the exchange of international event data. Complications arise because of the different regulatory regimes and the proprietary nature of much of the information relating to engineering activities that involve structural evaluation of flawed parts, root-cause evaluation and ISI technology. Another basic problem is that reporting levels vary from country to country and even over time in a given country. Experience has shown that this has resulted in considerable difficulties to provide detailed or complete information about very early events.

There is now an awareness of the value of compiling such information and thus for more recent events extensive information can often be collected. These difficulties are of course reflected in the quality of the data and can reflect the usefulness of the event database for some applications in which specific information is required but not available to the extent necessary.

2.2.5 Contents of the database

Event data

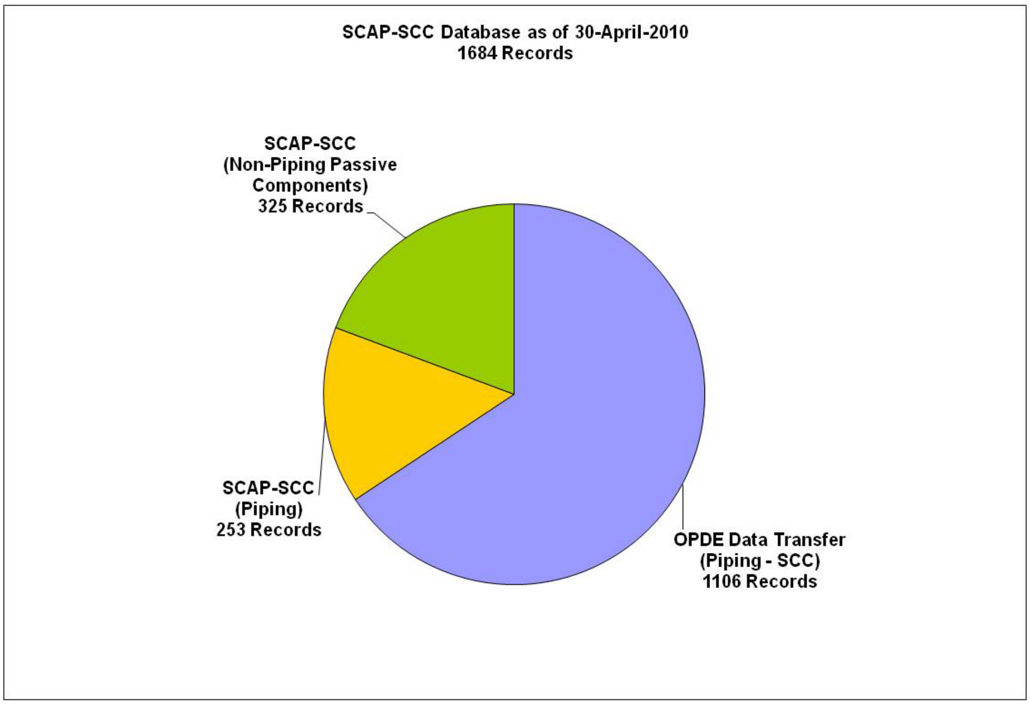
The general aim of SCC databases is to provide qualified data to form the basis for recommendations concerning commendable practices that will help to establish optimised and improved ageing management for light water reactors systems. Therefore priority should be directed to data on SSC in which degradation, defects or failure could affect safe long-term operation or could lead to a considerable burden regarding maintenance strategy, outage time or personnel exposure.

In order to develop the database and knowledge base for stress corrosion cracking, the event data were collected and populated as follows:

- SCC data from OECD/NEA Piping Failure Data Exchange Project (OPDE) were extracted and merged.
- Representative events were identified considering the importance of SSC such as reactor vessel and vessel internals, Class 1 including the reactor coolant pressure boundary (RCPB) components; Class 2 including systems or portions of systems important for safety designed for post-accident containment and removal of heat and fission products.
- Entries for the events were completed, in particular representative events.

The SCAP event database contains more than 1 600 data records (see Figure 2.2.5-1). The event database contains a number of representative events for which more extensive information has been made available. There are approximately 20 representative BWR records of a total of 1 156 and

Figure 2.2.5-1: Distribution of the data populated in the SCC database



15 representative PWR records of a total of 507. It should be noted that the contents of the OPDE and SCAP SCC event databases are not the same since the fields differ to some extent. The contents for BWR RPV internals in the SCC event database are shown in Table 2.2.5-1. The contents of the SCC event database on PWR non-piping passive components are shown in Table 2.2.5-2. The differences in the numbers in the text, tables and figures are due to the use of representative events. As described above, these have been chosen as events for which a relatively complete set of information is available, rather than for the most part the events for which much of the SCC specific information has not yet been entered, or is not available.

A representative event is typical for several identical or very similar events and is intended to contain all the relevant information and references for such events with information in the majority of the fields. This allows similar events to be entered in a simplified format with only the basic information and a note that they are coupled to the reference event.

Table 2.2.5-1: Contents for BWR RPV internals in SCC event database

Passive Component Category (P-C-C)	Passive Component Type (P-C-T)	No. of records
BWR RPV internals	Bolting	1
	Control rod	3
	Core shroud support	2
	Core shroud tie rod (X-750)	1
	Core shroud weld	51
	In-core monitor housing	4
	Core shroud head Hold-down bolt	1
	Core shroud access Hole cover	1
	Jet pump hold-down beam (X-750)	4
	RPV internals – pipe weld	7
Total		75

Table 2.2.5-2: Contents of SCC event database on PWR non-piping passive components

Passive Component Category (P-C-C)	Passive Component Type (P-C-T)	No. of records
PWR non-piping passive component	Bolting	7
	Control rod guide tube support pin	7
	CRDM housing/tube	13
	In-core instrumentation tube	3
	Pressuriser heater sleeve	63
	RPV bottom head – BMI nozzle	2
	RPV head penetration – VHP	126
Total		221

General information

General information includes both data and information on the following items for ageing management:

- regulation/codes and standards;
- inspection/monitoring/qualification;
- preventive maintenance/mitigation;
- repair/replacement;
- safety assessment;
- R&D (initiation/crack growth/fracture).

In addition, documents and reports published by regulators and technical support organisations (TSO) and by operators in member countries have been collected and compiled. Two consultant reports on IGSCC and PWSCC prepared by the expert consultants are included. One report covers stress corrosion cracking of stainless steels in BWR. The report includes a historical review of SCC, empirical correlations based on plant and laboratory experience, a quantitative understanding of the mechanism of cracking, parametric dependencies of SCC under irradiated and un-irradiated conditions, and a rationale for various mitigation actions. The other consultant report covers primary water stress corrosion cracking of nickel-based alloys. The report includes a brief history of PWSCC in PWR, a review of laboratory investigations on PWSCC of Alloys 600, 132, 182 and 82, an analysis of general information on PWSCC and mitigation techniques for PWSCC.

Country reports on knowledge are another source of information prepared by members of the SCC Working Group. Some country reports cover the general information on each SCC mechanism and others cover information on each item of regulation/codes and standards, inspection/monitoring/qualification, preventive maintenance mitigation, repair/replacement.

2.3 SCC knowledge base

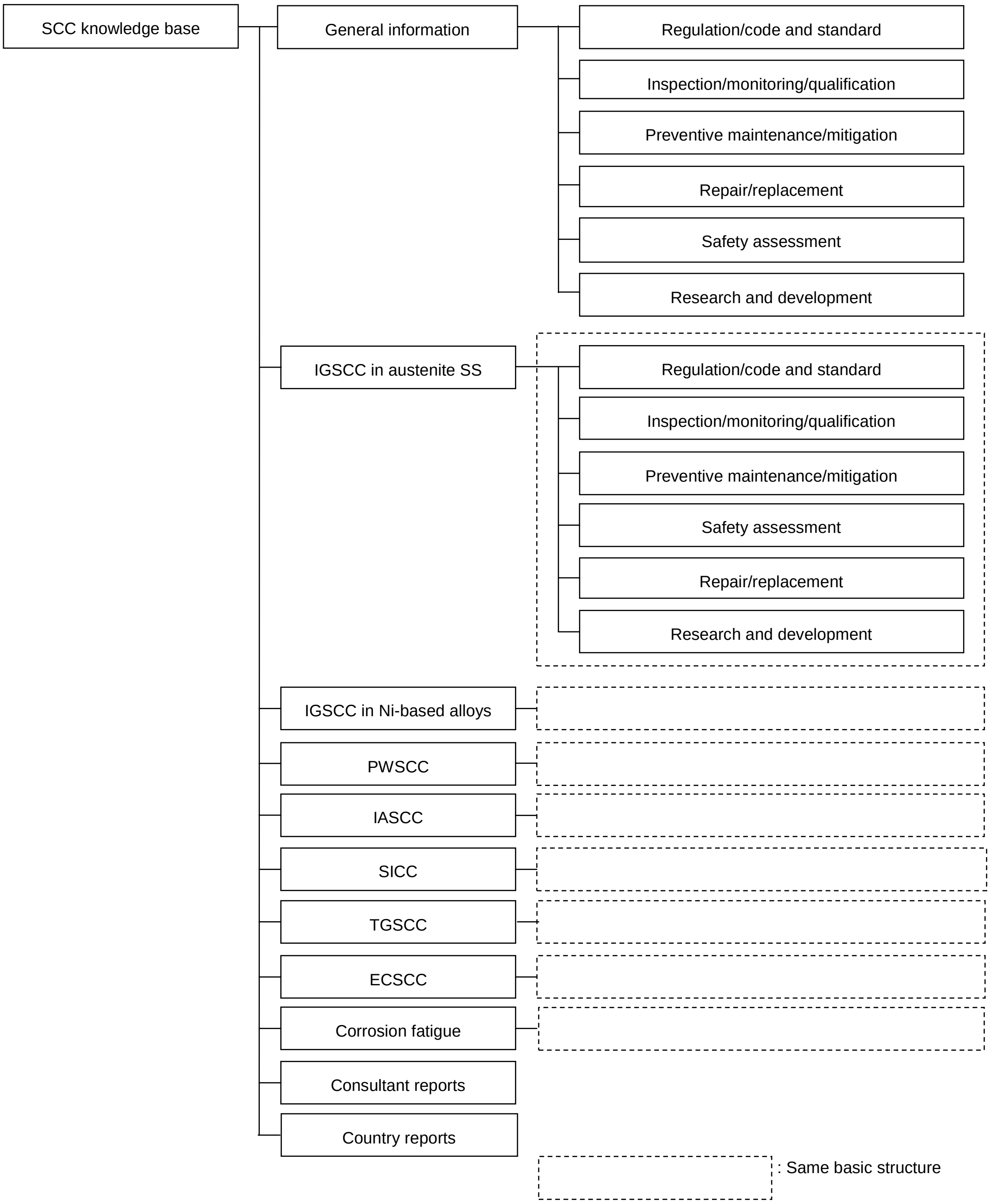
2.3.1 Introduction

The structure of the knowledge base is illustrated in Figure 2.3.1-1. The knowledge base consists of the event database, general information data and information on SCC mechanisms.

Country reports contain data and information on SCC operating experience and general information. Some country reports cover the general information on each SCC mechanism, while others cover information on individual SCC mechanism, regulatory framework and approaches, and operators' ageing management activities. The knowledge base also contains basic information on safety assessment approaches, regulations/codes and standards, inspection, monitoring and qualification practices, preventative maintenance, mitigation, repair and replacement, as well as R&D.

The future user of the SCC knowledge base can access the information regarding their home country immediately, whilst having access to all the other information.

Figure 2.3.1-1: SCC knowledge base web interface structure



The general aim of the knowledge base is to provide a state-of-the-art description of the SCC mechanisms treated and the main influencing factors, the affected locations of SSC, as well as the strategies available for mitigation and repair. This information provides the basis to identify the commendable practices useful for establishing AMP.

The knowledge base comprises information identified from the databases (events, operational experience and recent findings), which is analysed and evaluated from the viewpoint of the implementation of appropriate ageing management and maintenance activities.

2.3.2 Search capability

An example of a knowledge base search result is shown in the Figure 2.3.2-1. The user can further filter the search result by entering more specific search criteria using the advanced search option by the document title (specify words to be found in the title), description (specify words to be found in the document's description), full text, status and document type (see Figure 2.3.2-2).

Figure 2.3.2-1: An example of a search result (search with the keyword "IGSCC")

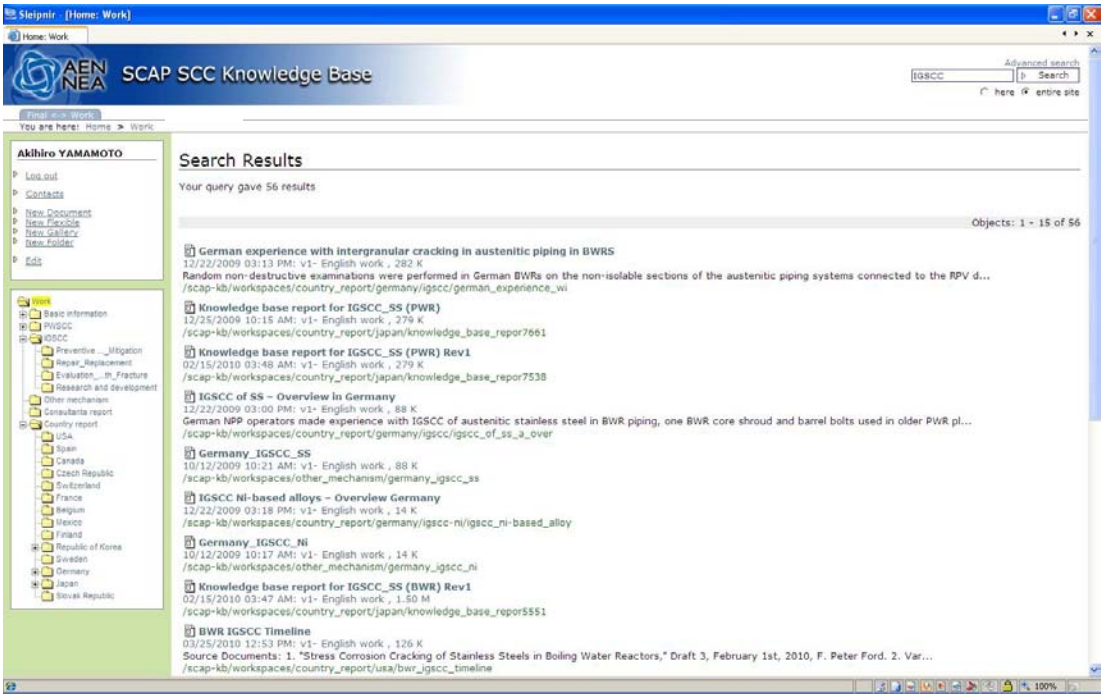
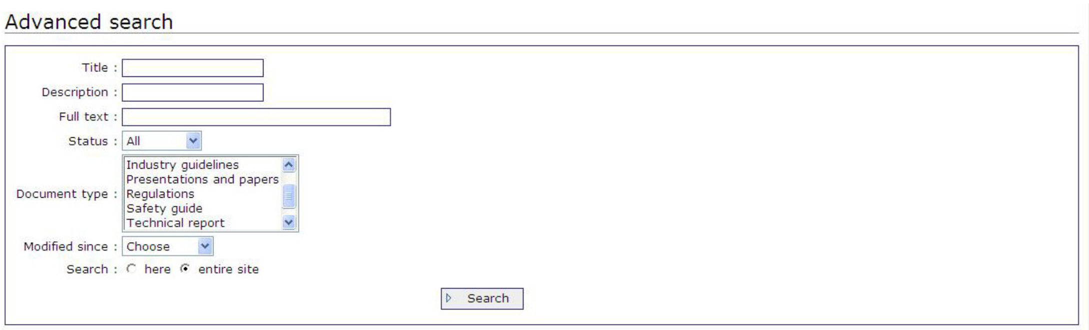


Figure 2.3.2-2: Advanced search function



2.3.3 SCC phenomena

SCC is an ageing mechanism that occurs at locations where there is a corrosive environment, where material is susceptible and where there is sufficient tensile. This section gives a short description of the different SCC phenomena that have occurred in LWR components.

Intergranular stress corrosion cracking (IGSCC) of stainless steel

Intergranular stress corrosion cracking (IGSCC) is the dominant type of SCC in LWR. It first occurred in the 1970s and continues to occur up to the present time.

The IGSCC morphology was associated with the temperature/time fabrication conditions that gave rise to thermal sensitisation and the formation of chromium carbide precipitation (*e.g.* M23C6) and chromium depletion at the grain boundary. The reduction in chromium concentration adjacent to the grain boundary gives rise to a reduction in passivity and makes the material susceptible to intergranular stress corrosion cracking.

Since the late 1970s, the importance of water purity control became increasingly apparent, especially with regard to creviced components (where the geometry and oxidising conditions in the bulk environment could give rise to increased anionic activity in the creviced region), even though the bulk water purity was acceptable at that time. This water purity aspect was of importance for environmentally-assisted cracking of stainless steel and low alloy pressure vessel steels and nickel-based alloys.

Subsequent to the introduction of low carbon and stabilised grades of stainless steel IGSCC has occurred in these materials that were clearly not in a sensitised condition. It has been shown that their susceptibility to IGSCC is due to cold work induced during fabrication. In many cases the initial cracking was found to be initially transgranular then changing to an intergranular cracking mode. The initial transgranular cracking is often associated with a surface layer of cold work induced by grinding.

Failures have also occurred where the occurrence of IGSCC was attributed to the presence of either severe bulk cold-worked material (cold bent piping). The mechanism by which cold work renders austenitic alloys susceptible to IGSCC in BWR environments is not fully understood and is still being investigated. It is possible that there is an unfavourable interaction between deformation-induced martensite, high residual stresses and strains, and localised deformation.

SCC seldom occurs in austenitic stainless steels under the PWR primary coolant water condition, as the dissolved oxygen content of the PWR primary coolant system is normally controlled at an extremely low level.

IGSCC of Ni-based alloy including PWSCC

In contrast to the IGSCC problems experienced in stainless steels in BWR systems, the same materials used in PWR systems have experienced few problems.

However, nickel-based alloys, particularly Alloy 600, and weld metals 82,132 and 182, have proven to be generically susceptible to IGSCC in normal specification PWR primary water systems. This is commonly known as PWSCC. Recent operational experience shows that the fabrication-induced residual stresses have a large influence on PWSCC in Alloy 600 weld metal. Examples of components affected include pressuriser, hot leg, cold leg, steam generator drain, and reactor coolant pump nozzle-to-safe end dissimilar metal welds, penetrations welded to the reactor vessel and reactor vessel head and steam generator.

PWSCC in the weld metal grows along the grain boundaries of columnar crystal dendrite packets. Initiation in the weld metal is often thought to be the result of typical and non-typical fabrication processes leading to locally high residual stresses, or surface stresses from, for example, grinding. To date, it has been found that the susceptibility to SCC of nickel-based alloy weld metal is higher than that of the base metal.

IGSCC of Ni-based alloys in BWR is believed to be attributed to Cr depletion at grain boundaries, similar to IGSCC in thermally sensitised stainless steels.

Corrosion-resistant Ni-based alloy has been used in the RV dissimilar metal weld because of the thermal expansion rate compatibility. A limited number of events has occurred after relatively long operational times (25 years or more) in BWR plants.

Irradiation-assisted stress corrosion cracking (IASCC)

As with SCC, irradiation-assisted stress corrosion cracking (IASCC) requires stress, environment and a susceptible material. In the case of IASCC, however, a normally non-susceptible material can be rendered susceptible above a fluence threshold due to the accumulation of neutron irradiation. There are increasing concerns that it might occur at high fluences if no mitigation measures are taken.

IASCC is an ageing mechanism that affects reactor vessel internals in both BWR and PWR plants. Neutron irradiation effects are primarily thermal but, in the case of gamma heating of thick section, the higher temperatures generated can have a significant effect on void swelling. In addition, neutron capture induces transmutation reactions and hence changes in chemical composition of the material. Irradiation hardening and radiation-induced segregation (RIS), resulting in chromium depletion and silicon enrichment at grain boundaries, are considered to be the most probable factors leading to IASCC susceptibility.

Transgranular stress corrosion cracking (TGSCC)

As with other types of SCC, transgranular stress corrosion cracking (TGSCC) requires stress, an aggressive environment and a susceptible material. It is often caused by chloride contamination or other halide anions such as fluorides, and it can occur in materials in the solution heat-treated condition. TGSCC occurs in all types of reactors.

It generally initiates on the outside surfaces of components mainly due to lack of attention to adequate cleanliness (also known as external chloride stress corrosion cracking).

TGSCC has also occurred from inner surfaces, mainly in pipe sections containing stagnant two-phase coolant, where evaporation and concentration of chlorides can occur. Wetting due to condensation or nearby water leaks allows an aqueous environment to form that leads to TGSCC, usually accompanied by pitting or crevice corrosion. The stress required for chloride-induced TGSCC is relatively modest, the threshold being close to the proportional yield strength of solution annealed austenitic stainless steels. Implementation of the known adequate procedures to ensure appropriate surface cleanliness is a continuing necessity that requires careful management attention at all stages of construction and operation of nuclear power plants.

External chloride stress corrosion cracking (ECSCC)

External chloride stress corrosion cracking (ECSCC) is normally a transgranular type of stress corrosion cracking initiated on the outside surface of components due to contamination from saline environments, high polymer products, etc., and even by human sweat.

Strain-induced corrosion cracking (SICC)

Strain-induced corrosion cracking (SICC) is used to refer to those corrosion situations in which the presence of localised dynamic straining is essential for crack formation to occur, but in which cyclic loading is either absent or restricted to a very low number of infrequent events. SICC have been observed in particular in pressurised components in German NPP made of higher-strength ferritic carbon steel.

Corrosion fatigue/environmentally-assisted fatigue

Corrosion fatigue or environmentally-assisted fatigue is the behaviour of materials under cyclic loading conditions and is commonly considered as consisting of two broad categories of material properties. One category relates to the cycling life for the formation of a fatigue crack in a smooth test specimen, the so-called S-N fatigue properties. The second relates to the growth of a pre-existing crack. Laboratory tests have shown that LWR coolant water can have a detrimental effect on both S-N fatigue properties and fatigue crack growth.

2.3.4 Analysis of the event database

The contents of the event database are summarised in Table 2.3.4-1 for the different stress corrosion cracking mechanisms in the different reactor types. The event database contains a number of representative events for which more extensive information has been made available. There are approximately 20 representative BWR records of a total of 1 156 and 15 representative PWR records of a total of 507.

The SCC event database contains in principle data connected to reportable events supplemented by additional technical information. To assist in the understanding of the flow of information the flow charts for reportable events have been contributed to the knowledge base by the participating countries.

Three factors are critical in SCC and it can be avoided by eliminating at least one of these three factors since there are some uncertainties in the specific mechanism; elimination of only one of the

Table 2.3.4-1: Matrix of SCC mechanisms and major components

SCC mechanism	Plant type			
	BWR	PWR	PHWR (CANDU)	VVER (PWR)
IGSCC (SS, BWR)	1 027	–	–	–
IGSCC (SS, PWR)	–	85	–	2
IGSCC (Ni-based alloy, incl. PWR)	17	323	–	–
ECSCC	27	63		–
TGSCC (SS, ferritic steel)	50	33	15	3
SICC	26	1	1	
IASCC	8*	2		1
Corrosion fatigue	–			
Total	1 155	507	16	6

* All event records are for control rods.

three factors may not be sufficient. Therefore it is recommended if possible to eliminate at least two SCC factors in order to improve the effectiveness of the mitigation method. This can also be achieved by applying a combination of mitigation methods.

The commendable practices which can be identified from an analysis of the event database and the general information comprising the knowledge base are discussed in Chapter 4 and not in Chapter 2, so as to avoid repetition.

2.3.4.1 IGSCC of stainless steel

BWR plants have experienced a large number of IGSCC events in stainless steels. Figure 2.3.4.1-1 shows the historical summary of IGSCC in stainless steel components in BWR. Various kinds of mitigation measures have been developed and applied. Figure 2.3.4.1-2 shows the number of reported cracking incidents in the SCC database for BWR, PWR and CANDU as a function of the calendar year.

Figure 2.3.4.1-1: Historical summary of IGSCC for stainless steel components in BWR

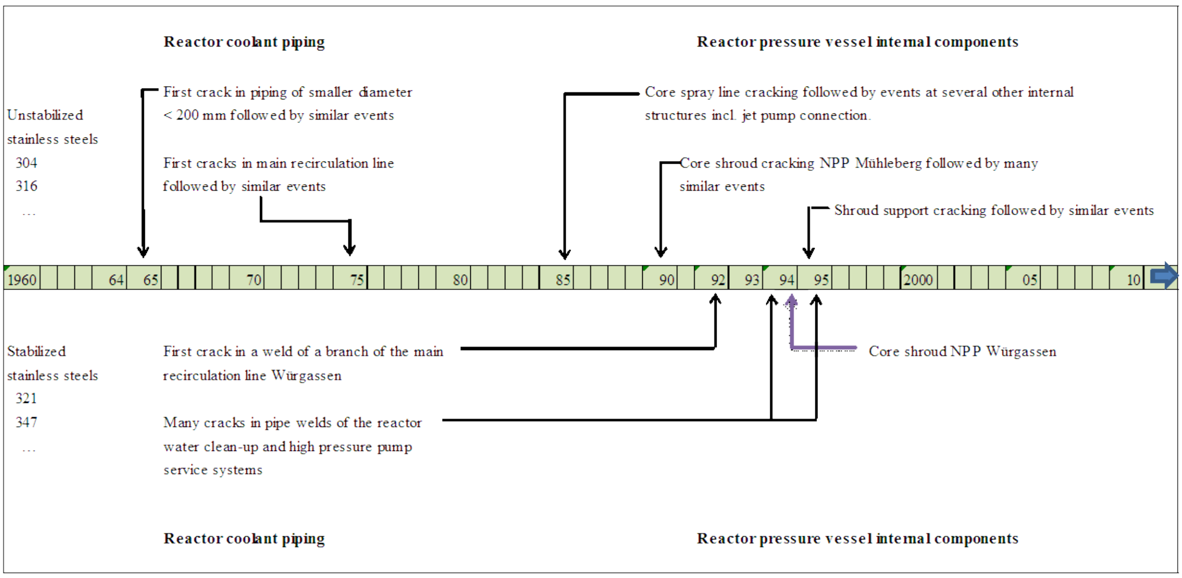
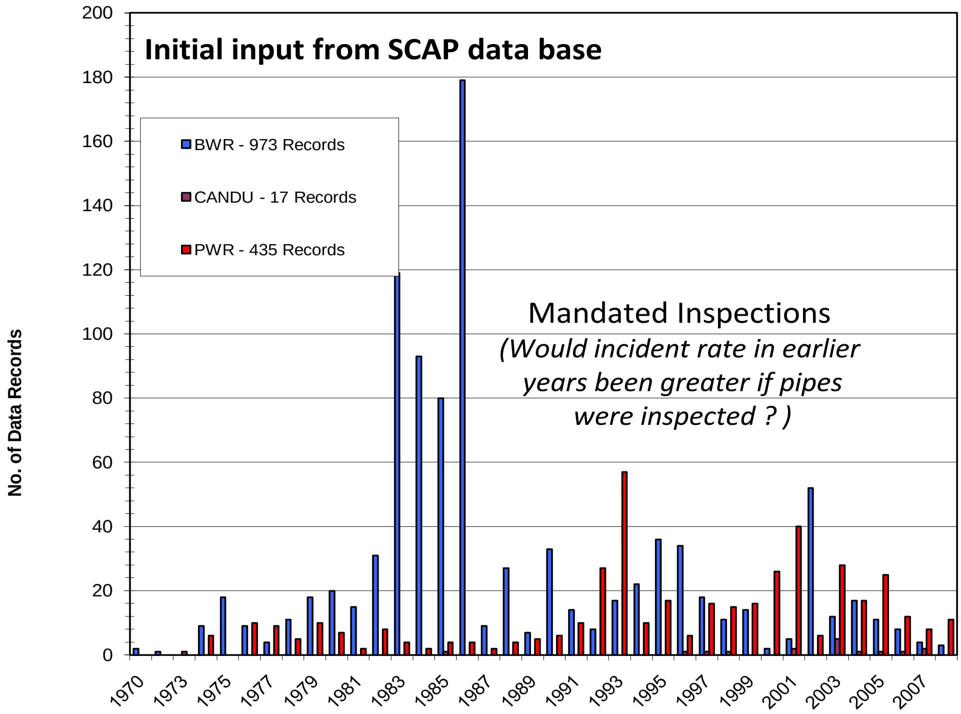


Figure 2.3.4.1-2: Number of reported cracking incidents in SCC database for BWR, PWR and CANDU as a function of calendar year



IGSCC is a cracking phenomenon occurring initially in 304 SS and other materials, which develops along the grain boundaries. It occurs under synergistic effects of three factors in austenitic stainless steel such as 304 SS with carbon content exceeding 0.03 wt.%. The three factors are: the material sensitisation by heat input during welding, surface hardening heat treatment such as nitrising, etc.; the residual stress caused by welding; and the water chemistry environment such as high dissolved oxygen content, etc. However, in recent years, IGSCC has also been observed in non-sensitised austenitic stainless steel, such as 316NG SS, 316L SS, etc., in nuclear components having improved IGSCC resistance compared with 304 SS, owing to material surface hardening by machining and cold work or surface residual stress resulting from cutting, grinding, etc.

Low-carbon stainless steel was developed and adopted considering the analysis results of IGSCC initiation in 304 SS due to sensitisation and the corrective measures taken. In those days, it was considered that the improvement of material only is sufficient to prevent SCC. However, many IGSCC incidents occurred in low-carbon stainless steel. Accordingly, a combination of several technologies was adopted instead of a single corrective measure to address potential SCC in a more effective manner. For example, regarding SCC in low-carbon stainless steel, it was determined to reduce residual stresses as well as to remove the surface hardened layer. By taking the measures to address two of three factors causing SCC, the SCC initiation was successfully controlled.

Regarding inspection, in sizing SCC which has developed into stainless steel welds, the requirements according to the Potential Drop (PD) system were imposed and thus a certain level of accuracy can be maintained. In addition, utilities' operating experience has shown that SCC may arrest or the SCC propagation rate may significantly decline in the middle of the depth depending on the residual stress and the configuration of the flaw.

It is desirable that SCC incidents at operating nuclear power plants and knowledge about SCC in other countries be widely shared among the regulatory authorities, academia and industry so that such experience and knowledge can be immediately incorporated into corrective actions, and that the regulatory authorities, academia and industry will work closely to verify the validity of an evaluation as soon as possible.

Regarding IGSCC, many events have been observed and much research undertaken to understand its mechanism on low carbon stainless steels. Based on the above background, preventive maintenance measures (recommended practical actions) considering the possibility of IGSCC initiation and crack growth are described below.

2.3.4.2 IGSCC of Ni-based alloy including PWSCC (PWR)

Historically, PWSCC first affected wrought Alloy 600 components and then spread to the compatible weld metals Alloys 132/182, and, to a limited extent, to Alloy 82. Cracking in thick section wrought Alloy 600 nozzles was first detected in pressuriser nozzles in the 1980s, spread to CRDM nozzles in the 1990s and then finally after 2000 to the compatible weld metals Alloys 132/182/82 (see Figure 2.3.4.2-1).

The SCC database contains over 200 records pertaining to PWSCC incidents in PWR primary circuits since the mid 1980s. PWSCC has mostly affected components fabricated from wrought Alloy 600 and its high strength analogue Alloy X750. However, increasingly the weld filler metals Alloys 132 and 182 are being affected by PWSCC together with a much small number of welds fabricated from Alloy 82. A further general comment that can be made on the basis of the representative PWSCC events in the database is that several different remedial techniques have been adopted that to some extent reflect the regulations in force in different countries.

Figure 2.3.4.2-1: Historical summary of PWSCC in components of wrought Inconel 600 based material and welds (Alloys 132, 182 and 82)

The PWSCC records in the database can be analysed to reveal the most vulnerable locations and occurrence rates in the PWR primary circuit. Tables 2.3.4.2-1 and 2.3.4.2-2 summarise the wrought Alloy 600 and Alloy 132/182/82 welds affected by PWSCC for, respectively, RPV, steam generators, pressurisers and primary circuit penetrations. The last group listed in Table 2.3.4.2-1 essentially only concerns B&W and combustion engineering design PWR. It is not immediately apparent from these tables how important a role surface finish has played in the incidence of PWSCC and this can only be deduced by reading the narrative accounts in the event database and associated publications. Consequently, mitigation methods for components that are not too severely cracked as to warrant repair or replacement concentrate on improving surface finish and residual stress state.

It can be seen immediately in Tables 2.3.4.2-1 and 2.3.4.2-2 with some reservations relating to CRDM nozzles in RPV upper heads, that there is a very strong influence of operating temperature on the incidence rate of PWSCC. Nevertheless, Alloy 600 and Alloys 182/132 have cracked at cold leg temperatures, albeit at a significantly lower incidence rate.

Table 2.3.4.2-1: Distribution of RPV Ni-based components and welds with known indications attributed to PWSCC

Component	Material affected		Temp °C	Number cracked
	Wrought	Weld		
CRDM nozzles	Alloy 600	–	288-318	44
CRDM nozzles and J-groove welds	Alloy 600	Alloy 182	289-318	191
CRDM J-groove welds		Alloy 182	307-317	8
Thermocouple nozzles and head vents	Alloy 600		311-317	19
BMI nozzles	Alloy 600		288	1
BMI nozzles and J-groove welds	Alloy 600	Alloy 182	294	2
Hot leg nozzle butt welds		Alloy 182/82	319-325	7
Cold leg nozzle butt welds		Alloy 132	290	7
Split pins	Alloy X-750		320	Multiple

Table 2.3.4.2-2: Distribution of steam generator Ni-based components with known indications attributed to PWSCC

Component	Material affected		Temp °C	Number cracked
	Wrought	Weld		
Tube sheet cladding		Alloy 82	321-323	1
Tube sheet cladding	Alloy 600	–	326	8
Channel head and manway drain in J-groove welds	–	Alloy 182/82	290-325	8
Partition (divider) plate (stub)	Alloy 600	–	321-323	5
SG inlet nozzle butt welds	–	Alloy 132	319-325	16
Tube to tube sheet welds	–	Alloy 82?	325	196

The reasons why the effect of temperature appears to be much less apparent in the case of CRDM nozzles and J-welds could be related in part to the uncertainty in the actual temperature of operation in so-called "cold" upper heads. The primary water that circulates into the upper head may, depending on the particular plant design, originate from the hot leg or the cold leg. In the former case, there is no uncertainty as to the temperature of the CRDM nozzles and J-welds on the underside of the upper head, but in the latter case, an experience has certainly indicated that there is significant uncertainty, which may be due to convection effects from the mass of hot leg water below the upper core internals support plate. Another location where there could be similar uncertainty in the actual temperature of the component is the steam generator channel head drain nozzle, which, depending on design, is often located beneath a small hole in the divider plate at the bottom of the channel head.

In any event, the apparent lack of a strong effect of temperature in the operating experience of Alloy 600 CRDM nozzle cracking was also significantly affected by other highly distributed parameters, such as intrinsic material susceptibility and residual fabrication stress.

Another observation that can be made on the data shown in Tables 2.3.4.2-1 and 2.3.4.2-2 is that, in Alloy 182/82 welds, cracking is usually believed to have initiated in the Alloy 182 part of a weld and then propagated, seemingly without difficulty, into the Alloy 82 part of the weld. However, as mentioned earlier, there are a very small number of PWSCC events that have without doubt initiated in Alloy 82. These cases appear to be characterised by design features or weld repairs that severely exacerbated residual fabrication stresses.

The evolution of PWSCC in Alloy 600 components with operating time and its dependence on stress, surface finish, material susceptibility and temperature have already been addressed. Similar predictive tools have not yet been developed for Alloys 132/182/82 although crack growth equations are available for conservatively evaluating the future growth of any defects found by non-destructive examination.

The event database can be used to examine the evolution of detectable cracking with operating time in Alloy 132/182/82 welds. However, some reliance has to be placed on judgments from the original failure analyses as to whether PWSCC initiated first in Alloy 600 or in Alloy 132/182/82 where both were present and cracked in the same affected component. This approach updates an analysis where 148 cases of PWSCC initiation in Alloys 132/182/82 welds were plotted as the cumulative number of cracking incidents as a function of effective full power hours. Interestingly, the continuous evolution of cracking revealed then and by the updated plots based on 161 cases shown in Figures 2.3.4.2-2 and 2.3.4.2-3 were uncorrected for operating temperature differences.

Figure 2.3.4.2-2: Operating times to observation of defects attributed to PWSCC in Alloy 132/182 welds with no correction for operating temperature

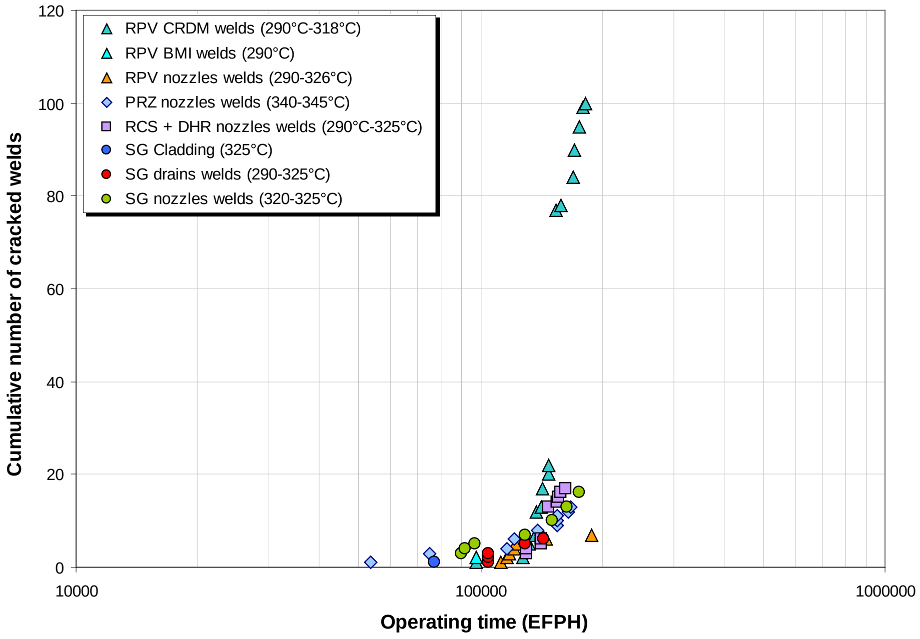
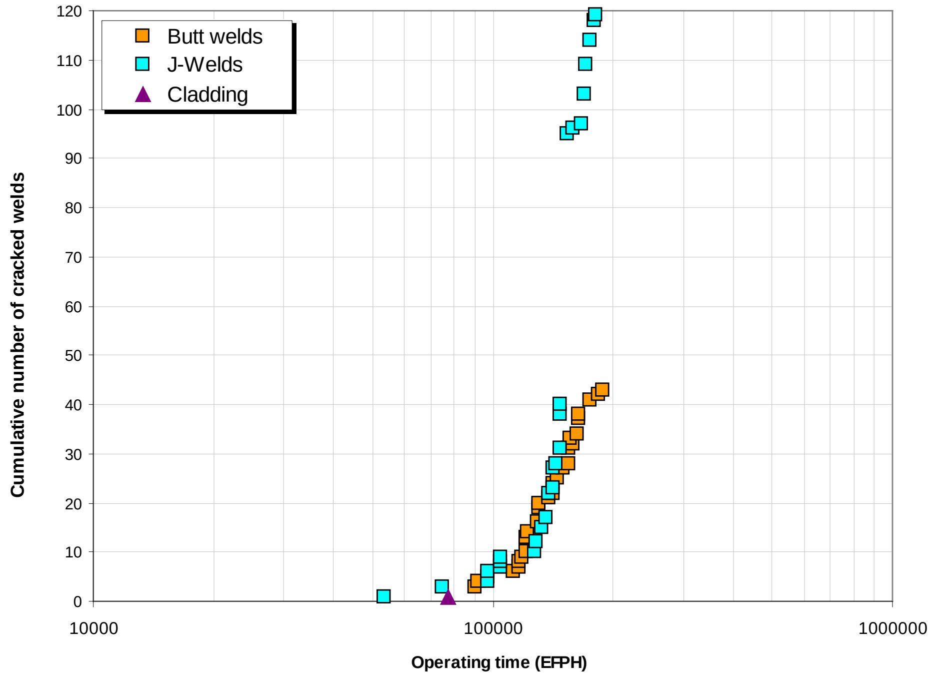


Figure 2.3.4.2-3: Operating times to observation of defects attributed to PWSCC in Alloy 132/182 welds by weld type with no correction for operating temperature



If temperature compensation is attempted, the development of cracking is then no longer a continuo function of operating time. This is probably due both to uncertainties in certain operating temperatures, as already discussed, and secondly to a dominating influence of the distribution of residual fabrication stresses that can mask the effect of temperature. Nevertheless, the conclusion from Figures 2.3.4.2-2 and 2.3.4.2-3 is that PWSCC of Alloy 132/182/82 weld metals is a slowly developing generic materials ageing problem.

Replacement of upper heads with Alloy 690 CRDM nozzles welded with Alloys 152/52 has been the almost universal response to PWSCC of the original upper heads with Alloy 600 CRDM nozzles welded with Alloy 182/132. Temporary internal overlay coatings have also been applied using Alloys 152/52 while new replacement upper heads are procured.

Small Alloy 600 nozzles attached by Alloy 182/132 J-welds in pressurisers and steam generators have generally been replaced where necessary using Alloy 690 or stainless steel nozzles and matching weld metals, a temper bead welding method being necessary for attachments to the low alloy steel. On the other hand, so-called half-nozzle repairs, again using Alloy 690 and Alloy 152, have been adopted to add a supplementary pressure boundary on the exterior surface of RPV for the small number of cases on BMI nozzle cracking that have been encountered so far.

External full structural weld overlays have been widely deployed as a preventative measure for nickel-based butt welds, particularly in the United States, where the overlay adds PWSCC-resistant material (Alloy 52/52M) as well as produces compressive residual stresses on the internal diameter. MSIP is another technique that produces compression on the inner surface of the component through a squeezing operation. To date operational experience in both BWR and PWR is that there have been no crack growths in welds containing flaws and repaired by weld overlay or MSIP. Spool piece inserts welded with Alloy 152/52 have been used to replace major RPV and steam generator nickel-based butt welds affected by PWSCC. Inlays using Alloys 152/52 are another method available to isolate original Alloy 132/182/82 butt welds from the PWR primary water environment.

Measures such as shot peening, water jet peening, and laser stress improvement to improve the surface stress condition of the original Alloy 132/182/82 welds when a flaw is detected but is not of a size that welding repair is necessary, can be used as mitigation techniques.

Finally, the widespread occurrence of PWSCC of CRDM guide tube split pins fabricated from the high-strength Alloy X750 have been replaced either with pins manufactured from the same material with an improved heat treatment plus design changes to reduce stress concentrations and improve surface finish, or with strain-hardened Type 316 stainless steel. The latter solution necessitates a design change to compensate for the lower strength of strain-hardened Type 316 stainless steel as compared to Alloy X750.

The terminology PWSCC has in some instances been used erroneously to describe cracking of a few stainless steel components where there was no direct evidence of chemical contamination of the PWR primary water environment. (Most cracking in stainless steel components in PWR has been attributed to the presence of anionic contaminants such as chloride and sulphate together with trapped air bubbles in dead legs after refuelling.) In fact, previous diagnoses of these incidents involving stainless steel, especially in pressurisers, were attributed to a combination of the presence of sensitised or cold-worked stainless steels with operational practices that involved injecting inadequately de-aerated boric acid and/or activity reduction practices involving oxygenation using hydrogen peroxide of the primary circuit during cool-down. It can be noted that around half the PWR fleets in the USA and France no longer de-aerate boric acid and demineralised water storage tanks as the original vendors intended. It is not clear that initiation of stress corrosion cracking can occur in PWR primary water, but it is clear that stress corrosion cracks can propagate in that environment. Work to further investigate this is in progress. Cleanliness and limiting surface abrasion are very important factors in addition to operational practices.

Recent operational experiences in Japan and overseas show that the residual stress from mechanical work and welding has big influence on PWSCC in Alloy 600 series weld metal.

Based on this standpoint, this report addresses commendable practices against PWSCC in 600 series nickel-based alloy weld metal of nozzles and penetrations which are welded to the reactor vessel (RV), reactor vessel head (VH), steam generator (SG) and pressuriser (Prz) of PWR.

The United States NRC defined the effective degradation year (EDY) as a function of effective full power years (EFPY) and RV head top operating temperature to estimate the timing of PWSCC occurrence. The NRC assumes that the susceptibility to PWSCC initiation increases as EDY becomes larger. In Japan, temperature-accelerated constant load tests of base metal were conducted and the test result was used to develop an equation to estimate the timing of PWSCC as a function of applied stress and RPV head top operating temperature. However, PWSCC in the weld metal grows along the grain boundaries of columnar crystal. PWSCC initiation in the weld metal is suspected to have been caused by defective welding application and high residual stress on the surface due to grinding. Therefore, the frequency of PWSCC initiation may increase depending on the operating years, service temperature and stress. It is expected that the susceptibility to SCC of nickel-based alloy weld metal is higher than that of base metal.

Considering that the possibility of PWSCC initiation in nickel-based alloy increases as a plant ages and that PWSCC susceptibility of weld metal is higher than that of base metal, it is desirable that inspection should be conducted in an assured manner and Alloy 600 components should be replaced with corrosion-resistant Alloy 690 components as necessary. At the same time, it is necessary to improve the inspection accuracy so that PWSCC can be detected at an early stage. For example, the detection ability of ultrasonic testing and eddy current examination techniques needs to be improved and technologies applicable to narrow gaps need to be developed. In addition, research on the estimation of the timing of PWSCC initiation both in base metal and weld metal should be promoted while collecting new knowledge about PWSCC world wide. At the same time, it is necessary to work on the establishment of evaluation techniques, including the improvement of detection ability and accuracy in estimating the propagation rate. In addition, repair and replacement methods, including peening and stress improvement, should be developed.

For feedback to another structure system and component, it is recommended to make analysis and evaluation based on the event database and knowledge base of OECD/NEA SCAP, to review inspection, evaluation, repair, replacement and preventive maintenance.

The SCAP SCC event database also allows the practical repair, replacement and maintenance strategies adopted in various countries to combat PWSCC in nickel-based components to be catalogued and compared. A summary is shown in Table 2.3.4.2-3. Two zones where repairs have either not been necessary to date or are not known to have been implemented at the time of writing are, respectively, core radial supports (no cracks reported) and steam generator divider plates, even if cracking has been observed in divider plates.

2.3.4.3 IASCC

Irradiation-assisted stress corrosion cracking (IASCC) is a mode of the ageing mechanism in which cracking occurs in BWR and PWR core internals when the component is irradiated to high fluence.

PWR

As shown in Figure 2.3.4.3-1 and Table 2.3.4.3-1 only a few events attributed to IASCC have been reported in the event database. In the 1980s, inspections of older French PWR revealed cracking in baffle-former bolts. The bolts were made of type 316 cold-worked (CW) stainless steels. They were damaged by intergranular cracking. Normally, AISI 316 steel is not prone to IGSCC in the PWR primary coolant environment and all the bolts cracked were predominantly located in the second and third rows from the bottom, where the highest neutron fluence is expected. This demonstrated that the neutron irradiation played a significant role in this cracking, even though the detailed mechanism is as yet unknown. Until now, baffle-former bolt cracking was mainly observed in the “down-flow” design of reactor vessel internals (RVI) either before or after the conversion to “up-flow”.

As a proactive countermeasure against baffle-former bolt damage, two different measures were applied to PWR plants. The first measure was to replace baffle-former bolts by bolts with a better mechanical design. All the baffle-former bolts made of AISI 347 stainless steel have been replaced in Japan by new ones made of 316 CW stainless steel. The second approach was to replace all the reactor vessel internals. Lower internals including baffle-former bolts and upper internals were replaced in four PWR plants in Japan.

Table 2.3.4.2-3: Summary of repair, replacement and preventative techniques for PWSCC recorded in the SCAP SCC event database

Component / Technique	Reactor pressure vessel				Core internals	Pressuriser		Steam generator			
	RPV nozzles	Upper head	Lower head nozzles	Radial supports	Split pins	Nozzles	Heater sleeves	Complete steam generators	Inlet/outlet nozzles	Channel head nozzles	Divider plate
Replacement		F J S U			F J K S U	F	F U	F G J K S U		K S U	
Spool piece replacement	U					J			J		
Cutting, drilling, grinding	J S U					U			J		
Temper bead welding	J U		J U			F J				S U	
MSIP						U					
Overlay clad (external, full structural)	U	U				U					
Overlay clad (internal)		U				U					
Inlay clad (internal)	J S								J		
Half nozzle repair			U								
Seal welding		J U									
Shot peening									J	U	
Water jet peening	J		J								
Laser stress improvement						J					

Key: Technique applied in F = France, G = Germany, J = Japan, K = Korea, S = Sweden, U = USA.

Figure 2.3.4.3-1: One of the field experiences of IASCC

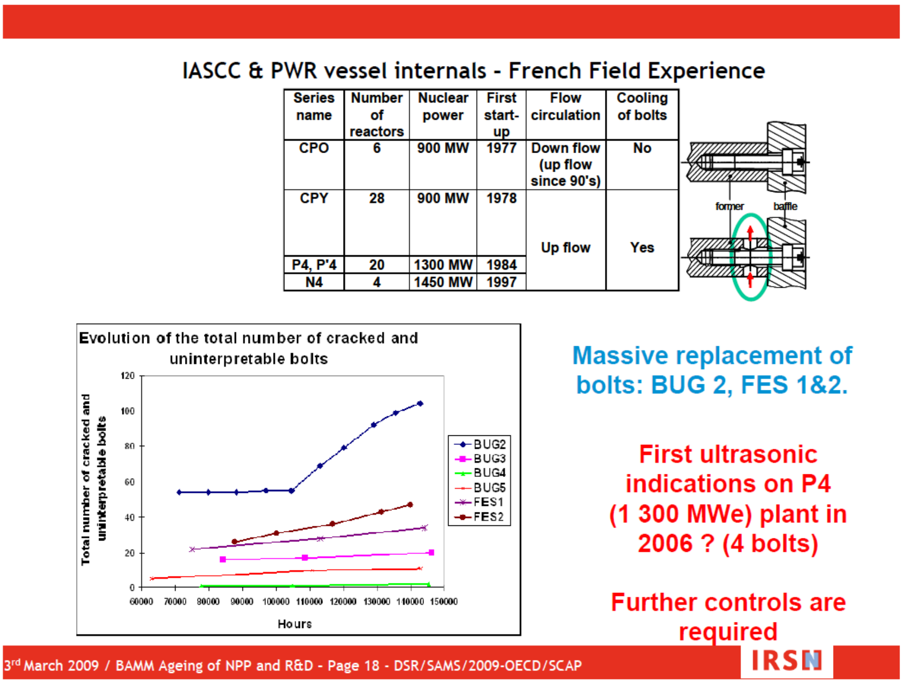


Table 2.3.4.3-1: IASCC experiences example in France, USA and Korea

Degradation History

	Facilities / Power Plant	Operating year	Generating power (MW)	Loop number	BFB number	BFB material	Defect number	Defect position
FRANCE	Bugey-2	20	955	3	960	316SS	87	#2,3 Former
	Bugey-3	20	955	3	960	316SS	18	-
	Bugey-4	-	937	3	960		3	-
	Fessenheim-1	-	950	-	960		28	-
	Fessenheim-2	-	950	-	960	347SS	47	-
USA	Farley-1	22	860	3	1024		No	-
	Point Beach-2	27	510	2	728	347SS	55	#1,3,8 Former
	Ginna	-	600	2	728		59	-
KOREA	Kori-1	21	600	2	728	316SS	2	#1,6 Former
	Kori-2	16	650	2	800	316SS	-	-
	Kori-3·4 YoungGwang-3·4	12 ~ 14	950	3	960	316SS	-	-
	Ulchin-1·2	10 ~ 11	950	3	1029	316SS	-	-

BWR

As shown in Table 2.3.4.3-1, in BWR, several events involving IASCC in control rods have been reported in the event database but no events of IASCC in core shrouds have been reported. It is well known that several plants in the US and in Japan have experienced cracking in their core shrouds, and it is to be expected that some of these will be added to the event database in the future.

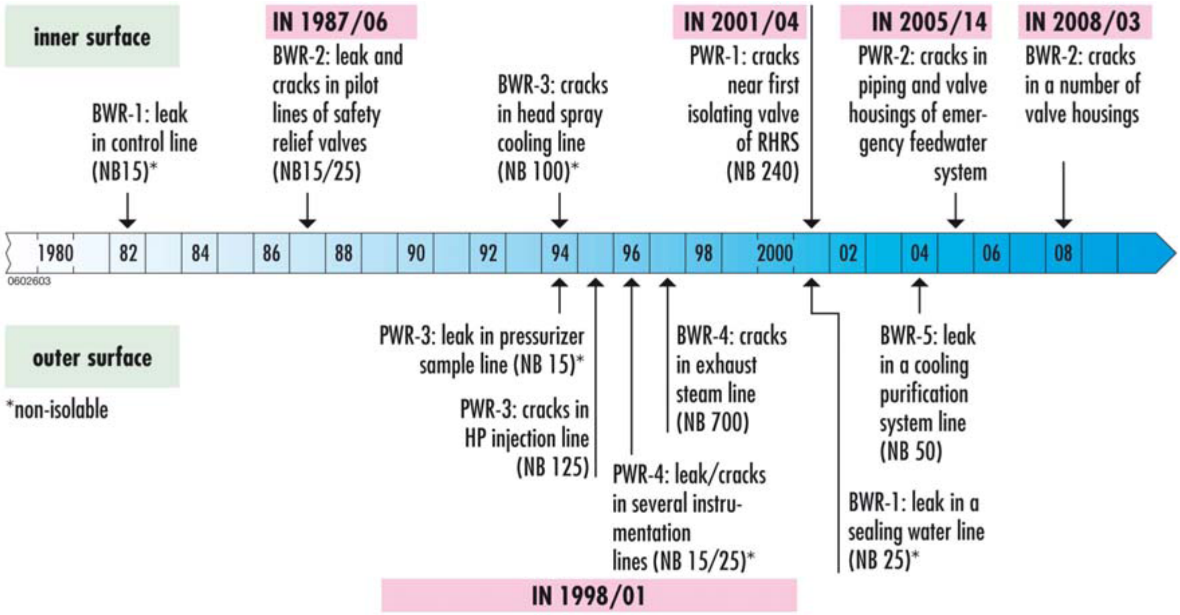
2.3.4.4 Transgranular stress corrosion cracking (TGSCC)

Chloride-induced TGSCC can also initiate on the inner surface, generally in regions of stagnant fluid, such as dead legs, due to concentration of chloride and oxygen ions. An area that has been damaged rather frequently in PWR is the CRDM seal, known as the canopy seal. It seals the pressure boundary of threaded joint of the control rod drive housings located above the reactor pressure vessel upper head. Trapped air bubbles, when the reactor pressure vessel is open during refuelling, are unable to be removed by inadequate de-oxygenation processes since the pathway to the reactor vessel is complicated. Plant start-up procedures to eliminate these air pockets may vary among operators, but completing the final fill of the primary loop after connection of a vacuum pump to the upper head penetration has been acknowledged to be the most reliable method. Small bore piping is another component that has been subjected to TGSCC (see Figure 2.3.4.4-1).

2.3.4.5 External chloride stress corrosion cracking (ECSCC)

External chloride stress corrosion cracking initiates on the outer surface of austenitic stainless steel components mainly due to lack of attention to adequate cleanliness. It is called external chloride SCC (ECSCC). Wetting due to condensation or nearby water leakage allows developing an aqueous environment that leads to TGSCC which is usually accompanied by pitting or crevice corrosion. The stress level required for initiation of chloride-induced TGSCC is relatively modest; the threshold being close to the elastic limit of solution annealed austenitic stainless steels. The only known measure to maintain sufficient surface cleanliness is continuous attention by management at all stages of construction and during operation of the nuclear power plant.

Figure 2.3.4.4-1: Historical summary of TGSCC in German plants



ECSCC is transgranular type stress corrosion cracking (TGSCC) initiated for the reason that chlorine in the sea salt grains, high polymer products, etc., attach to the material surfaces. It has been initiated for the reason that the outside surfaces of components are not kept properly clean, then further contaminated with chlorides. These chlorides are then dissolved by dewing, deliquesce with the humidity in the air, etc., and the environment which contains chloride ions on the material surfaces is produced.

With this in mind, and as mentioned above, preventive maintenance measures (recommended practical actions) with respect to ECSCC initiation and crack growth are as follows.

Materials used for storage or curing during construction or maintenance and repair work have also been reported to cause TGSCC. Therefore the amount of halides should be controlled within the allowable level. In particular poly vinyl chloride (PVC) tape should not be used in contact with stainless steels. If PVC tape is used, special care should be taken to clean the surface after it has been removed. The selection of lubricants for bolt tightening should be made taking into account the possibility of SCC initiation. Other factors of importance are the presence of halides and the prevention of dew condensation inside the containment.

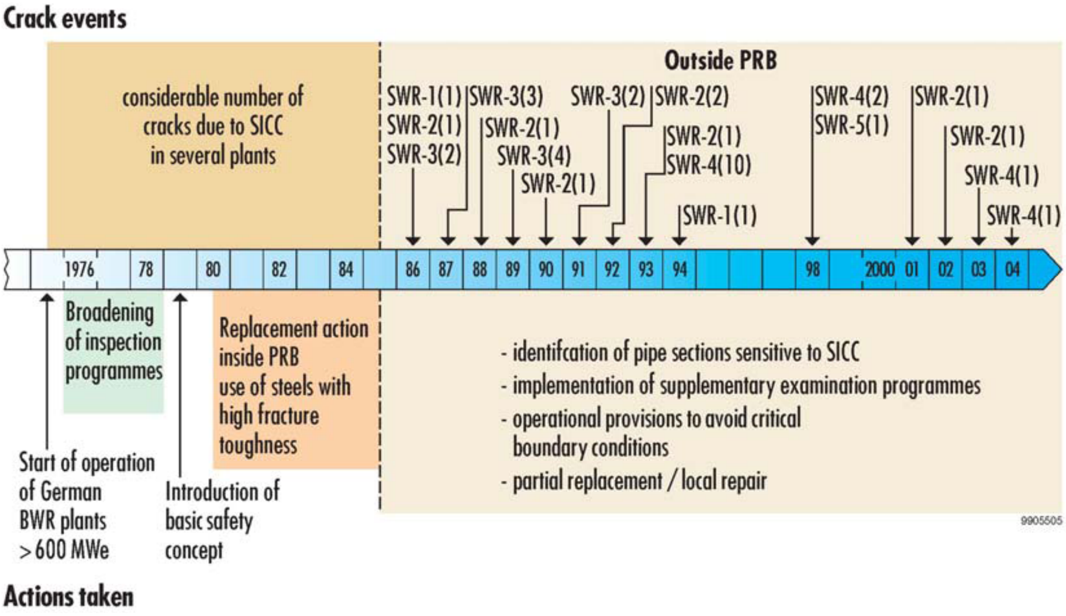
2.3.4.6 Stress-induced corrosion cracking (SICC)

In contrast to practice in other countries, relatively high-strength steels were widely used for the construction of steam and feedwater piping in the 1970s. From the mid-1970s cracks were found in several BWR plants, which were categorised as SICC (see Figure 2.3.4.6-1). This led to the introduction of the basic safety concept in Germany, *i.e.* the use of steels with high fracture toughness and moderate strength is required among others.

An extensive replacement action was performed in five operating German BWR for the piping inside the pressure-retaining boundary. Between 1980 and 1985 all high-strength steels were replaced there by steels of high fracture toughness with restricted chemical composition. Up to now the new piping has been behaving as expected, *i.e.* no further cracking has occurred.

In contrast, piping systems in the turbine hall made by high-strength ferritic steels were not replaced at that time. In the 1980s a programme was performed to investigate the critical boundary conditions for SICC. Based on the results of this programme, pipe sections in the turbine building sensitive to SICC were identified. These pipe sections are the object of supplementary examinations. Several operational and systems engineering provisions were implemented to avoid the critical

Figure 2.3.4.6-1: Historical summary of SICC in ferritic steel piping of German plants



boundary conditions, such as reduction of the oxygen content in the water phase during start-up and avoidance of corrosion during the shutdown period. In most cases, partial replacement or local repair was performed.

However, some incidents of minor safety significance occurred due to SICC outside the containment even in the 1990s, which indicates that the issue has not yet been completely solved.

2.3.4.7 Corrosion fatigue/environmentally-assisted fatigue

Results from laboratory tests generally reveal a detrimental effect of BWR and PWR water environments on the fatigue lives of specimens made from carbon steels, low-alloy steels, austenitic stainless steels and nickel (Ni)-based alloys. The parameters predominantly affecting the fatigue life of laboratory specimens are strain rate, temperature, dissolved oxygen concentration in the water and sulphur content of the material, the latter of which is only applicable for carbon steels and low alloy steels.

The detrimental effects of reactor environments on fatigue lives have been known for more than 30 years. Reactor coolant pressure boundary components exposed to the reactor water environment have exhibited degradation due to environmentally enhanced fatigue in service. In all these cases, unacceptable component fabrication, material selection, or plant operation (and combinations of these) were identified as root causes leading to the degradation. Significant large-scale, generic degradation due to environmental fatigue has not been observed in service even though environmental effects due to the impact of light water reactor (LWR) coolant were not explicitly considered in current design rules. NRC investigation of the risk associated with corrosion fatigue in the Fatigue Action Plan concluded that there was no inherent risk to core damage frequency for operating nuclear reactors, although increased probability of leakage indicates this issue requires management for extended plant operation [16-18].

Limited observations of cracking due to corrosion fatigue stand in contrast to significant occurrences of stress corrosion cracking in stainless steels and Ni-based alloys, which have been observed more systematically in reactor coolant pressure boundary welds and reactor internals from LWR plant operational experience world wide.

The lack of significant observed degradation in plant components with regard to corrosion fatigue is attributed, at least in part, to the generally conservative design requirements adopted within the ASME Code and applicable regulations (*e.g.* the NRC's requirement to keep the cumulative usage factor less than 0.1 for break exclusion locations). Margins in the design requirements appear to compensate for the detrimental environmental effects.

Another consideration when comparing the environmental effects between laboratory and service components is the applied loading associated with pressure and thermal transients. Laboratory testing typically relies on simple mechanically-controlled loading transients (*e.g.* artificially shaped waves), and may arguably include some amount of compensation for the effects of more complex thermal transient loading. Additionally, plant components are often subjected to thermal transients with long-lasting hold times at almost constant load or temperature corresponding to steady-state operating conditions which may lead to some strain recovery within the component. These differences should be taken into consideration because they can significantly affect fatigue lives.

Currently, different regulatory approaches are being pursued or initiated to explicitly address environmental effects as part of component fatigue design in countries with operating nuclear power plants (NPP), especially as longer-term operation of NPP is considered. These approaches are based on use of existing environmental factors derived from previous laboratory test programmes, reduced environmental factors based on tests that attempt to more accurately simulate plant loading and environmental conditions, or predicted "attention factors" of cumulative usage factor (CUF). Once a component exceeds the attention factor, additional measures are applied.

2.3.5 Analysis of general information

2.3.5.1 Regulations/codes and standards

In most countries information regarding regulations, codes and standards is fairly general with respect to requirements to prevent corrosive influences. Detailed information is partly contained in regulatory documents which have often been generated over the course of event evaluations, such as information notices, generic letters, bulletins, decisions, etc. These documents are of prime importance, because they contain actual regulatory positions and are updated if new information becomes available, for example in the course of root cause analysis or further events.

The national systems to incorporate lessons learned from experience in revised codes and standards are different, and it sometimes takes more than one revision period to incorporate the new information. This reflects the situation that codes and standards are based on consolidated knowledge and broad acceptance within the technical community.

The analysis of the information supplied by the participants and complementary sources like NEA and IAEA reports can be summarised as follows.

Prevention and mitigation of component degradation caused by SCC mechanisms has been dealt with in general terms during the construction and licensing phase of NPP. Today's regulatory documents and safety standards as listed in the knowledge base country reports reflect the operating experience and component degradation observed. Most countries have national regulatory documents requiring or giving guidance for ageing management and/or safe long-term operation.

Event reporting systems are in place in all countries. Flow charts describing the information flow within a country and interactions between the parties involved as well as the international organisations are stored in the KB. Regarding the reporting criteria it is well known that there are different practices in the different countries with changes over the time to adapt to experience and regulatory developments. SCC events in Class 1 components will be reported in all countries.

2.3.5.2 Inspection/monitoring/qualification

Inspection and monitoring techniques applied in the past have undergone many developments. Today most countries use ultrasonic and eddy current testing techniques and have established qualification procedures to ensure reliable detection of cracks. Some countries require test pieces with "natural" crack morphology in the qualification process, reflecting the experience that artificial cracks or notches may not be adequate to achieve the detection capability desired.

The applicable codes and standards and regulatory documents for the inspection of pressure boundary components and internal structures for SCC mechanisms are identified in the knowledge base in the country reports. The information regarding the qualification of non-destructive examination (NDE) systems are summarised as a separate table in the knowledge base web interface.

The operators shall perform inspection of vessels, pipes and other components specified by the codes and standards of academic societies in order to comply with the requirements in the laws and regulations. Besides those inspections, and in the event SCC is detected, the regulatory authority may require the operators to implement special inspection by specifying the concerned sections made of material susceptible to SCC, timing of inspection and inspection methods considering previous actions taken by domestic and overseas NPP to address detected SCC.

The determination of special inspection depends on incidents of SCC in material for which SCC has been hardly observed, validation of the applicability of new flaw detection and evaluation methods, and safety significance of a flaw which is different from previous cases. However, the regulatory authority may take flexible actions by specifying the timing and start of inspection period according to the extent of measures to be taken to address SCC in the concerned section. In order to ensure such flexibility in special inspection for SCC, both the regulatory authority and operators should work on collecting data from operating experiences which show the countermeasures against SCC and their validity.

Regarding regulatory requirements, operators conduct inspections of vessels and piping based on the code and standards. In some cases, extended inspections are required based on the regulatory practices that instruct location, timing, scope and methods of inspection as the feedback of operating experiences of observation of SCC domestically and overseas.

In these cases, materials and characterisation for which SCC has not yet been observed, application and validity of detection and evaluation methods on SCC and so on are considered. However, there are some exemptions that flexible actions for location, timing, scope and methods of inspection are applied by the extent of SCC countermeasure. In these cases, it is of importance to populate the operating experience data to confirm the SCC countermeasures and their validity and effectiveness.

It is necessary to make the AMP appropriate to the ageing mechanism (*e.g.* those common to member countries or key to the maintenance of individual ageing mechanism) and the improvement of AMP (*e.g.* six attributes of an effective ageing management programme).

It is one of key points that more sophisticated ageing management is to undertake timely and adequate maintenance activities in compliance with the age specific to each plant. Another one is harmonisation of maintenance and ageing management activities from plant construction. The following ageing management is conducted:

- ageing management from the early stage of NPP operation;
- ageing management every 10 years within the framework of PSR;
- ageing management before the operation for 30 years and following every 10 years.

The inspection methods for each SCC mechanism are described in the paragraph of inspection/ monitoring/ qualification in Section 2.3.3 (SCC mechanism).

For the development of an adequate inspection and ageing management programme the extent of information available from surveillance and monitoring systems as well as the population of systems, structures and components involved is to be considered. Specifically for a small population of systems, structures and components being operated under similar conditions the international exchange of operating and inspection experience as it is achieved by SCAP is of importance to broaden the knowledge of the issue.

Regarding methods for inspection, signal processing, evaluation and interpretation of results, further improvements are to be expected in the coming decades as was the case in the past driven by technological developments in the NDE area. The inspection of areas where previous inspections have been performed with different techniques may deliver results which are difficult to evaluate without knowing the details of different inspection techniques. Therefore the documentation of the inspection method applied and qualification procedures used is important and need to be maintained.

2.3.5.3 Preventative maintenance/mitigation and repair/replacement

It has been requested to conduct ageing management based on combination of check/ inspection and preventive maintenance in a planned manner and to prepare the plan based on data of past inspection and the results of research and development. Evaluation items such as the philosophy of check/ inspection and preventive maintenance, methods, plan and its performance are addressed in the ageing management related manual. Therefore, current knowledge on these items was analysed and summarised.

For preventive maintenance, mitigation, repair and replacement, it is effective that one of three SCC factors at least is excluded. However, it is not sufficient to exclude only one of the three SCC factors because the mechanisms of some SCC are uncertain.

When selecting the combination of preventive maintenance, mitigation, repair and replacement measures, it is desired to consider the extent of understanding of related SCC, technology level applicable to plants, operating experiences and regulatory framework. In many countries, however, ageing management is conducted taking these factors into account.

Table-2.3.5.3-1 shows an example of the preventive maintenance, mitigation and repair techniques. As described in the section on inspection/monitoring/qualification, there are practices based on the latest knowledge such that flexible practices on inspection timing and scope are conducted by the extent of preventive maintenance, replacement and repair applied.

2.3.5.4 Safety assessment

Regarding ageing management on SCC, it is necessary to conduct periodic evaluations taking into account the characteristics of ageing mechanisms; inspection from the early stage of plant operation, about a ten-year evaluation within the framework of periodic safety reviews and in some countries an evaluation before the end of 30 years' operation and then every 10 years.

As for inspection from the early stage of operation, flaw evaluating or integrity evaluation is carried out when the SCC is detected in the inspection. For ageing technical evaluation, especially IASCC, ageing mechanism trend evaluations are conducted to assess the occurrence of prediction and the postulated flaw on the location where the cumulative neutron fluence is predicted to exceed the threshold at 30 years operation. In this case, maintenance performance over the past ten years, maintenance at 30 years and current maintenance at 40 years are compared.

There are some cases, based on the prediction of possibility of occurrence, the location and frequency of inspection were prepared and other cases crack growth evaluation for the crack detected were conducted. In these safety evaluations, it is important to predict and identify the location where the possibility of SCC occurrence may be, and of importance to collect and analysis of data and to develop the prediction and evaluation methods on SCC.

In operating nuclear power plants active degradation mechanisms exist, although they were not anticipated when the plants were designed. Regulations in the different countries define the principles and provide guidance concerning the steps to be taken to assure the integrity of components. The principles vary considerably from detailed clarification of the root cause of any defects detected, expanded inspection programmes to identify defects in comparable components, or to confirm that no are present, assess and implement repair methods, flaw evaluation using standardised analysis methods.

When defects are found the first step is the identification of possible root causes. In some cases, use of different inspection methods can help in this process by better defining the flaw. Final clarification is often only achieved by taking boat samples or a micro-printing method called SUMP and carrying out metallography. Previous inspection records and operating experience are often reviewed as part of this process. It is then necessary to determine whether the component can remain in operation for a limited period of time or whether it must be repaired or replaced immediately.

In most countries, defect assessment methods are addressed in regulatory documents (codes, standards, guides, etc.) as partly referenced in the knowledge base. The documents provide guidance regarding evaluation of acting stresses, including residual stresses, the estimation of flaw geometry based on the inspection results, crack growth laws and duration of the operating time the assessment should cover. Physically based models are still at the stage of scientific development. Due to the number of variables involved as well as dependencies and correlations, further R&D efforts are ongoing as shown in Section 2.3.5.5. Common to all such evaluation methods is the need for propagation rates in the specific material/environment combination. It is generally accepted that propagation rates should be based on quality assured data whenever possible.

Table 2.3.5.3-1: Preventive maintenance, mitigation and repair techniques

Technique		BWR	PWR
Material change		• From 321 to 347 NG • From 304 to 316 L/NG • Improved heat treatment of X-750 • Alloy 600: Modified Alloy 600 (Nb added) • XM19 • Austenitic stainless steel with low-carbon grades	• SG: from 600/CS to 690 • Instrument tubes • Pressuriser nozzles for instrumentation Alloy 600 to 316 • Pzr heater sleeves 600 to 690 • RV nozzle (VC summer) • X-750 AH to improved version X-750 (support pin of CI) • X-750 to CW316 SS • Minimum ferrite of 7.5% in weld metal and CASS
Isolation technique		• Corrosion-resistant cladding	• Sleeving 690 SG (SG tube) • Plating (Ni) • Coating (cold spray) • Cladding • Weld inlay
Weld material change		• From Nb containing 182 to high Cr and Nb containing 82 • High ferrite content weld metal (more than 5%)	• From 82/182 to 52/152 • Inlays
Design change		Reducing number of welds by using induction bend pipes or forged materials	• Design changes of core internals • Down-flow to up-flow conversion • Bolting design changes, *e.g.* head-shank radii, thread details • Water pass holes in baffle formers • Shape of the head-shank curve of baffle bolts
Stress improvement		• Induction heat stress improvement (IHSI) • Peening [laser peening (LP), water jet peening (WJP), shot peening (SP), ultra sonic peening (USP)] • Mechanical stress improvement process (MSIP) • Polishing (N stripping) • Post-weld heat treatment (PWHT) • Solution heat treatment (SHT) • Improved weld preparation (including narrow gap welding, heat sink welding)	• MSIP • Peening (WJP, SP, USP) • Laser • Cavitations/water jet • L-SIP • *In situ* heat treatment • Surface buffing
Environment improvement		• Hydrogen water chemistry (HWC) • Noble metal chemical addition (NMCA) • Noble metal cladding (NMCL) • TiO_2 injection	• Inhibitor application (secondary system) • Zinc addition • Optimised hydrogen xx • Temperature reduction
Repair	Mechanical repair	• Tie rod • Clamp • Bracket • Roll repair (tube expansion) • Crack removal [electric discharge machining (EDM), drilling, grinding] • Shape memory alloy (SMA) coupling • Supplement wedge	• Half nozzle repair (BMI nozzle) • Tie rod repair • Repair weld • Drilling (BMI nozzle) • Capping (BMI nozzle) • Crack removal (EDM) • Weld over lay (WOL)
	Weld	• Weld over lay (WOL) • Repair weld	

For stainless steels and nickel-based alloys the flaw acceptance criteria are generally based on limit load analysis reflecting the ductile characteristics of the materials involved. Following the concept applied in the design to demonstrate the integrity of components of operational and design basis accident loads.

Regarding IASCC, it is considered that possibility of occurrence is less if the fluence does not exceed the threshold value, it is requested to conduct maintenance taking cumulative fluence into account. For PWR, it is anticipated that some location (*e.g.* baffle former bolts) with the high potential of IASCC might occur in the extent of no affect to the intended function of components when the neutron fluence exceeds the threshold. Therefore, as with the ageing management, it is requested to conduct evaluations and predictions of the cumulative fluence at locations with a high potential of occurrence and to perform appropriate measurement near the threshold value.

2.3.5.5 Research and development

Research and development efforts are ongoing in most countries with the goal to improve understanding of the initiation and propagation processes associated with the different types of stress corrosion cracking and also for preventive maintenance/mitigation and repair/replacement. Research and development is carried out by industry, regulators and academia. Extensive efforts are also under way to develop inspection and monitoring techniques, repair and replacement methods, mitigation and preventative maintenance programmes, component integrity assessment technologies, and the necessary data to enable the performance of safety analyses. The outcomes of research and development should be the seeds for the knowledge base and further commendable practices may be identified from the knowledge base. The results of such research and development will be reflected in the commendable practices and also be further reflected in revisions of regulations and codes and standards in the different countries.

To ensure that research and development is relevant and cost effective it is important that experience feedback is incorporated into the planning. One of the main sources of such relevant information can be obtained from a thorough root cause analysis. This is particularly true when stress corrosion cracking is found in an unexpected location or material/environment combination. With the establishment of the SCC event and knowledge databases new sources of experience feedback are available for participants to analysis and review.

For the preventive maintenance of SCC, research of crack initiation prevention, cracking growth suppression, proper inspection, clarification of crack initiation and crack growth mechanisms, and root cause investigation, etc., are important. In addition, it is important to identify the subjects such as verification for long-term service, and to perform safety-related R&D and feed the results back into the ageing mechanism management of components. The long-term operation of plants is also an important area of research with regard to stress corrosion cracking. Experience has shown that initiation times can be very long, up to decades, and thus it is to be expected that new problems will arise in the future.

Because the degradation due to IASCC proceeds with neutron irradiation with long-term operation, it is important to conduct a comprehensive evaluation taking into account the high fluence comparable to the operating period, irradiation hardening (decrease in plasticity), segregation, accumulation of helium, stress relaxation/creep, and swelling. Therefore, it is of importance to reflect on maintenance practices of the results of understanding of mechanism, modelling and its verification due to research for irradiated materials from ex-plants, and to develop the degradation prediction for the components in service.

Regarding these approaches, it is necessary to promote research co-operation and ties within the international framework. Below are some examples of ongoing safety-related research regarding stress corrosion cracking being performed by industry, regulators and academia in the areas of inspection, monitoring, qualification, preventative maintenance, mitigation and repair/replacement.

Mechanisms

- Mechanism studies for PWSCC and IASCC initiation to improve the prediction of initiation.
- Development of disposition curves for safety assessment using quality-assured crack propagation data.

- Methods for in-service measurement of residual stress.
- Development of disposition curves for safety assessment using quality-assured crack propagation data.

Materials

- Studies of the long-term resistance to stress corrosion cracking of replacement materials such as Alloy 690.

Inspection of SCC

Regarding inspection on SCC, there are technical issues of detectability and sizing of SCC of Ni-based alloys and PWSCC from experiences of inspections at the fields. It is necessary to study to reduce the area inaccessible for inspection and to study inspection at location conducted repair welding such as weld overlay methods because of remaining inspectability issues. From the viewpoints of reasonable inspection, it is necessary to develop inspection technology applied during operation. It is necessary to develop and verify the non-destructive inspection combined of UT and ECT technology in the future when it is expected that inspection may be conducted with good efficiency and high accuracy. The themes of research and development are listed as follows:

- to verify the UT inspection accuracy;
- to develop inspection equipments and its technology to improve the inspection accessible area and to improve UT inspection accuracy;
- to prepare codes and standards reflecting the results of research and development in a continuous manner.

Technology of in-service monitoring

It is necessary to improve the accuracy of current monitoring technology from the viewpoints of promotion of application to operating plants. The following themes are desirable:

- to develop advanced inspection methods, some for specific applications or complicated component geometries and robots for application under water;
- to develop on-line monitoring techniques for crack initiation and propagation;
- to promote the further development of fundamental technology on in-service monitoring technology and environment evaluation technology;
- to develop and verify the in-service monitoring technology.

Preventive maintenance technology

For BWR, it is necessary to prepare codes and standards on plant ageing maintenance based on review of currently applied technologies and on verification of applicability and effectiveness for preventive maintenance methods. For PWR, it is necessary to develop and verify mitigation technology on PWSCC. The following themes are desirable:

- to promote the preparation of codes and standards on environmental isolation methods and related welding methods to prevent SCC propagation;
- to verify the effectiveness of long-term integrity after repair welding;
- to study effectiveness of new chemistries such as the effect of zinc, titanium oxide, changes in hydrogen concentrations (PWR).

Chapter 3: Cable Working Group

3.1 Introduction

Electrical cables are installed approximately 1 000 to 2 000 km for each nuclear power plant (NPP). They transmit instrument and control signals, supply power to electric components such as motors, actuators for power-operated valves. Certain cables for safety-related systems are required to remain operational until the last stage of an in-service period, during and following a design basis event such as a loss of coolant accident (LOCA).

Typical construction of the cable consists of conductor, insulator, filler, tape and jacket and is shown in Figure 3.1-2. The filler and tape are used for fitting them into the cable, and the jacket is for the protection of the insulator from external force that might be applied during cable installation. The insulator has the function of assuring electric independence between the conductor and ground or between the conductors and is made of polymer material such as XLPE, EPR and SIR.

Figure 3.1-1: Example of cable layout in NPP (PWR)

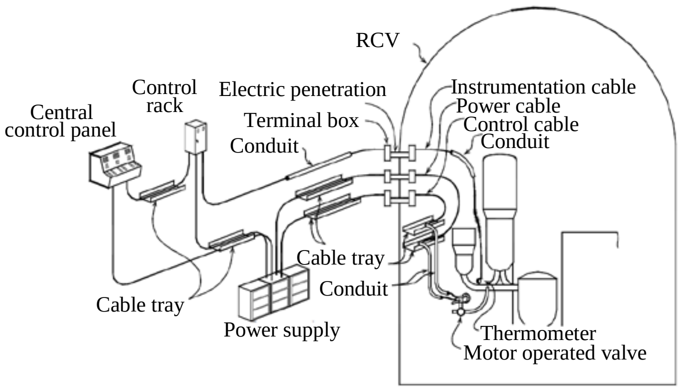
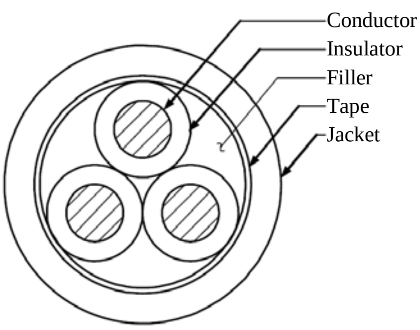


Figure 3.1-2: Typical cable construction



The insulator degrades during normal operation due to various environmental factors, such as temperature, radiation, moisture, etc., and their combined effects. In general, degradation of polymer is said to be attributed to oxidation. It causes chain scission or cross-linking among chains and the oxidative products such as carbonyl and hydro-peroxide accumulate in polymer matrix. The electrical property and mechanical property degrade with the degree of oxidation and such phenomenon is generally referred to as insulation degradation and it progresses with time.

In addition to the degradation which occurs during normal operation, rapid degradation of cable insulator progress can occur under the severe environments such as the design basis event (DBE) and it will cause failure. Therefore appropriate ageing management for cables, especially for safety-related cables, is important for prevention of potential failures and for safe long-term operation of NPP.

For maintenance, inspection and condition monitoring for cables is performed to detect degradation of the insulation. When degradations are detected, cables are repaired or replaced. However, it is difficult to evaluate the integrity of safety-related cables in a LOCA environment based on the results of the inspection or condition monitoring techniques currently used. The type of test that simulates degradation of the cable during normal operation and the LOCA environment is performed as an environmental qualification (EQ) test. In addition, environmental condition monitoring during normal operation is performed to confirm that the environmental conditions of the place where the cables are installed are within the design basis assumptions and the result of the monitoring is also used to requalify the service life of the cables. If the environment condition is very severe, mitigation of the cable's installed condition may be applied.

A few countries have introduced a requirement to perform a flame propagation test on aged cables. This requirement is to ensure that the flame retardant capabilities have not degraded due to operational ageing. A representative sample of cables has to be tested for their fire properties to confirm its continued suitability for nuclear plant application. Generally, XLPE, EPR, EVA based cables show no changes after the ageing. PVC cables improve their fire retardant properties due to the loss of flammable plasticisers during the ageing.

3.2 Cable database

3.2.1 Scope

The cable database covers safety-related cables that support emergency core cooling, safety cables (required to prevent and mitigate a design basis event) and cables important to plant operation (cables whose failure could cause a plant trip or reduction in plant power). The scope of the database includes cables with voltage levels up to 15 kV AC and 500 V DC, including instrumentation and control (I&C) cables.

3.2.2 Development of structure

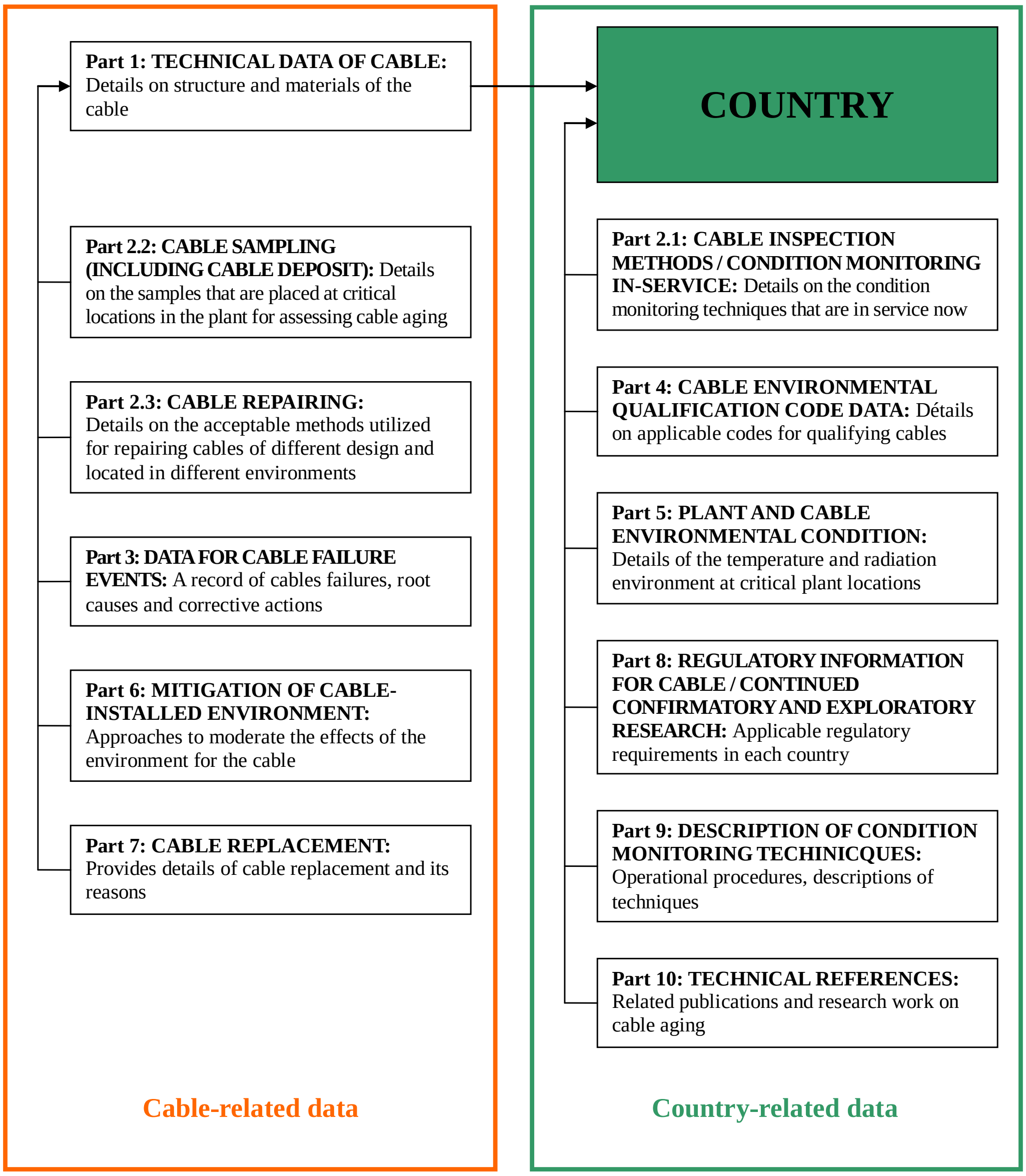
The electrical cable systems are a vital part of the plant because of the communication they provide for the operation of every system, monitoring, controlling and shutdown. The cables primarily fall into two categories: cables with operational significance, and those with nuclear safety significance. The failure of cables with operational significance could trip a plant, cause a transient, reduce output power, or cause other conditions that affect power production. Therefore, the cables that have operational significance receive prompt attention for corrective and preventive care because of the direct impact on plant availability. Most of the cable failures in this category are remedied through a simple replacement of the cable; an improved design is sought only when there are repeated failures. The steady advancement in cable chemistry has further contributed to better performance of cables that were manufactured in the 1990s and later.

The cables with nuclear safety significance are needed to mitigate the effects of an accident, monitor critical parameters to ensure safety, monitor the effectiveness of emergency core cooling, and monitor and prevent the potential for containment breach and the consequent off-site releases. Most of the cables in this category are in standby mode and never subjected to an accident environment. A significant per cent of the cables in this category are required to be operational during and following an accident to preserve nuclear safety. The required confidence in the performance of these cables during an accident environment is achieved through the testing of cables in simulated accident environments. While the simulation of accident environment has improved over the years, it has not offered a continuing affirmation on the qualification process that prevailed in the 1980s. Therefore, the working group added the qualification of the safety-related cable into the scope of the database.

The operating experience shared by the participating countries indicated that there were other environmental factors such as moisture intrusion, chemical and adverse environment that could degrade the cable during its normal service (non-accident) conditions. Therefore, it is important to monitor the operational readiness of the cable periodically or in response to unanticipated service conditions. The cable condition monitoring techniques have been evolving and the industry has not yet developed a single test that could evaluate the condition of all the variety of cable designs. In order to share the evolving techniques applicable to the variety of cables, the working group decided to include the cable condition monitoring techniques and the related research, study and publications to the database.

The database is organised in 10 parts to document all the details of the cable including design, qualification, maintenance, condition monitoring and continuing research. Data entry in the SCAP cable database is managed via tables and roll-down menus. Database searches and applications can be performed through user-defined field entries. The data entry tables are organised to capture essential cable insulation failure events along with information regarding environmental qualification and condition monitoring. The 10 data entry sheets are as shown in Figure 3.2.2-1.

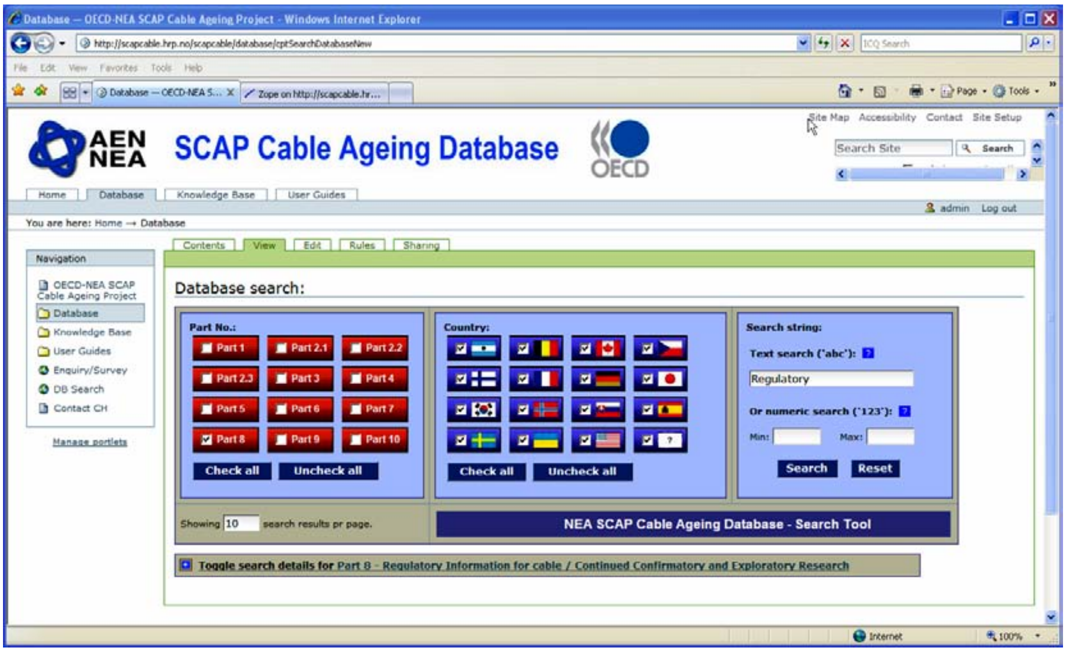
Figure 3.2.2-1: Cable ageing database structure



3.2.3 Search capability

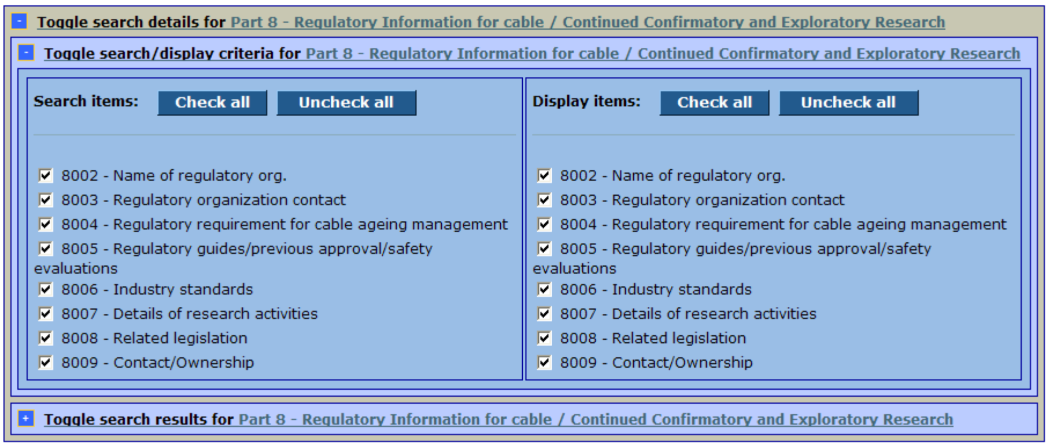
The cable database search tool allows fast access to desired information in any part of the database. The search tool opens up for multi-section/multi-country searches, with possibilities for tailor-made search and display criteria. Figure 3.2.3-1 shows the entry point of the search tool, with panels for selecting main search criteria as parts, countries and search strings.

Figure 3.2.3-1: Entry point for the database search tool, showing panels for selecting the main search criteria, *i.e.* parts, countries and search strings



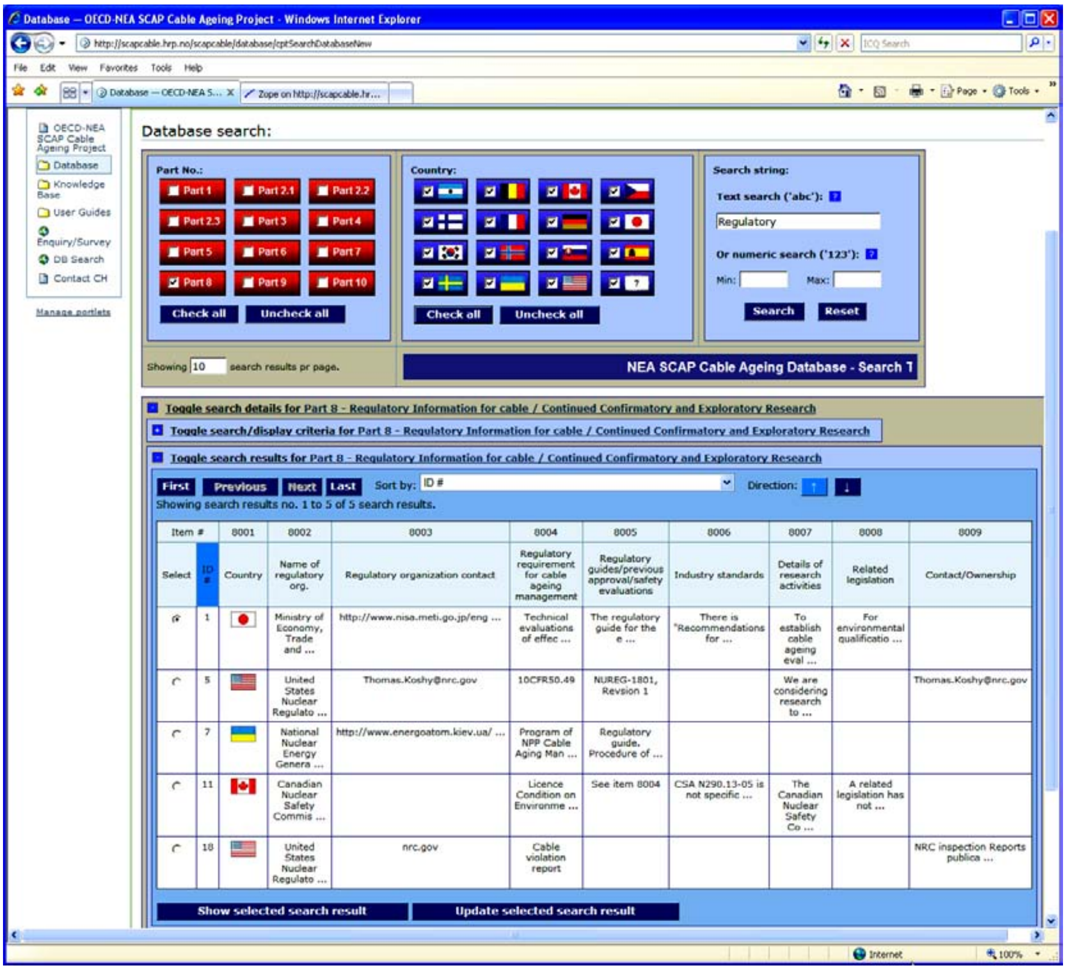
To narrow the search down to one or more fields within one part, each part has a dedicated search/display criteria panel allowing the user to specify in detail the actual field(s) to be considered in the search. In the same way the user can choose which field(s) of that particular part shall be displayed in the search result list. See Figure 3.2.3-2 for an example from "Part 8: Regulatory information for cable/Continued confirmatory and exploratory research".

Figure 3.2.3-2: Detailed search and display criteria panel for a given database part, allowing detailed field-specific selections



An example of a database search and its outcome is shown in Figure 3.2.3-3. Here, searching for the word “Regulatory” within “Part 8: Regulatory information for cable/Continued confirmatory and exploratory research”, given any country and a list of applicable cable regulations, organisation, a contact name for the regulator, etc., are displayed in the figure. Viewing or editing detailed record information based on the search results list can be done by choosing “[Show/Update] selected search result”.

Figure 3.2.3-3: Search results (example)



3.3 Cable knowledge base

The knowledge base was built from the data that was collected from the member participants. The data included all the subject areas discussed in Section 3.2.2. The working group evaluated the information in the database and identified subject areas that were of great significance that would be of great use to current nuclear operators and regulators and classified them as the knowledge base. Because of the wide variety in the classification of importance the assessment began with safety classification and proceeded to identify the essential elements of the knowledge base.

The Cable Working Group has examined and reviewed the data collected in the database and extracted information, leading to a certain number of conclusions. There is a wide variation of cable classification among the different countries. These classifications form the basis for the required level of qualification to meet its specific performance requirements. Table 3.3-1 provides a relative comparison.

Table 3.3-1: Comparison of different classification systems

National or international standard	Classification of the importance to safety			
IAEA	Systems important to safety			Systems not important to safety
	Safety	Safety related		
IEC 61226	Systems important to safety			Unclassified
	Category A	Category B	Category C	
France N4	1E	2E	IFC/NC	
European utility requirements	F1A (automatic)	F1B (automatic and manual)	F2	Unclassified
Japan	PS1/MS1**	PS2/MS2**	PS3/MS3**	Non-nuclear safety
USA	1E, safety related, or safety	Non-safety related		
		Important to safety		Balance of plant (BOP)
CANADA (CSA Std N209.14)	Important to safety			Not important to safety
	Safety, safety related			Not safety, not safety related
	Category 1	Category 2	Category 3	Category 4

* DBE: design basis event. (There is also Class 1 in Russian regulations, but it is not relevant to I&C systems and does not correspond to IAEA Safety, IEC Category A and IEEE Class 1 categories.)

** PS: prevention system, MS: mitigation system. There is no cable classified as PS1.

The Ukraine systems on nuclear power plants are categorized into four classes based on the influence on plant safety:

- Class 1: Fuel elements and systems of NPP for which failures led to fuel element damage exceeding the limits specified for the design basis event (DBE).
- Class 2: Systems for which failures led to fuel element damage within the limits of DBE. Elements of safety systems for which failures led to failures in operation of such systems.
- Class 3: Elements of safety systems not belonging to 1 and 2 safety classes. Systems for protection of plant personnel and residential population from radiation.
- Class 4: System for normal operation of NPP which are not relevant for safety. Other systems not belonging to1, 2 and 3 safety classes.

3.3.1 Technical data of cable

General information

This section addresses the variety of cables that are currently in use at various nuclear stations. The most common type of insulation materials are cross-linked polyethylene, ethylene propylene rubber (EPR), and ethylene propylene diene monomer (EPDM) as shown in Figure 3.3.1-1. The most common type of jacket material is chloro-sulphanated polyethylene (CSPE) as shown in Figure 3.3.1-2. However, there are some unique applications with unique materials with distinct properties essential for certain plant applications.

Polyvinyl chloride insulation

The general use of polyvinyl chlorides (PVC) has been discontinued at nuclear stations because of its undesirable nature with regard to fire propagation.

In the Czech Republic, new NPP installations are significantly restricted with regard to the use of certain chemical elements. European Utility Requirements ("European Utility Requirements for LWR Nuclear Power Plants", Revision C, Vol. 2, Chapter 2.6: Material-related Requirements, April 2001) quotes general rules and requirements concerning new NPP projects, *e.g.* the amount of halogens should not be higher than 200 ppm. There are also requirements concerning other elements like sulphur, zinc, lead, mercury, asbestos, etc. These requirements disqualify PVC cables for future installations.

Figure 3.3.1-1: Material of cable insulation

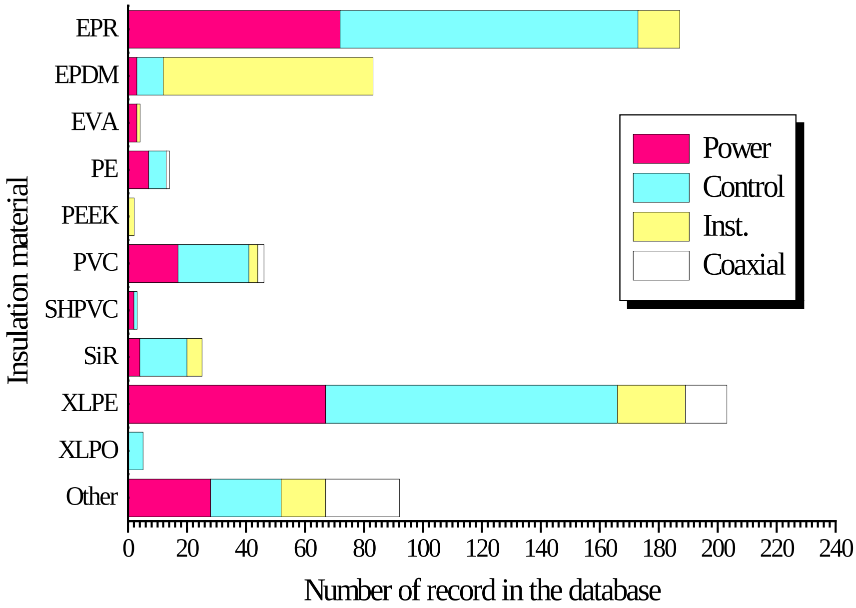
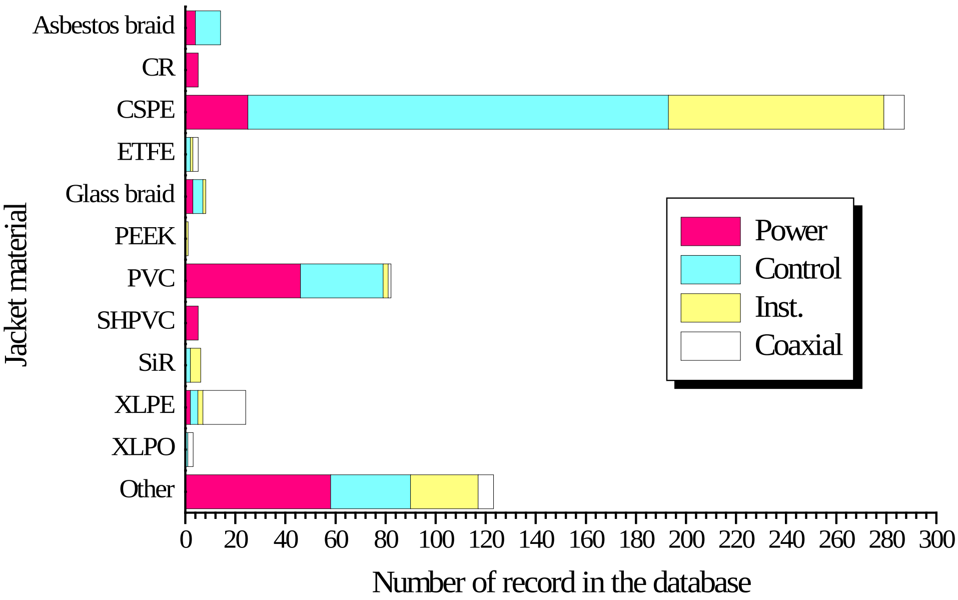


Figure 3.3.1-2: Material of cable jacket



The design temperature and dose rate of the cable recorded in the database are shown in Figures 3.3.1-3 and 3.3.1-4.

In Spain, PVC insulated cables are not commonly used in NPP. Only two plants have used them in safety-related applications.

In a BWR plant, 1E, PVC insulated, instrumentation and control cables have been used, but these cables were located in mild environmental conditions and, as a consequence, were not required to be qualified for DBE conditions. The cables have recently been visually inspected and tested, according to the “Cable Ageing Management Program” developed in support of the plant “Life Extension Application”. No significant degradation has been found in these cables after 40 years of operation.

Figure 3.3.1-3: Design temperature of cable

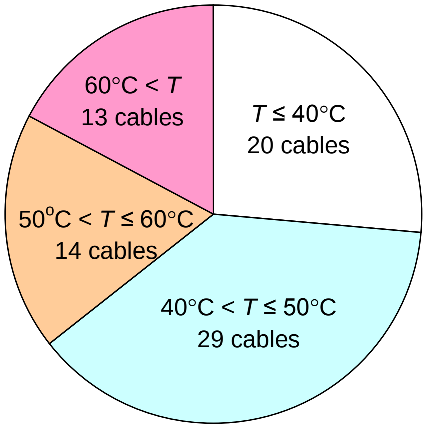
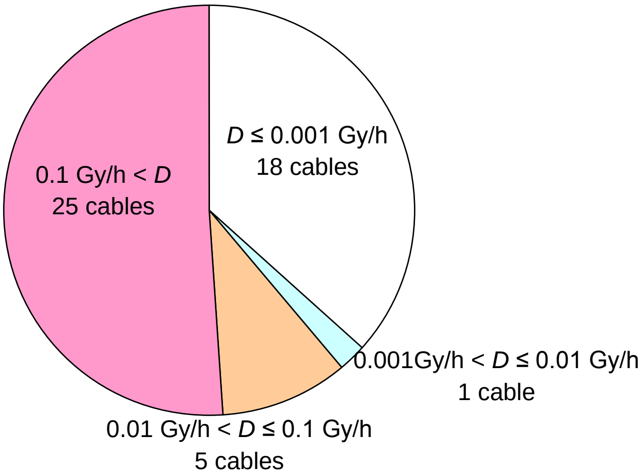


Figure 3.3.1-4: Design dose rate of cable



Another PWR plant has installed 1E instrumentation and control cables with PVC insulation and a polyethylene jacket. These cables are located outside reactor containment (containment annulus) and they are required to function under a harsh environment (HELB conditions). These cables "Pirelli Radiflam" (see database cable ID 511) have been qualified according to IEEE-383-74. Pirelli specified a maximum continuous duty temperature of 70°C for these cables. During the qualification process the cables were aged at 110°C for 8 days. Maximum DBE qualified temperature was 112°C. After 23 years of operation, no ageing degradation has been reported by the plant for these cables.

In Japan, special heat-resistant PVC (SHPVC), an improved type of PVC using heat-resistant plasticisers (which withstand up to 80°C) and polyvinyl chloride resins with a high degree of polymerisation, is used mainly for the cables installed in high-temperature area. The maximum temperature of continuous duty for the cables with SHPVC, specified in Japanese standards for electric power utilities, is 80°C. Some cables installed outside the containment vessel of BWR are required to perform their function under DBE environment, and thus they are qualified by EQ test.

The Argentinean programme to qualify in-containment cables for the CAREM advanced reactor project concluded against the use of PVC insulated cables. In this context, and based on international experience, the life cycle of cable ageing management was developed with a special focus on environmental qualification. The first efforts concerned the evaluation of material characterisation and thermal endurance properties. The first set of materials to be studied was PVC cable insulation. Plastiser characterisation and content evaluation were done and the activation energy evaluation was done based on IEEE standards.

This last set of tests was carried out with elongation at break curves vs. time curves at three different temperatures. Once this step was completed, the value of the intersection between 50% elongation at break (relative to the un-aged material) and time was stored for each curve. Finally from the slope of the linear regression of the graphics log time vs. reciprocal temperature, activation energy was derived.

The results indicated that, even though it was possible to evaluate activation energy in this manner, such property has a strong dependence on the value chosen for elongation at break. It was concluded that commercial PVC formulation does not follow Arrhenius behaviour and therefore, it is undesirable to use this model to predict future ageing characteristics and cable performance.

Environmental qualification of cable

Environmental qualification (EQ) was required of almost all the cables recorded into the database as shown in Figure 3.3.1-5. As for these cables, EQ test was carried out based on the IEEE Std. 323 and 383 or a standard similar to IEEE standard.

The qualification period (qualified life) of the EQ cable recorded in the database is shown in Figure 3.3.1-6. The qualification period of more than the half of the cables is 60 years.

Figure 3.3.1-5: EQ cable or non-EQ cable

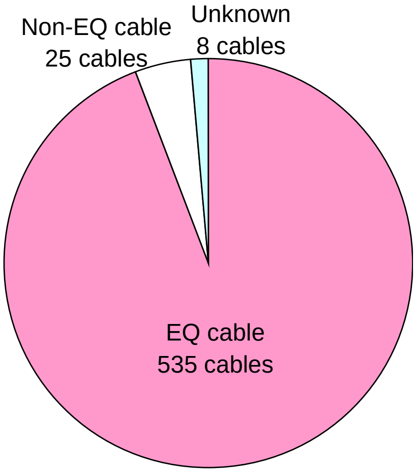
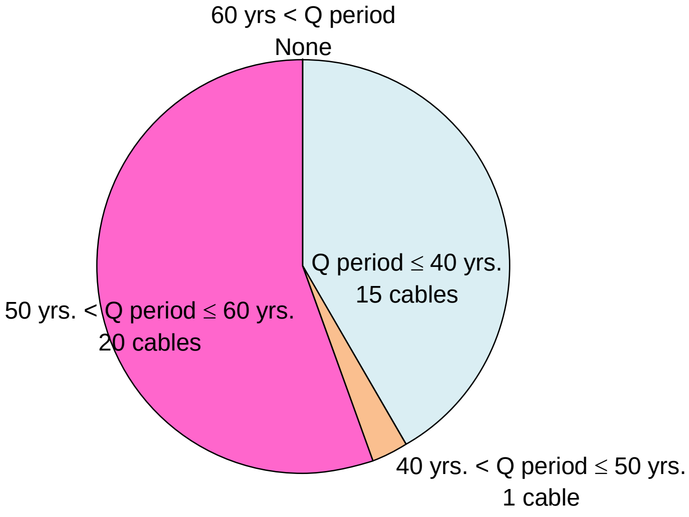


Figure 3.3.1-6: Qualification period of EQ cable



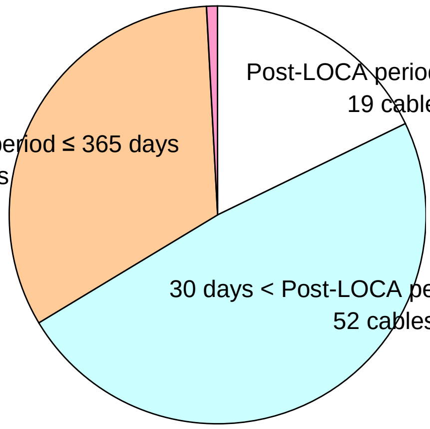
Cables with 60 years life

Wyle performed a cable qualification programme on cables that were naturally aged in a nuclear power plant for 29 years. Samples were removed and a test programme performed on all of the required ageing to simulate another 31 years prior to DBE radiation and accident simulation. These cables are now qualified for 60 years. The subject cables were Kerite FR and Kerite FR3 unjacketed power and control cables. This qualification would apply only to plant-specific conditions because the cable was naturally aged at a specific nuclear station.

Qualified post-accident mission time

The qualification post-LOCA period for the EQ cables, as recorded in the database, is shown in Figure 3.3.1-7. The qualification of about half of the cable post-LOCA period is 30 to 100 days.

Figure 3.3.1-7: Qualified post-accident mission time



The post-accident mission time in the United States varies widely, ranging from two weeks to 180 days and in certain cases the duration is more. The approach to post-accident mission and the corresponding qualification period evolved from the early 70s and marked differences were noted following the Three Mile Island Accident in the US.

The post-accident temperature profile of the core, containment environment, radiation, etc., forms the basis for mission time for each of the accident mitigation systems. While specific mission time may offer some relief on the duration and level of environmental qualification of certain components, the general practice had been to qualify the common cables for the worst-case environment. Certain operators have opted to specify component specific mission time based on its specific mission and the associated cable would need only the same mission time. Certain specialty cables may be qualified for a limited mission time because of specific chemical property limitations.

In Japan, the "Assessment of Electrical Equipment Ageing for Nuclear Power Plants (AEA)" project started in 2008. One of its purposes is the re-evaluation of the mission time required for cables to maintain their functions during the post-LOCA period. The period for BWR is 100 days and for PWR one year.

In Spain there are six operating NPP, four PWR and two BWR, but all of them present specific differences in their designs. Therefore, the required qualification profiles (temperature/time and pressure/time) are different and plant-specific. Selected examples are given below:

- CN Vandellos II PWR. Combined LOCA + MSLB qualification profile. Post-LOCA period of 120 days with temperature decaying from 300°F to 140°F.
- CN Santa Maria de Garoña BWR. LOCA qualification profile. Post-LOCA period of 400 days with temperature decaying from 185°F to 101°F.

Conclusion

While it may not be possible to identify a fixed period for post-accident mission time for all reactor designs, certain basic principles can be observed in arriving at that period. The primary basis for environmental qualification is that the equipment has to perform its safety function in a harsh environment where it is practically impossible to perform any equipment replacement or corrective maintenance. Therefore, the minimum duration for qualification is the time that is needed for accessing the area when the risk is relatively low and the environment is suitable for people to enter and perform services with reasonable precautions. For most containments, this period is considered to be a minimum of 180 days. Based on the worst-case design basis event, a plant-specific evaluation is necessary to arrive at a post-accident mission time in each case. The containment size, reactor core size, containment spray systems, clean-up systems and its effectiveness will determine the final value for this duration.

Assessment of paper-oil insulated cables for life extension using double kink number

In Ukraine, cables with paper-oil insulation (BMI) have a specified service life term of 25 years. Further life extension is accepted based on conditions of cable insulation and protective shields. Measurement of mechanical characteristics of cable paper is one of the methods of cable condition assessment. It is known that during the ageing process the degree of paper polymerisation decreases, the elongation at break indicator and the number of double kink fall down. To define the degree of polymerisation it is necessary to prepare a cellulose solution and to measure its viscosity.

The advantage of this method is that it requires only a small amount of paper sample. On the other hand, high labour for the solution preparation and toxicity of solvents are the disadvantages. Additionally, it is difficult to measure elongation at break because even at the initial condition the ratio for a paper is only 2% in relation to polyethylene, at 300 to 500%.

The number of double kinks is a more convenient indicator for the technical condition monitoring of the cables with paper-oil insulation. From one section of a cable with a length of about 0.2 m it is possible to prepare 20 to 70 strips of a paper with the size 140 $\times$ 15 mm^2. This number of samples is enough for the reliable definition of the specified parameter. The samples of cables for laboratory researches are selected during repair.

Characteristics of cable paper in initial condition are specified in the standard documentation. The breaking effort of strips of the specified size should make at least 7-13.5 kg (depending on a thickness of paper type K080-K120), relative elongation at break – 2%, number of double kinks – at least 2 000, irrespective of a paper thickness.

3.3.2 Cable maintenance data/condition monitoring

Generally, cable was considered a maintenance-free item and its failures were considered to be random events and therefore the failed cable was replaced without much of a post-mortem analysis. The United States nuclear industry noticed a number of failures and questioned the randomness of the failure. These failures appear to be linked to ageing and monitoring the condition of insulation was recognised to be a rising need to mitigate in-service failures.

3.3.2.1 Background for condition monitoring

The interest of safety aspects of cable ageing is increasing world wide because of their impact on several industrial fields, like power generation, transportation and defence. Although the environmental conditions and degradation mechanisms of installed cables can be different in each application, the negative consequences of cable failures, both from a safety and performance standpoint, are so important that almost all the countries in the industrialised world have some research project in progress for this area. In the nuclear field, where cables are normally qualified before installation for an expected life of 40 years, there are a number of issues that are not adequately solved today. These issues include:

- The effect of the particular adverse environment conditions (high radiation, humidity and temperature), especially during and after a design basis event (DBE).
- Extending the plant life after 40 years involves the requirement to assess and qualify the cable conditions for a longer time.
- Many cable condition monitoring techniques do exist today, but none of them are considered accurate and reliable enough for all the cable materials and types in use at their installed applications. In addition to that, only few of them are non-destructive techniques and are applicable *in situ*.
- Accelerated ageing techniques, for qualification purposes under DBE conditions, are often not conservative and should be complemented with reliable condition monitoring methods.

The United States White House National Science and Technology Council Committee on Technology issued a report in 2000 [19] in which safety issues on wire systems were addressed. The conclusions of this report are important to understand the weak points of the current status and which topics should be addressed in future research. The recommendations of the Committee can be summarised as follows:

- increase co-operation between industry and research institutes, also internationally;
- improve design and functionality of wire systems;
- develop advanced wire system techniques including condition monitoring.

Research efforts from the IAEA [20] and the OECD NEA [21] led to similar conclusions.

In the United States, the NRC published in May 2003 a Regulatory Issue Summary (RIS) [22], wherein it reported the conclusions of qualification tests on I&C cables. Here a particular concern was posed on cables status assessment needs when extending the plant life and the need to have reliable qualification methods for LOCA and post-LOCA conditions. Basically, the NRC concluded that current I&C wire system qualification methods provide a high level of confidence that the installed cables will perform adequately during a design basis event, as required by 10 CFR 50.49. However, some LOCA test failures indicate that under certain conditions, the accepted conservatisms in the qualification tests is less than expected. Moreover, no single monitoring technique was found to be adequate to reliably detect I&C cable failures. Two recommendations are significant, among others:

- Environment conditions should be monitored during plant operation, to ensure that they do not exceed those applied for the qualification tests.
- A combination of condition monitoring techniques is suggested, to overcome the limits existing in each single method.

3.3.2.2 Cable inspection/condition monitoring method in-service

In January 2010 the US NRC published a report entitled “Essential Elements of an Electric Cable Condition Monitoring Programme” (NUREG/CR-7000). This publication is based upon the results of the NRC’s electric cable and equipment research programmes, industry guidance and standards, and the experience and observations of other electric cable condition monitoring and qualification testing. The programme methodology presented in this report provides guidance on the selection of cables to be included in the programme, characterisation and monitoring of cable operating environments and stressors, selection of the most effective and practical condition monitoring techniques, documentation and review of cable condition monitoring testing and inspection results, and the periodic review and assessment of cable condition and operating environments.

The cable inspection/condition monitoring methods applied in-service recorded into the database are shown in Table 3.3.2.2-1. This table shows that there are few countries which are conducting condition-monitoring of the cable under normal operation.

In the Czech Republic, the basic condition monitoring technique is visual and tactile inspection. The inspection is performed mainly in the hot spots. Nevertheless, all safety-related cables are successively inspected within the whole length of their installations. For cables where the calculated service life time is very short and for some cables located in hot spots, other condition monitoring techniques are applied. The method commonly used in the Czech Republic is the OIT/OITP method.

Table 3.3.2.2-1: Cable inspection/condition monitoring methods applied in-service

Method	Applied number of country	Applied cable type					
		Power	Control	Inst.	Coaxial	Fibre optic	Hybrid
Combined AS high potential and partial discharge extinction level tests	1	√					
Current leak rate testing	1	√					
Elongation at break.	2		√	√			
Indenter	2	√	√	√	√		
Insulation resistance	2	√	√	√	√		
Line resonance analysis (LIRA)	2	√	√	√	√		
Oxidation induction temperature (OITP)	1	√	√	√	√	√	
Oxidation induction time (OIT)	1		√	√			
Potential decay	1	√					
System of electrical characterisation and diagnosis	1	√	√	√			
Tangent delta	1	√					
Thermogravimetry	1	√	√				
Visual/tactile inspection	5	√	√	√	√	√	√

In Japan, all cables installed in NPP are inspected periodically and the acceptance criteria for respective techniques are shown in Table 3.3.2.2-2. However, these acceptance criteria cannot be used for the evaluation of integrity in the DBE environment.

3.3.2.3 Cable sampling/cable deposits

In order to carefully monitor insulation degradation, it is recommended to have cable samples exposed to actual nuclear plant service environment. A novel idea, practiced by several plants, is to deposit additional pieces of cable at service locations and expose them to actual harsh service environment. Portions of these cables can in turn be conveniently made available ongoing qualification, destructive examination, and other requalification efforts based on the evolving needs.

Cable deposits in Czech NPP

In the Czech Republic, NPP have the following cable deposits:

- On the primary loop line with the temperature and radiation higher than other locations within NPP. Safety related 1E LOCA cables.
- Inside the NPP containment, different locations where the cables age under the same conditions as the cables in service. Safety related 1E LOCA and non-LOCA cables.
- Outside the containment. This is a good source of cables for future testing.

The cable deposits on the primary loop line is most important where the cables age under higher temperature and dose rate than in other NPP locations. Six such deposits have been installed in Czech Republic; four in Temelín NPP (8 years in operation) and two in Dukovany NPP (23 years in operation).

The deposited cables are representatives of the cable types used in containment. The number of samples and its total length must be adequate for scheduled and unscheduled removal of samples over a period of up to 40 or 60 years. There are three 8 meters long samples for each cable type. These samples are prepared for:

Table 3.3.2.2-2: Acceptance criteria for inspection in Japan NPP

Cable inspection method	Insulation resistance	Visual inspection	Current leak rate testing	Potential decay	Tangent delta
Cable type	All cable types except for fibre optic.	All cable types except for fibre optic.	Power	Power	Power
Rated voltage	No limitation	No limitation	6 600 or 7 000 V	6 600 or 7 000 V	6 600 or 7 000 V
Operating voltage	No limitation	No limitation	6 600 or 6 900 V	6 600 or 6 900 V	6 600 or 6 900 V
Insulation material	No limitation	No limitation	No limitation	No limitation	No limitation
Assumed ageing mechanisms	Insulation resistance degradation caused by thermal and radiation ageing	Physical damage, thermal ageing, etc.	Water tree degradation	Water tree degradation	Water tree degradation
Industry standard	JIS C 3005*[1]	No standard	Technical document 116C, *Maintenance Guideline for CV Cable for Medium Voltage*, Dec. 2007, Japan Electric Wire & Cable Makers' Association	No standard	JIS C 3005*[1]
Details of test	1) Disconnected from power source. 2) Keep connected with load or disconnected with load. 3) Voltage for megger testing: 1 000 V megger for power cable (6 600 or 7 000 V) 500 V megger for power cable (600 V)	Colour change of an insulator/jacket is checked visually	With a DC-voltage generator, DC voltage is applied between each conductor and shield layer, and the temporal response of leakage current is measured	The discharge time after DC charging is measured by special devise	AC 3.8 kV is applied between each conductor and shield layer, and the tangent delta is measured by the Schering bridge method
Management criteria	Case by case, *e.g*. power circuit > 5 M ohms, control circuit > 2 M ohms	None	*e.g.* 6.6 kV cable: $< 1 \times 10^{-7}$ A (at 5 kV, 10 min.)	*e.g.* 6 kV cable: applied voltage 5.0 kV, judged voltage: 4.0 kV, decay time: > 230 sec.	*e.g.* 6.6 kV cable: < 0.1% (at 3.8 kV)
Additional information	Carried out once during a 1 to 6 time periodical inspections	Quantitative ageing assessment cannot be performed	Carried out to examine the water tree for power cables installed outdoors once every 4 to 6 times periodical inspections	Carried out to examine the water tree for power cables installed outdoors once every 4 to 6 times periodical inspections	Carried out to examine the water tree for the power cable installed outdoors once every 4 to 6 times periodical inspections

*[1] JIS: Japanese industrial standards.

- mechanical testing (elongation at break, strength at break), determination of physico-chemical properties (oxidative-induction time or temperature, density);
- testing of electrical properties (insulation resistance, loss factor tangent delta and capacity at different frequencies);
- verification of the LOCA and post-LOCA survival.

Temelín NPP is a new nuclear power plant of PWR 1000 type, in operation since 2001 and the deposits were installed before the plant start-up. The cable deposits were installed at the cold leg of the primary loop line between reactor pressure vessel and the main recirculation pump. Each construction skeleton is assembled around the loop line. The structure has a width of 120 cm and the distance from the thermal insulation of the loop line is 10 cm. The selected cables are wound around this structure.

Nine types of cables performing safety functions of Temelín NPP have been chosen for the deposit. The different types of cables include low voltage control, instrumentation, power cables and one medium voltage power cable.

The cables are not powered. The joule heating due to current flow in power cables may heat up the insulation by about 30°C. Important safety-related power cables are operated in Temelín NPP at a relative low temperature of 30°C (the data comes from our long-term temperature measurements). On the deposit, the temperature is about 56°C. The cable deposits are protected from the environment and from undesirable manipulation with a fireproof blanket and sheet steel. This protection does not restrict the air flow and supply of oxygen from the environment.

At the Dukovany NPP, the situation was more complicated. The deposits were realised after 20 years of NPP operation. Therefore, old safety-related cables had been taken at first from the store and subjected to an accelerated ageing to bring them to the same condition as naturally aged cables in the containment. To perform the simulation as realistically and reliably as possible, the accelerated ageing conditions were very mild. The cables were aged simultaneously at elevated temperature (70°C) and radiation (2.5 Gy/h).

Cable deposits in Canada NPP

In Canada, since 1995, Ontario Power Generation (OPG) has been using the cable deposits at OPG Darlington Nuclear Generating Station. Various cable insulations are deposited for a period of 40 years on specific areas so that they can be exposed to the actual environment inside the NPP. Every 10 years, these cables are removed for testing to verify whether their insulation materials are deteriorating. The last test verification has shown that there was no degradation of the insulation materials. This cable deposit approach is considered part of Darlington Station's cable condition monitoring programme.

Factors to be considered for cable deposits:

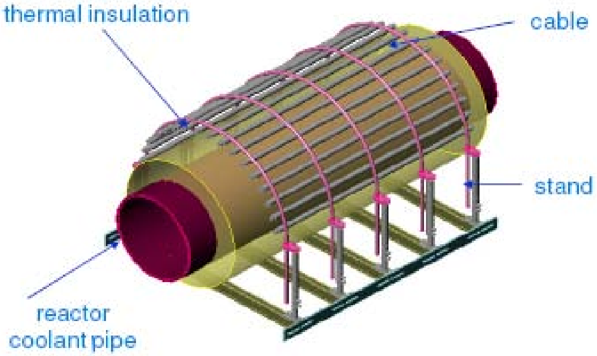
- anticipated plant life including potential life extension;
- critical locations where the cables are likely to be subjected to extreme environmental effects;
- samples to represent all the cables that have potential age-sensitive locations;
- hot spot locations for radiation and temperature;
- adequate length of cable with due consideration for plant life, continuing qualification and research that could involve destructive examination;
- documentation that supports traceability to production, historic test records, stock no. (sample labels, etc.);
- appropriate protection of cable ends;
- impeccable sample;
- convenience for retrieving the required samples;
- self-heating that raises the temperature should be considered while evaluating the cable insulation degradation (since deposited cables are not energised with an electrical load).

Cable deposits in Ukraine NPP

Sampling and preparation of reference specimens for deposit

- Reference specimens are sampled from the cables which are available at the plant storage or from set of spare parts
- Reference specimens are sampled from the operable cables the storage conditions of which meet the requirements of specifications.
- The length of reference specimens shall be 2.00 ± 0.05 m.
- Reference specimens should have no deformation or mechanical damages on the jacket.
- The ends of reference specimens shall be properly sealed preventing the paper oil insulated cables from ingress of moisture and leakage of dipping compound or hydrophobic compound.
- Ends of cables with a rubber, PVC or PE cover shall be sealed with the caps, wrapped with the tapes of similar to the cable jacket material, or sealed using the "hot" method.
- "Hot" sealing of the cable ends could be performed using melted polyethylene providing hermetic sealing of the cable.
- Ends of cables in a metal jacket should be sealed with a "hot" method (soldered) or using metal caps.
- Each reference-specimen shall be labelled.
- Labels shall be made of flame-retardant material standing operation conditions of the equipment, specified by regulations for reference specimen locations.
- Metal labels shall be corrosion-proof.

Figure 3.3.2.3-1: The stand for deposit of reference specimen in the reactor containment at a Ukraine NPP



Deposit of reference specimens in the plant

Reference specimens are to be deposited in "hot spots" identified in the list of deposited reference specimens of the plant. The method of reference specimen deposit in cable installations and the premises of the plant meets the requirements of cable normative documents and can be similar to the method of allocation of operational cables. The minimum bending radius of cable reference specimens shall be provided according to the cable's technical specifications.

In a case of reference specimens, deposit in the containment is expected to place them on a separate tray or stand.

- During reference specimen deposit it is necessary to consider:
  - periodic sampling of specimens;
  - identical distribution of influence of the damaging effects on the surface of the specimens.

- The data of the reference specimen deposit shall be recorded in the register of cable reference specimens of the plant.
- The reference specimens were deposited in the 1000 WWER plant in the containment, turbine room and reactor containment in accordance with this programme.

Cable deposits in Spain NPP

Vandellos II NPP (PWR) is performing an ongoing qualification programme covering the safety-related cables installed in the plant. The cables have been qualified according to IEEE-383-74 requirements. The programme activities started in 1988 and include the following groups of 1E plant cables:

- medium voltage power cables (6 600 V);
- low voltage power and control cables (600/1 000 V);
- instrumentation, coaxial and triaxial cables (300/1 500 V);
- instrumentation cables (300 V);
- thermocouple cables (300 V);
- mineral insulation cables were excluded from the programme;
- cables from different manufacturers are included in each group.

Programme activities

The programme is based on the comparison of mechanical and electrical test results, applied on two sets of identical cable samples: *i)* naturally aged cables located in selected plant areas (cable deposit) and aged during plant operation; *ii)* new cables existing in plant stores and later artificially aged in laboratory. Both sets of cable samples are aged to the same time intervals: 5, 10, 15, 20…40 years. The programme includes the following basic activities:

- *Cable sample selection*. Thirteen different cable types (existing in plant stores) were selected, as a representative sample of the cable population in the plant. For each sample, two pieces 34 meters long were selected and insulation-resistance base line tests values were obtained.
- *Cable deposit area selection*. The following plant areas were selected to install cable samples for naturally ageing:
  - Area 1. Reactor containment, near AFW turbine-pump, with the following design normal operation conditions (dose values corresponding to 40 years): 49°C, 10 Mrad, 95% humidity.
  - Area 2. Reactor containment, near containment liner: 49°C, 0.3 Mrad, 95% humidity.
  - Area 3. Reactor containment, near a steam generator: 49°C, 10 Mrad, 95% humidity.
  - Area 4. Auxiliary building: 40°C, 0.1 Mrad, 80% humidity.
- *Environmental monitoring of deposited cable samples*. Cable samples in each area were installed (de-energised) in cable trays, together with the existing operational cables.
  - Temperature monitoring in each area is performed using specific thermocouples, (one per tray) located near cable trays. Measured values are sent outside reactor containment and registered weekly.
  - Radiation monitoring is performed using specific “high-temperature designed TLDs” (one per area). Dose accumulated values are measured and registered yearly.
  - Humidity is measured continuously inside containment, using the existing design plant devices. Values inside the auxiliary building were taken weekly using portable devices.
  - Sequentially, groups of all the naturally aged cable samples are extracted every five-year period.

- *Simulated cable-ageing process*. Cable samples selected for artificial ageing simulation (previously stored under controlled environment), were subjected to radiation and thermal ageing test, simulating expected degradation in the programme time intervals (5, 10, 15,…, 40 years).
- *Post-ageing tests*. The following tests are applied to the different groups of artificially and naturally aged cable samples (5, 10, 15,…, 40 years):
  - Electrical tests: insulation resistance and polarisation index.
  - Mechanical tests: elongation at break and tensile strength.
  - High potential tests: each cable sample was subjected to the post-LOCA test required by IEEE Std. 383-74.
- *Programme results evaluation*. For each cable sample, the results of the above tests are registered and plotted in graphics, to be compared for the two groups of cables (naturally and artificially aged). Evaluation of the programme results is performed every five years by the plant staff. Presently (2010) the programme is still ongoing but already there are some evaluation data (plant property), referring to the 5, 10, 15 and 20 years of cable-ageing groups.

As a preliminary conclusion, the plant has informed CSN that after 20 years, most of the naturally aged cables seem to be less degraded than the corresponding artificially aged cables.

3.3.2.4 Cable repairing

Cable repairing is not a common maintenance practice in the Spanish NPP; when damaged cables are found the usual practice is to replace them as soon as possible. Before replacement, the damaged cables could be temporarily repaired, using insulating tapes or Raychem heat-shrink sleeves.

In the 90s, a BWR NPP, following cable manufacturer recommendations, has used special BOPIR (auto-vulcanised butyl rubber) and NABIB adhesive tapes, for repairing its Pirelli EPR Radiflam cables.

Plant repairing procedures recommend applying two layers of BOPIR tape over cable conductor, followed by two layers of NABIB tape with 50% overlap. NABIB and BOPIR tapes are certified up to 100°C thermal endurance but are not qualified for accident conditions.

Generally these repairs are done to cable jackets. Raychem, a manufacturer of cable termination and splicing products provides detailed instructions on conducting a cable repair: *www.tycothermal.com/assets/NorthAmerica/English/Documents/Installation_Operation_and_Maintenance_Manuals/Products/287/H57286.pdf*. These are special materials that shrink on to cable surface and form a protective jacket. Certain models of this heat shrunk materials are qualified for safety application and in certain cases for harsh environments. On multi-conductor cables, the repair is recommended only when the individual conductors indicate 20 M ohms when tested with a 2 500 V DC megger.

3.3.3 Data for the cable failure events

Most of the cable failures recorded in the database are from the United States experience. The US experience was further analysed to gather insights from the reported events. On 7 February 2007, the NRC issued Generic Letter (GL) 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients". Specifically, the NRC issued this GL to obtain from its operators information in two areas:

1) A history of inaccessible or underground power cable failures for all cables within the scope of Title 10, Section 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (the Maintenance Rule), of the Code of Federal Regulations (10 CFR 50.65) and for all voltage levels with the following characteristics indicated for each cable failure: type, manufacturer, date of failure, type of service, voltage class, years of service, and the root causes for the failure.
2) A description of the inspection, testing and monitoring programmes to detect the degradation of inaccessible or underground power cables that support emergency diesel generators, off-site power, emergency service water, service water, component cooling water and other systems that are within the scope of 10 CFR 50.65.

The operators provided information for 269 cable failures, of which 125 occurred while the cable was in service. Operators reported 114 instances of cables failing to meet testing and inspection acceptance criteria (referred to as "testing failures" in this report). These include cables that did not fail while in service and cables that failed visual inspections. Operators reported 30 cable failures that the NRC could not conclusively infer from the responses as being in-service or testing failures.

Of the 269 cable failures reported by the operators, 209 cables were normally energised, and 15 cables were normally de-energised. The operators did not supply this information for 45 cables. Of the failures, 93% (209 of 224) occurred on normally energised cables. Of the 15 normally de-energised cable failures, 3 failures were identified while the cables were in service and 10 were identified during testing; operators did not indicate this information for 2 of the normally de-energised cable failures. Of the 15 normally de-energised cable failures, the root causes included water/moisture, manufacturing defects, installation error and digs. The operators listed water or moisture as a contributing or root cause for 4 of the 15 normally de-energised cables, and 3 of these 4 failures were of 480 V (service)/ 600 V (rated) cables.

It was noted that the majority of failures occurred on normally energised cables. Testing identified 66.7% of the normally de-energised cable failures, and water/moisture was a contributing factor to the failures of some de-energised cables.

In Japan, four cable failure events have occurred since 1970, three of which were caused by initial flaws or defects given in cable installation processes.

Figure 3.3.3-1: Cables that failed while in service

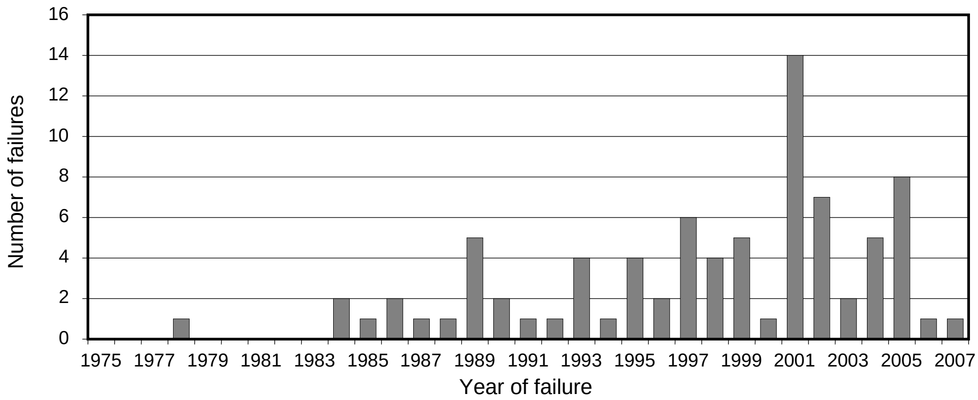


Figure 3.3.3-2: Cables that failed to meet testing and inspection acceptance criteria

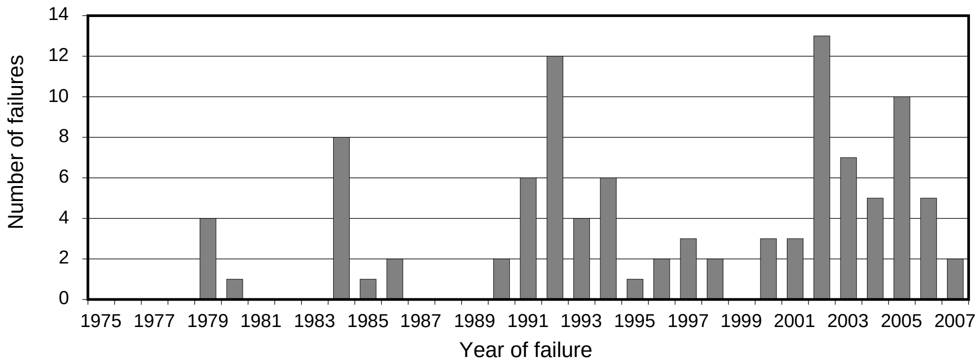


Figure 3.3.3-3: Number of failures per ten-year service intervals

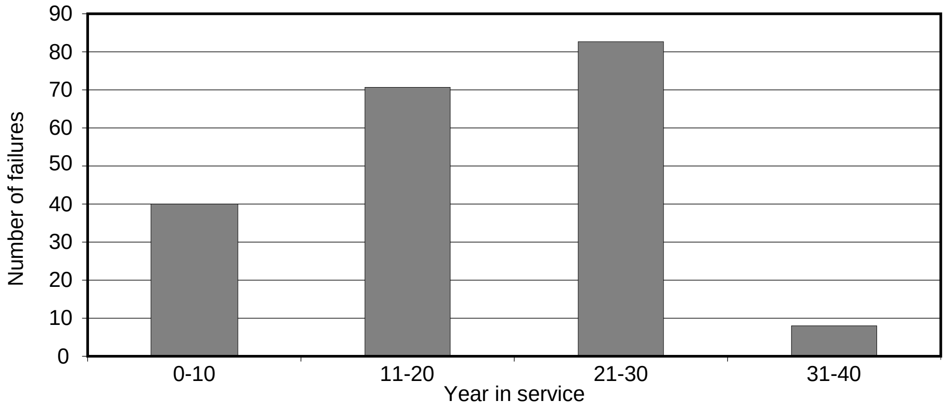


Figure 3.3.3-4: Failures per insulation type

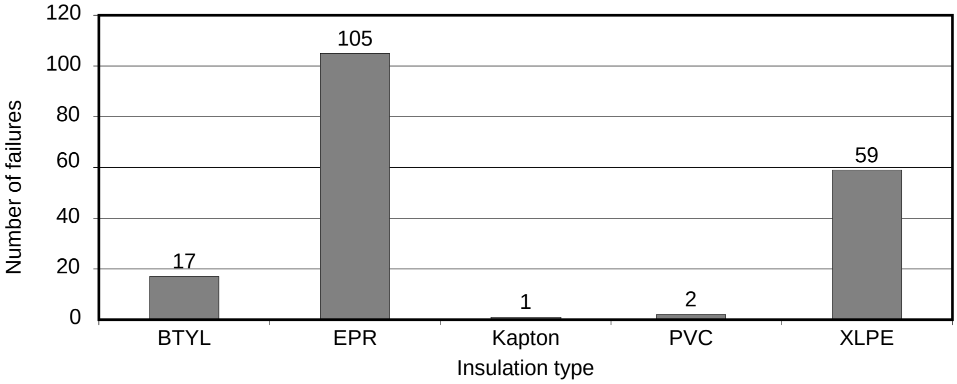
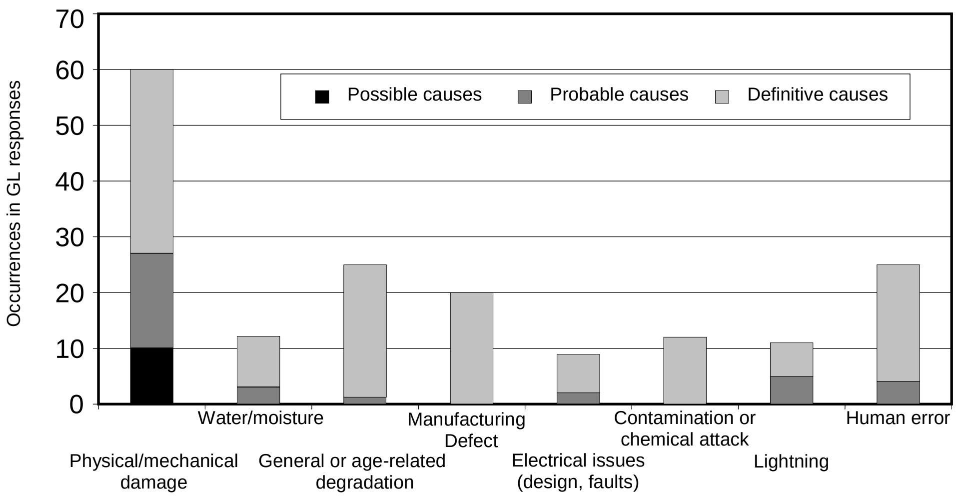


Figure 3.3.3-5: Causes and causal factors for all cable failures



In Ukraine, the analysis of cables defects was performed using the data received for the period 1984-2008. For the AASHv cables three types of failures were identified:

- The failure occurred as the result of contact of the cut part of the cable and the sharp corner of metalware. Failure is caused by an error in the cable laying connecting cells KRU-6kV.
- The failure was the result of the power cable damage during earth excavations.
- The failure was the result of short circuit of the cable conductor and the jacket of the power cable at a bending point of the cable. Two failures of PvSG cable were registered.

The failure occurred as the result of the cable mechanical damage (breaks, dents, cuts) during the construction. For cable KVVG three failures were registered:

- The failure was the consequence of rubbing against control cable laid between control panels.
- The failure was the result of mechanical damage of a cable during repair of fire-resistant partitions.
- The failure was the result of short circuit which occurred because of rubbing against an operating valve.

The reasons of failures:

- errors in cables installation;
- errors at cable routes and channels installation;
- degradation of cable insulation caused by the wrong cable application;
- mechanical damage of a cable during work not relevant to cable operation;
- change of cables' electric characteristics (decrease in insulation resistance) caused by insulation ageing.

A few other countries reported cable failures that had root causes as listed in Table 3.3.3-1..

Table 3.3.3-1: Root causes of cable failures

Cable type	Root cause	Number of events
Power	Mechanical ageing. The cables were connected to a running motor. Due to vibrations coming from the motor, the cables wore out on edges.	20
Power	Poor quality cable. The manufacturer did not use the same material as was used in the qualification programme. Moreover, not all technological conditions were maintained during the manufacturing. After one year at 60°C the cable sheath cracked.	1 batch (some km)
Signal	Poor quality of the whole batch. Some cables from this batch were installed. No failure, but extremely low insulation resistance measured soon after installation. All cables were immediately changed.	1 batch (some km)
Power	During the cable installation, the workers pulled a cable with very high force. The cable crossed another power cable. At the crossing, the cable insulation wore out. Two years later, this position had a short circuit.	1
Power	Very low insulation resistance was measured. It was a cable in the middle of a cable bundle and under a fire protective layer. After cutting out, it appeared that the cable was extremely brittle.	5
Power, signal	During the visual inspection some cables with cracked sheath were found. Conditions: 20 years in operation, 60°C, 0.001 Gy/h, PVC cable. Cracking due to loss of plasticiser. No loss of functionality, but such cables were replaced as needed following LOCA.	10

Other topics

Removal of heat insulation

In various locations of the plant, a variety of heat insulation materials (reflective metallic insulation, asbestos fibre, wool wrap, etc.) are used either to shield the plant environment from high temperature or to preserve the heat within the pipes and components for thermal efficiency. These insulation materials are often removed for maintenance activities. Since most of the heat insulation activities are performed by non-electrical staff, the vulnerabilities of cable insulation to high heat is often not considered. As a result, the removal of heat insulation leads to overheating of cables in the immediate proximity.

Based on the level of excess heat exposure received the cable may have a delayed failure or a prompt failure when a service is required. The problems are often identified following a cable failure. Therefore, it is important to evaluate the impact of heat insulation removal on electrical components located in the proximity.

Cables wrapped within heat insulation

The piping systems associated with steam systems and other hot process systems have motor-operated valves and other electrical components connected to the piping systems. During construction or maintenance, the heat insulation wrap work is the last step of the work completion. Therefore, the insulation workers mistakenly wrapped the hot pipe with the cable inside. This type of wrapping results in exposing the cable to an unacceptable level of heat produced by the piping system. The cable undergoes accelerated ageing, leading to premature failure. These problems are often identified only with system failures and could lead to system unavailability or plant trip. Therefore, it is prudent to have oversight on heat insulation work to prevent deleterious effects on electrical cables.

3.3.4 Cable environmental qualification code data

Cable ageing programmes in all countries require managing the effects of ageing so that the cables and connections can perform their intended function during the period of operation. Qualification tests include the verification that wire systems can maintain electrical and mechanical properties also during and after a design basis event (DBE), which is normally a large break LOCA accident.

These requirements have evolved as operating experience accumulated and the ageing process was better understood. Initially, qualification was based on the “high industrial quality” of electrical components. For plants constructed after 1971, a more formal approach was adopted: qualification was judged on the basis of Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1971, “IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations” [23].

Although IEEE Std. 323-1971 did not address ageing or service life determination issues, the standard did call for a systematic programme of analysis, testing and quality assurance. It specified that qualification might be achieved through type testing, analysis, operating experience or a combination of these methods. In the United States, for plants with NRC construction permits dated 1 July 1974 or later, the NRC endorsed the 1974 version of IEEE Std. 323, IEEE Std. 323-1974, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations” [23], and IEEE 383-1974, which specifically addressed cable qualification [24].

In 1983, the NRC established federal regulations with specific environmental qualification (EQ) requirements in 10 CFR 50.49 [25] for nuclear stations. In addition to the general design requirements, 10 CFR Part 50 Appendix A, Criterion 4, 10 CFR 50.49 specifically requires that an EQ programme be established to demonstrate that electrical safety components located in harsh plant environments [that is, areas of the plant that could be subject to the harsh environmental effects as a consequence to a loss of coolant accident (LOCA), high-energy line breaks (HELB) or post-LOCA radiation] are qualified to perform their safety function in those harsh environments during and following the event while the component is in service. 10 CFR 50.49 requires that consideration must be given to all significant type of degradation which can have an effect on the functional capability of the equipment and this includes ageing mechanisms to be addressed as part of environmental qualification.

In Europe, the international standards normally used, in addition to IEEE-323 is IEC-780 [26] and IEC-60780 [27]. For the effects of radiation on insulation materials, common guidelines are IEC-544 [28] and IEC-1244. France and Germany have developed their own national standards, *i.e.* RCC-E and RCC-M (France) and KTA 3706 (Germany) [29]. In Sweden, the Swedish utilities have developed their own guidelines, KBE EP-154 [30]. These standards include procedures using accelerated ageing to assess cable capability to survive the environmental conditions existing in a power plant during and after a DBE.

In most areas within a nuclear power plant, the actual ambient temperature, pressure and humidity is lower than that used for the cable qualification, providing a high degree of confidence during the expected 40 years of plant life. In many countries it is then assumed that the qualification process (especially for I&C cables) provides a reasonable assurance that the cables will perform the intended safety-related function during the qualified life. In consequence, ageing assessment requirements can be summarised as follows:

- Accuracy and limitations of accelerated ageing mechanisms utilised for the qualification. The issue is to evaluate to what extent the ageing mechanisms can be simulated by models and tests. It is normally assumed that thermal ageing in polymer materials can be modelled using the Arrhenius equation (explained in the next paragraph) and that the thermal effect due to long exposure to working environment conditions can be achieved in a much shorter time frame for testing (accelerated ageing).
- Life extension. It is normally assumed that the difference between the real operating environment and the original qualification environment (worse) can justify the requalification for the extended period, usually 20 years, by reanalysis [22]. However, some LOCA tests performed at Wyle Labs [31] showed that most cables aged 60 years (accelerated ageing) during LOCA tests exhibited high leakage currents and failures. These results suggest that monitoring of environmental conditions and methods to assess cable conditions are needed to improve confidence during the extended plant life. Moreover, the new generation of plants and the digital upgrades to the existing designs are increasingly less tolerant of leakage currents.

The considerations above suggest that some requalification process or update might be needed under specific circumstances. However, plant license renewal is only one of the reasons that could require actions in this direction. Recently [32], a list of cases where an installed component needs to be updated regarding its qualification for long-term effects of ageing conditions was presented. This list can be summarised as follows:

- The environment conditions deviate from those used in the qualification. As specified above, this requires environment condition monitoring procedures during the whole qualified life. An important aspect of this case is the consideration of “hot spots”, defined as points where temperature and/or dose rate are higher than what was assumed in the early qualifications tests.
- Reconsideration of the previously assessed qualified life, due to the use of non-conservative factors such as very high acceleration factors for ageing, depletion of oxygen in the LOCA test chambers in the early hours of the test or lack of adequate consideration of dose rate effects.
- New knowledge in the field in consideration of the long qualification period (40 or 60 years).
- The end of the qualified period is approaching.
- Installed life is longer than previously expected. Reanalysis of an ageing evaluation to extend the qualification life of a component is performed routinely in the context of an EQ programme.

In the US, the licensing basis for environmental qualification is 10 CFR 50.49, “Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants”. A Regulatory Guide (RG) 1.89 “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants” provides practical approaches to complying with the code requirement. The RG endorses IEEE Std. 323, “Qualifying Class 1E Equipment for Nuclear Power Generating Stations” [23] with certain exceptions. Further guidance is provided in the “Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors” or NUREG 588 “Interim Staff Position on Environmental Qualification of Safety-related Electrical Equipment” for the plants that were operating when this guidance was issued. All the new plants are required to fully comply with 10 CFR 50.49.

The IEEE standard provides a recommended procedure for conducting the qualification test. A further caution is given to confirm that the selected procedure is conservative for the particular materials that are under test. Figure 3.3.4-1 is taken from IEEE 323 to indicate the development of a test profile with adequate consideration for margin and extended period for post-accident service period. The 2003 version of IEEE 323 does not have the pressure and temperature curve given below. The standard directs the use of plant-specific profiles with suggested margins for qualification tests.

Figure 3.3.4-2 shows the predicted accident profile for a plant in Korea. The qualification test profile has to envelope the worst case environment with recommended margins.

Tables 3.3.4-1 and 3.3.4-2 show the test profile of accident pressure and temperature for a plant in Germany.

Figure 3.3.4-1: Typical LOCA/HELB temperature and pressure illustrating additional peak transient to account for margin

Note: A later revision to CFR and endorsement of IEEE 323 of 2003 is anticipated in 2010-12

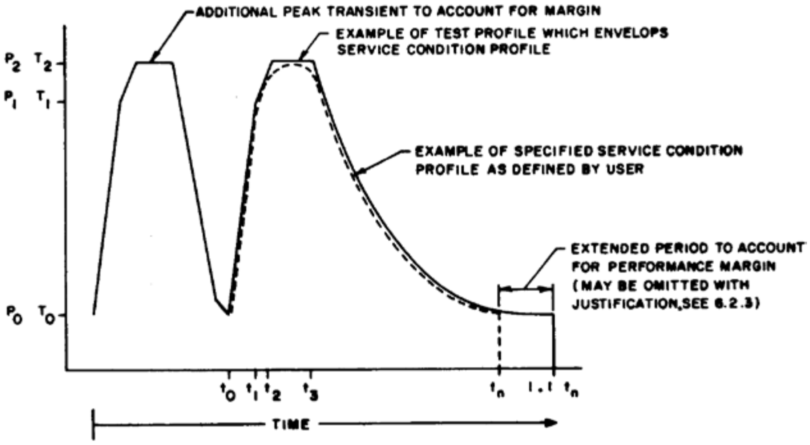


Figure 3.3.4-2: DBE temperature profile of inside containment for a plant in Korea

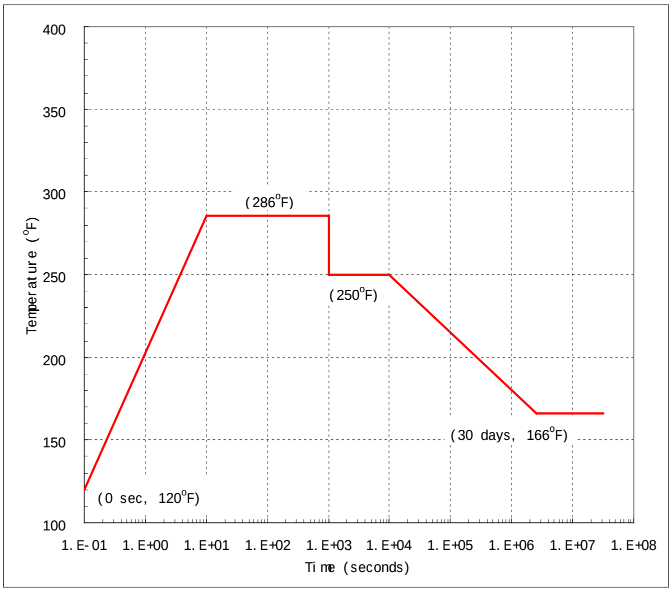


Table 3.3.4-1: Pressure profiles in Germany

Test profile location	Peak	Peak increase time	Peak duration
Test profile of devices in the valve compartment of a 1 300 MW (electric) type PWR light water reactor	1 750-2 000 mbar	~1 s	~4 s
Test profile of devices in the annulus (within the region of the break compartment) of a 1 300 MW (electric) type PWR light water reactor	100-120 mbar	~100 s	~200 s
Test profile of devices in the annulus (outside the region of the break compartment) of a 1 300 MW (electric) type PWR light water reactor	100-120 mbar	~100 s	~100 s

Table 3.3.4-2: Temperature profiles in Germany

Test profile location	Peak	Peak increase time	Peak duration	Short-term decrease	Short-term decrease duration	Long-term decrease	Long-term decrease duration
Test profile of devices in the containment vessel of a 1 300 MW (electric) type PWR light water reactor	155-165°C	~10 s	~1 h	Up to 95-105°C	~1.3 h	Up to 50-60°C	~21.7 h
Test profile of devices in the valve compartment of a 1 300 MW (electric) type PWR light water reactor	150-160°C	~40 s	~1 h	Up to 95-100°C	~1.8 h	Stable	~2.2 h
Test profile of devices in the annulus (within the region of the break compartment) of a 1 300 MW (electric) type PWR light water reactor	95-100°C	~50 s	~1 h	–	–	Up to 25-35°C	~3 h
Test profile of devices in the annulus (outside the region of the break compartment) of a 1 300 MW (electric) type PWR light water reactor	50-60°C	~100 s	~1 h	–	–	Up to 25-35°C	~3 h

Input from several countries has introduced several experiences that challenge the early approaches to qualifying electrical cables using accelerated ageing. The areas of new insight include: the influence of oxygen content in LOCA chamber, variation in activation energy, the influence of dose rate in ageing and the procedure for accelerated ageing.

New insights are reported from Japan on activation energy and the procedure for accelerated ageing in the environmental qualification tests.

Activation energy used for accelerated ageing

When the activation energy of each insulation material is calculated from thermal ageing test data acquired at this time, most of the data is of the order of 100 kJ/mol (20 or more kcal/mol), and some is also 40 or more kJ/mol (a little more than 10 kcal/mol). In addition, a large portion of thermal ageing test data used for calculation of activation energy is data ranges from 100 to 120°C [33].

When the sampling data in actual operating plants and the data acquired in this project are compared, the activation energy in actual operating plants' temperature region (from 50 to 60°C) follows a lower trend other than the value calculated from the thermal ageing test data acquired in this project.

Also, although it differs from elongation at break, some literature states that around 60 kJ/mol (approximately 15 kcal/mol) it can be assumed to be appropriate for the activation energy in a region of uniform oxidisation acquired with chemo-luminescence analysis.

Based on the above, the principles of calculation and application for the activation energy used for future assessment as shown below were determined to be appropriate.

- Applicable region of activation energy calculated by thermal ageing tests is limited up to the minimum temperature in thermal ageing tests. However, when the calculated activation energy is less than 62.8 kJ/mol (15 kcal/mol) the value can be applied up to the operating temperature region of actual operating plants.
- Activation energy in the region between the minimum temperature in thermal ageing tests and the temperature of actual operating plants are evaluated from the investigation results of ageing in actual operating plants (sampling inspection) and thermal ageing characteristics at the minimum temperature in thermal ageing tests.
- When activation energy cannot be evaluated from the investigation results of ageing in actual operating plants, 62.8 kJ/mol (15 kcal/mol) is used as a tentative value for the activation energy in the region between the minimum temperature in thermal ageing tests and the temperature of actual operating plants.

Recommended procedure for accelerated ageing

Accelerated ageing equivalent to ageing during normal operation is defined as simultaneous ageing based on the results of the "Assessment of Cable Ageing for Nuclear Power Plant (ACA)" project [33]. The temperature and dose rate that enable confirmation of degradation progress uniformly into the full thickness of the insulation material should be selected for simultaneous ageing.

In the ACA project, progress of degradation into the inside of the insulator was confirmed with a dose rate of 100 Gy/h and a temperature of 120°C (175°C for silicone rubber) for cross-linked polyethylene, flame-retardant cross-linked polyethylene, ethylene-propylene rubber, flame-retardant ethylene-propylene rubber, silicone rubber and special heat-resistant polyvinyl chloride.

Qualification procedures in Ukraine for the harsh environment of the control cables in the containment of the WWER 1000 plant are performed according to The "Procedure of the Control Cables Qualification for Harsh Environment in Containment of the VVER 1000 Plant". The scope and the sequence of cable qualification procedures include the following steps:

1) sampling of cables;
2) testing in the cable initial condition;
3) accelerated thermal ageing;
4) radiation ageing;
5) testing after the accelerated thermal and radiating ageing;
6) testing in simulated LOCA and post-LOCA conditions;
7) qualification testing;
8) technical report on the qualification of the control cables for harsh environment in containment of the VVER 1000 plant.

Steps 3) and 4) should be evaluated to confirm that sequential ageing is conservative and acceleration factors utilised are in fact suitable for homogeneous changes.

In the above sequence, Step 5) conducted following the ageing can be a reference point for periodically monitoring the component to verify if the degradation has gone beyond the condition that was adequate to endure a LOCA environment. If several tests can be conducted to identify the degradation threshold at which the component can endure the LOCA environment, this condition can be monitored in the plant as the maximum allowable degradation. If the operating environment remained milder than the test assumptions, or if the accelerated ageing was more demanding, the component service life can be extended by condition monitoring until the degradation threshold to withstand a LOCA environment is reached.

3.3.5 Plant and cable environmental condition

Another finding from the international database is that the variations in plant environment have influenced the life of the cables. An increasing temperature decreases the service life and a decrease in temperature could be of benefit to increasing the life of the cable in relation to the assumptions in the qualification phase of the cable.

There is yet another benefit to be gained by conducting periodic inspections in plant areas. A visual inspection of the cable in accessible areas could reveal surface discoloration, flaking, browning, etc., that would be indicative of localised overheating that degrades insulation. A periodic inspection at potential hot spot locations could avoid several in-service failures.

Environmental monitoring in the Czech Republic

Environmental monitoring

It is a predominant consideration to have environmental monitoring in the NPP containment to obtain a detailed knowledge of temperature, dose rate, humidity, etc. Inside and outside the NPP containment, more than 200 locations have been monitored. The temperature, humidity and dose rate have been monitored for many years at the same locations.

Temperature and humidity monitoring

A proven approach for local monitoring is the use of self-powered temperature, temperature/humidity data loggers (black boxes). The recorders are equipped with internal or external (up to 15 m long) sensors. The recording interval is usually set on 2, 4 and/or 6 hours, respectively. During the reactor shutdown, the results from the data loggers can be evaluated.

Radiation monitoring

The dose rate distribution is measured with the aid of Alanine dosimeters. The selected location has been fitted with hermetically sealed dosimeters from Bruker Company, Germany, all in the form of small cylinders with a diameter and height of 5 mm. The dosimeters are evaluated in the laboratory using electron spin resonance (ESR) spectroscopy. Since the irradiation temperature influences the ESR response of Alanine dosimeters, the true absorbed dose is calculated from the measured dose by using the temperature coefficient. The dose rate is calculated from the known irradiation time.

Fluency of neutrons

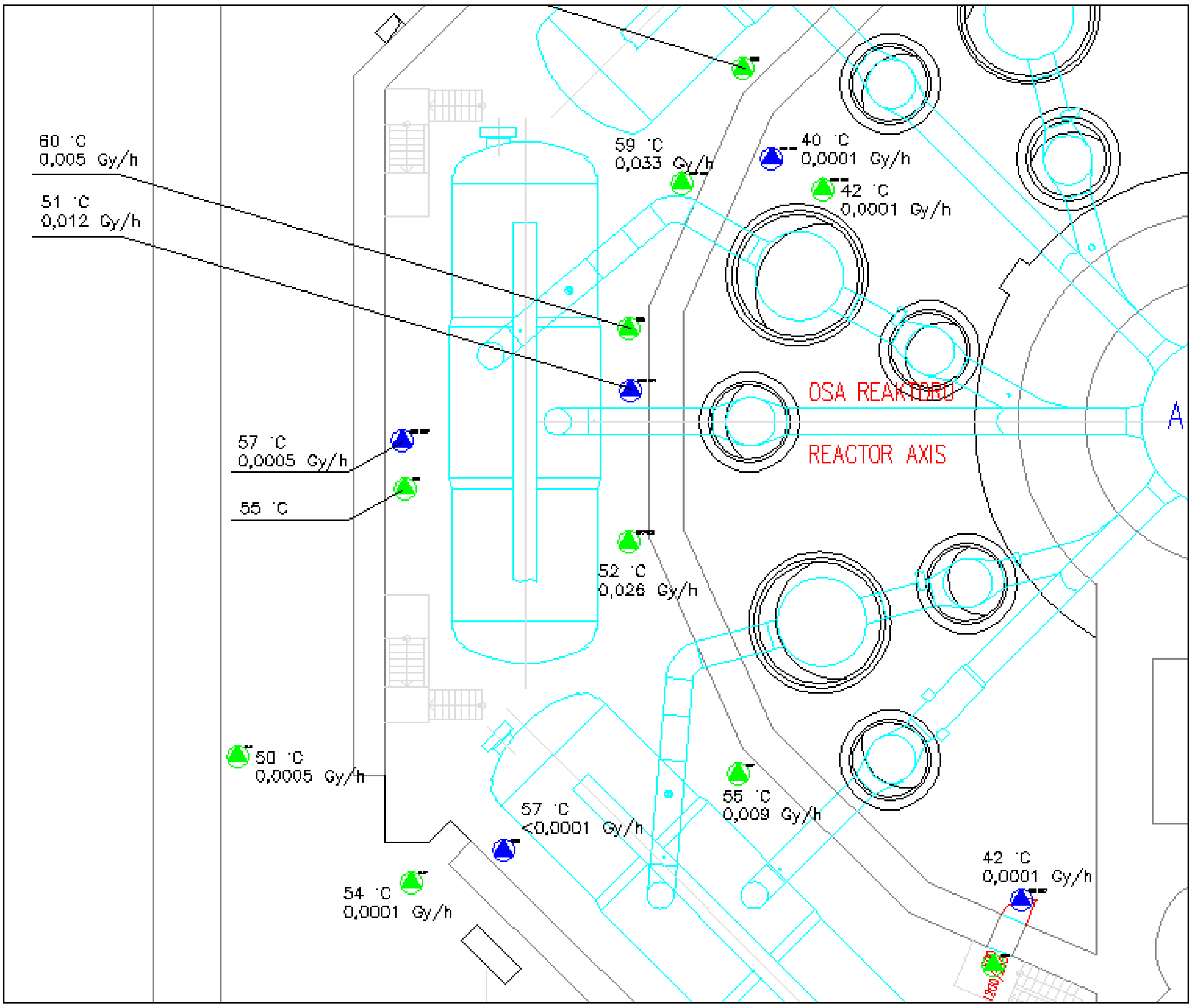
The fluency of thermal and fast neutrons is measured using cobalt and nickel foils, respectively. The evaluation of these dosimeters consists in measuring their induced activity after neutron irradiation.

At some locations the temperature exceeded the projected calculations. The dose rates at all locations were markedly lower than what was projected.

Table 3.3.5-1: Summary of plant environmental condition for Czech NPP

Czech plant	Temperature			Dose rate		
	Tool	No. of points	Temperature	Tool	No. of points	Dose rate
PWR 440	Data logger	180 per unit	75°C at one position with cables, otherwise 60°C and less	Alanine	200 per unit	0.4 Gy/h, but in most cases under 0.1 Gy/h
PWR 1000	Data logger	200 per unit	51 °C	Alanine	230 per unit	0.17 Gy/h, usually less

Figure 3.3.5-1: Measured points for a PWR 440



Plant environmental condition in Japan

The electric power utilities are required to investigate the environmental conditions for cables installed inside reactor containment vessels in all operating commercial nuclear power plants, based on an administrative document issued by the Japan Nuclear and Industrial Safety Agency (NISA) in October 2007.

In the investigation programme, the temperature and radiation dose in cable-installed areas must be measured once during periodic inspection or during refuelling outage. Devices for the temperature measurement must include a data logger and an Alanine dosimeter for the radiation measurement.

Investigations of 35 nuclear power plants have been completed as of January 2010 and their results are shown in Table 3.3.5-2. On the whole, temperature and radiation dose rate investigated showed lower values than the design value except in some nuclear power plants. Especially, the values of radiation dose rate were considerably low. Based on the results, the qualified life of cables is being revaluated and cable replacement is planned in some nuclear power plants where the environment was more severe.

Table 3.3.5-2: Summary of plant environmental condition for Japanese NPP

Investigated plant	Temperature			Dose rate		
	Measuring tool of temperature	Number of measurement points	Maximum temperature	Measuring tool of dose rate	Number of measurement points	Maximum dose rate
BWR, ABWR	Data loggers and labels for thermometry	9 (data loggers) 110 (labels)	59.5°C (data logger) 65°C (label)	Alanine dosimeters	110	0.147 Gy/h
BWR, BWR-4		9 (data loggers) 59 (labels)	70.8°C (data logger) 75°C (label)		56	0.179 Gy/h
BWR, BWR-5		96 (data loggers) 117 (labels)	59.8°C (data logger)		112	0.234 Gy/h
BWR, BWR-3	Data loggers	74	65.7°C	Alanine dosimeters	75	0.093 Gy/h
BWR, BWR-4		58	74.8°C		73	0.581 Gy/h
BWR, BWR-4		74	67.0°C		82	0.301 Gy/h
BWR, BWR-4		65	75.7°C		73	1.016 Gy/h
BWR, BWR-5	Data loggers and labels for thermometry	34 (data loggers) 38 (labels)	64.5°C (data logger)	Alanine dosimeters	38	0.546 Gy/h
BWR, BWR-5		84 (data loggers) 56 (labels)	74.2°C (data logger)		95	0.265 Gy/h
BWR, BWR-5		80 (data loggers) 86 (labels)	76.2°C (data logger)		83	0.305 Gy/h
BWR, BWR-5		89 (data loggers) 94 (labels)	84.8°C (data logger)		93	0.139 Gy/h
BWR, BWR-5		72 (data loggers) 88 (labels)	66.8°C (data logger)		87	0.192 Gy/h
BWR, BWR-5		135 (data loggers) 148 (labels)	60.4°C (data logger)		142	0.156 Gy/h
BWR, BWR-3		11 (data loggers) 78 (labels)	54.0°C (data logger) 70°C (label)		68	0.563 Gy/h
BWR, BWR-5	Labels for thermometry	90	70°C	Alanine dosimeters	88	0.152 Gy/h
BWR, BWR-2		100	80°C		95	0.484 Gy/h
PWR, WE-2	Data loggers and labels for thermometry	22 (data loggers) 16 (labels)	60.8°C (data logger)	Alanine dosimeters	30	0.288 Gy/h
PWR, WE-2		28 (data loggers) 15 (labels)	73.2°C (data logger)		30	0.288 Gy/h
PWR, WE-2		25 (data loggers) 21 (labels)	74.6°C (data logger)		21	0.190 Gy/h
PWR, WE-3		17 (data loggers) 21 (labels)	50.0°C (data logger)		15	0.085 Gy/h
PWR, 3-loop		35 (data loggers) 40 (labels)	51.8°C (data logger)		18	0.146 Gy/h
PWR, 3-loop		33 (data loggers) 34 (labels)	61.4°C (data logger)		23	0.099 Gy/h
PWR, 3-loop		34 (data loggers) 34 (labels)	44.5°C (data logger)		25	0.385 Gy/h
PWR, 3-loop		38 (data loggers) 39 (labels)	47.5°C (data logger)		15	0.090 Gy/h
PWR, WE-4		34 (data loggers) 10 (labels)	66.1°C (data logger)		15	0.144 Gy/h
PWR, WE-4		29 (data loggers) 9 (labels)	64.0°C (data logger)		20	0.129 Gy/h
PWR, WE-4		35 (data loggers) 8 (labels)	39.4°C (data logger)		30	0.078 Gy/h
PWR, WE-2		37 (data loggers) 25 (labels)	79.8°C (data logger)		25	0.307 Gy/h
PWR, WE-2		37 (data loggers) 36 (labels)	75.8°C (data logger)		25	0.337 Gy/h
PWR, 2-loop		25 (data loggers) 39 (labels)	53.3°C (data logger)		26	0.240 Gy/h
PWR, 2-loop		20 (data loggers) 37 (labels)	67.8°C (data logger)		29	0.286 Gy/h
PWR, 4-loop		21 (data loggers) 29 (labels)	45.8°C (data logger)		30	0.241 Gy/h
PWR, 4-loop		27 (data loggers) 33 (labels)	43.6°C (data logger)		33	0.240 Gy/h
PWR, 3-loop		20 (data loggers) 36 (labels)	43.5°C (data logger)		29	0.354 Gy/h
PWR, 3-loop		17 (data loggers) 27 (labels)	47.0°C (data logger)		22	0.310 Gy/h

Plant and cable environmental monitoring in Spain

According to 10 CFR.49 the Spanish NNP are required to develop a plant-specific “Equipment Environmental Qualification Programme” on the “electrical equipment important to safety”.

Environmental monitoring programmes are not specifically required by the applicable qualification rules and standards, but most of the plants, according to CSN recommendations, have decided to implement these programmes, to complement their equipment qualification programmes.

Each monitoring programme is plant-specific but, as a general rule, the plants have historically monitored the operating temperature and radiation values in selected areas where qualified equipment are located. Measuring periods and devices vary from plant to plant. Usually, instrumentation of temperature and radiation plant control systems have been used, but some plants have also installed additional temperature sensors (thermocouples, RTD, thermometers) and radiation monitors (TLD), or “Westinghouse Time Life Monitors” in selected areas.

The “Time Life Monitors” supplied by Westinghouse, are self-integrated devices, qualified according to IEEE-323-83 and IEEE-344-87 that, as customers choose, can make different environmental measurements:

- high range gamma dose;
- low range gamma dose;
- beta particle dose;
- neutron flux exposure;
- peak temperature on a period;
- integrated temperature in a period (based on Arrhenius methodology).

In order to make the above measurements, the monitors are supplied with different environmental “block sensing materials”. At the end of the measuring period, the blocks are removed from the monitor and sent for analysis to Westinghouse facilities.

The following summarises the main aspects of the “Environmental Monitoring Programme on 1E Equipment” developed by some Spanish plants:

PWR Westinghouse design (2 units)

The temperature monitoring programme is based on 60 Westinghouse Time Life Monitors (33 installed in Unit 1, 23 in Unit 2 and 4 monitors in common plant areas). The monitors are located outside reactor containment in the “safeguards building” (including the steam tunnel), auxiliary building, electric building and fuel building. Inside the electric building, five specific cable rooms are monitored. The installed monitors only include peak and integrated temperature sensors. The measures have been made during three operation cycles (18 months/cycle). Results of the programme show some peak temperatures higher than design values (10°C in steam tunnel) but the integrated temperature values always remained below design values.

BWR (Mark 1 containment)

The operating temperature inside the drywell has historically been measured using the “drywell temperature control system”. This system includes 14 thermocouples that send continuous temperature values to data loggers located in the control room. Thermocouples are located in different drywell elevations:

- Elevation 514: 4 thermocouples.
- Elevation 520: 2 thermocouples.
- Elevation 525: 4 thermocouples.
- Elevation 536: 4 thermocouples.

In 2007, according to the recent "Life Extension Application" of this plant, CSN required an analysis of the existent temperature data in the different drywell elevations corresponding to a 20-year period (1986-2007). For each elevation, the following medium-temperature values during the period were estimated, considering conservative margins:

- Elevation 514: 42°C.
- Elevation 520: 57°C.
- Elevation 525: 65°C.
- Elevation 536: 66°C.

The maximum cable operating temperature was established at 65°C. According to the above temperature values, a "Time Limited Ageing Analysis" was performed for all qualified equipment and cables existing inside the drywell.

BWR-6 (Mark 3 containment)

The programme is based on temperature and radiation values, measured in selected areas inside the reactor building, auxiliary building (including steam tunnel), turbine building and diesel generator building over a 13-year period (from 1991 to 2003).

- *Temperature monitoring.* The programme is based on the following types of measures:
  - Local temperature monitoring on selected 1E equipments. Accessible and non-accessible equipment, located inside the reactor building, steam tunnel and auxiliary building.
    - Temperature sensors located on the equipment or as close as possible.
    - Maximum, minimum and average temperature values were measured and registered yearly for each piece of equipment. Maximum temperature deviations from the design temperature were also calculated.
  - Temperature monitoring on relays located on cabinets and panels located in different buildings.
    - Measures were taken avoiding heat dissipation coming from relays. Maximum, minimum and average temperature values were registered yearly for each cabinet/panel.
    - Deviations from design temperature in each cabinet and area were calculated.
  - Area temperature monitoring.
    - Ambient temperature monitoring on selected areas inside the reactor building, steam tunnel, auxiliary building, turbine building and diesel generator building.
    - Different temperature sensors (RTD and thermocouples) were used in each area (22 sensors inside the reactor building and 9 sensors inside the steam tunnel).
    - Maximum/minimum/average temperature values for each sensor were registered yearly.

  For each equipment, cabinet and area, the measured yearly values over the 13-year period were plotted on graphs.

- *Radiation monitoring.*
  - Operating gamma doses were measured yearly on equipments in several locations:
    - 1 measurement point inside the reactor building (internal MSIV);
    - 2 measurement points inside the steam tunnel (external MSIV);
    - 2 measurement points inside the turbine building) (1 limit switch, 1 switch).
  - Accumulated doses over 8 years were calculated for each location. These values were extrapolated to a 40-year value and later compared with design values in each location.
  - Final results show that design estimated doses in each location were greater than the expected 40-year actual doses.

- *Significant temperature monitoring results*. Measured local temperatures on some accessible equipment inside the reactor building show some years where peak values are over the design temperature (57°C), but the average values remain below the design temperature during the entire measured period.

  Measured local temperatures on SRV and MSIV inside and outside reactor containment show several maximum values over design temperature. Some temperature peak values reached 100°C (full range of the sensor).

  Maximum measured temperatures on relays (cabinets and panels) remained below design temperature during the 13-year period. The average temperatures, for all locations, remained at least 10°C below the design temperature.

  Area temperature monitoring showed the following results:

  – Maximum values inside the reactor building always remained over the design temperature (57°C) reaching peak values of 72°C, but the average temperature remained below the design temperature.

  – Maximum temperature values inside the steam tunnel always remained over the design temperature (49°C). The peak values near the main steam lines reached 90°C some years. The average temperature in areas near the main steam lines always remained about 10°C over the design temperature. The average temperature in other areas inside the steam tunnel remained below the design temperature.

  – Inside the auxiliary building and turbine building the maximum temperature values in some areas remained over the design temperature (40°C and 49°C respectively), but the average temperature remains below the design temperature.

- *Conclusions*. After evaluation of the monitoring programme results, the plant reached the following conclusions.

  – The high local temperature peak values (about 100°C) measured on MSIV and SRV located inside the reactor building and steam tunnel were not considered representative of actual average temperature in the area. Plant analysis concluded that these peak values could have been a consequence of “steam leaks” coming from valves located near the equipment temperature sensors, and also of “thermal radiation beams” coming from defects in thermal insulation of “high-energy lines” and equipments.

  – The area surrounding the main steam lines were considered as a “hot spot” with an average temperature of 60°C (more than 10°C over design temperature).

  – The plant considered that postulated design temperature values were acceptable inside the reactor building and the rest of the steam tunnel.

- *Plant corrective actions*. As a consequence of the above conclusions, the plant implemented (in 2005) the following corrective actions.

  – Performance of walkdowns in the steam tunnel, using thermography to detect hot spots.

  – Replacement or repair of damaged thermal insulation on high-energy lines and equipments inside the steam tunnel and reactor building.

  – Isolation of thermal hot spots detected by thermography.

  – Performance of visual inspection on cables and equipments affected by steam leaks inside the reactor building and steam tunnel.

  – Reactor building and steam tunnel HVAC systems units reviewed and improved.

  – Replacement of local temperature sensors on MSIV and SRV with new devices (temperature range up to 250°C).

  – Performance of an analysis of qualified life on cables and equipment located inside the steam tunnel.

Cable hot spot identification

In accordance with regulatory requirements, in 2003 Spanish plants began to develop ageing management programmes (AMP) on structures, systems and components important to safety, including electrical cables (see database, Part 8). General criteria for cable critical environmental and service conditions identification are described in Technical Procedure ES13/IT-03-0903, developed in the UNESA/CSN Research Co-ordinated Project PCI-13 (see database field 8007).

Based on the above document, preliminary basic recommended activities for cable hot spots and critical service conditions identification are the following:

- meetings with plant technical staff, for obtaining actual plant environmental and service information;
- review of historical equipment and cable maintenance reports;
- review of plant equipment location drawings.

As a result of the above activities, it is possible to identify areas inside the plant with potential hot spot indicators like the following:

- presence of liquids or fluids on floors or on plant structural elements, particularly chemical solutions;
- degraded or "burned" paintings on walls and structures;
- high-temperature water leaks;
- presence of condensation;
- high-temperature areas;
- high vibrations on equipments and structures;
- presence of oil or fuel steams;
- signals of chemical spray or steam impingement on cables or equipments;
- equipment subjected to frequent maintenance activities;
- cable trays and conduits presenting sharp surfaces or loosened supports;
- degraded fire barriers;
- degraded high-temperature equipment or piping isolation;
- areas where previous cable failures have been detected.

Based on the above indicators specific potential hot spot plant areas can be selected for walk downs and visual cable inspection.

According to CSN recommendations, some Spanish plants have performed specific plant inspections and walk downs based on the ES13/IT-03-0903 Technical Procedure criteria for identifying cable hot spots.

- *BWR Mark 1 containment*
  - 2003: Visual inspection on qualified cables located in drywell (10 cables).
  - 2005: Visual and tactile inspection on qualified cables located in drywell and steam tunnel.
  - 2007-2009: Cable ageing management activities related to the plant "Extended Operation Permit Application":
    - Plant walk downs for cable hot spot identification.
    - Inspection on qualified and non-qualified representative cable sample (233 cables), located on drywell, steam tunnel, reactor building, turbine building and other specific hot spot areas, previously identified.

- *PWR Westinghouse design, 2 units*

  The plant began to implement its cable ageing management programme activities in 2006.

  Walk downs on cable routings, using thermography, were performed in selected areas inside safeguards building (steam tunnel), electric building and outside areas (building terraces). Some temperature values, higher than those measured by the existing area temperature sensors, were detected. During walk downs visual and tactile inspection, and indenter modulus technique, were also applied to accessible cables.

- *PWR KWU design*

  The plant began its cable ageing management activities in 2007. A walk down using visual inspection and thermography was performed for cable hot spot identification on the following plant areas:

  – valve chamber (steam tunnel);

  – electric building (cable rooms);

  – galleries ZX, ZA;

  – turbine building;

  – containment annulus.

Walk down results did not identify any cable hot spot. Complementing the walk down, a detailed inspection on a representative sample of cables in the above areas was performed, using visual and tactile inspection, and the indenter modulus technique.

Service life time assessment based on environmental monitoring in the Czech Republic

Accelerated testing methods used during the qualification may not be enough to simulate operational cable ageing. Moreover, the cables in containment age at different rates based on their location. For example, the temperature and the dose rate may vary within the containment from 25 to 70°C and/or from 0 to 0.5 Gy/h, respectively. As the cables age in different locations at different rates, the service life can vary. Hence, it is necessary to know the temperature and the dose (dose rate) to ensure the qualification of polymeric materials. Representative cable samples are aged at different temperatures, at different dose rates and/or are simultaneously irradiated at elevated temperature. Functional properties were measured.

The results are transformed into the mathematical equations that convert the rate of degradation at different conditions (temperature, dose rate) with ageing time. The equations exist for all 1E cable types, sheath as well as insulation and this approach helps to extend cable life based on actual temperature and radiation doses.

The rate of degradation in individual locations in a NPP depends on the surrounding temperature and on the dose rate. To assess the service life time in individual locations more easily, a special software system has been developed. The input parameters are the following:

- equations describing the rate of degradation at specific conditions;
- information on the temperature and dose rate in individual locations in NPP;
- information on cables: type, trace (room, location, etc.), qualification.

The output is:

- the cable service life time in individual locations;
- much more additional information like temperature distribution and dose rate along the cable trace, connected device, etc.

Canadian experience on service life

It was observed that some specific insulation materials failed after radiation doses much lower than predicted. Research work confirmed that there are a few insulation materials which are very sensitive to low dose rate. IAEA-TECDOC-1188 states (Vol. I, pg. 28) that very low dose rates (20-30 Gy/h) have been found necessary for testing certain materials particularly sensitive to the dose rate [20]. It is important to note that the use of such insulation materials in radiation environments where the effects of dose rate at high dose rate exposure may not be an indication of ageing experience in a low dose environment for material irradiated to a common dose. Therefore, materials "qualified" for a 40-year service life may fail sooner than expected [34].

3.3.6 Mitigation of cable-installed environment

Another approach to ageing management, endorsed by three or four countries, is to manage the environment that cables are subjected to in their service. The normal operating environment for the cable could be provided with additional shielding from heat, radiation and other chemical effects to reduce the impact of operational ageing and extend cable life.

3.3.7 Cable replacement

In 1991, one US licensee experienced a cable failure at 4 160 V level. The root cause was identified to be "treeing" from a damp environment. This cable failure resulted in the shutdown of both units. The license replaced all the 4 160 V and 13 kV underground cables of the same vintage to prevent any future problems in this area.

In order to fulfil the objective of preventing/mitigating operational failures, it is essential to know the conditions that would prompt a cable replacement. An early knowledge of potential cable failure scenario could help to stage the replacement cable and schedule a convenient plant outage time for replacement. This assessment is more challenging for cables that are required to endure an accident environment and remain functional.

These conditions are identified by condition monitoring against the previously known threshold on cable degradations or using new information gained from operating experience.

In the Czech NPP, old safety-related cables required to function following a LOCA were re-qualified after 25 years of operation. The real operating conditions were used for qualification. Some cables that did not pass the LOCA test were replaced.

In order to extent the NPP life for another 30 years, the potential of an increasing number of cable failures also has to considered for non-safety-related cables. The cables are protected by fire protection foam as shown in Figure 3.3.7-1. Therefore, it is impossible to change a single cable that failed. Therefore, new cables are added to the current cable trays and the old ones are not removed. In many cases there is no room left for further addition of cables. Therefore, it has been decided to change the majority of the NPP cables. A massive cable replacement is supposed to be performed within about 10 years (2020).

Figure 3.3.7-1: Fire protection layer for cables

In one Spanish BWR plant, built in the 60s (operation permit in 1971), many originally installed cables supplied by General Electric were not qualified for functioning under accident conditions. In 1983, the plant was required by CSN to develop a Safety Evaluation Plan (SEP). According to the SEP, an Equipment Environmental Qualification Programme, following NRC IE-Bulletin 79-01-B (DOR Guidelines), was required for cables and equipments. As a consequence of the SEP, in 1985 the plant replaced power and control cables in safety-related equipment located inside the drywell (RV, SRV, MSIV, MOV, thermocouples, etc.). About 140 cables, with lengths varying from 10 to 150 meters, were replaced.

New cables from Spanish manufacturers Pirelli (EPDM, EPR and Afumex insulation) and Saenger (EPR insulation) were qualified according to IEEE-383-74 and installed inside the drywell. Afterwards, a CSN inspection (2002) on the plant EQ programme, found that many power and control cables located inside the steam tunnel and reactor building were not properly qualified (lack of documentation) according to 10 CFR 50.49 requirements. Following a specific assessment of the issue, 34 cables inside the steam tunnel and 122 cables inside the reactor and turbine building were replaced in 2003 and 2005, respectively. New Pirelli (EPR and Afumex insulation) and Saenger (EPR insulation) cables, qualified according to IEEE-383-74 were installed.

In another Spanish BWR plant, original cables installed in the plant were qualified according to IEEE-383-74 requirements. As result of maintenance activities on MOV, in 1994 and 1995 plant personnel found degradation (burned and cracked jackets) in some Pirelli EPR/CSPE, control and power cables feeding MOV (Limitorque actuators). Subsequent cable failures occurred in 1996, 2000 and 2002 resulted in three safety-related MOV failures.

A further plant evaluation in 2005 determined that temperature hot spots inside the reactor building and steam tunnel could have been the cause of the cable degradation. As a consequence, the plant decided to replace all Pirelli EPR cables in 39 MOV located in high-temperature areas. A plant design modification was applied to a total of 176 actuators (1E, non-1E) including those located in high-temperature areas. Intermediate connection boxes were installed in cable routings, and control and power cable portions, from box to actuator, were replaced with new Nucletef (ETFE insulation) power and control cables. Quick connectors were also installed in some actuators. Cable replacement began during a 1997 outage and will be probably be finished in 2012 outage.

In Canada, the PVC cables were replaced inside containment at Ontario Power Generation (OPG) nuclear power plants for several reasons: The qualification of PVC for use in high-radiation fields was a concern because OPG's original qualification tests for PVC did not perform well for the bounding environmental profiles, and a concern with regard to dose rate effects related to PVC cables. In addition, some original manufacturers were out of business or had no detailed information needed to demonstrate the environmental qualification of their cables. In light of these issues, a decision was made to remove PVC insulated cables from qualified applications inside containment. At that time, OPG owned most of the nuclear power plants in Canada.

Currently, Gentilly-2 (owned by Hydro-Québec), another Canadian nuclear power plant, is also planning to replace PVC cables inside containment based on similar reasons to those encountered by OPG. For instance, even though a few of the PVC cables successfully passed one of the simulated design basis events (DBE), such as main stream line break (MSLB), they did not pass the loss of coolant accident (LOCA). Thus, these cables will not perform their safety-related functions when exposed to harsh environments resulting from a LOCA.

As this nuclear power plant is scheduled for refurbishment, a decision was made to replace the PVC insulated cables inside containment by the end of refurbishment in 2012.

3.3.8 Regulatory information for cables

Advanced EQ test guide

In Japan, an "Assessment of Cable Ageing for Nuclear Power Plants (ACA)" was performed from FY2002 to FY2008 and the advanced guide for cable EQ test "Guide for Cable Environmental Qualification Test for Nuclear Power Plants" was formulated based on the results of the project. The guide is a part of the final report JNES-SS-0903 [33]. In the guide, simultaneous thermal and radiation ageing is adopted as a pre-ageing method instead of the conventional method of sequential ageing. In addition, activation energy used to set the accelerated ageing condition is determined considering the condition in operating plants. The several techniques for setting accelerated condition are also provided in the guide.

The outline of the advanced guide is as follows:

- A cable specimen with 3 m length is used. The cable with the thinnest insulation is selected as cable specimen for the cables with the identical specifications in insulation.

  Even if the insulation specification is identical, a separate test is carried out for the cables supplied by different manufacturers.

- Accelerated ageing equivalent to ageing during normal operation is conducted as simultaneous ageing. The temperature and dose rate that enable confirmation of degradation progress into the full depth of the inside of the insulation are selected for simultaneous ageing.

  In the ACA project, degradation progress on inside of the insulator was confirmed with dose rate 100 Gy/h and temperature 120°C (175°C for silicone rubber) for cross-linked polyethylene, flame-retardant cross-linked polyethylene, ethylene-propylene rubber, flame-retardant ethylene-propylene rubber, silicone rubber and special heat-resistant polyvinyl chloride.

- Activation energy used for accelerated ageing is as follows:
  - Applicable region of activation energy calculated by thermal ageing tests is limited up to the minimum temperature in thermal ageing tests. However, when the calculated activation energy is less than 62.8 kJ/mol (15 kcal/mol) the value can be applied up to the operating temperature region of actual operating plants.
  - Activation energy in the region between the minimum temperature in thermal ageing tests and the temperature of actual operating plants are evaluated from the investigation results of ageing in actual operating plants (sampling inspection) and thermal ageing characteristics at the minimum temperature in thermal ageing tests.
  - When activation energy cannot be evaluated from the investigation results of ageing in actual operating plants, 62.8 kJ/mol (15 kcal/mol) is used as a tentative value for the activation energy in the region between the minimum temperature in thermal ageing tests and the temperature of actual operating plants.

- The condition of accelerated simultaneous ageing equivalent to ageing during normal operation is established by the techniques of “superposition of time dependent data” and “superposition of dose to equivalent damage data” or the technique of “same acceleration factor” (which has been verified to be equivalent to the former) based on the condition of actual operating plants. However, it is recommended not to use the technique of “superposition of time dependent data” to establish the condition for silicone rubber insulation cable.

  Further, if the cable has an insulation made of cross-linked polyethylene, flame-retardant cross-linked polyethylene, ethylene-propylene rubber, flame-retardant ethylene-propylene rubber, silicone rubber and special heat-resistant polyvinyl chloride, the condition of accelerated simultaneous ageing may be established by a simplified method, that of using the technique of “superposition of dose to equivalent damage data”.

- The withstand-voltage test in Japanese Industrial Standards (JIS) is used as a method for determination of integrity after LOCA test.

Future trend of cable environmental qualification test in Japan

NISA issued the regulatory instruction on 30 October 2007 based on the ACA project interim report (issued in December 2006). To meet this instruction, electric utilities are conducting environmental conditions and performance evaluations of safety-related cables installed in the reactor containment. The performance evaluations are conducted with LOCA tests, using cable samples from actual plant applications. Tensile tests are also conducted using actual cables from the plant.

The results of these investigation and evaluations will be used to confirm the appropriateness of applying the advanced guide for cable environmental qualification tests. Therefore, the advanced guide is planned to be used in order to evaluate the validity of ageing management in the near future.

Requalification plan for stored cables

In Argentina, Atucha II NPP is delayed in construction. Like Atucha I, it is a pressure vessel pressurised heavy water reactor (PV-PHWR), but was planned to have a higher power. Its construction started in 1980 under a contract with Siemens AG, but it was never finished and it remained around 80% completed.

Non-safety related cables were purchased during '90. Some of them were installed and the rest were stored in good condition. Cable ageing management programme (CAMP) for stored cables, and EQ maintenance programme (EQMP) are planned for the end of 2009. The plan includes the following phases:

- *Screening of cables involved in systems and components relevant for the long-term operation*. The scope involves the non-safety-related cables but still relevant for the power generation of the NPP and the long-term operation.
- *Condition assessment of stored cable based on original standards*. Several studies are being carried out. The studies include material characterisation, mechanical properties, thermal stability, etc. The methodology and acceptance criteria are based on the original standards of qualification for cables (DIN-VDE).
- *Determination of ageing related mechanism and its effect on current conditions*. This phase includes: environmental condition assessment, determination of ageing mechanism based on material and environmental condition (based on design parameters). Finally, the determination of the acceptance criteria of each cable will be developed and the chosen property to be evaluated is the elongation at break.
- *Data collection of baseline for ageing management programme (including activation energy)*. Development of master curves of elongation at break at different temperature and correlation of elongation at break with non-destructive assessment of properties.
- *Condition monitoring and in-service inspection procedures development and their implementation*.
- *EQ maintenance programme*:
  - *Qualified life determination and initial qualification assessment*. The proposed qualified life is 30 years. Based on design data establish the environmental condition in order to carry out accelerated ageing and testing.
  - *Sampling location determination*. The location selected should be representative of the cable environment condition and accessible to carry out the periodic qualification maintenance test. The set of qualification reference location and measurements for maintenance are still under discussion and the frequency is now proposed to be 10 years.
  - *Sample testing procedures elaboration and implementation*. The scope of the quality maintenance test is still under discussion.

Canadian environmental qualification programme

The regulatory information below is related to the Canadian Nuclear Safety Commission (CNSC) license conditions concerning the environmental qualification programme. This also includes components such as cables.

License condition

"The licensee shall have an environmental qualification programme in accordance with the requirements of CSA Standard N290.13: Environmental Qualification of Equipment for CANDU Nuclear Power Plants."

Preamble

The purpose of this license condition is to ensure that all required systems, equipment, components, protective barriers, and structures in a nuclear facility are qualified to perform their safety functions if exposed to harsh environmental conditions resulting from credited design basis events (DBE) and that this capability is preserved for the life of the plant.

Compliance verification criteria

As part of the CNSC regulatory compliance activities, CNSC staff shall verify compliance of the environmental qualification (EQ) programme with the requirements of CSA Standard N290.13 and the following acceptance criteria:

- The licensee shall have a documented environmental qualification programme in place. The environmental qualification programme shall:
  - identify managerial roles and responsibilities;
  - be consistent with the safety report, design basis documentation and abnormal accident procedures;
  - list processes and procedures in place to identify equipment that require environmental qualification;
  - identify methods used to establish environmental qualification of equipment;
  - list procedural controls in place to preserve environmental qualification of equipment for the life of the plant.
- The licensee shall provide evidence that the processes and procedures related to the environmental qualification programme meet the requirements of recognised industrial standards.
- The licensee shall have available, at the plant, all documentation related to environmental qualification.

  This documentation includes (but it is not limited to) policies, procedures, environmental qualification list, environmental qualification assessments, type test reports, maintenance, procurement and storage of replacement parts for inspection by CNSC staff.
- The licensee shall have in place a monitoring programme to measure degradation and failures of qualified equipment.

  The monitoring programme shall contain elements of condition monitoring and environmental monitoring. Condition monitoring shall measure variables that indicate the physical state of the equipment, including cables, and assess its ability to perform its intended function following the period of observation. Environmental monitoring shall measure environmental stressors, such as temperature, radiation and operational cycling during normal operating conditions. This monitoring programme will allow periodical re-evaluations of equipments' qualified life.
- The licensee shall provide evidence that all EQ work, including processes for establishing and preserving EQ, meet the requirements of the licensee's quality assurance programme.
- The licensee shall provide evidence that the personnel in the EQ work have received training on EQ principles and related procedures. This shall include both in-house and contract personnel.

Programme of nuclear power plant cable ageing management in Ukraine – PM-T.0.08.121-07

The "Programme of NPP (Nuclear Power Plant) Cable Ageing Management" specifies the requirements for the development, implementation and context of methodical, organisational and technical activity related to NPP cable ageing management. This programme is developed taking into account the recommendations of IAEA experts.

The purpose of ageing management is to assure cable operational safety, reliability and cost-effectiveness under normal operating conditions over the specified cable lifetime and under design accident conditions.

The main purposes of this document are:

- to specify the requirements for NPP cable ageing management;
- to specify the main principles of cable ageing management, the order of works and requirements for the procedures of ageing management.

The programme is applied to control cables (instrumentation, signal, block cables) with plastic and rubber insulation, low- and medium-voltage cables with plastic and paper-oil insulation operating in NPP. The programme requirements are obligatory for the ageing management of safety-related cables, which are classified as 2nd and 3rd safety classes according to the "General Provisions for NPP Safety" NP 306.2.141-2008.

The programme requirements can be used for the ageing management of the cables for systems of normal operation (4th safety class). The programme specifies:

- procedures of ageing management activity;
- programme requirements for NPP ageing management for specific plants;
- programme requirements for cable technical condition inspection;
- requirements and principles of cable lists compiled for the inspection of cable technical condition in order to define the possibility of lifetime extension;
- procedures of cable environment condition monitoring and identification of hot spots;
- methods of cable technical condition assessment;
- requirements to cable technical condition inspection for the purpose of the cable lifetime extension;
- the requirements to cable database and content of cable data for the integration into general cable database;
- registration of cable ageing management works;
- content of scientific and technical support activity and maintenance;
- quality assurance.

The requirements of the programme are obligatory for the implementation by the NPP staff and management and the management of the national nuclear energy-generating company Energoatom and specialised organisations involved in cable ageing management activity. The work on cable ageing management is performed by the NPP personnel involved in the specialised organisation when needed.

3.3.9 Description of condition monitoring technique

There are 18 cable condition monitoring techniques recorded in the database and their classifications of application are shown in Table 3.3.9-1. It has varying levels of capabilities and weaknesses. Each technique has its own specialties that are better suited for certain kinds of cables.

US records

In January 2010 USNRC published NUREG/CR-7000 "Essential Elements of an Electric Cable Condition Monitoring Programme". Table 3.3.9-1 of this publication provides an exhaustive listing of all cable monitoring techniques, applicable cable categories and materials, applicable stressors, ageing mechanisms detected, advantages and limitations.

Applicability tests of non-destructive degradation diagnostic technologies for cables

In Japan, several condition monitoring methods for cables installed in nuclear power plants was tested their applicability. The outlines of the methods tested are as follows [33].

- *Indenter*. The indenter is to measure the indenter modulus as the non-destructive degradation diagnostic parameter. The Indenter modulus is calculated as the slope of load versus penetration curve and its unit is Newton per millimetres (N/mm). Load at the start of measurement and load at the end of measurement are specified for each material. An improved type of indenter modulus device, which was originally manufactured by AEA Technology Inc. in (UK) and improved jointly by Institute of Nuclear Safety System, Inc. (INSS, Japan) and AEA Technology Inc., was used to measure indenter modulus.

Table 3.3.9-1: Condition monitoring techniques

Position	Destructive/ non-destructive	In service/ disconnect	On site/ laboratory	Condition monitoring techniques
Local	Destructive	Disconnect	Laboratory	Elongation at break
				Thermo-gravimetry
	Non-destructive	In service	On site	Indenter
			Laboratory	Oxidation induction time
				Oxidation induction temp.
				Density analysis on polymer samples
		Disconnect	On site	Indenter
				Optical diagnosis
				Surface hardness measurement
				Ultrasonic diagnosis
				Microwave diagnosis
Full length	Non-destructive	In service	On site	Time domain reflector
		Disconnect	On site	Current leak rate testing
				Insulation resistance
				Loss factor
				Potential decay
				Tan delta
				Broadband impedance spectroscopy

- *Ultrasonic degradation diagnostic method*. The ultrasonic diagnostic method is the technology developed by Mitsubishi Cable Industries, Ltd. as the degradation diagnostic technology for insulation or jackets of the cables and to measure ultrasonic propagation velocity for axial direction of materials as the non-destructive degradation diagnostic parameter. The diagnostic equipment for low-voltage cables in NPP has been jointly developed by Mitsubishi Heavy Industries, Ltd. and Mitsubishi Cable Industries, Ltd. The ultrasonic probes of this equipment moves automatically by the sequential control to measure accuracy in a short time and to decrease exposure dose of the operators.
- *Optical diagnostic method*. The optical diagnostic method has been developed by Hitachi, Ltd. as a degradation diagnostic method for polymer materials used for the cables. The basic concept of this method is the colour-change of polymer materials due to ageing are exposed to two optical beams with different wavelengths and their absorption is quantitatively evaluated. In the test, the difference of absorption for 405 nm and 1 310 nm were measured as a diagnosing parameter.
- *Surface hardness measuring method*. The surface hardness measuring method is to press a cylindrical needle against the surface of the object with a spring, and its compressed depth is measured as a surface hardness in relative value. In the test, the device with a simplified micro-hardness meter developed by Mitsubishi Cable Industries, Ltd. for rubber materials of cable insulation was used.

Results of investigation

Correlation between the diagnostic data of the non-destructive methods listed above and elongation at break were obtained by round robin tests. A summary of the results is shown in Table 3.3.9-2. Based on the result, the applicability of the methods to cables installed in operating NPP was evaluated as follows:

- The indenter is applicable to the EPR family, SIR, SHPVC and certain kinds of XLPE.
- Though the ultrasonic degradation diagnostic technology is applicable to the EPR family, an improvement is needed as the result showed a somewhat large dispersion.
- Though the optical diagnostic method is applicable to white colour insulation of the XLPE and EPR families, an improvement is needed as the result showed a somewhat large dispersion.
- Though the surface hardness measuring method is applicable to the EPR family and SIR, an improvement is needed as the result showed a somewhat large dispersion.

Table 3.3.9-2: Correlations of the diagnostic data

	The indenter	Ultrasonic diagnostic method	Optical diagnostic method	Surface hardness measuring method
XLPE #1	R = 0.94*1	No definite correlation	No definite correlation	–
XLPE #2	R = 0.85	No definite correlation	No definite correlation	–
FR-XLPE #1	No definite correlation	No definite correlation	R = 0.90	–
FR-XLPE #2	R^2 = 0.94*2	R = 0.71	No definite correlation	–
EPR	R^2 = 0.93	R = 0.75	No definite correlation	R = 0.72
FR-EPR #1	R^2 = 0.75	R = 0.73	R = 0.63	R = 0.76
FR-EPR #2	R^2 = 0.75	R = 0.61	R = 0.70	R = 0.58
FR-EPR #3	R^2 = 0.96	R = 0.62	R = 0.80	R = 0.73
SIR #1	R^2 = 0.99	–	–	R = 0.72
SIR #2	R^2 = 0.97	–	–	R = 0.59
SIR #3	R^2 = 0.98	–	–	R = 0.86
SHPVC #1	R^2 = 0.87	–	–	–
SHPVC #2	R^2 = 0.91	–	–	–

*1 R: Coefficient of correlation for linear regression.

*2 R^2: Coefficient of determination for nonlinear regression.

Line impedance resonance analysis (LIRA)

The current techniques to evaluate ageing properties of electric cables include electric properties tests [21,22]. While known to be difficult, advancements in detection systems and computerised data analysis techniques may allow ultimate use of electrical testing to predict future behaviour and residual life of cables.

Line impedance resonance analysis (LIRA) was developed by the Institute for Energy Technology (IFE), Halden, Norway and a new company, Wirescan AS, was founded in 2005 to further develop and market the product. In 2005, IFE contacted EPRI concerning the use of LIRA. By mutual agreement, EPRI and IFE chose to explore the use of LIRA for nuclear cable applications. EPRI developed specimens with localised thermal damage that caused hardening of the cable materials but did not cause cracking of the insulation and IFE brought their equipment to EPRI and tested the samples to determine if LIRA could identify and locate the thermal damages to the otherwise undamaged cables. LIRA successfully identified and located the damaged segments of the cables both when cables were tested individually and when connected in series. Given the success of this proof of principles effort, EPRI chose to fund a larger, more formal effort addressing the capabilities of LIRA to identify cuts, gouges and thermal ageing. This article highlights the results of the larger programme that began in 2006 and was completed in April 2007.

Two basic types of thermal and radiation ageing of cables are of concern at nuclear power plants: Bulk ageing where an entire room or space within a plant has elevated temperature or radiation conditions, and local ageing where a localised heat or radiation source such as a pipe is close to a cable tray or conduit. Identification of bulk area conditions is generally easy in that the temperature or radiation levels in an entire room are known with a reasonable level of precision. Localised ageing is somewhat more of a problem in that identifying all possible localised adverse conditions is time consuming and somewhat difficult. In addition, determining whether the localised condition has significantly affected a cable may be difficult if the cable is located inside a conduit or located in a tray that requires scaffolding or other access means to allow the condition to be assessed. Accordingly, a means of assessing the condition of a cable from its terminations by electrical means is desirable.

Until recently, the changes in the electrical characteristics of low-voltage insulations caused by thermal and radiation damage have been too subtle to be detected electrically from the terminations until the damage is so severe that cracking or powdering has taken place. Even at that point, good insulation resistance readings may occur as long as the insulation remains dry and the circuit is not physically disturbed. Accordingly, ageing characterisation methods have concentrated on mechanical and chemical properties. Tests such as the indenter measure modulus (a form of hardness) and many chemical tests are available for laboratory assessment. Depending on the type of insulator and jacket polymers, thermal and radiation ageing causes chemical changes in the material that can be easily measured and compared to trending data from accelerated laboratory ageing. These tests are useful when the surface of the cable is accessible or the insulator can be evaluated at the terminations. They cannot be used for cable contained in a conduit unless the cable is pulled out.

The advent of LIRA has provided a means to detect thermal and radiation damage to cables because it can detect small changes in electrical properties of insulator materials on the order of 1 pf. This detection level allows localised and bulk thermal ageing to be identified well before the material has aged to the point of cracking or powdering. The tests described here indicate that LIRA can identify damage below the point where a cable can no longer pass a LOCA test. The results also indicate that trending of the severity of damage is possible if LIRA tests are performed periodically.

The results indicate that LIRA may be used to assess the condition of cable circuits that traverse multiple rooms with different environmental conditions and circuits with intermediate termination points such as splices and terminal blocks. LIRA will also be a useful troubleshooting tool if there is a concern that significant installation damage has occurred. The tests proved that LIRA can identify cuts and gouges in the insulation system as well as identify thermal or radiation damage. While this research used 30.5 m (100 ft) cables, other assessments performed by IFE have evaluated much longer cables and in one case a 128 km (~78 mile) undersea cable.

LIRA presents a significant addition to the tools available to evaluate cable condition and ageing. Because the system allows the location of the adverse condition to be identified, the position of the hot spot along the length of a cable circuit can be reviewed to determine if a heat or radiation source is present or if another damage type is present in the cable. Conversely, if a heat source is identified adjacent to a conduit system, LIRA may be used to determine if significant damage has occurred adjacent to the heat or radiation source.

Conclusion

The tests at EPRI in November 2006 showed that LIRA could identify localised thermal damage to insulator that had not progressed to the point where the insulation had totally failed. LIRA could locate the damage even though the insulation could still function adequately under normal and accident conditions. These tests indicated that LIRA could identify ageing before the end of the qualified life. The results indicate that LIRA will be useful in assessing the condition of cables located in conduits that are suspected of having been subjected to localised thermal/radiation ageing.

Similarly, LIRA could be used to assess cables in trays that are difficult to access. The May 2006 EPRI tests indicate that LIRA can identify cuts and gouges to one or more conductors of multi-conductor cables. In-plant tests or simulation thereof may be necessary to determine if cuts and gouges to a single conductor can be identified under plant conditions.

An important issue is the assessment of the condition of installed cables that have been exposed for a long time (more than 30 years) to relative high temperature and gamma radiation (the condition of cables inside the reactor containment). Several techniques have been proposed to monitor and identify cables that are close to the end of their qualified life. The purpose of this work was to evaluate three well known techniques and finding the correlation among them. These techniques are the elongation-at-break (EAB), the indenter and the line resonance analysis (LIRA). The first one is the reference technique, for which a limit of 50% absolute was set by several international standards. The indenter is a local technique that has produced good results, mainly with EPR insulated cables. LIRA is an emerging technique based on the evaluation of electrical properties and their trends with the ageing conditions.

The cables tested are low-voltage, EPDM insulated cables produced by the Swedish Lipalon. The reason for this choice is that this type of cable is widely in use in all the Swedish nuclear power plants. Samples 5 m long of three Lipalon cable types were globally aged artificially for different times and their condition was analysed using the three methods mentioned above. This report describes the findings and results if this analysis.

In this report, three techniques for cable global ageing assessment were tested and evaluated. The EAB technique is a destructive, local technique that is often used as a reference for other methods. The indenter is a local, *in situ* mechanical technique that is currently quite often used in NPP. LIRA can be an electrical method, full line, *in situ*. LIRA correlated quite well with EAB and both tend to flatten when the ageing time reaches 40 years. The only cable type that was difficult to assess for all three methods was the medium type in an air environment.

These tests considered only thermal ageing up to 50 years and should be completed by also considering gamma irradiation ageing. This work was performed with the financial support of Nordic Nuclear Safety Research (NKS), Ringhals AB and Forsmark AB. The cable samples used in these tests were supplied by Ringhals AB.

Hot spot identification

The general term "hot spot" normally references any local damage caused on a cable by temperature, radiation or mechanical stress. The local degradation extends from a few cm to several meters and most often are difficult to spot because they develop in inaccessible areas of the primary containment or in underground medium-voltage cables. The main causes of hot spots can be:

- Local adverse environment conditions (temperature and/or radiation). This is usually due to the presence of hot pipes or components nearby. In such conditions, the local part of the cable degrades at a considerably faster rate than the rest of the cable and eventually fails.
- Installation or maintenance errors. This usually results in mechanical damages due to friction or mishandling. Aged and brittle cables are particularly exposed to this kind of damage.
- Manufacturing errors.

The main problem with hot spots is that, although the average cable condition is still good with a residual life expectation of several years, the cable would eventually fail at the hot spot position much earlier than expected, if the hot spot is not properly identified and the cause corrected or the cable replaced/repaired.

Thermal hot spot identification in non-accessible areas can be performed with the following methods:

- Environment monitoring, to identify containment areas where the temperature/radiation is higher than the average (higher than that used in the cable qualification process). In this case, actions to remove the cause of possible hot spots can be implemented.
- Cable deposits. Deposits of sacrificial cable samples can be installed in areas where the environment temperature/radiation is higher than average and periodically tested with local condition monitoring techniques (EAB, indenter, OIT and so on).
- Full length, on-line, electrical condition monitoring. Methods such as LIRA can be used to periodically test cables *in situ* to identify the presence of hot spots along the cable.

Mechanical damages in non-accessible positions (underground or inside containment, in conduits) can only be identified using electrical tests.

Figure 3.3.9-1 shows an example of mechanical damage and its identification and localisation (Figure 3.3.9-2) using the condition monitoring tool LIRA. Figure 3.3.9-3 shows an example of a thermal hot spot and its identification in LIRA. Here, the black trace represents the cable signature before the development of the hot spot degradation, while the red trace shows the effect on the signature of the developing damage (high temperature).

Leakage current technique utilised in Ukraine

The control of leakage current is carried out for DC voltage power cables. Absolute value of leakage current is not an index of defect. For cables with paper-oil insulation at 10 kV the leakage current should be no more than 300 microampere at the maximum test pressure, and asymmetry factor on phases no more than 2.5. The indicator of insulation defect is the slow decrease of leakage current or even its increase, especially at full test voltage.

Partial discharge measurement in Ukraine

Partial discharge is a breakdown in air cavities of insulation or oil films in paper-oil insulated cables. It is accompanied by a spasmodic voltage change in insulation from 0.1 mV to 1 mV, that it is difficult to notice against high operating voltage. At AC voltage partial discharge could be observed each half cycle. Then their frequency will make over 100 Hz. Long-term influence of partial discharges causes insulation damage.

Figure 3.3.9-1: Mechanical damage on a low voltage cable

Figure 3.3.9-2: Identification of a mechanical damage spot (LIRA, EPRI experiment, 2007)

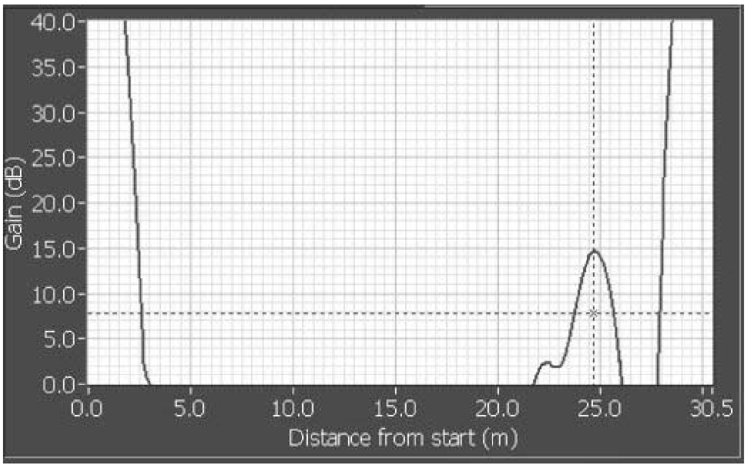
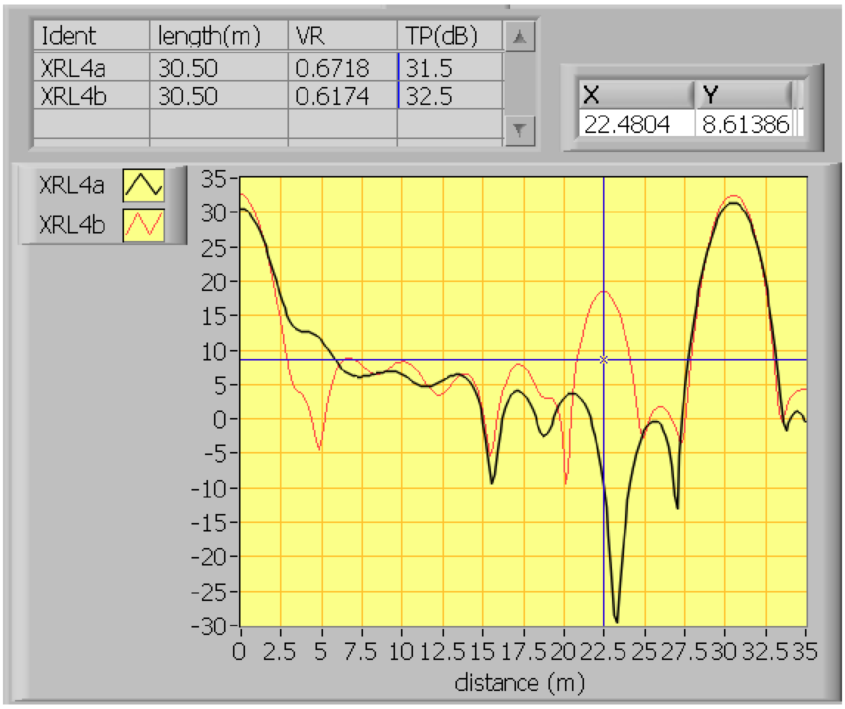


Figure 3.3.9-3: Identification of a thermal hot spot in a low voltage cable (LIRA, EPRI experiment, 2007)



Partial discharge influence leads to occurrence of short current impulses in an external electric chain (lasting less than 1 microsecond) and to charge (q) transfer. This charge is called "seeming" and could be measured. Admissible level of seeming charge at the testing of new paper-oil insulated cables makes q ≤ (10 – 20) • 10-12 coulomb = from 10 picocoulomb to 20 picocoulomb and is defined by comparison of an actual cable resource to the initial level of partial discharges measured in a new product. For cables with PE insulation this level is essentially lower – from 2 to 5 picocoulomb, since polymeric insulation is less resistant to the influence of partial discharges than paper-oil insulation.

3.3.10 Future advancement for the report

OECD/NEA carried out the research on the wire system ageing and published the report *Research Efforts Related to Wire System Ageing in NEA Member Countries* [NEA/CSNI/R(2004)12], in which three collaborative researches related to the SCAP Cable Working Group activity were recommended as the high-priority near-term needs to be pursued. The Cable Working Group priority evaluation for the recommended collaborative research is as follows:

- *"Collaborative research is recommended for the development of an electrical diagnostics and condition monitoring method that can scan the entire length of an installed wire system and determine its current condition. In this regard, advanced electrical, optical, ultrasonic and aerospace technologies should be evaluated and developed for nuclear plant applications."*
  - The LIRA system being developed by Wirescan in Norway is one of the potential methods of condition monitoring for cables installed in nuclear power plants. This technique may need further refinement to enhance its applicability to actual plant needs.
- *"Collaborative research is recommended to establish the correlation between wire system condition indicators and the functional performance of the wire system during design basis events."*
  - The result of the ACA project of Japan showed the correlation between elongation at break (EAB) of cable insulator and the results of the test for design basis event (DBE). In addition, the management value of condition indicator for ageing cable was suggested based on the results (see Section 4.3.6).
- *"Collaborative research is recommended to provide a technical basis for developing and/or updating qualification methods and standards to reflect past operating experience and realistic plant operating conditions. Further focus and emphasis is needed on condition-based qualification. It needs demonstration and ongoing support of the international community."*
  - Advanced environmental qualification method was suggested in the guide entitled "Guide for Cable Environmental Qualification Test for Nuclear Power Plants", which was made based on the result of ACA project of Japan. In the guide, simultaneous thermal and radiation ageing is recognised as the most suitable accelerated ageing method (see the item concerning advanced EQ testing in Section 4.3.5).
  - The impact of continued oxygen supply in the LOCA chamber, dose rate effects and appropriate use of Arrhenius equation require further collaborative evaluation and standardisation.

Chapter 4: Commendable practice

4.1 Introduction

The objective of this internationally co-ordinated project is to share the corporate knowledge and operating experience to understand the failure mechanisms and identify effective techniques and technologies to effectively manage and mitigate active degradation in nuclear power plants.

The specific objectives of the project are to: *i)* establish a complete database with regard to major ageing phenomena for SCC and degradation of cable insulation through collective efforts by NEA members; *ii)* establish a knowledge base in these areas by compiling and evaluating the collected data and information systematically; *iii)* perform an assessment of the data and identify the basis for commendable practices which will help regulators and operators to enhance ageing management.

Commendable practices are derived from collected knowledge and experience as shown in Figure 1.5-2. The SCAP database and knowledge bases provide the extensive information underlying the commendable practices outlined in Sections 4.2 and 4.3. These, in turn, provide the technical and organisational elements which can inform ageing management programmes in the areas of SCC and cable ageing for safe long term operation. To use accumulated knowledge and a co-operative approach is to help improve the efficiency and effectiveness of ageing management and safety of long-term operation, so that it will be benefit for all stakeholders including designers, operators, regulators and ultimately the public.

4.1.1 Benefits in the area of SCC

Operating experience shows that the frequency of SCC events is generally decreasing. This clearly indicates that the mitigation measures developed and applied in the past decades to control SCC, incorporated into ageing management programmes for safe long-term operation, have been successful. While it is possible that new mechanisms may surface and that there could be an increase in SCC type failures involving some components or materials (*e.g.* IASCC), the overall trend is positive.

It is now understood that effective ageing management of SSC can be hindered by several factors which include an insufficient understanding or predictability of ageing, lack of data for ageing management and inappropriate use of reactive ageing management. The latter can lead to either unexpected or premature ageing (*i.e.* ageing mechanism that occurs earlier than expected). These factors or weaknesses need to be considered and the work of the SCAP project will help in this regard.

The broad knowledge base which now exists in the area of SCC resulted from experience gained from past challenges. Currently there are many experts in this field, but this may not be the case in the future due to demographic, practical (less opportunity to gain experience) and educational trends. So, efforts will have to be made to change these trends and the SCAP-SCC event data and knowledge bases, as well as commendable practices, will be a good source of information for those entering the field of SCC or needing to assess new SCC events.

4.1.2 Benefits in the area of cable ageing

The project has yielded crucial knowledge in the area of cable ageing related to qualification procedure for harsh environments, as well as the predictive capability to estimate the remaining qualified life. As part of the work, an up-to-date encyclopaedic source of data on unique cables and condition-monitoring techniques has been gathered and this can now be applied to help monitor and predict cable performance. While the project did not yield a great deal of data on cable failures, this, in and of itself was a finding. It was noted that cable failure events were strongly related to past manufacturing and installation practices. When corrected or when the cable was replaced, the problems were resolved. This information also forms part of the database and can be used to inform future practices.

The insights collected in this project offer a greater level of knowledge of ageing mechanisms, cable types and technical basis for addressing life extension and continued qualification of cables. Besides the technical data and operating experience, the SCAP cable ageing data- and knowledge base provide up-to-date information on environmental qualification of cables that need to remain functional during and following a design basis event. The database incorporates information on recent research results on ageing mechanisms and on continued efforts in enhancing condition-monitoring capability.

Countries regulate the long-term operation of their NPP in accordance with their legal and regulatory frameworks. Information supporting AMP is of vital importance given that ageing management is an essential and important aspect to be taken into consideration in connection with safe long-term operation. The activities of both working groups, the databases generated and the knowledge gained for SCC and degradation of cable insulation will be very valuable in this regard. The commendable practices being outlined in this report demonstrate practices that enhance safety or performance, and provide information on how to mitigate or avoid problems now known.

4.2 SCC

4.2.1 Introduction

Many stress corrosion cracking events have occurred in different structures, systems and components of nuclear power plants as early as the 1970s and have continued to the present time. The causes of these events were sensitisation of material by high heat input, local high residual stress, surface finishing and hardness associated with certain environments (such as high-temperature water). For these reasons, it is necessary to carry out maintenance activities, inspections, monitoring, preventive maintenance/mitigation, repair/replacement and safety assessment to minimise the occurrence of future events.

Therefore, knowledge should be extracted from the database (operational experience and recent findings) by analysing and evaluating the data from the viewpoint of the implementation of appropriate ageing management and maintenance activities beneficial to both regulators and operators.

To achieve this, a well grounded technical information basis (TIB) is needed for both the operator and the regulator. The ageing management and maintenance activities for SCC are to be performed taking into account ageing management from the early stage of nuclear power plant operation for the safe long-term operation of the plant.

In many countries comprehensive ageing management activities are now included from the early stages of plant operation and are reviewed regularly both through routine supervision and as part of the Periodic Safety Review (normally every ten years). Some countries perform a first extensive review in connection with the Periodic Safety Review associated with 30 years operation and others as part of the license renewal process.

As shown in Figure 2.1-1, it is first necessary to define an ageing management programme (AMP) for the counter measure regarding SCC by selecting the structures, systems and components (SSC), determining the safety important specific SCC mechanisms, and determining the criteria for evaluating that the intended safety function of the SSC are maintained. It is necessary to perform evaluation consistent with the long-term operation (*e.g.* 40 or 60 years), and to include the evaluation results in a preventive maintenance plan.

In order to perform optimal preventive maintenance for SCC, utilities should establish a long-term maintenance management plan. For safe long-term operation a well-grounded TIB is needed for each of the SCC mechanisms. To construct the TIB, comprehensive activities, including research and development, establishment of codes and standards, and consolidation of maintenance activities (PDCA), need to be continuously reviewed.

There are several factors which may hinder effective ageing management and may lead to either unexpected or premature ageing. These factors or weaknesses need to be identified and addressed in a proactive approach involving experts.

As shown in Figure 1.5-1 and Table 2.1-1, effective ageing management for SCC throughout the service life requires the use of a systematic approach to manage ageing. It is helpful to use the basic idea of a framework for co-ordinating all ageing management programmes and activities based on the understanding of preventive maintenance, mitigation, repair and replacement, inspection/monitoring/ qualification, safety assessment, and research and development on the ageing mechanisms and/or ageing effects of the SSC.

As described in Section 1.5, the IAEA Safety Standard Series No. NS-G-2.12 "Ageing Management for Nuclear Power Plants" explains how to extract and identify commendable practices appropriate for ageing management programmes (AMP) for SCC.

The guide describes nine generic attributes of an effective ageing management programme: scope of the ageing management programme, preventative actions to minimise control and ageing mechanisms, detection of ageing effects, monitoring and trending of ageing effects, mitigation of ageing effects, acceptance criteria, corrective actions, operating experience feedback and feedback of research and development, and quality management.

It is also important to establish the TIB to identify commendable practices from the knowledge base data which should contain international harmonisation of at least the following five items, excluding quality management which is not within the scope of SCAP:

- inspection/monitoring/qualification;
- preventive maintenance/mitigation;
- repair/replacement;
- safety assessment (flaw evaluation/fracture);
- R&D (initiation/crack growth/fracture).

As an example the Japanese approach to the ageing management of stress corrosion cracking for safe long-term operation is presented in Appendix 1.

4.2.2 Inspection/monitoring/qualification

Inspection and qualification

The locations and components to be inspected should be chosen on the basis of their susceptibility to the various types of stress corrosion cracking. This should be based upon a systematic analysis of the plant SSC. The choice of components and methods in the inspection programme can be affected by exposure to irradiation of the inspection personnel, for example visual techniques such as underwater cameras should be used for reactor vessel internals, and other mechanised methods developed for other components. When setting up an inspection programme, the initiation times should be taken into account. For example PWSCC under normal chemistry conditions and IASCC have been found to have much longer initiation times than IGSCC and ECSCC.

The inspection interval and technique used depends upon the type of stress corrosion cracking to be detected. The inspection interval should be related to the propagation rate of the cracking to be detected and the probability of detection using the inspection performance qualification. The inspection interval should be revised periodically, in particular if a defect is found and left in place during continued operation, or if suitable methods are available to monitor the crack propagation.

Inspection techniques should be qualified for the purpose in accordance with the applicable regulations, codes and standards. The applied process should be qualified (procedure, equipment and personnel) using real SCC or, if this is not possible, artificial defects (ex. EDM notch).

Different inspection methods can be qualified for detection and/or sizing the defect. In some cases, one method is used to detect a crack and a second method to size a defect. For some SCC degradation mechanisms a surface examination is sufficient, and for others volumetric methods must be used. It should also be noted that the crack morphology is mechanism-dependent. For example, PWSCC cracks are much tighter than the more oxide filled IGSCC in BWR, and this should be considered in the qualification procedure. In order to provide reliable input data for safety assessments of cracked components, both the probability of detection and the accuracy of the sizing method must be known.

As plants become older, it will be necessary to revise and expand the inspection programmes. There needs to be continuous development of new monitoring and inspection techniques. Improved understanding of some of the degradation mechanisms not fully understood today will aid in the identification of new mitigation possibilities.

Monitoring

Several chemical parameters that can affect the initiation or propagation of stress corrosion cracking such as harmful impurities should be monitored on a regular basis and should be included in plants' chemistry programmes. Many of these measurements are now performed on-line so that monitoring is carried out on a continuous basis. Some specific monitors have been developed, such as electrodes to measure the electrochemical corrosion potential (ECP) *in situ* in BWR to determine if sufficient hydrogen is being dosed to the system. The cleanliness of stainless steel surfaces should be monitored regularly to ensure that the prescribed levels are maintained and eliminate the initiation of ECSCC.

Since there is susceptibility fluence for IASCC which depends on the stress level of the component the fluence levels of vessel internal components should be monitored, and/or calculated, so that inspection and maintenance can be carried out in a timely manner. The cumulative damage in materials and changes in stress, including radiation creep and swelling in PWR, should also be taken into consideration.

Once a crack has been detected and is left in place during continued operation some countries permit that it be monitored directly or indirectly. Direct *in situ* monitoring techniques are not yet being used other than in an experimental or verification manner. Some advanced inspection/monitoring techniques could become available, but need further development. There are a number of established methods for leak detection. In many countries leak detection is not an approved method of detection, but can be considered to be part of the defence in depth.

4.2.3 Preventative maintenance/mitigation

The maintenance process should consider maintenance systematisation in order to prevent SCC initiation, mitigate crack propagation and perform proper inspection, flaw evaluation, preventive maintenance, repair and replacement. In the operation stage, preventive maintenance consists of period surveillance (ISI time/cycles) and predictive maintenance (based on the residual life assessment). For the selection of appropriate inspections, the safety functions of the components, the anticipated degradation based on the pertaining environment, and the field experience should be considered.

If the possibility of stress corrosion cracking cannot be ruled out, preventative maintenance or predictive maintenance is applicable. An example of a maintenance plan for ageing due to stress corrosion cracking is shown in Figure 4.2.3-1. Based on past operating experience and the latest research results, mitigation against crack growth, partial or complete replacement, can be carried out as preventative maintenance.

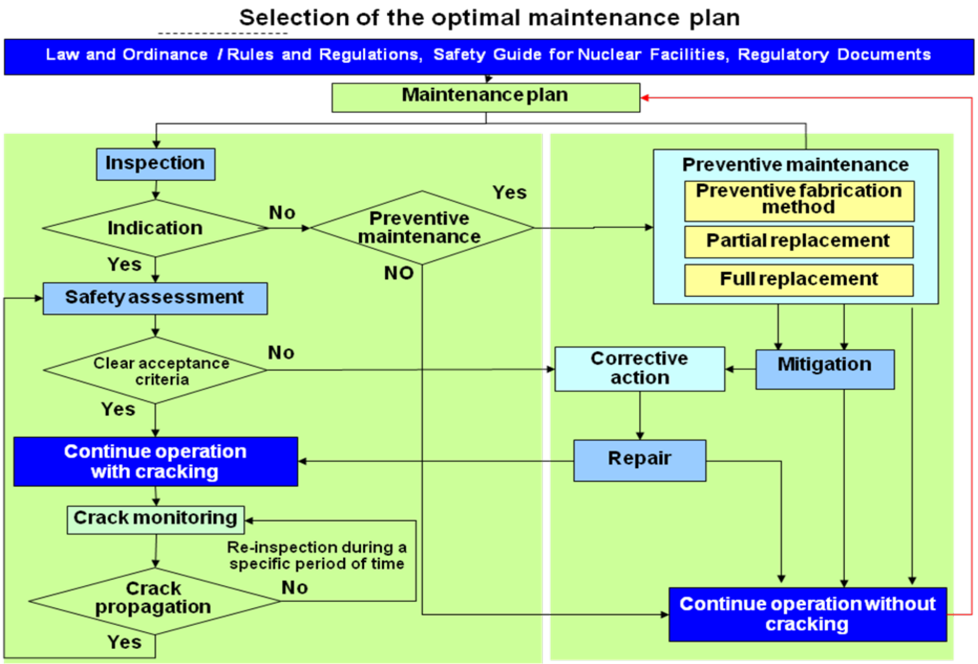
Three factors (susceptible material, aggressive environment and stress) are essential for SCC, and it can be avoided by eliminating at least one of these three factors. Therefore mitigation of several SCC factors is recommended to improve the effectiveness of mitigation methods. Early application of mitigation methods is recommended since mitigation is often more effective when applied prior to SCC initiation or when the degradation is limited.

For the safe long-term operation of nuclear power plants, it is important to maintain a proactive ageing management programme. This should be based on the lessons learned from past experience. It is important that past experience be regularly incorporated into new regulations, codes and standards, and shared and harmonised internationally.

Material

It should be noted that it is important to eliminate at least one of the factors to prevent SCC. In terms of prevention, the choice of material is important and is one of the key factors. Since the first plants were built, new materials have been designed in an attempt to eliminate some of the causes of cracking based on the experience of stress corrosion cracking events.

Figure 4.2.3-1: Example of a maintenance plan for ageing due to stress corrosion cracking



To prevent stress corrosion cracking in BWR components, low carbon grades of austenitic stainless steel containing up to 0.03% carbon are selected as material while for PWR components, even though carbon level control is not needed for weld metal and casting with duplex structures according to Regulatory Guide 1.44, material with a carbon content ranging between 0.03% and 0.05%, and weld metal with a maximum carbon content of 0.035%, and often a specified minimum ferrite content weld metal and cast stainless steel (CASS) have demonstrated good operational experience and are recommended.

The Alloy 600 series has successively been replaced by the less susceptible Alloy 690 series because of its higher chromium content, in particular in PWR components. Care however is needed when using its weld materials (Alloys 52 and 152) since they are prone to hot cracking. In the case of IASCC, 316L stainless steel can be used in BWR since the susceptibility fluence has been found to be higher than that of 304/304L stainless steels.

If reactor vessel internals are to be replaced as preventative maintenance it is worthy to note that experience has shown that the susceptibility fluence for IASCC in BWR is higher for 316 than for 304 austenitic stainless steels. It is also advisable to avoid welds in the highest flux regions of the components.

Stress

Stresses introduced during manufacture should be minimised to reduce the risk of SCC initiation. For this purpose new welding procedures such as narrow gap and heat sink welding have been developed.

In the BWR reactor water environment, there have been many incidents in which cracks initiated in the transgranular mode in the hardened layer due to the heavy machining, and have then grown in the intergranular mode. It is therefore important that the surfaces of components do not contain high tensile stresses and a number of methods (for example peening and polishing) have been developed to ensure that the surface finish is acceptable in this respect free from cold work. Some of these methods can also be applied to existing components to reduce residual stresses or introduce compressive stresses, for example induction heating stress improvement and laser stress improvement processes. When applying any of these techniques, it is recommended that a qualification process is followed, and if appropriate, that the work be supervised by an independent third party. It is also important that the component can be inspected as part of the ordinary inspection programme. Weld overlays and MSIP are accepted as mitigation techniques in some countries.

In PWR, it is recommended that a parabolic-shaped radius be introduced for the head-shank curve of baffle former bolts, in order to reduce the stress concentration factor compared to the circular curve. Elongation of head-shank length is also effective to reduce the stress of the head-shank.

With regard to IASCC the threshold stress decreases as the dose increases. Stress improvement surface finishing techniques can be applied during preventative maintenance. However it is necessary to take into account irradiation-induced stress relaxation, creep and void swelling, when applying stress improvement techniques to highly irradiated components.

Environment

Research on the root cause of IGSCC in BWR led to improved water chemistry and chemistry control programmes with low levels of impurities measured indirectly by the conductivity of the water. Lowering the electrochemical corrosion potential by the addition of hydrogen [hydrogen water chemistry (HWC)] is a widely applied mitigation technique. Noble metal chemical addition (NMCA) has also been applied in cases where cracks have been found in reactor vessel internals, such as core shrouds, to increase the effectiveness of HWC in the more oxidising core environment.

Chloride contamination levels must be controlled to avoid ECSCC. It is also advisable to control the cation contamination, and it can be noted that sodium is more benign than magnesium, zinc and copper. The choice of non-chloride-containing marking pens and tapes is important and it should also be noted that personnel are a major source of chloride contamination. Lubricants and gaskets should not contain substances known to cause TGSCC. It is also important to pay attention to such things as coversheets used for temporary storage on site, and also to dew condensation in the containment during construction or maintenance and repair work, etc. (installation of an air dryer can alleviate this problem).

In general reactor vessel internals are intended to be used for the entire plant lifetime and, in consequence, will in particular in PWR be exposed to very high radiation doses, typically up to 100 displacements per atom (dpa), assuming 60 years operation. With such high radiation doses, the material microstructure and mechanical properties can change considerably, which could have a significant impact on the stress corrosion cracking susceptibility. Radiolysis is suppressed in PWR by the addition of hydrogen. There are moves to reduce the amounts of hydrogen dosed and it is recommended that this be considered with care with regard to the potential effect on IASCC and PWSCC. Fuel management can affect the final dose of the material and may offer a method of mitigating IASCC.

All the environmental conditions of SSC must be reviewed periodically with respect to normal, local (crevice, two-phase) or accidental environment (polluted) conditions.

Other approaches to preventative maintenance and mitigation

When replacing piping and similar components weld configuration and welding procedures can be optimised.

When replacing baffle bolts, changes in design can be made to reduce the temperature due to gamma heating and thus to reduce stress by swelling in the high fluence range. Other changes have been made to improve the shape of the head-shank curvature and remove some of the regions of stress concentration in the original design. The bolt heads should be evaluated during the design and construction phase.

4.2.4 Repair/replacement

Many of the considerations for preventative maintenance and mitigation are applicable for repair and replacement. The difference between these two concepts is that the latter is carried out after a crack is detected and the former, as the name implies, is carried before a crack is found in order to prevent stress corrosion cracking. When a component is to be replaced the concepts presented in Section 4.2.3 are applicable. This section will therefore concentrate on repair methods rather than replacement.

If the flaw detected does not meet the acceptance criteria in the safety assessment, the component should be repaired or replaced in accordance with the regulations, codes and standards. When a repair/ replacement method that is not in accordance with the regulation standard is to be applied, the basis for the validity of the structural integrity must be approved and verified.

Repair normally implies that the crack is removed. Care should in such cases be taken to ensure that the crack tip does not remain in the component but is fully removed. If the crack is removed by grinding, the surface finish should not result in unnecessary cold work or residual stresses which could lead to initiation. Thus surface polishing can be applicable. If the crack is removed by electrode discharge machining the choice of electrode and process parameters should prevent hot cracking of the component surface. The surface condition should permit further inspection of the component.

Welding is another possible repair method. This can be applied after the crack has been removed to restore the original dimensions and structural integrity to the component. In this case an inlay of compatible material is introduced. A method acceptable in some countries is the weld overlay. In this case additional weld material is applied to the outside of the component (nozzle or pipe) and the crack may be left in place. Welding is also a central component in the half-nozzle repair technique which has been applied in some cases of reactor vessel penetrations in which cracking has been detected.

Repair methods should normally be chosen so that the possibility of recurrence is minimised. This can mean that the component is not repaired immediately after the crack has been evaluated, but that the repair is performed after extensive planning and verification procedures and associated safety assessments is carried out. It can therefore be cost-effective to develop repair techniques for specific essential components in advance. It is recommended that repair techniques be qualified for the specific application.

Regarding replacement, it is advisable to make a judgement taking into account the following factors: comparison of merits and demerits of repair and replacement, feasibility study of replacement in accordance with the original design or reflecting latest design, needs for technology development, applied experience, the need for additional licensing procedures, access limitation during field work, working environmental conditions (*e.g.* underwater), location, requirements of installation accuracy, working period, personnel radiation dose, costs, requirements of inspection after replacement and so on.

Opportunities for modernisation should be taken to replace systems and components using materials less susceptible to stress corrosion cracking with optimised weld configuration and welding processes. When replacing systems and components, quality control is essential not least because of the shortage of experienced workers, such as welders, in the nuclear field. The detail of on-site storage of new components is important to ensure that they are not degraded prematurely, *e.g.* by ECSCC.

4.2.5 Safety assessment

When defects are detected during in-service inspection the extent of the inspection programme should be expanded to include other components which could also be affected. It is important that a root cause analysis be performed so that appropriate repair and replacement procedures can be applied to limit the risk for repercussion. If the defect is to remain during continued operation then this should be justified by a safety evaluation. This is illustrated in Figure 4.2.3-1.

When defects are detected during in-service inspection, a safety assessment must be performed to determine if the flaw size will be within the acceptance criteria at the end of the evaluation period and if continued operation can be permitted for the evaluation period. The component integrity assessment can be made using either a deterministic or a probabilistic approach, or a combination of the two in accordance with the regulations in the country.

The evaluation methods used in different countries for the safety assessment of the acceptability of limited defects are mostly based on empirical data. As described in the consultant reports physically based models are still at the stage of scientific development. Due to the number of variables involved, as well as dependencies and correlations, further R&D efforts are ongoing as shown in Section 4.2.6. Common to all such evaluation methods is the need for propagation rates in the specific material/environment combination. It is generally accepted that propagation rates should be based on quality assured data whenever possible. It should be noted that propagation rates in irradiated materials are normally several orders of magnitude higher than for un-irradiated materials.

Operating experience has shown that stress corrosion cracking may arrest, or the propagation rate may decease significantly in the centre of the component depending on the residual stresses and the configuration of the flaw.

In the safety assessment, consequences of leak and break on plant safety must be assessed for normal and accidental conditions.

4.2.6 Research and development

Research and development (R&D) efforts are ongoing in most countries with the goal to improve understanding of the initiation and propagation processes associated with the different types of stress corrosion cracking. Extensive efforts are also under way to develop inspection and monitoring techniques, repair and replacement methods and the necessary data to enable the performance of safety analyses. The results of such research and development should be reflected in revisions of regulations and codes and standards in the different countries.

Although it is not an R&D activity in itself, one very important source of such efforts lies in experience feedback not least in the root cause analysis of events, in particular stress corrosion cracking events which are found for the first time in a specific material or material environment combination. Such findings should lead to research to ensure that there is full understanding of the specific conditions and that a suitable strategy can be developed regarding mitigation.

Another important aspect of R&D is international co-operation that has been developed within the SCAP SCC Working Group where regulators, operators, vendors and academia have collaborated in an open and productive manner.

4.2.7 New plants

For new plants ageing management of amongst other things stress corrosion cracking should be part of the maintenance programme from the first operation of the plant. The principles discussed in this report are as applicable to new plants as they are to the current fleet.

When designing and building new plants all of the comments concerning long-term operation are relevant; there are however greater possibilities to minimise the risk for many of the degradation mechanisms discussed here. This can be done through choice of materials (including insulation), optimising coolant chemistry and weld configuration and procedures. It is also possible to avoid positioning welds in regions which have been demonstrated to be sensitive, such as the core region for reactor vessel internals.

It is also important that inspection needs and monitoring be considered during the design process both with respect to accessibility but also to ensure that doses to personnel can be as low as reasonably possible. With regard to maintenance possibilities of on-line maintenance, condition-based monitoring and maintenance-free design should be considered. A surveillance programme for internals as well as the reactor pressure vessel can be introduced.

4.3 Cable

4.3.1 Introduction

Cables installed in nuclear power plants have the functions of transmitting instrument and control signals and supplying power to electric components. Some of the cables are required to perform their function under the condition of the design basis event and they are classified as Environmentally Qualified (EQ) cables.

In general, the cables have been regarded as a maintenance-free component. However, cable insulation performance will gradually decrease due to the surrounding environment such as heat, radiation, moisture, etc. Therefore appropriate ageing management needs to be carried out from the early stage of plant operation to prevent potential failure events.

The commendable practices for cable ageing management identified through the activity of the SCAP Cable Working Group are presented in following sub-sections. In addition, the items to be considered especially for new plants are described in Section 4.3.8, entitled "New plants". It is desirable that all the items of the commendable practices in this section be applied to EQ cables. For non-EQ cables, the items related to environmental qualification are excluded; however certain elements of the programme may be adopted based on specific needs.

4.3.2 Specification

Power cables with shield

The United States experience indicates that the condition monitoring of power cables has become a necessity in order to ensure the operational readiness of the cable. The medium-voltage cables that power most of the emergency core cooling pumps and feeds to 1E power supplies serve a significant function for nuclear safety. Some of these cables are normally energised and some are in standby conditions during the power operation of the plant. As the plants continue to age, several factors could contribute to insulation degradation.

Such degradation has to be monitored and corrective actions have to be implemented to prevent operational failures that could trip the plant or limit core cooling capability. The cables that have a continuous shield as part of the cable jacket provide much wider and better options in condition monitoring. Therefore, power cables for new plants or replacement cables for the operating plants should consider using only cables with a metallic shield.

Undesirability of PVC cable insulation

New NPP installations in the Czech Republic are significantly restricted as regards the use of certain chemical elements. European Utility Requirements (European Utility Requirements for LWR Nuclear Power Plants, Revision C, Vol. 2, Chapter 2.6: Material-related Requirements, April 2001) quote general rules and requirements concerning new NPP projects [35], *e.g.* the amount of halogens should not be higher than 200 ppm. There are also requirements concerning other elements like sulphur, zinc, lead, mercury, asbestos, etc. These requirements disqualify PVC cables for future nuclear installations.

Preservation of cable technical data

Cable specifications, normally required and acquired during cable procurement are very important for preserving the qualification of the cable. During plant life (40 years), design modifications and maintenance activities could change the original plant cable population or its routing. Additionally, cable manufacturers could disappear or change, causing problems to collect cable information for any future assessments.

In Spain there are examples where additional detailed designs of cable information were needed by the plants:

- *Plant life extension applications*: One plant was required by CSN, for collecting additional cable design details (identification, routings, material characteristics) to supplement its cable ageing effects analysis.
- *Cable qualification*: After manufacturer disappearance, one type of cable was qualified by the plant using the “analysis method”. The cable design and its materials (chemicals) were compared with a similar qualified cable and found to be identical.
- *Field cable identification*: One plant had problems for identifying many installed cables that were found without visible marks in the jacket.

In the above examples the corrective actions applied by the plants to solve the lack of cable information were expensive and time consuming and could have been avoided if the proper cable design and qualification information was available.

For these and many other reasons, preserving cable design details from the beginning of plant operation could be a valuable practice.

4.3.3 Inspection

Condition monitoring inspections and tests can provide the means for evaluating the level of ageing degradation of electric cables. The cables are exposed to a variety of environmental and operational stressors throughout their service life. Environmental stressors can include elevated temperatures, high radiation, high humidity, moisture intrusion, accumulation of dirt and dust and exposure to chemicals or other reactive contaminants. Operational stressors can include external interference, installation and maintenance damage, high voltage stress, materials defects, water treeing and electrical transients. Over time, the ageing and degradation mechanisms caused by these stressors can eventually lead to early failure of the cable. It is therefore important to have periodic condition-monitoring inspection and testing of electric cables in the assessment of cable ageing and degradation. Severely damaged or degraded cable insulation can then be identified and repaired or replaced to prevent unexpected early failures while in service.

In addition, the benefits of periodic cable condition-monitoring inspections and testing can be further complemented by monitoring cable operating environments. Environmental stressors, especially temperature, moisture/flooding and radiation, can contribute to significant ageing and degradation of electric cable insulation and jacket materials. Monitoring and management of the environmental conditions in which cables are operated can help operators to identify adverse stressors so that measures can be taken to control or reduce ageing and degradation.

4.3.4 Maintenance

Maintenance activities interact in various ways with the specified function of components and environmental conditions within the area where the activities are performed. Procedures are established to assure that conditions at the end of maintenance activities are within specified limits. Regarding cable ageing, past experience has demonstrated the importance of the following areas.

Acceptance criteria for condition monitoring inspection

In general, cables installed in nuclear power plants have been regarded as a maintenance-free component. However, for long-term operation, maintenance activities for cables important to safety should be done appropriately to ensure their capability for the required function. In Japan, all cables installed in NPP are inspected periodically and the acceptance criteria are shown in Table 3.3.2.2-2. These acceptance criteria cannot be used for the evaluation of integrity in a DBE environment.

Applicability of indenter

For maintenance, the level of degradation of cable insulation needs to be measured appropriately. Elongation at break (EAB) is considered to be a good condition indicator for degradation of polymer and a lot of EAB data have been accumulated.

In the ACA project, the applicability of the indenter modulus method for cables was tested by checking its correlation with EAB. In the test, reliability of the diagnostic was confirmed by round robin test with an improved type of indenter modulus device with bearing of the probe actuator made in Japan and higher supply voltage, developed by Institute of Nuclear Safety System, Inc. (INSS, Japan) and AEA Technology Inc. (UK). As a result, it was found that there is a good correlation between indenter modulus and EAB (shown in Figure 4.3.4-1) and therefore the method is proven to be applicable to the condition-based EQ.

Environmental condition monitoring

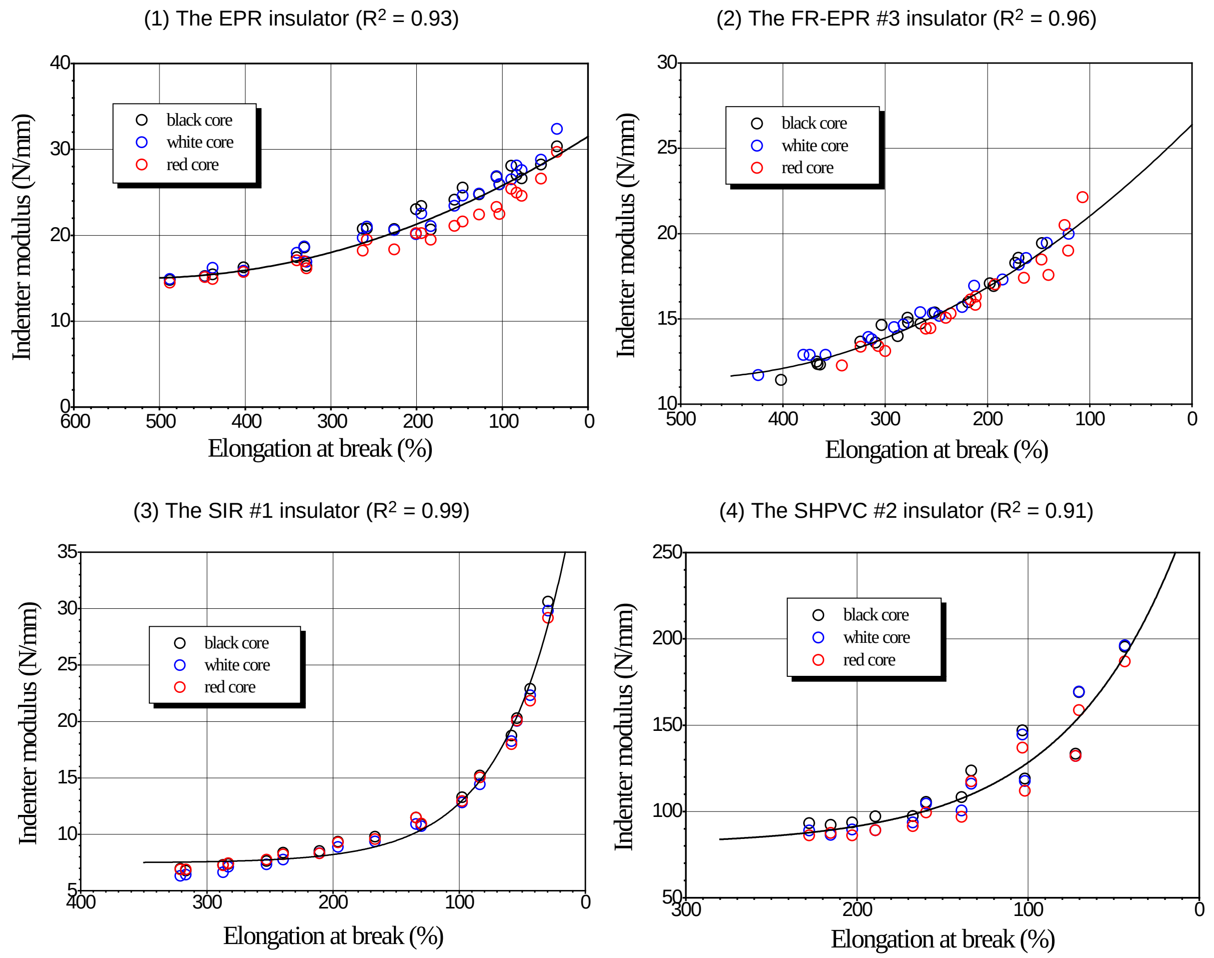
In order to ensure the qualification life or to predict the life of EQ cable, monitoring of a cable's installed environment is very important.

In Japan, electric power utilities are required to investigate the environmental conditions for cables installed inside reactor containment in all operating commercial nuclear power plants, based on an administrative document issued by the Nuclear and Industrial Safety Agency (NISA) in October 2007. In the investigation programme, the temperature and radiation dose in cable's installed areas must be measured once during periodical inspection period or during refuelling outage period. Devices for the temperature measurement must include a data logger and an Alanine dosimeter for the radiation measurement.

Investigations of 35 nuclear power plants have been completed as of January 2010 and their results are shown in Table 3.3.5-2. On the whole, temperature and radiation dose rate investigated showed lower values than the design value except for some nuclear power plants. In particular, the values of radiation dose rate were considerably low.

For power uprate of NPP, the possible change of the environment condition where cables are installed should be evaluated and further environmental monitoring should be carried out after the power uprate.

Figure 4.3.4-1: Correlations between indenter modulus and elongation at break



Hot spot identification

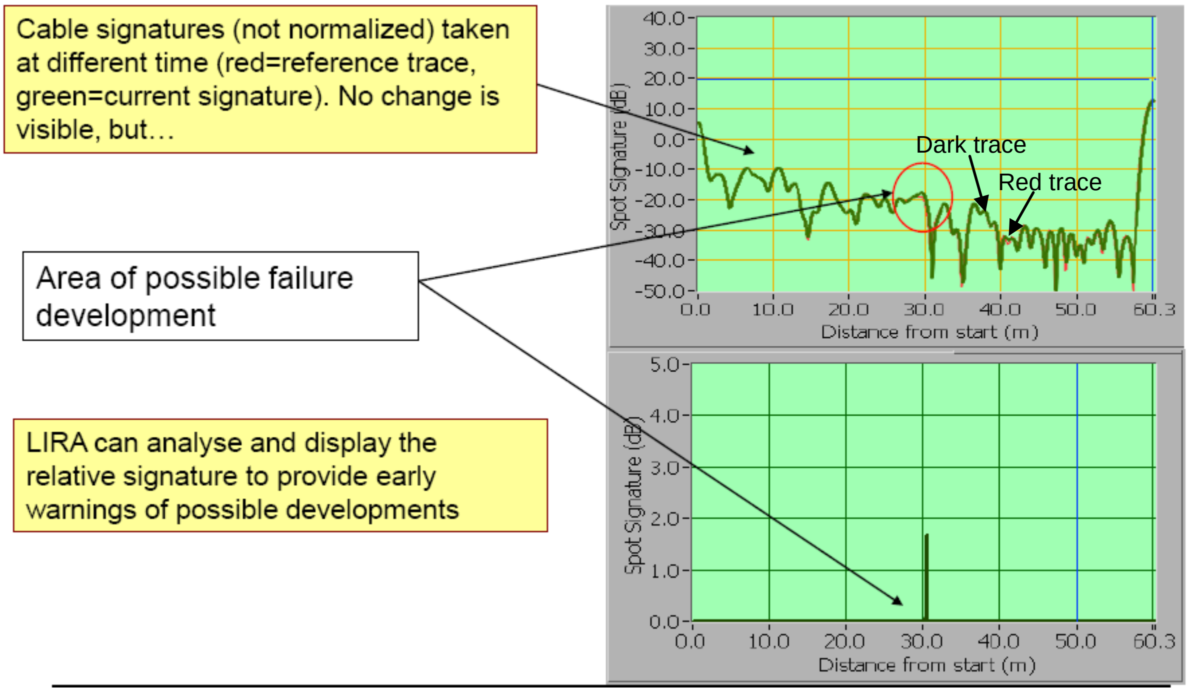
The main problem with hot spots is that although the average cable condition is still good with a residual life expectation of several years, the cable would eventually fail at the hot spot location much earlier than expected, if the hot spot is not properly identified and the cause corrected or the cable replaced/repaired.

Thermal hot spot identification, in inaccessible areas, can be performed with the following methods:

- Environment monitoring, to identify containment areas where the temperature/radiation is higher than the average (higher than that used in the cable qualification process) and remove the cause of possible hot spots or replace the cable with greater capability to withstand the environment.
- Use of cable deposits. Deposits of sacrificial cable samples can be installed in areas where the environment temperature/radiation is higher than average and periodically tested with local condition monitoring techniques (EAB, indenter, OIT…) to schedule corrective actions.
- Full length, on-line, electrical condition monitoring. Methods such as LIRA can be used to periodically test cables *in situ* to identify the presence of a hot spot along the cable.

Figure 4.3.4-2 (upper graph) shows a LIRA signature taken on a good cable (red trace) and another measurement taken a year later on the same cable (dark trace). The two traces are almost identical

Figure 4.3.4-2: Baseline analysis with reference signatures (LIRA)



and no action would be taken by the analysis of only the current, most recent measurement. However, a comparative analysis of the two signatures would discover a developing change that should be addressed, see the lower graph on Figure 4.3.4-2.

This approach is called Baseline Analysis of Installed Cables and is particularly suited for new or good cables, when a reference signature can be taken before any damage or degradation has occurred.

Removal of heat insulation

In various locations of the plant, a variety of heat insulation materials (reflective metallic insulation, asbestos fibre, etc.) are used either to shield the plant environment from high temperature or to preserve the heat within the pipes and components for thermal efficiency. These insulation materials are often removed for maintenance activities. Since most of the heat insulation activities are performed by non-electrical staff, the vulnerabilities of cable insulation to high heat is often not considered. As a result, the removal of heat insulation leads to overheating of cables in the immediate proximity.

Based on the level of heat exposure received the cable may have a delayed failure or a prompt failure during its service. The problems are often identified following a cable failure. Therefore, it is important to evaluate the impact of heat insulation removal on electrical components located in the proximity.

Cables wrapped within heat insulation

The piping systems associated with steam systems and other hot process systems have motor-operated valves and other electrical components connected to the piping. During construction or maintenance, the heat insulation work is the last part of the work. Therefore, the insulation work has mistakenly wrapped the hot pipe with cable inside. This type of wrapping results in exposing the cable to an unacceptable level of heat produced by the piping system. The cable undergoes accelerated ageing, leading to premature failure. These problems are often identified only with system failures and it could lead to system unavailability or plant trip. Therefore, it is prudent to have oversight on heat insulation work to prevent deleterious effects on electrical cables.

4.3.5 *Environmental qualification (focusing on EQ cables)*

New insight from Japan

Activation energy used for accelerated ageing

When the activation energy of each insulation material is calculated from thermal ageing test data acquired at this time, most of the data is of the order of 100 kJ/mol (20 or more kcal/mol), and some are also 40 or more kJ/mol (a little more than 10 kcal/mol). In addition, a large portion of thermal ageing test data used for calculation of activation energy is data ranging from 100 to 120°C [33].

When the sampling data in actual operating plants and the data acquired in this project are compared, it is found that the activation energy in actual operating plants' temperature region (from 50 to 60°C) is smaller other than the value calculated from the thermal ageing test data acquired in this project.

Although it differs from elongation at break, some literature states that around 60 kJ/mol (approximately 15 kcal/mol) can be assumed to be appropriate for the activation energy in a region of uniform oxidisation acquired with chemo-luminescence analysis.

Based on the above, the principles of calculation and application for the activation energy used for future assessment as shown below were determined to be appropriate.

- Applicable region of activation energy calculated by thermal ageing tests is limited up to the minimum temperature in thermal ageing tests. However, when the calculated activation energy is less than 62.8 kJ/mol (15 kcal/mol) the value can be applied up to the operating temperature region of actual operating plants.
- Activation energy in the region between the minimum temperature in thermal ageing tests and the temperature of actual operating plants are evaluated from the investigation results of ageing in actual operating plants (sampling inspection) and thermal ageing characteristics at the minimum temperature in thermal ageing tests.
- When activation energy cannot be evaluated from the investigation results of ageing in actual operating plants, 62.8 kJ/mol (15 kcal/mol) is used as a tentative value for the activation energy in the region between the minimum temperature in thermal ageing tests and the temperature of actual operating plants.

Recommended procedure for accelerated ageing

Accelerated ageing equivalent to ageing during normal operation is most accurate when conducted as simultaneous thermal and radiation ageing based on the results of ACA project [33]. The temperature and dose rate that enable confirmation of degradation progress uniformly into the full thickness the insulation material should be selected for simultaneous ageing.

In the ACA project, progress of degradation into the inside of the insulator was confirmed with a dose rate of 100 Gy/h and a temperature of 120°C (175°C for silicone rubber) for cross-linked polyethylene, flame-retardant cross-linked polyethylene, ethylene-propylene rubber, flame-retardant ethylene-propylene rubber, silicone rubber and special heat-resistant polyvinyl chloride.

Advanced EQ testing [differences from IEEE document (IEEE383-2003)]

It is important that the EQ test be performed with a method which can appropriately simulate ageing under normal operation.

In Japan, the "Assessment of Cable Ageing for Nuclear Power Plants (ACA)" was carried out from FY2002 to FY2008 and the advanced guide for cable EQ test "Guide for Cable Environmental Qualification Test for Nuclear Power Plants" was formulated based on the result of the project [33]. In the guide, simultaneous thermal and radiation ageing is adopted as an accelerated ageing method to simulate ageing under normal operation instead of the conventional method of sequential ageing. In addition, activation energy used to set the accelerated ageing condition is determined considering the conditions in operating plants. Several techniques for setting accelerated conditions are also provided in the guide.

The outline of the advanced guide is as follows:

- A cable specimen 3 m long is used. The cable with the thinnest insulation thickness is selected as the specimen for the cables with identical insulation specifications.

  Even if the insulation specification is identical, a separate test is carried out for the cables produced by different manufacturers.

- Accelerated ageing equivalent to ageing during normal operation is conducted as concurrent temperature and radiation ageing to reflect actual plant conditions. The temperature and dose rate are chosen such that a homogeneous change that penetrates the full depth of the insulation is selected for simultaneous ageing.

  In the ACA project, degradation progress into the inside of the insulation was confirmed with a dose rate of 100 Gy/h and a temperature of 120°C (175°C for silicone rubber) for cross-linked polyethylene, flame-retardant cross-linked polyethylene, ethylene-propylene rubber, flame-retardant ethylene-propylene rubber, silicone rubber and special heat-resistant polyvinyl chloride.

- Activation energy used for accelerated ageing is as follows:

  - Applicable region of activation energy calculated by thermal ageing tests is limited up to the minimum temperature in thermal ageing tests. However, when the calculated activation energy is less than 62.8 kJ/mol (15 kcal/mol) the value can be applied up to the operating temperature region of actual operating plants.

  - Activation energy in the region between the minimum temperature in thermal ageing tests and the temperature of actual operating plants are evaluated from the investigation results of ageing in actual operating plants (sampling inspection) and thermal ageing characteristics at the minimum temperature in thermal ageing tests.

  - When activation energy cannot be evaluated from the investigation results of ageing in actual operating plants, 62.8 kJ/mol (15 kcal/mol) is used as a tentative value for the activation energy in the region between the minimum temperature in thermal ageing tests and the temperature of actual operating plants.

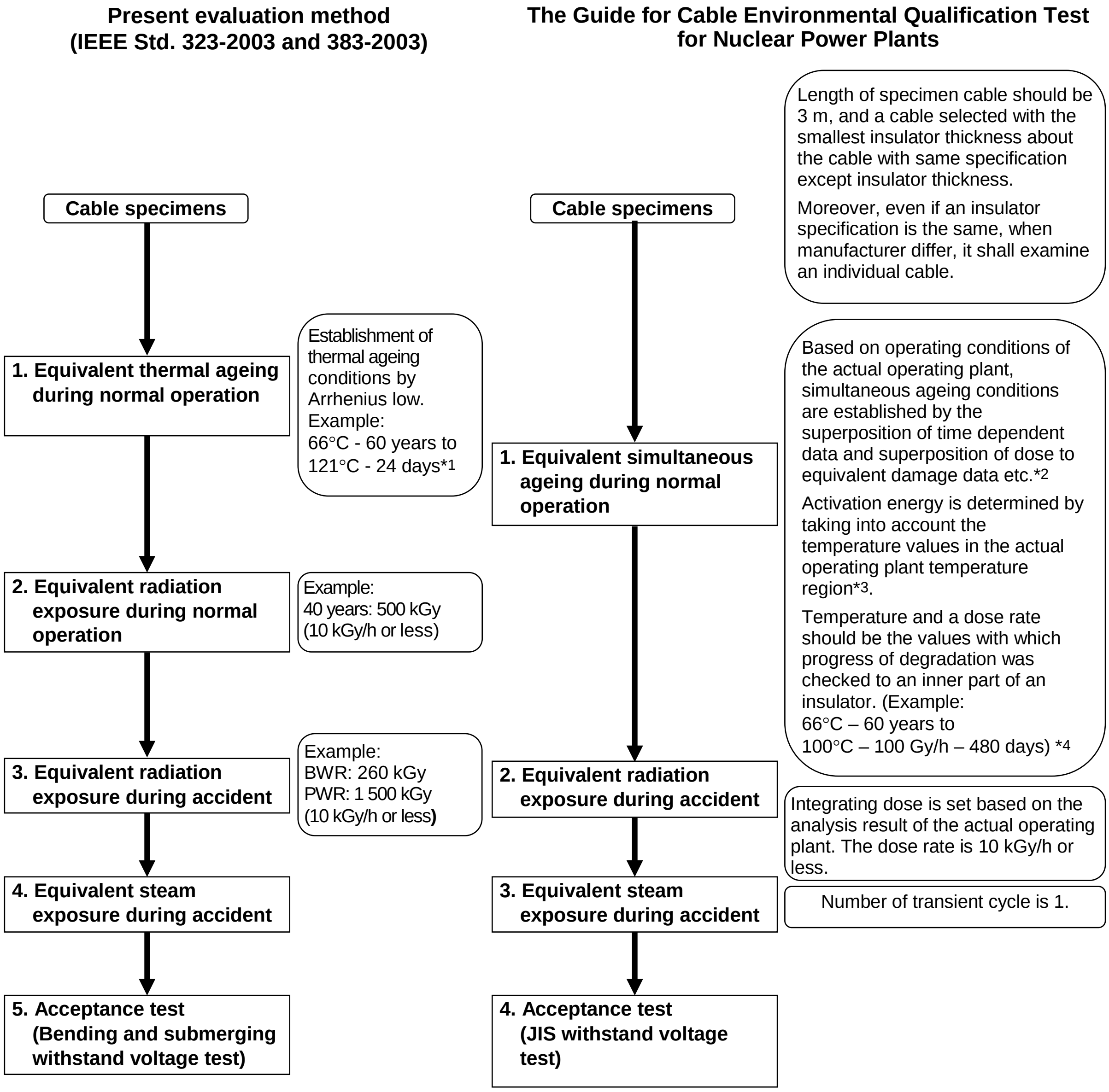
- The condition of accelerated simultaneous ageing equivalent to ageing during normal operation is established by the techniques of “superposition of time dependent data” and “superposition of dose to equivalent damage data” or the technique of “same acceleration factor” (which has been verified to be equivalent to the former) based on the condition of actual operating plants. However, it is recommended not to use the technique of “superposition of time dependent data” to establish the condition for silicone rubber insulation cable.

  Further, if the cable has an insulation of cross-linked polyethylene, flame-retardant cross-linked polyethylene, ethylene-propylene rubber, flame-retardant ethylene-propylene rubber, silicone rubber and special heat-resistant polyvinyl chloride, the condition of accelerated simultaneous ageing may be established by a simplified method, that of using the technique of “superposition of dose to equivalent damage data”.

- The withstand-voltage test in Japanese Industrial Standards (JIS) is used to determine integrity of insulation after LOCA test.

The differences between IEEE document IEEE383-2003 and the advanced guide are shown in Figure 4.3.5-1.

Figure 4.3.5-1: Comparison of the methods for Cable Environmental Qualification Test



*1: Extrapolation of activation energy evaluated from degradation characteristics in a high temperature region up to the actual operating plant temperature is permitted.

In the above example, 33 kcal/mol is used for activation energy.

*2: There is the technique of same acceleration factor, such as making equal the accelerating factor of temperature and radiation other than such technique.

*3: When the activation energy in the actual operating plant temperature region cannot be estimated, 41.8 kJ/mol (10 kcal/mol) for silicon rubber and 62.8 kJ/mol (15 kcal/mol) for materials other than silicone rubber may be used.

*4: It set up with the superposition of time dependent data by making activation energy into 62.8 kJ/mol (15 kcal/mol).

4.3.6 Recent discovery from cable ageing evaluation test

In the "Assessment of Cable Ageing for Nuclear Power Plants (ACA)" project, EQ tests for various types of cables was carried out with the advanced environmental qualification test method based on the new insight obtained in the project. Based on the result, the acceptance criteria for elongation at break (EAB) of cable ageing indicator was suggested as shown in Table 4.3.6-1.

In the EQ test, pre-ageing tests with several accelerated conditions were carried out for each type of cable to find the maximum degree of degradation that pass the LOCA test. The pre-ageing conditions and the results of the LOCA tests for XLPE #1 insulated cables are shown in Table 4.3.6-2.

The management values of elongation at break were determined by adding an approximate 10% safety margin to the original value measured. They are considerably higher than conventional ones. This is attributed to the fact that homogeneous degradation in samples is given by low accelerated ageing while ageing with high accelerated rate causes non-homogeneous degradation, which leads to the small value of EAB.

Table 4.3.6-1: Acceptance criteria for endurance in LOCA environment

Insulator of cable	Initial value (EAB)	Management value (EAB)	Remarks
XLPE #1 insulated cable	557%	310%	Confirmed to endure LOCA to follow at 276%
XLPE #2 insulated cable	488%	90%	Confirmed to endure LOCA to follow at 76%
XLPE #3 insulated triaxial cable	307%	70%	Confirmed to endure LOCA to follow at 58%
FR-XLPE #1 insulated cable	396%	250%	Confirmed to endure LOCA to follow at 226%
FR-XLPE #2 insulated cable	562%	240%	Confirmed to endure LOCA to follow at 215%
EPR insulated cable	571%	110%	Confirmed to endure LOCA to follow at 92%
FR-EPR #1 insulated cable	405%	70%	Confirmed to endure LOCA to follow at 60%
FR-EPR #2 insulated cable	515%	230%	Confirmed to endure LOCA to follow at 204%
FR-EPR #3 insulated cable	479%	210%	Confirmed to endure LOCA to follow at 184%
SIR #1 insulated cable	357%	30%	Confirmed to endure LOCA to follow at 24%
SIR #2 insulated cable	517%	40%	Confirmed to endure LOCA to follow at 30%
SIR #3 insulated cable	420%	40%	Confirmed to endure LOCA to follow at 29%
SHPVC #1 insulated cable	248%	110%	Confirmed to endure LOCA to follow at 99%
SHPVC #2 insulated cable	245%	220%*	Confirmed to endure LOCA to follow at 219%

* Determined by adding safety margin of small per cent because the value of EAB with 10% margin is nearly equal to the initial value of EAB.

Table 4.3.6-2: The pre-ageing conditions and the results of the LOCA tests

	Pre-ageing condition	LOCA test result
Case 1	100°C – 89.4 Gy/h – 591 hrs	Passed
Case 2	100°C – 89.4 Gy/h – 734 hrs	Passed
Case 3	100°C – 89.3 Gy/h – 805 hrs	Passed
Case 4	100°C – 99.3 Gy/h – 852 hrs	Failed
Case 5	100°C – 99.6 Gy/h – 988 hrs	Failed

The acceptance criteria can be used for condition-based EQ, in which cables are periodically tested after they are installed by comparing the EAB value with the management values. For condition-based EQ, management values should be determined based on the result of the LOCA test of cable samples that are pre-aged by low accelerated simultaneous thermal and radiation ageing test. If management values are set using high accelerated pre-ageing test, it will be non-conservative value.

However, the condition indicator of EAB is obtained using a destructive test. Condition indicators that can be obtained non-destructively are desirable for condition-based EQ. Other non-destructive condition indicators such as indenter modulus can be used as an alternative condition indicator if their correlation with the condition indicator of EAB is confirmed. The correlation between the indenter modulus of improved type and indicator of EAB has already been confirmed in the ACA project and its outline is described in Section 4.3.4.

4.3.7 Cable deposits

In order to carefully monitor insulation degradation, it is recommended to have cable samples exposed to actual nuclear plant service environment. A novel idea, practiced by several plants, is to deposit additional pieces of cable at service locations and expose them to the actual service environment. Even for operating plants that have not made cable deposits should consider keeping good sections of old cables in plant locations when cable replacements are done.

Portions of these cables can in turn be conveniently made available for ongoing qualification, destructive examination and other requalification efforts based on the evolving needs. Factors to be considered for cable deposits:

- anticipated plant life including potential life extension;
- critical locations were the cables are likely to be subjected to extreme environmental effects;
- samples to represent all the cables that have potential age-sensitive locations;
- hot spot locations for radiation and temperature;
- adequate length of cable with due consideration for plant life, continuing qualification and research that could involve destructive examination;
- documentation that supports traceability to production, historic test records, stock no. (sample labels, etc.);
- appropriate protection of cable ends;
- impeccable sample;
- convenience for retrieving the required samples;
- self-heating that raises the temperature should be considered while evaluating the cable insulation degradation (since deposited cables are not in service).

4.3.8 New plants

The following topics are specially selected because of their applicability to new plants. Considerations of these aspects in the plant design phase would lead to significant benefits.

- *Cable deposits*. It is recommended to have cable samples exposed to actual nuclear plant service environment.
- *Area monitoring for environmental conditions*. In order to ensure the qualification life or to predict the life of EQ cable, monitoring of a cable environment is essential.
- *Cable condition monitoring (base line and periodic)*. Baseline analysis of installed cables is particularly suited for new or good cables to better assess degradation or damage.
- *Shielded jackets for cables*. Power cables for new plants should consider using cables with a continuous metallic shield.
- *Environmental qualification method*. It is important that the EQ test be performed with a method which can appropriately simulate ageing in the normal operation.

Chapter 5: Recommended future activities

5.1 Lessons learned from the SCAP process and potential follow-up activities

The working groups have brought together representatives of regulators, operators, vendors and academics working in the field, and this combination has been found to be invaluable for the successful execution of the project.

The products of each working group are a database, a knowledge base and commendable practices which will support both regulators and operators. The use of the products will strengthen technical approaches to optimise ageing management in the areas of SCC and cable ageing. The products are useful tools and documents for technical experts including the younger generations of engineers, and they are of even greater use when they are continuously enhanced and updated.

The working process of SCAP has also provided an important example to demonstrate how such a challenging task can be effectively addressed and therefore could be used as a basis for other topics in ageing management. Vital elements of the working process have been the identification of priority items of common interest, the assignment of a dedicated project co-ordinator, chairperson and clearing house with expert knowledge and lead organisations providing input to start the discussion and giving orientation.

The international knowledge that was collected in this project should help industry organisations to revise existing standards or develop new standards. IEEE and IEC have just started such activities and this foundation should help the regulators to implement requirements and programmes to support safe long-term operation of nuclear power plants.

The US is launching new research in light of the lessons learned from this project. The impact of concurrent ageing, dose rate effects, manufacturing tolerance and oxygen-starved chambers for LOCA tests became areas of interest that require further research.

Research efforts are also ongoing in other countries. An important aspect of the research will be international co-operation such as has been developed within this project, during which regulators, operators, vendors and academics have collaborated in an open and productive manner.

5.2 SCC Working Group and Cable Working Group

5.2.1 SCC Working Group

The SCAP SCC Working Group has demonstrated that an international project to develop a common source and understanding of information on stress corrosion cracking is an invaluable method to share important experience in a timely manner. The products of the working group are an event database, a knowledge base and a report describing the commendable practices that the group recommends in the ageing management and safe long-term operation of stress corrosion cracking. The working group has brought together representatives of regulators, operators, vendors and academics working in the field and this combination has been found to be invaluable for the successful execution of the project. Members of the working group have agreed that the efforts expended should continue for a number of reasons.

It is important that the knowledge base be maintained, as this is a central source of information concerning both general information of the different approaches that member countries have for ageing management and specific information about stress corrosion cracking. This is a unique collection of data that will provide an invaluable source of readily accessible information and its transfer, and the training of the next generation of engineers. Several of the stress corrosion cracking mechanisms are the subject of ongoing research, as are the inspection and mitigation techniques. This ongoing work should be added to the knowledge base as it becomes available.

The SCAP SCC event database is a development of the OPDE database. It covers more types of components and the information about stress corrosion cracking is much more extensive and detailed. The database is not complete and new events will occur and need to be added. The approach taken by which the working group concentrated upon ensuring that the representative events were as complete as possible means that for many purposes it is necessary to expand the information for many of the other events, including those transferred from OPDE. It is important that this collection of operational experience continue.

After an investigative meeting together with the OPDE Programme Review Group it was agreed that the two groups should work together to establish a new project under the CSNI. This new project should combine the best of the two projects with respect to content, scope and working methods. Since OPDE has an agreement which expires at the end of May 2011, the interim period after the completion of the SCAP project will be used to define the new project in more detail whilst maintaining the important network of contacts established within the SCAP SCC Working Group.

5.2.2 Cable Working Group

The current report along with the web-based database and commendable practices has created an encyclopaedia on cable based on the historic work of participating countries. Continued research and developments should take place with interested participation to further increase the safe long-term operational availability and reliability of cable systems.

The next phase of work should involve the collection of more data, advancement of the knowledge base and the extraction of further commendable practices.

It is recommended to carry out benchmarking procedures for cable life extension with international co-operation to refine the qualification process and other areas of interest. A round robin testing should be considered to build further consistency and refinement of the process. Expanding the scope with a similar database to other electrical equipments which are commonly used in the NPP across the globe could be another valuable addition to the project.

The Cable Working Group should take into consideration the recommendations described in Section 3.3.10 (“Future advancement for the report”) to define the scope in the first meeting to proceed with the second phase.

References

[1] Nuclear Energy Agency (NEA), "Regulatory Aspects of Ageing Reactors", *1998 Committee on Nuclear Regulatory Activities (CNRA) Special Issue Meeting*, OECD/NEA, Paris, NEA/CNRA/R(99)1, March 1999.

[2] NEA, *Technical Aspects of Ageing for Long-term Operation*, OECD/NEA, Paris, NEA/CSNI/R(2002)26.

[3] NEA, *Research Efforts Related to Wire System Ageing in NEA Member Countries*, OECD/NEA, Paris, NEA/CSNI/R(2004)12.

[4] NEA, *Status Report on Developments and Co-operation on RI-ISI and NDT Qualification in OECD/NEA Member Countries*, OECD/NEA, Paris, NEA/CSNI/R(2005)9.

[5] NEA, "Better Nuclear Plant Maintenance: Improving Human and Organisational Performance", *CSNI Technical Opinion Papers*, No. 11, OECD/NEA, Paris, NEA/CSNI/R(2006)8.

[6] International Atomic Energy Agency (IAEA), *Safety Aspects of Nuclear Power Plant Ageing*, IAEA-TECDOC-540, IAEA, Vienna (1990).

[7] IAEA, "Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing", *IAEA Safety Series*, No. 50-P-3, IAEA, Vienna (1991).

[8] IAEA, "Methodology for Ageing Management of Nuclear Power Plant Components Important to Safety", *IAEA Technical Report Series*, No. 338, IAEA, Vienna (1992).

[9] IAEA, "Implementation and Review of Nuclear Power Plant Ageing Management Programme", *IAEA Safety Report Series*, No. 15, IAEA, Vienna (1999).

[10] IAEA, *Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Steam Generators*, IAEA-TECDOC-981, IAEA, Vienna (1997).

[11] IAEA, *Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Vessel Internals*, IAEA-TECDOC-1119, IAEA, Vienna (1999).

[12] IAEA, "Ageing Management for Nuclear Power Plants", *IAEA Safety Guide*, NS-G-2.12 (2009).

[13] IAEA, "Proactive Management of Ageing Management of Ageing for Nuclear Power Plants", *IAEA Safety Report Series*, No. 62, IAEA, Vienna (2009).

[14] IAEA, "Safe Long-term Operation of Nuclear Power Plants", *IAEA Safety Report Series*, No. 57, IAEA, Vienna (2008).

[15] IAEA, *IAEA Safety Glossary Terminology Used in Nuclear Safety and Radiation Protection*, 2007 ed., IAEA, Vienna (2007).

[16] Taylor, James M., *Completion of the Fatigue Action Plan*, SECY-95-245, United States Nuclear Regulatory Commission (US NRC), Washington, DC, 25 September 1995.

[17] United States Nuclear Regulatory Commission (US NRC), *Fatigue Analysis of Components for 60-year Plant Life*, NUREG/CR-6674 (PNNL-13227), June 2000.

[18] Thadani, Ashok C., "Fatigue Evaluation of Metal Components for 60 Year Plant Life", Memorandum from Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, to William D. Travers, Executive Director for Operations, Closeout of Generic Safety Issue 190, US Nuclear Regulatory Commission, Washington, DC, 26 December 1999.

[19] National Science and Technology Council Committee on Technology, *Review of Federal Program for Wire System Safety*, United States White House, November 2000.

[20] IAEA, *Assessment and Management of Ageing of Major NPP Components Important to Safety*, IAEA-TECDOC-1188, IAEA, Vienna (2000).

[21] NEA, *Research Efforts Related to Wire System Ageing in Member States*, OECD/NEA, Paris (2002).

[22] US NRC, *Environmental Qualification of Low-voltage Instrumentation and Control Cables*, RIS 2003-09 (2003).

[23] Institute of Electrical & Electronic Engineers (IEEE), *Qualifying Class 1E Equipment for Nuclear Power Generating Stations*, IEEE-323, IEEE, New York (1974, 1983).

[24] IEEE, "Type Test of Class 1E Electric Cables, Filed Splices and Connections for Nuclear Power Generating Stations", IEEE-383, IEEE, New York (1974).

[25] US NRC, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants", Code of Federal Regulations (CFR), Title 10, Part 50.49.

[26] International Electrothecnical Commission (IEC), *Qualification of Electrical Equipment of the Safety System for Nuclear Power Plants*, IEC 780, IEC, Geneva (1984).

[27] IEC, "Qualification of Electrical Equipment of the Safety System for Nuclear Power Plants", IEC 60780, IEC, Geneva (1997).

[28] IEC, *Guide for Determining the Effects of Ionizing Radiation on Insulation Materials*, IEC 544, IEC, Geneva (1977, 1985, 1991).

[29] Kerntechnischer Ausschuss (KTA, German Nuclear Safety Standards Commission), *Ensuring the Loss-of-coolant-accident Resistance of Electrotechnical Components and of Components in the Instrumentation and Controls of Operating Nuclear Power Plants*, KTA 3706, KTA, Germany (2000) (in German).

[30] Svenska karnkraftverken, *Environmental Qualification for Accident Conditions*, KBE EP-154, Sweden (1996).

[31] US NRC, *Assessment of Environment Qualification Practices and Condition Monitoring Techniques for Low Voltage Electric Cables*, NUREG/CR-6704 (2001).

[32] SCK•CEN (Belgian Nuclear Research Centre), *Topical Day on Cable Ageing in Nuclear Environments*, SCK/CEN-BLG-969, Mol, Belgium (2003).

[33] Yamamoto, T., T. Minakawa, *The Final Report of the Project of Assessment of Cable Ageing for Nuclear Power Plants*, JNES-SS-0903, Japan Nuclear Energy Safety Organization (JNES), Tokyo, Japan (2009).

[34] Sindelar, Robert L., "Nuclear Radiation", presentation paper of short course of Introduction to Nuclear Fuel Cycle Chemistry of Vanderbilt University School of Engineering, Nashville, TN, USA, 4-6 August 2009.
www.cresp.org/NuclearFuelCycleCourseII/Presentations/05_SINDELAR_CRESP_%20Sindelar_Bob_August_2009.pdf

[35] European Utility Requirements (EUR), European Utility Requirements for LWR Nuclear Power Plants (website), Revision C, Vol. 2, Chapter 2.6: Material-related Requirements, *www.europeanutilityrequirements.org*.

Appendix 1: Example of ageing management programmes for stress corrosion cracking

This appendix describes the Japanese approach to the ageing management of stress corrosion cracking. The tables describe, for each stress corrosion cracking mode, appropriate methods for preventative maintenance, the inspection, mitigation, repair and replacement, safety assessment and research based on the information in the knowledge base.

Introduction

Many stress corrosion cracking (SCC) events occurred in different structures, systems and components (SSC) of nuclear power plant as early as the 1970s and have continued to the present time. The causes of these events were sensitisation of material to high heat input, local high residual stress, surface finishing and hardness, associated with certain environments (such as high-temperature water). For these reasons, it is necessary to carry out maintenance activities, inspections with the appropriate intervals, monitoring and collection of relevant data, preventive maintenance/mitigation, repair/ replacement and safety assessment to minimise the occurrence of future events.

Therefore, knowledge should be extracted from the database (operational experience and recent findings) by analysing and evaluating the data from the viewpoint of the implementation of appropriate ageing management and maintenance activities beneficial to both regulatory authorities and operators.

To achieve the goal for both operator and regulators, a well-grounded technical information basis (TIB) is needed. The ageing management and maintenance activities for SCC are to be performed under the following three stage considerations:

1) *Ageing management from the early stage of nuclear power plant operation*. Effective ageing management can be achieved if performed regularly from the early stage of operation.

2) *Ageing management every 10 years within the framework of PSR*. The subject of investigation should include those ageing mechanisms for which mid-term trend monitoring is effective.

3) *Ageing management before the operation for 30 years and following every 10 years*. The ageing mechanisms which require the ageing management technical assessment (AMTA) shall be identified and specified. SCC should be managed in accordance with trend monitoring which can be performed through inspections, and thus the ageing management programme (AMP) should be established based on such results.

As shown in Figure 2.1-1 it is first necessary to define an ageing management programme (AMP) for the countermeasure of SCC by selecting the structures, systems and components (SSC), determining the safety-important specific SCC mechanisms, and determining the criteria for evaluating if the intended safety function of SSC are maintained. It is necessary to perform evaluations consistent with long-term operation (*e.g.* 40 or 60 years), and to include the evaluation results in a preventive maintenance plan (or AMP contained inspection, repair/replacement, monitoring/surveillance).

In order to perform optimal preventive maintenance for SCC, utilities should establish a long-term maintenance management plan (LMP). For safe long-term operation a well-grounded TIB is needed for each of the SCC mechanisms. To construct it, comprehensive activities, including the results of research and development, establishment of codes and standards, and consolidation of maintenance activities (PDCA), need to be continuously reviewed.

There are several factors which may hinder effective ageing management and may lead to either unexpected or premature ageing (*i.e.* IGSCC, PWSCC). These factors or weaknesses need to be identified and addressed in a proactive approach by experts.

As shown in Figure 1.5-1 and Table 2.1-1, effective ageing management for SCC throughout the service life requires the use of a systematic approach to manage ageing. It is helpful to use the basic idea of a framework for co-ordinating all ageing management programmes and activities based on the understanding of preventive maintenance, mitigation, repair and replacement, inspection/monitoring/ qualification, safety assessment, and research and development on the ageing mechanism and/or ageing effects of the SSC.

As described in Section 1.5, the IAEA Safety Standard Series No. NS-G-2.12 "Ageing Management for Nuclear Power Plants" explains how to extract and make commendable practices equal to appropriate ageing management programmes (AMP) for SCC.

The guide outlines nine generic attributes of an effective ageing management programme: scope of the ageing management programme, preventative actions to minimise control and ageing mechanism, detection of ageing effects, monitoring and trending of ageing effects, mitigation of ageing effects, acceptance criteria, corrective actions, operating experience feedback and feedback of research and development, and quality management.

It is also important to establish the TIB to establish commendable practices from knowledge base data which should include at least the following five items, excluding the international harmonisation of quality management not within the scope of SCAP.

- inspection/monitoring/qualification;
- preventive maintenance/mitigation;
- repair/replacement;
- safety assessment (flaw evaluation/fracture);
- R&D (initiation/crack growth/fracture).

Table A-1: Commendable practices for IGSCC of stainless steel in Japan

Items	Commendable practices
Ageing management	IGSCC is a degradation mechanism of which a number of events were observed tens of years in the pasts. Also, this IGSCC mechanism is not sufficiently understood yet at present. Therefore, it is required to perform systematic ageing management combining inspection and preventive maintenance from the early stage operation of nuclear power plant. The ageing management programmes (AMP) should be developed based on the updated and evaluated previous inspection data and research results. It is recommended to select the appropriate preventive activities described in this table, taking into account related regulation, the evaluated structural integrity on the maintenance of SSC functions, and assumption of the safe long-term operation. In the core internals, it is necessary to consider the influence of the irradiation in addition to the material, the stress, the environment. Because IASCC susceptibility appears over a threshold of the cumulative neutron irradiation fluence, it is recommended to refer the tables of commendable practices for irradiation-assisted corrosion cracking (IASCC) on preparing or revising of specified ageing management programme (AMP).
Preventive maintenance/ mitigation	It is commendable to improve more than two factors among three factors as shown in material, stress and the environment on performing repair weld, mitigation and replacement. Special attention should be paid to the reason why IGSCC is observed at location with high hardness due to surface finishing of stainless steels, by grinding, machining and welding shrinkage of the inside diameter near the HAZ. Material: • For BWR components, low-carbon grades austenitic stainless steel containing up to 0.03% carbon is selected as material. • For PWR components, material with a carbon content ranging between 0.03% and 0.05%, and weld metal with a maximum carbon content of 0.035 wt.%. • For BWR and PWR, certain amount of delta ferrite should be included in weld metal and cast austenitic stainless steel (CASS) (*e.g.* in minimum ferrite of 7.5% in weld metal for PWR, 5.0% for BWR). Residual stress: • Induction heating stress improvement (IHSI). • Peening (shot peening, US peening, water jet peening, laser peening). • Polishing. • HSW (heat sink welding). Water chemistry: • Hydrogen water chemistry (HWC), noble metal chemical addition (NMCA) (It is recommended to pay an attention to demonstrate the effectiveness of HWC and NMCA to high dose material experiments). Because IASCC susceptibility appears over a threshold of the cumulative neutron irradiation fluence, it is recommended to refer to the tables of commendable practices for irradiation-assisted corrosion cracking (IASCC) for preparing or revising of specified preventive maintenance activities.
Inspection, monitoring, qualification	It is recommended to select appropriate inspection to maintain the safety function of the SSC: • *Specified inspection*. VT, UT or ECT is recommended for inspection as required by the range, method, commencing time and frequency under the considering of the assumed IGSCC initiation and propagation in service operation. • *General inspection*. VT is recommended for inspection as checking method to supplement the individual above-specified inspection just in case from the viewpoint of depth protection or double checking. The methods to detect the SCC include visual inspection capable of discriminate 0.0254 mm wires, liquid penetrant inspection, phased array ultrasonic inspection (detection and sizing). Because IGSCC propagates along heat affected zone (HAZ) or the weld line, it needs to consider the attenuation of UT signal when the sizing of the cracks by UT inspection.

Table A-1: Commendable practices for IGSCC of stainless steel in Japan (*cont.*)

Items	Commendable practices
Safety assessment (flaw evaluation/ fracture)	For example, when IGSCC is detected in the core internals, it is recommended to conduct flaw evaluation (crack growth evaluation and fracture evaluation) in accordance with the applicable regulatory guidelines and to confirm that the components satisfy the required safety functions throughout the pre-determined evaluation period. • *Crack growth evaluation*. Based on the crack growth rate of the relevant material/environment combination, the crack size at the end of the pre-determined evaluation period should be estimated. IGSCC of BWR occurs at the grain boundaries of the HAZ, so that it needs to take into consideration the high propagation rate for hardening HAZ on performing the crack growth evaluation. • *Fracture evaluation*. Determine whether or not the estimated cracking size at the end of the evaluation period could reach the allowable limit which might cause fracture of the component.
Repair and replacement	*Same as preventive maintenance/mitigation.* Many of the considerations for preventative maintenance and mitigation are applicable for repair and replacement. The difference between these two concepts is that the latter is carried out after a crack has been detected and the former as the name implies is carried out before a crack is found in order to try and prevent stress corrosion cracking. It is recommended that the validity and reliability of the methods in terms of the long-term operation be verified by a third-party organisation: • To select proper methods based on investigating the structures, workability, etc. • It is recommended to apply one or more mitigation procedures to the heat affected zone with high tensile strength by repair weld. • To pay a special attention to helium content in stainless steel induced due to neutron irradiation for a long-term operation.
R&D	It is recommended to sophisticate and understand the SCC phenomena at SSC in the operation plants in precision manner and to study and prepare inspection method and the reasonable frequency. The research and development themes are as follows: • To obtain the SCC initiation data for materials machined out or used at operating plants and their database under the water chemistry at operating plants. • To preparation and development of codes and standards on evaluation methods of SCC initiation and growth. • To research the evaluation of effectiveness of optimised PWR and BWR water chemistry.

Table A-2: Commendable practices for IGSCC of Ni-based alloy including PWSCC (PWR) in Japan

Items	Commendable practices
Ageing management	IGSCC is a degradation mechanism that a number of events tackled tens of years in the past. Also, the IGSCC mechanism is insufficiently clear. Therefore, systematic ageing management combined with inspection and preventive maintenance are required to be performed from the early stage of nuclear power plant operation. The ageing management programmes (AMP) should be developed based on the update and evaluated previous inspection data and research results. It is recommended to select the appropriate preventive activities from the following items in this table based on the applicable regulation, the structural integrity evaluation on the maintenance of SSC functions, and assumption of the safe long-term operation.
Preventive maintenance/ mitigation	It is recommended to improve more than two factors among three factors as shown in material, stress and the environment to do repair weld, mitigation, and replacement, as the IGSCC mechanism of Ni-based alloy including PWSCC is not sufficiently clear. Material: • Alloy 690 base metal and Alloy 52 and 152 weld metal are resistant for PWSCC. These materials should be employed for replacing materials or repair weld or cladding materials to isolate the existing Alloy 600. Residual stress: • Peening (shot peening, US Peening, water jet peening, laser peening). • Outer surface irradiated laser stress improvement process (L-SIP). • Polishing. Water chemistry: • Optimisation of dissolved hydrogen concentration. Peening, outer surface irradiated laser stress improvement process (L-SIP), surface residual stress improvement by polishing are recommended to reduce residual stress caused by welding or to improve tensile stress on material surface caused by severe plastic deformation during fabrication into compressive stress.
Inspection, monitoring, qualification	It is recommended to select appropriate inspection to maintain the safety function of the SSC: • To identify the cause of cracking, it is desired to observe crack morphologies by visual inspection, ECT, printed replicas and SUMP observation and to compare them with those observed in past events. • To detect cracking, bare metal inspection within a required period is recommended in addition to a leak test conducted during every refuelling outage. The methods to detect the SCC include visual inspection capable of discriminate 0.0254 mm wires, liquid penetrant inspection, phased array ultrasonic inspection (detection and sizing). Because PWSCC propagates along columnar (dendrite) microstructure in the weld metal, when the sizing of the cracks by UT inspection, the attenuation and scattering of UT signal due to grain of weld metal need to be considered.
Safety assessment (flaw evaluation/ fracture)	When PWSCC is detected, flaw evaluation (crack growth evaluation and fracture evaluation) should be conducted in accordance with the applicable regulatory guidelines and to confirm that the components satisfy the required safety functions throughout the pre-determined evaluation period. • *Crack growth evaluation*. Based on the crack growth rate of the relevant material/environment combination, the crack size at the end of the pre-determined evaluation period should be estimated. • *Fracture evaluation*. Determine whether or not the estimated cracking size at the end of the evaluation period could reach the allowable limit which might cause fracture of the component.
Repair and replacement	*Same as preventive maintenance/mitigation.* Many of the considerations for preventative maintenance and mitigation are applicable for repair and replacement. The difference between these two concepts is that the latter is carried out after a crack has been detected and the former, as the name implies, is carried out before a crack is found in order to try to prevent stress corrosion cracking. It is recommended that the validity and reliability of the methods in terms of the long-term operation are verified by a third-party organisation. Especially, if applying Alloy 690 and its weld metals, Alloys 152/52 as a countermeasure, special consideration should be paid to the susceptibility to micro-cracking due to local residual stresses along the grain boundaries in the repair weld.

Table A-2: Commendable practices for IGSCC of Ni-based alloy including PWSCC (PWR) in Japan (*cont.*)

Items	Commendable practices
R&D	From now on, it is necessary to promote further development of evaluation methods taking conditions in the operating plants into account. It is also necessary to verify the data of effects of water chemistry on PWSCC, in order to apply the SCC countermeasures from environmental aspects to the operating plants. It is necessary to verify the long-term integrity for Alloy 690 applied as countermeasures to PWSCC. The themes for the research and development are as follows: • To promote further reliability of SCC growth data and SCC growth evaluation methods. • To study evaluation of effectiveness of optimised PWR water chemistry. • To verify the long-term integrity for Alloy 690 and its weld metals. • To study the SCC mechanism for development new materials in future applications.

Table A-3: Commendable practices for IGSCC of Ni-based alloy including NiSCC (BWR) in Japan

Items	Commendable practices
Ageing management	IGSCC is a degradation mechanism that a number of events tackled tens of years in the past. The IGSCC mechanism is insufficiently clear. Therefore, systematically ageing management combined inspection and preventive maintenance are required to be performed from the early stage of nuclear power plant operation. The ageing management programmes (AMP) should be developed based on the updated and evaluated previous inspection database and research results. It is recommended to select the appropriate preventive activities from the following items in this table based on the applicable regulation, the structural integrity evaluation on the maintenance of SSC functions, and assumption of the safe long-term operation.
Preventive maintenance/ mitigation	It is commendable practice to eliminate more than two factors among three factors as shown in material, stress and the environment to do repair weld, mitigation and replacement, as the IGSCC mechanism of Ni-based alloy including NiSCC is not sufficiently made clear. Material • To use SCC resistant Ni-based alloys, such as Alloy 82, modified Alloy 600 and modified Alloy 182. Alloy 82 contains higher Cr than conventional Alloy 182 and has shown excellent field performance. Modified Alloy 600 and 182 contain niobium (Nb) to stabilise carbon by suppressing chromium carbide precipitation especially. Stress/environment • Same as the IGSCC of stainless steels. • The recommended methods to reduce residual stress caused by welding or to improve tensile stress on material surface caused by severe plastic deformation during fabrication into compressive stress are peening, surface residual stress improvement by polishing.
Inspection, monitoring, qualification	It is recommended to select appropriate inspection to maintain the safety function of the SSC: • To identify the cause of cracking, it is desired to observe crack morphologies by visual inspection, ECT, printed replicas and SUMP observation and to compare them with those observed for past events. • To detect cracking, visual inspection in addition to leak test conducted during every refuelling outage. Because NiSCC propagates along columnar (dendrite) microstructure in the weld metal, when the sizing of the cracks by UT inspection, the attenuation of UT signal should be considered.
Safety assessment (flaw evaluation/ fracture)	It should conduct flaw evaluation (crack growth evaluation and fracture evaluation) in accordance with the applicable regulatory guidelines and confirm that the components satisfy the required safety functions during the pre-determined evaluation period. • *Crack growth evaluation*. Based on the crack growth rate of the relevant material/environment combination, the crack size at the end of the pre-determined evaluation period should be estimated. • *Fracture evaluation*. Determine whether or not the estimated cracking size at the end of the evaluation period could reach the allowable limit which might cause fracture of the component.
Repair and replacement	*Same as preventive maintenance/mitigation.* Many of the considerations for preventative maintenance and mitigation are applicable for repair and replacement. The difference between these two concepts is that the latter is carried out after a crack has been detected and the former, as the name implies, is carried out before a crack is found in order to try to prevent stress corrosion cracking. • It is recommended that the validity and reliability of the methods in terms of the long-term operation be verified by a third-party organisation. • To select proper methods upon investigating the structures, workability, etc.
R&D	• It is necessary to promote further development of reliable methods on SCC initiation, growth evaluation and integrity evaluation. It is necessary to understand and manage the SCC phenomena in operating plants in a precise manner and to prepare and study rationalised inspection frequency. The following themes are desirable: • To develop modelling and UT simulation technology on SCC initiation and its propagation. • To study an evaluation of SCC propagation behaviour of weld fusion line of dissimilar weld joint between ferrite steel and Ni-based alloys.

Table A-4: Commendable practices for irradiation-assisted corrosion cracking (IASCC) in Japan

Items	Commendable practices
Ageing management	Irradiation-assisted stress corrosion cracking (IASCC) has less experience for structural materials of components in nuclear power plants. IASCC is characterised by the threshold of irradiation level related to susceptibility. There are increasing concerns that IASCC might occur in the high fluence if no countermeasure~~s~~ could conducted. Susceptibility of IASCC is induced by the accumulation of neutron irradiation to austenite stainless steels with normally non-susceptible and IASCC has high time dependency compared to other SCC mechanisms. Since these changes in the material and stress due to neutron irradiation are all cumulative, IASCC should be under the management as an irradiation level dependent, that is, a time-dependent degradation phenomenon. IASCC also requires structural evaluation since material and stress continuously and complicatedly changed with time and also the allowable level changed due to the irradiation. The inspection programme and interval should take all these factors into account. Therefore, systematically ageing management combined inspection and preventive maintenance are required to be performed from the early stage of nuclear power plant operation. The ageing management programmes (AMP) should be developed based on the updated and evaluated previous inspection database and research results. It is recommended to select the appropriate preventive activities from the following items in this table based on the applicable regulation, the structural integrity evaluation on the maintenance of SCC functions, and assumption of the safe long-term operation.
Preventive maintenance/ mitigation	**Preventive maintenance** *PWR* Preventive maintenance measure/mitigation with respect to material, stress and environment aspect, It is important to take into account that the degradation is cumulative nature, *i.e.* time-dependent nature. The possibility that any specific locations would reach the threshold value for IASCC susceptibility during the lifetime should be evaluated. • Approximately 1×10^{25} n/m^2 or more (E > 0.1 MeV). • The threshold stress for IASCC initiation decreases as dose increases, and Type 316 cold-worked stainless steel over approximately 30 dpa showed IASCC initiation threshold, even at the relatively low stress condition approximately to 0.6 σy. As examples of preventive maintenance, all baffle former bolts were replaced, as was the whole reactor vessel internal (RVI). *BWR* Preventive maintenance measure/mitigation with respect to material, stress and environment aspect. It is important to take into account that the degradation is cumulative nature, *i.e.* time-dependent nature. To plan maintenance relevant location before cumulative neutron exposure reaches 1.51×10^{25} n/m^2 (E > 1 MeV)(at H4 weld for 60 years) to take into account the threshold IASCC susceptibilities are as follows: • Type 304 SS: approximately 5×10^{24} n/m^2 or more (E > 1 MeV). • Type SUS316 SS: approximately 1×10^{25} n/m^2 or more (E > 1 MeV). Preventive technologies to IGSCC are believed to be effective for IASCC. However, it is necessary to demonstrate the effectiveness experimentally such as the improved water chemistry and stress improvement technologies to the relevant components. It is recommended to take into account irradiation-induced stress relaxation phenomenon in the application of the stress improvement technologies to the relevant components. As concerns the application of hydrogen water chemistry (HWC), the effectiveness of HWC and NMCA to high dose material should be demonstrated by experiments. It is recommended to select proper methods by investigating the structures, working conditions, etc.

Table A-4: Commendable practices for irradiation-assisted corrosion cracking (IASCC) in Japan (*cont.*)

Items	Commendable practices
Preventive maintenance/ mitigation (*cont*.)	**Mitigation** *PWR* It is important to take into account that the degradation is cumulative nature, *i.e.* time-dependent nature, and to perform countermeasures. Examples of changes in design are as follows: • Upflow conversion. • Improving the shape of the head-shank curve of baffle bolts. • Low leakage fuel loading. *BWR* As with the preventive maintenance to IGSCC are recognised to be effective for IASCC, but it is necessary to demonstrated the effectiveness experimentally.
Inspection, monitoring, qualification	*PWR* The inspection method and interval should take the IASCC's nature into account. • Visual inspection and/or UT inspection to inspect baffle-former bolts. • Dissolved oxygen concentration, dissolved hydrogen concentration and pH are for main water chemistry. *BWR* It is recommended that IASCC should be detected in order to maintain structural integrity and required function of components. So that, it is necessary to prepare the method and range of surface inspection involving visual inspection and volumetric inspection. It is recommended to select inspection methods and time interval based on the results of structural integrity with assumed crack considering the irradiation effect on the crack growth rate and fracture toughness of the respected location. As detection methods for IASCC, the following are recommended. • Visual inspection (underwater camera): The visual inspection capable of discriminating 0.0254 mm wires can be recommended. • Liquid penetrant inspection, ultrasonic inspection, etc. As with ultrasonic inspection for the purpose of flaw sizing, phased array method is recommende
Safety assessment (flaw evaluation/ fracture)	*PWR* It is recommended for baffle former bolts to manage the allowable number of damaged bolts considering plant's operational condition such as operation years, temperature, neutron fluence and applied stress considering swelling/creep effect due to irradiation. *BWR* For safety assessment, it is necessary to take into account the irradiation effect both for material, stress, and environmental aspect. In case of flaw evaluation, it is recommended to take into account the following items: • Crack growth evaluation: Neutron fluence dependency. • Residual stress distribution: Neutron fluence dependency. • Fracture toughness evaluation: Neutron fluence dependency.
Repair and replacement	*Same as preventive maintenance/mitigation*. Many of the considerations for preventative maintenance and mitigation are applicable for repair and replacement. The difference between these two concepts is that the latter is carried out after a crack has been detected and the former, as the name implies, is carried out before a crack is found in order to try to prevent stress corrosion cracking.
R&D	Fundamental study on IASCC mechanism at the high irradiation fluence region, in order to reflect the outcomes to long-term operation and construction of LWR in next generation: • To develop the database on IASCC initiation and propagation and to prepare the simulation technology based on the results of mechanism understanding researches. • To further develop the database including data from components from ex-plants and swelling data from the components. • To develop the simulation technology for evaluation on IASCC initiation and growth based on the results of research and development mentioned above.

Table A-5: Commendable practices for external chloride corrosion cracking (ECSCC) in Japan

Items	Commendable practices
Ageing management	ECSCC is a degradation mechanism of which there were a number of events except for the nuclear energy power plants. The outside material surface of components for the reason that chlorine in the sea salt grains, high polymer products, human sweat, etc., is attached. Systematic plans are required with respect to the quantity of adhered salt particles such as sea salt particles. As this mechanism depends on halogen ion, especially chlorine ion. So, it is needed to perform periodical cleaning to avoid the chlorine concentration. It is necessary to manage a hygroscopic-moisture of not bringing the equipment to deal with halogen ion by the good practice to this mechanism, and manage needs to control chlorine concentration quantity regularly below the tolerance. In case ECSCC is detected, there is a way of evaluating the breakage depth by assuming the depth to be half of the crack length.
Preventive maintenance/ mitigation	It is necessary to keep free of halide such as the sea salt particles and keep cleanliness of component surface during manufacturing and installation stage. As for the outdoor components, which are painted or worked out waterproof measures, it is necessary to confirm the integrities of painting protections or waterproof measures.
Inspection, monitoring, qualification	Inspection manuals, inspection equipments and personnel should be defined based on codes and standards. The methods to detect ECSCC include visual inspection capable of discriminate 0.0254 mm wires, liquid penetrant inspection, phased array ultrasonic inspection (detection and sizing). As to the monitoring/trending of salt adsorption to the surfaces of piping, etc., it is necessary to periodically measure the surface-salt concentration and prescribe a proper control criteria (for example, 70 mg Cl/m^2). If the measured concentration exceeds the control criteria, surface clean-up (washing) and examination of SCC occurrence (*e.g.* VT, PT, ECT, UT) should be implemented.
Safety assessment (flaw evaluation/ fracture)	It is recommended that the crack depth be assumed to be half of the maximum crack length and the integrity of the component is evaluated using this assumption.
Repair and replacement	If the crack length and depth do not meet the safety requirements in the codes and standards, repair/replacement is necessary.
R&D	Not necessary

Appendix 2: SCAP Management Board, SCC Working Group and Cable Working Group

SCAP Management Board

Member states	Organisation	Experts
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	IRSN/DSR/SAMS/BAMM (Institut de Radioprotection et de Sûreté Nucléaire)	Marc Le Calvar
	ASN (Nuclear Safety Authority)	Rachel Vaucher
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		Kentaro Morita
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Cable Working Group

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