IAEA-TECDOC-1147

Management of ageing of I&C equipment in nuclear power plants

Report prepared within the framework of the International Working Group on Nuclear Power Plant Control and Instrumentation

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

June 2000

The originating Section of this publication in the IAEA was:

Nuclear Power Engineering Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

The IAEA does not normally maintain stocks of reports in this series. They are however collected by the International Nuclear Information System (INIS) as non-conventional literature. Should a document be out of print, a copy on microfiche or in electronic format can be purchased from the INIS Document Delivery Services:

INIS Clearinghouse
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

Telephone: (43) 1 2600 22880 or 22866
Fax: (43) 1 2600 29882
E-mail: chouse@iaea.org

Orders should be accompanied by prepayment of 100 Austrian Schillings in the form of a cheque or credit card (VISA, Mastercard).

More information on the INIS Document Delivery Services and a list of national document delivery services where these reports can also be ordered can be found on the INIS Web site at http://www.iaea.org/inis/dd_srv.htm.

MANAGEMENT OF AGEING OF I&C EQUIPMENT IN NUCLEAR POWER PLANTS
IAEA, VIENNA, 2000
IAEA-TECDOC-1147
ISSN 1011–4289

Printed by the IAEA in Austria
June 2000

FOREWORD

Experience has shown that ageing and obsolescence have the potential to cause the maintainability and operability of many I&C systems to deteriorate well before the end of plant life. An I&C ageing management strategy is therefore required to control and minimize this threat. This report gives guidance on how to develop such a strategy and provides examples and supporting information on how established and recently developed maintenance, surveillance, and testing techniques may be employed to support the strategy. In some cases, equipment refurbishment may be necessary and guidance on this subject is given in a companion publication (IAEA-TECDOC-1016, Modernization of Instrumentation and Control in Nuclear Power Plants, IAEA, Vienna, 1998).

The International Working Group on Nuclear Power Plant Control and Instrumentation (IWG-NPPCI) of the IAEA proposed in 1995 that a technical report be prepared to provide general guidelines on the management of ageing of important I&C equipment in nuclear power plants. The purpose of the report would be to guide the worldwide nuclear industry on potential effects of I&C ageing on plant safety and economy, and the means that are available to help minimize or eliminate any detrimental consequences of ageing.

In response, a consultants meeting of five experts from Finland, France, Germany, the United Kingdom and the USA was held by the IAEA in Vienna in September 1997 to exchange national experience on the subject and to discuss the possible content of the report. The group of experts was tasked with bringing together all the information that is available on I&C ageing and ageing management methods. After a thorough discussion and analysis of the available information, an extended outline of the report on the subject was produced. The purpose of the extended outline was to identify a structure of the report, bring together information available at the moment and to provide guidance in the developing of a full-scale report.

The extended outline was then elaborated by a working team into a first draft report during the period September 1997 to February 1998. This draft served as a working document during an IAEA advisory group meeting held Vienna in April 1998. In this meeting, in addition to the five original experts, advisors from the Czech Republic, India, the Russian Federation and Slovakia participated, reviewed the draft report, and made recommendations as to what material should be added or deleted. Participants of the meeting provided additional material that has been included in the report.

The final consultants meeting was held in Vienna from 28 September to 2 October 1998. This report provides a review of ageing characteristics of representative nuclear power plant instrumentation and control (I&C) equipment and describes some of the technologies and procedures that are currently available for management of ageing of this equipment.

Special thanks are due to H. Hashemian (USA) who chaired the working meetings and co-ordinated the work. V. Neboyan and A. Kossilov of the Division of Nuclear Power were the IAEA officers responsible for this publication.

CONTENTS

1. INTRODUCTION

Experience throughout the utility and process industries has shown that the increasing age of instrumentation and control (I&C) systems in older plants has the potential to cause deterioration of operability and maintainability. This problem is also shared by nuclear power plants (NPPs), many of which have been operating in excess of 20 years. The need to address this problem is now becoming more pressing as I&C systems provide vital support for the safe and economic operation of NPPs and their functions must be sustained throughout plant life.

Ageing and obsolescence problems have already been encountered and dealt with by most utilities on a case-by-case basis. However, it is now becoming clear that all I&C systems are potentially vulnerable to these phenomena and that a formal management strategy is needed to ensure that limited resources are targeted on priority cases to minimize the overall threat. Regulators are interested because a sudden degradation of I&C performance could place increasing demands on plant protection systems with possible detriment to nuclear safety.

The problems arise from two root causes, ageing and obsolescence, which has different characteristics, but are closely related. In the context of this report, ageing is defined as the process by which physical characteristics of a system, structure, or component change with time or use. Obsolescence is defined as a condition when an item or system can no longer be procured by normal means. There are two important stages in an item becoming obsolete. Obsolescence begins when a manufacturer removes it from his marketed product range. The manufacturer may continue to offer a supply of replacement items or a repair service for some time after this event. Absolute obsolescence occurs when a supplier withdraws all forms of support or service from an item.

Activities concerning NPP ageing have been in progress at the IAEA over ten years. Work in the field of safety aspects of plant ageing started in 1985 and since then the IAEA has organized a number of meetings in the field of ageing management aimed at exchanging information and documenting experiences, practices, and research. The IAEA has published recommendations on data collection and record keeping for ageing management and a report on the safety aspects of NPP ageing [1, 2]. In 1989, the IAEA initiated work on pilot studies on management of ageing of I&C cables [3]. In 1993, the IAEA organized a specialist meeting on ageing, maintenance, and modernization of I&C systems [4].

This report is solely concerned with the ageing management of I&C systems. It draws together experience from various nuclear utilities across the world, examining ageing of specific components and also ageing management techniques. This information is distilled into a suggested ageing management strategy and several practical steps are suggested. I&C ageing management is a developing field and, as yet, there is no one accepted and definitive solution. However, the increasing severity of the problem and on-going work justifies the production of this report, which documents the best current practices. Replacement and upgrading of equipment will form a part of any ageing management strategy. This topic has been considered by another IAEA working group, whose findings have been reported in 1997 [5].

Based on the I&C ageing research completed to date, four typical I&C components were selected as examples to be discussed in this report. The ageing characteristics of these components are described here and the current practice for their ageing management is identified to illustrate how I&C equipment may age, the consequences of the ageing, and how ageing may be managed by testing of the I&C equipment. The material provided in this report will be useful not only to the nuclear power industry, but also to other power production facilities and industrial processes. For example, fossil power plants are learning that better calibration, response time testing, and other maintenance efforts on process instrumentation systems can help improve plant efficiency. Furthermore, trending of instrument calibration stability and response time is proving to be an effective means of preventive maintenance and monitoring for the health, reliability, and longevity of instrumentation systems in process industries, aerospace, and other industrial applications.

The structure of this report is as follows: I&C ageing standards are discussed in Section 2, including a listing of standards related to I&C ageing, qualification, and performance monitoring.

In Section 3, current efforts in I&C ageing, and ageing management techniques are reviewed. This section gives background information on ageing research conducted to date, a listing of I&C components subject to ageing, and examples of test methods that are used for verifying the performance of electrical and I&C equipment as they age in a NPP.

Section 4 identifies the I&C components that are covered in this report and presents their ageing characteristics and test techniques that are used for their ageing management. This is followed by Section 5 in which broad-base ageing management strategies and techniques are described. The report ends with Section 6, the concluding remarks. Following the conclusion, references and bibliography are included for additional information and reading material related to the ageing of critical equipment in NPPs.

The annex contains country reports on how ageing of I&C and other critical plant equipment are evaluated and managed.

2. I&C AGEING STANDARDS

In most IAEA Member States, it is obligatory for the design and operation of nuclear power plant components, systems, and structures, to ensure that safety requirements are maintained throughout the plant's service life, including any extended life.

As of September 1998, there is no specific standard on management of ageing of nuclear power plant electrical and I&C equipment. On the other hand, many organizations in Member States are developing a technical basis for manageing ageing and in some Member States the regulators require plants to be periodically assessed to identify ageing problems. For example, the NRC has used the reports of its ageing research programmes to formulate inspection guidelines for nuclear plant I&C equipment. More specifically, the NRC's Standard Review Plan, NUREG-0800, includes a section on I&C with specific appendices that address test methods for I&C equipment such as temperature sensors [6]. That is, in Appendix 13 of Section 7 of NUREG-0800, the calibration of resistance temperature detectors (RTDs) in the safety systems of NPPs is addressed in detail. The document discusses laboratory calibration, in-plant calibration (using the RTD cross-calibration technique), and response time testing using the loop current step response (LCSR) method. There are also European Standards developed by the International Electrotechnical Commission (IEC) which address the testing of nuclear power plant RTDs and other sensors. For example, the IEC Standard 1224 is written specifically for insitu response time testing of nuclear plant RTDs using the LCSR and noise analysis techniques [7].

A large volume of reports on ageing of nuclear power plant equipment are available that can provide the foundation for many of the standards that may be written over the next few years. A listing of some of these reports is provided at the end of the report as bibliography.

There are many standards for surveillance testing and evaluation of safety related components in NPPs, including electrical and I&C components. Although these standards are not written in the context of equipment ageing, they are nevertheless very relevant in management of ageing of nuclear power plant equipment. For example, the ISA 67.06 standard provides guidelines for monitoring the health of safety related sensors in NPPs. The title of this standard is "Performance Monitoring for Nuclear Safety Related Instrument Channels in Nuclear Power Plants". This standard is being developed under the auspices of the International Society for Measurement and Control, which was formally called the Instrument Society of America or ISA. The original version of this standard was developed in the 1978 to 1984 period and was published in 1984 with the title "Response Time Testing of Nuclear Safety-Related Instrumentation Channels in Nuclear Power Plants" [8].

Ageing, qualification, and testing of nuclear plant electrical and I&C equipment have been addressed directly, or indirectly, in a number of IEEE standards. Some examples are IEEE Standards 336, 338, 323, 497 and 627 [9–13]. These standards give general guidelines on how safety related and other important plant equipment, including I&C equipment, may be qualified or tested. The qualification involves experiments to verify that safety related equipment will perform satisfactorily in post-accident environments at the end of their designed life. This is accomplished by testing of a representative sample of the equipment that is artificially aged to bring it to the end of its designed life. The artificial ageing conditions are specified based on the environments to which the equipment may be exposed in a NPP during and after a design-basis accident, and how the environment may affect the safety function of the equipment. For example, if a piece of equipment that is designed for 40 years of service, is normally used at a temperature of 20 degrees Celsius, then the qualification procedure involves ageing the equipment in a heating chamber for a period of time that simulates 40 years of operation at 20 degrees Celsius. Obviously, this experiment can not be conducted for a 40-year period. Thus, the Ahrrenius theory is used to reduce the length of the ageing experiment while simulating 40 years of life. For thermal ageing, the Ahrrenius theory is very useful for a lot of equipment, especially electrical and electronic equipment. Of course, most equipment in safety related and other NPP applications are exposed to many environments in addition to high temperature. For example, in-containment equipment such as pressure sensors, which have built-in electronics, are normally exposed not only to heat, but also to humidity, vibration, radiation, etc. Furthermore, there is the possibility for mechanical and thermal shock, temperature and pressure cycling, etc. Most of these conditions must be included in the qualification testing of equipment for nuclear service. In the IEEE standards for qualification of safety related equipment (also called Class 1E equipment), test sequences are given on how the equipment may be aged and tested to verify that it will operate as expected at the end of its qualified life in a post-accident environment.

Standards similar to the IEEE standards mentioned above exist for ageing of nuclear power plant equipment in Germany, France, and other countries. For example, a standard exists on accident resistance proof in Germany - KTA Rule 3706 [14]. The title of this standard is "Repeating Proof of the Accident Resistance of Electrical and Instrumentation and Control Components of the Safety Relevant Systems." Other relevant KTA standards are included in the bibliography section at the end of this report.

Ageing of I&C equipment for qualification testing is covered by a sequence of tests, which may be different in different countries. However, the purpose is the same. For example, a typical sequence in the French qualification approach for an I&C component may contain the following tests:

— temperature (heat, cold, and cycle);
— humidity;
— vibration;
— radiation;
— seismic.

Each of these tests covers one of the environmental stresses to which I&C equipment may normally be exposed in a NPP. Very often, the test itself is based on an international standard. For I&C equipment, the IEC has produced a number of standards. A listing of some of these standards including two other European standards is provided in Table I. Also included in Table I are two other IAEA safety documents which provide a general guidance on NPP activities relevant to the management of ageing.

TABLE I. EXAMPLES OF RELEVANT EUROPEAN STANDARDS AND DOCUMENTS ON QUALIFICATION AND AGEING OF NUCLEAR POWER PLANT COMPONENTS

IEC STANDARDS

IEC 68-2-1	Environmental testing – Part 2: Tests – Test A: cold
IEC-68-2-2	Basic environmental testing procedures. Part 2: Tests B: Dry heat
IEC-68-2-3	Basic environmental testing procedures. Test methods. Test Ca: Damp heat, stead state
IEC-68-2-6	Environmental testing - Part 2: Tests – Tests Fc: Vibration (sinusoidal)
IEC 68-2-14	Basic environmental testing procedures. Test methods. Test N: Change of Temperature
IEC 68-2-30	Basic environmental testing procedures. Test methods. Test Db and guidance. Damp heat, cyclic (12+12 hour cycle)
IEC 255-7	Electric relays. Part 7: test and measurement procedures for electromechanical all – or nothing relays
IEC 1000-4-2	Electromagnetic compatibility (EMC) – Part 4: Testing and measurement techniques – Section 2: Electrostatic discharge immunity test - basic EMC publication
IEC 1000-4-3	Electromagnetic compatibility (EMC) – Part 4: Testing and measurement techniques – Section 3: Radiated radio-frequency, electromagnetic field immunity test - Basic EMC publication
IEC 1000-4-4	Electromagnetic compatibility (EMC) – Part 4: Testing and measurement techniques – Section 4: Electrical fast transient / burst immunity test - Basic EMC publication
IEC 1069-5	Industrial-process measurement and control – Evaluation of system properties for the purpose of system assessment – Part 5: Assessment of system dependability
IEC 1131-2	Programmable controllers. Part 2: Equipment requirements and test
IEC 1812	Requirement and test
EN 60529	Degrees of protection provided by enclosures (Code IP)
EN 60068-3-3 or IEC 68-3-3	Environmental testing. Part 3: Guidance seismic test methods for equipment

IAEA DOCUMENTS

Safety Series No. 50-C-O (Rev.1)	Code on the safety of nuclear power plants: operation
Safety Series No. 50-SG-O7 (Rev. 1)	Maintenance of nuclear power plants: A safety guide
Safety Series No. 50-SG-O8	Surveillance of items important to safety in nuclear power plants: A safety guide
Safety Series No. 50-SG-O12	Periodic safety review of operational nuclear power plants: A safety guide
Safety Reports Series No. 3	Equipment qualification in operational NPPs
Safety Series No. 50-P-3	Data collection and record keeping for the management of nuclear power plant ageing: A safety practice

3. PAST AND PRESENT EFFORTS IN I&C AGEING AND AGEING MANAGEMENT

3.1. HISTORY

The interest in ageing of NPP equipment dates back to the mid-1970s. Prior to 1975, most of the R&D efforts in the nuclear power area were focused on plant design characteristics, improvements in plant operations, and ageing of only large components of the plant such as the reactor pressure vessel.

Between 1975 and 1985, the nuclear industry began to address the issue of equipment ageing that included all structures, systems and components (SSCs) that are important to the safety of the plant. In particular, a meeting was sponsored by the NRC in 1982 to exchange information and ideas on ageing of NPPs [15]. The discussions and debate that began in this meeting stimulated the NRC and others to establish a formal programme to address the ageing of all SSCs in NPPs that can affect plant safety, and identify testing and evaluation guidelines to manage the ageing of the important SSCs. As a result, the NPAR Programme was initiated by the NRC in the mid-1980s. This programme is described in NUREG-1144 that was first published in 1985 and later revised twice to reflect the comments of the nuclear industry experts [16]. The last revision of NUREG-1144 (Rev. 2) was published by the NRC in 1991. The NPAR programme had the following objectives:

— Identify the SSCs whose ageing may impact the safety of NPPs.
— Determine the ageing mechanisms of SSCs that are important to safety and the consequence of this ageing on safety.
— Identify surveillance, testing, and evaluation criteria for timely detection of ageing of safety related SSCs.
— Establish objective testing and replace schedules for SSCs to ensure that the plant will operate safely throughout its designed life in spite of ageing of SSCs.

The NPAR programme covered many components, large and small, including the electrical and I&C equipment.

Most of the NRC ageing research projects under the NPAR programme have been conducted at ORNL, SNL, Idaho National Engineering Laboratory (INEL), and other US government facilities. The NRC has also sponsored research under the NPAR Programme in private firms in the area of nuclear plant electrical and I&C equipment. Examples of private organizations which have performed ageing research on nuclear plant I&C equipment include Analysis and Measurement Services Corporation (AMS), Kalsi Engineering Company, and the Pentak Corporation. Furthermore, US nuclear utilities began their own ageing research in the mid-1980s, mostly through the Electric Power Research Institute (EPRI) [17, 18]. As a result of all of these efforts, a number of ageing research projects have been completed on a variety of SSCs including the nuclear plant electrical and I&C equipment.

Ageing research and development of ageing management strategies, procedures, and technologies have also been conducted in Germany, France, United Kingdom, Sweden, Finland, Spain, Netherlands and other countries. There have also been significant developments in the I&C equipment market which have influenced this area. In Germany, for example, manufacturers have announced that spare part availability for all existing analog components cannot be guaranteed for the time after the year 2005. Transition to digital I&C systems has therefore been envisioned and is already under way for two units in Germany and six in France. These replacements can be seen as mid-life refurbishments for plants that are about 20 years old.

Several I&C replacement projects are already under way on NPPs of Soviet design, namely in the Czech and Slovak Republics, in Hungary and in Ukraine, whereas the other mid- and eastern European countries are likely to follow in the short to medium term.

In the UK the nuclear regulators are beginning to express concern about ageing in the older plants and what may have to be done to manage the consequences of ageing. The UK nuclear site licenses require operators to conduct periodic reviews of safety cases, which must include consideration of the impact of equipment ageing question on future operations. In Germany, new standards that are being written on digital I&C also address the ageing question.

The European Community is expected to include ageing research when formulating the work that will be done in the future to improve the safety of NPPs. Furthermore, the Organisation for Economic Development and Co-operation (OECD) has addressed the nuclear plant ageing issue. In an OECD report published in 1997, a number of issues are addressed that relate directly or indirectly to the ageing question. Prepared by the Nuclear Energy Agency (NEA) Committee of OECD on Safety of Nuclear Installations, this report is entitled "Nuclear Safety Research in OECD Countries — Capabilities and Facilities" [19].

3.2. CURRENT AGEING RESEARCH

The growing interest in the subject of I&C ageing has stimulated most utilities to begin development of appropriate management strategies. However, the subject is relatively immature and the knowledge base is incomplete.

Ageing research programmes have been performed extensively in many countries. In some countries, the concern on ageing of NPPs is stronger than in other countries. As a result, many projects have been promoted in these countries and supported by the national, international, and local governments, private firms, and others, to address ageing and other safety needs of the plants. For example, in Slovakia, the problem of equipment ageing and ageing management programme is the concern of the day for the Bohunice NPP. Adding to this ageing problem is the fact that several vendors of essential plant components have disappeared since 1989. Presently, efforts are continuing at Bohunice on equipment qualification and this work is expected to end in December 1998. Current activities at Bohunice are focused on the review of existing processes and on collecting data for ageing management activities and to establish policy documents. This work is also due for completion by the end of 1998. For the second Slovak power plant on the Mochovce site, in order to successfully license the plant for operation, a number of safety measures have been implemented, including equipment qualification. Despite the fact that Mochovce is a new plant, a group was assigned to create a plant ageing management policy and to apply it right from the beginning of the plant operation.

A number of conferences have been held in the last five years on the ageing of Soviet-designed reactors. The proceedings from some of these conferences are listed in the bibliography section at the end of this report. These activities include ageing studies on both analogue and digital I&C equipment.

Plant ageing and life extension issues have been the subject of a European conference called Plant Life Management and Extension (PLIM&PLEX) which has been held biennially in Europe under the auspices of the Nuclear Engineering International Magazine of the UK. The last PLIM+PLEX conference was held in Prague, Czech Republic, in December 1997 [20]. This conference is among the few forums of this type that are left for the researchers, regulators, utilities, suppliers, and others to discuss their research efforts and express their views on plant ageing, life extension, life management, and related issues.

The Nuclear Plant Ageing Research (NPAR) programme of the NRC (see above) which was very active in the 1985–1995 period, focused predominantly on ageing effects on analogue equipment for the purpose of evaluating the important SSCs for plant life extension beyond the current design life of 40 years. The NPAR programme has slowed down in recent years and the NRC's ageing research activities have not been actively promoted beyond the work on the analogue equipment covered by the NPAR programme. There is some NRC work under way at the Sandia National

Laboratory (SNL) on the effect of smoke on digital circuits. Also, the Oak Ridge National Laboratory (ORNL) has worked for the NRC in recent years to study the effect of electromagnetic and radio-frequency interference (EMI/RFI) on digital equipment. Since most digital equipment in critical applications in NPPs are less than ten years old, their ageing is not as much of a concern today as their potential obsolescence, software qualification (often referred to as software verification and validation, V&V), environmental effects, separation, common mode failure, licensing issues, etc. As such, for digital I&C equipment, these issues are currently the focus of most activities, and ageing concerns are not a priority issue at this point with digital equipment. In the UK the Industry management committee is considering sponsoring of further work in the area of I&C ageing. UK nuclear utilities are also beginning to develop life-cycle management plans for key I&C systems. Much of this work is based on the family of I&C life-cycle management methodologies developed by the Electric Power Research Institute (EPRI).

Table II gives a listing of most of the electrical and I&C equipment on which ageing research has been conducted, is under way, or may be needed in the future. Representative references on the ageing research that has been completed are provided in the bibliography at the end of this report.

3.3. AGEING MANAGEMENT TECHNOLOGIES

Some of the ageing research projects have included development efforts to provide ageing management technologies and guidelines. Table III provides a summary of testing and evaluation tasks that can be used for maintenance and management of ageing of electrical and I&C equipment in NPPs. Some of these techniques are described later in this report.

A number of ageing management techniques that have been developed during ageing research projects are actively used in NPPs for management of ageing of electrical and I&C equipment. Although most of these techniques are described and used in the context of plant maintenance, some techniques were developed under ageing projects. For example, the motor current signature analysis (MCSA) Technique was developed under an ageing research project [21, 22]. This technique is used for testing motor operated valves (MOVs) in NPPs (see Annex).

Another example of an ageing management technique is a cable testing system that was originally developed under an ageing research project sponsored by the NRC. The EPRI has also developed cable testing equipment as a result of their cable ageing research. Referred to as cable indentor, the EPRI's equipment measures the ductility of cables to determine if the cable insulation has hardened due to heat, radiation, and other environments in NPPs [18].

On the ageing of sensors and related equipment, the NRC has sponsored a number of research projects. In addition to ageing research, these projects have resulted in the development of a number of ageing management techniques for electrical and I&C equipment in NPPs, including cables, connectors, RTDs, thermocouples, and transmitters. Among these developments were testing based on the noise analysis principle to measure the dynamic performance of pressure, level, and flow transmitters as installed in NPPs, as well as detection of voids and blockages in pressure sensing lines. Also, the noise analysis technique was validated for condition monitoring and preventive maintenance of reactor internals [23]. The annex includes a description of some of these techniques.

Also developed under an ageing research programme, is a new technique that was validated for on-line verification of calibration of process instrumentation channels in NPPs [23]. This method is applicable to pressure, level, and flow transmitters, RTDs, thermocouples, neutron detectors, and other sensors and can include not only the sensors, but also the rest of the instrumentation components in an entire channel. The calibration of RTDs and thermocouples can also be tested remotely using a method called the cross-calibration technique which is being implemented in many NPPs around the world (see Annex). This technique is discussed in the NRC's Standard Review Plan, NUREG-0800 [6]. Also mentioned in NUREG-0800, is the Loop Current Step Response (LCSR) method that is

TABLE II. EXAMPLES OF I&C EQUIPMENT VULNERABLE TO AGEING DEGRADATION ON WHICH AGEING RESEARCH HAS BEEN CONDUCTED, IS UNDER WAY, OR MAY BE NEEDED

Component	Status of ageing research		
	Work conducted	Work under way	Work needed
SENSORS			
RTDs	✓		
Thermocouples	✓		
Pressure transmitters (Including level and flow)	✓		
Pressure switches			✓
Sensing Lines	✓		
Thermowells	✓		
Neutron detectors	✓		✓
Ultrasonic level measurement			✓
Capacitance level measurement			✓
Conductivity Probe			✓
CABLES			
I&C cables	✓		
Power cables	✓		
Connectors/penetrations/terminal blocks	✓		
Cable sleeving	✓		
ELECTRONICS/ELECTRICAL			
Relays, relay logic microswitches	✓		
Circuit breakers	✓		
Transformers	✓		
Switches	✓	✓	
Valves actuators	✓		
Motors	✓		
Capacitors	✓	✓	
Resistors, diodes, transistors, integrated circuits	✓		✓
Fire detectors	✓	✓	
Analog & digital electronics		✓	
Potentiometers	✓		
Printed circuit boards	✓	✓	
Elastomers: Anti-vibration	✓		
Batteries, battery chargers	✓		
Control room equipment (displays)			✓

TABLE III. EXAMPLES OF AGEING MANAGEMENT TECHNIQUES FOR NUCLEAR PLANT I&C EQUIPMENT

1	Calibration (manual calibration, on-line calibration, cross-calibration)
2	Response time testing
3	Vibration analysis and noise analysis
4	Condition monitoring
5	Trending
6	Record keeping
7	Inspections
8	Maintenance programmes
9	Surveillance, periodic testing, diagnostic testing
10	In situ testing and on-line performance measurements
11	In-service inspections
12	Channel checks
13	Operational feedback
14	Environmental monitoring
15	Replacement
16	Spares management
17	Supplier's QA
18	Periodic safety reviews
19	Self/auto testing
20	Accident resistance proof

used by the nuclear power industry in over 100 NPPs worldwide for in situ response time testing of RTDs. This method has been in use in the nuclear industry for 20 years and has revealed RTD and thermowell problems due to inadequate design, installation problems, and ageing effects. In pressurized water reactors (PWRs), in particular, the response time of primary coolant RTDs is important to safety and is usually measured once every fuel cycle to comply with the regulatory requirements, as well as the plant technical specifications [24].

The LCSR method provides the actual in-service response time of nuclear plant RTDs accounting for all installation and process condition effects on response time. These effects include the thermowell, the air gap in the thermowell, the fluid flow rate, the process temperature, and other conditions that can influence the RTD response time. The LCSR test is performed on RTDs remotely from the control room area while the plant is operating.

For preventive maintenance and ageing management of reactor internals, detection of core flow anomalies, loose parts monitoring, and identification of shaft cracks in pumps, the noise analysis technique has proven very useful and effective in NPPs. Although the noise analysis technique was not originally developed as an ageing management tool, it is a very effective method for both preventive maintenance and ageing management of sensors, transmitters, and other instruments, as well as heavy equipment. The method is being used in the context of preventive maintenance and ageing management in many plants now. In Germany, for example, the noise analysis technique is used in most plants to meet the KTA 3204 requirements for measurement of motion of reactor internals, loose parts monitoring, etc. [25]. In some cases, the German plants perform the noise tests as often as every three months while the plant is at normal operating conditions [26, 27].

In France, some of the PWR plants are equipped with noise recording equipment to sample the data, which are stored on magnetic tapes and other media for later retrieval in case of a problem. The noise information that was taped in the mid-1980s was instrumental in identifying the root cause of a problem that caused several trips of a French reactor during load following. Using the noise analysis technique, Electricité de France (EdF) traced the problem to faulty pressure transmitters due to clogging of the sensing lines which bring the pressure signal from the process to the transmitter. This problem later occurred in other French plants and was quickly identified based on EdF's first experience and the availability of the noise data.

In Germany, all research and development (R&D) in the field of electrical and I&C component ageing in NPPs is performed by technical service organizations (TÜV) and by the manufacturers, mostly on order of the licensing authorities and the utilities, the latter through their association that is called VGB. In addition to loss of coolant accident (LOCA) resistance proof tests (which are mandatory for all safety relevant components), on-line and recurrent testing, R&D is performed on issues of specific importance like spray testing on terminals and ageing of cables. In Germany, ageing of electrical and I&C components are not regarded as a first-line-issue, because the maintenance strategy requires replacement in time; e.g., before ageing phenomena become important.

4. COMPONENTS COVERED

Four components representing I&C systems of NPPs were selected as examples to write about in this report. These components are: sensors, electronics, relays, and cables and connectors. For each of these components, a separate section is devoted in this section. Each section covers some or all of the following as applicable: (1) environmental conditions that affect sensor performance as they age; (2) ageing affects on performance; and (3) ageing management methods that are available for verifying the performance of the component.

4.1. SENSORS

Temperature sensors, pressure transmitters (including level and flow transmitters), and neutron detectors constitute most of the important I&C sensors that are used in NPPs. As such, this section is devoted to these sensors.

4.1.1. Temperature and pressure sensors

4.1.1.1. Environmental condition effects

Nuclear plant temperature and pressure sensors are subject to a number of stressors (environments) that can cause degradation in both the calibration and response time of the sensors. Examples of these stressors and how they may affect sensor performance are as follows:

Heat. Long term exposure to heat affects material characteristics. For example, heat can cause insulation breakdown in RTDs and thermocouples. In pressure transmitters, long term exposure to heat is not a major problem for the sensing cell, but it can affect the electronics in the transmitter housing. The heat can dry out the transmitter seal and allow moisture into the electronics, causing performance degradation.

Humidity. Humidity reduces insulation properties, causes rusting of metallic components, creates shunts or short circuits, and can cause a variety of other problems which can affect the performance of sensors.

Temperature cycling. Temperature cycling causes expansion and contraction of materials producing stress and strain in sensors or their materials, affecting both the performance and the longevity of the sensors.

Vibration and mechanical shock. Vibration and mechanical shock produces metal fatigue in the sensor components and cold working in wires and sensor materials. Vibration and mechanical shock can displace or redistribute the sensor insulation and other materials, especially at high temperatures, and result in performance degradation.

Ionising radiation effects. Like heat, radiation affects material properties and changes sensor output characteristics. Gamma radiation is most detrimental to equipment performance.

Chemical attack. Damage to the sheaths of cables during installation can lead to the ingress of foreign substances. This is of concern for thermocouple cables (especially minerally insulated cables) where parasitic junctions may be formed which will cause incorrect indications.

4.1.1.2. Ageing effects on sensor performance

Ageing affects both the steady state (calibration) and dynamic (response time) performance of sensors. For example, RTD and thermocouple seals can fail (dry out, shrink, or crack) and allow moisture into the sensor causing a reduction in insulation resistance. The low insulation resistance can result in temperature measurement errors. This error will often be temperature dependent because a low insulation resistance can change drastically with temperature. Moisture in temperature sensors can also cause noise at the output of the sensor, the magnitude of which depends on the temperature and the amount of moisture in the sensor.

The calibration of pressure sensors can change by ageing due to heat and humidity. If these stressors cause failure of the transmitter sealing materials (which protect the transmitter from the environment), and moisture enters the transmitter housing, it can cause calibration shifts and may also produce high frequency noise at the output of the transmitter. In the long run, this problem can render the transmitter inoperable or unreliable. Table IV provides a listing of ageing effects and their consequences on the performance of nuclear plant pressure transmitter. Note in particular that those transmitters in which a fill fluid, such as oil, is used for signal transmission, long term exposure to high pressure can cause the oil to leak out of the sensing cell, which can result in both a significant drift, as well as response time degradation. In some plants, oil loss has caused the response time of some transmitters to increase from less than one second to more than an hour [28].

Another performance problem in nuclear plant pressure transmitters is the clogging of sensing lines that bring the pressure signals from the process to the transmitter. Sensing lines typically have a length of 30 to 300 meters. These lines can become partially or totally blocked due to sludge, boron solidification, and other debris in the reactor coolant and result in sluggish dynamic performance in the pressure sensing system. The problems can be detected while the plant is on-line using the noise analysis technique as described in the Annex.

TABLE IV. POTENTIAL EFFECTS OF AGEING ON PERFORMANCE OF NUCLEAR PLANT PRESSURE TRANSMITTERS

Degradation	Potential cause	Affected Performance	
		Calibration	Response time
Partial or total loss of fill fluid	– Manufacturing flaws – High pressure	✓	✓
Degradation of fill fluid	– Viscosity changes due to radiation and heat		✓
Wear, friction, and sticking of mechanical linkages (especially in force balance transmitters)	– Pressure fluctuations and surges – Corrosion and oxidation		✓
Failure of seals allowing moisture into transmitter electronics	Embrittlement and cracking of seals due to radiation and heat	✓	
Leakage of process fluid into cell fluid resulting in temperature changes in sensor, viscosity changes in fill fluid, etc.	– Failure of seals – Manufacturing flaws – Rupture of sensing elements	✓	✓
Changes in values of electronic components	– Heat, radiation, humidity – Changes in power supply voltages – Maintenance	✓	
Changes in spring constants of bellows and diaphragms	– Mechanical fatigue – Pressure cycling	✓	✓

Vibration and mechanical shock can cause cold working in sensing wires, shake the sensor components loose, and break the sensor away from the process. All of these affect the sensor's performance or can cause catastrophic failures. For example, vibration causes cold working in RTD sensing elements, which result in an increase in the sensor resistance and an erroneously high temperature indication. This problem can sometimes be resolved by annealing the wire. Annealing is accomplished by heating the wire, either internally by a large electrical current, or externally by heat treating the sensor in a furnace near the maximum temperature to which the sensor can be exposed.

Vibration and thermal shock can also affect the response time of temperature sensors. In particular, thermowell mounted RTDs and thermocouples respond properly only if they are fully inserted into their thermowell. Experience in NPPs, as well as laboratory research, has shown that if an RTD or a thermocouple has even a millimeter of air gap in its sensing tip, the response time can increase drastically. For this and other reasons, RTDs and thermocouples are often mounted, secured, and kept in thermowells by high-tension spring loading and other mechanisms to ensure a tight fit.

Also, vibration and thermal shock are said to cause redistribution of air gaps within the insulation materials in the sensors, resulting in response time changes. Sometimes the response time increases, and in other times it decreases depending on how the vibration or thermal shock has affected the sensor internals, especially in the presence of high temperatures involved in NPPs.

4.1.1.3. Ageing management methods for temperature and pressure sensors

The performance of temperature and pressure sensors in NPPs is dependent predominately on their calibration accuracy and response time. Therefore, ageing management of temperature and pressure sensors is accomplished by periodic calibration and response time testing. To facilitate this work, new methods have been developed to verify sensor calibration and response time while the plant is on-line. For pressure transmitters, blockages in sensing lines can cause a delay in detecting a change in process pressure, level, or flow. Therefore, on-line tests using noise analysis have been developed to detect any blockages.

Whilst not strictly an ageing management technique, it is also important to ensure that all items are correctly installed when they are first fitted onto the plant. This is especially important for thermocouple cable sheaths to ensure they have not been damaged to allow the ingress of foreign material. Also the installation of thermocouples and RTDs that are fitted into thermowells must be verified to ensure that the sensing junction of the sensor is at the tip of the thermowell. In situ tests such as the loop current step response (LCSR), time domain reflectometry (TDR) and others (described elsewhere in this report), are available and can be used to verify proper cable and sensor performance and functionality.

Pressure transmitters in most plants are calibrated at least once every fuel cycle. The calibration involves hands-on work in the field and is now being automated to reduce test time and personnel radiation exposure. In particular, on-line drift monitoring techniques as described in the Annex, may be used to identify the transmitters that have drifted out-of-tolerance. This helps limit the calibration effort to those transmitters that are drifting beyond the acceptable band. Currently, all transmitters are calibrated periodically because until now, there has been no validated method to know in advance if a transmitter has drifted. With the new on-line calibration verification technique, those transmitters that have drifted are identified and calibrated as opposed to calibrating all transmitters. This approach improves safety because it identifies calibration problems as they occur.

The calibration of RTDs and thermocouples may be periodically verified on-line in NPPs using a method called cross-calibration. This technique has been developed to verify the calibration of RTDs and thermocouples while they remain installed in a NPP. The work is performed remotely from the control room while the plant is at isothermal conditions. Basically, the cross-calibration technique involves a systematic intercomparison of redundant temperature sensors to identify any outliers. The method is currently used in many PWR plants at isothermal conditions during startup or shutdown to verify the calibration of the primary coolant RTDs and the core exit thermocouples. To improve the reliability and accuracy of the cross-calibration tests, analytical methods have been developed as described in the Annex to account for plant temperature stability and uniformity problems during the cross-calibration tests. Furthermore, it has been shown that new calibration tables can be generated for any outliers if the cross-calibration data is collected at three or more widely spaced temperatures during plant startup or shutdown periods.

For management of ageing effects on response time of RTDs, thermocouples and pressure transmitters, in situ response time testing methods have been developed and validated for use in NPPs. These methods are described in the annex. For RTDs and thermocouples, an in situ response time testing method called the loop current step response (LCSR) test was developed in the mid-1970s. This method is used in many NPPs around the world for response time verification of temperature sensors [29]. In addition, TDR measurements are made on RTDs and thermocouple wires periodically to check for circuit problems.

The LCSR test is based on heating the sensor internally with an electrical current. The current causes Joule heating in the sensor and results in an internal temperature transient. This transient is monitored and analyzed to give the response time of the sensor. The LCSR method has been approved by the NRC for testing of nuclear plant RTDs. It provides the actual 'in-service' response time of RTDs and thermocouples. This is important because the response time of RTDs and thermocouples are highly dependent on installation into thermowells and the process conditions such as fluid flow rate, temperature, and pressure. In addition to response time testing, the LCSR method can be used for sensor diagnostics and to verify proper sensor-in-thermowell installations. For example, in WWER reactors, long thermocouples are used in long thermowells for measurement of temperature in various regions of the core. If these thermocouples do not reach the bottom of their thermowells when they are installed in the plant, they will not indicate the temperature of the correct region of the core. The LCSR method has successfully been used in WWERs to verify the installation of core thermocouples.

To detect the presence of moisture in an RTD or thermocouple, an Insulation Resistance (IR) measurement may be made. See for example ASTM standard E644 [30]. Also, the LCSR method may be used to determine if there is moisture inside the sensor.

For pressure sensors, the response time may be tested using the noise analysis technique. The noise analysis technique is used while the plant is on-line and is based on monitoring the natural fluctuations (noise) that exist at the output of most process sensors while the plant is operating. These fluctuations can be analyzed in frequency domain (using Fast Fourier transformation) or in time domain (using autoregressive modeling) to obtain the response time of the pressure transmitters. For those transmitters that can not be tested by noise analysis, a method called "ramp test" may be used. The ramp test involves a hydraulic ramp generator to perform the response time measurements.

The noise analysis technique for response time testing of pressure transmitters provides the response time of not only the transmitter, but also the sensing line. Noise analysis is the only method in widespread use for on-line detection of sensor line blockage.

Another method has also been developed for response time testing of pressure transmitters. This method is referred to as the power interrupt (PI) test and is applicable only to force-balance pressure transmitters. The PI test is based on analyzing the transient output of force-balance transmitters when the current loop is momentarily interrupted. For a detailed description of the PI test, the reader may refer to NUREG/CR-5383 or NUREG/CR-5851 referenced in the bibliography section at the end of this report.

Table V summarizes the ageing management technologies for sensors in NPPs. Included in this table is ageing management methods for neutron detectors. Experience has shown that the response time of neutron detectors usually increases with ageing. As such, neutron sensor response times are measured in some NPPs to monitor for ageing effects on the performance of these sensors. The measurements are made using the noise analysis technique in the same manner as for pressure transmitters (see Annex). Another technique is available for response time testing of neutron detectors. This method was developed in the late 1970s and early 1980s in response to the NRC's recommendation in Regulatory Guide 1.118 on sensor response time testing [24].

TABLE V. TEST METHODS FOR VERIFYING THE PERFORMANCE OF SENSORS AND MONITORING FOR SENSOR AGEING (see Annex for more information)

SENSOR	PERFORMANCE INDICATOR	TEST METHOD
RTD	– Calibration accuracy/stability – Response time – Electrical parameters	– Cross-calibration – LCSR test – Insulation resistance, loop resistance, capacitance
I&C cables/connectors	– Cable conductor characteristics – Cable insulation/jacket material properties	– TDR and LCSR tests – TDR, DC resistance, AC impedance, ductility, chemical analysis
Pressure, level, and flow	– Calibration accuracy and stability – Response time	– On-line calibration verification – Noise analysis and PI tests
Pressure impulse line/sensing line	– Blockages, voids, leaks – Calibration accuracy/stability	– Noise analysis – On-line calibration verification, trending, empirical and physical modelling, neural networks
Neutron detectors	– Calibration accuracy/stability – Response time – Cables and connectors	– Calorimetric calculations and conventional calibrations with a source – Noise analysis – TDR, DC and AC impedance measurements
Thermocouples	– Calibration accuracy/stability – Response time – Inhomogeneity, parasitic junction, reversed connection – Cables and connectors	– Cross-calibration – LCSR, noise analysis – LCSR test, insulation resistance tests, Loop resistance test – TDR, LCSR, DC and AC impedance measurements

4.1.2. Neutron flux detectors

Measurement of neutron flux, in-core and ex-core, is important to reactor control and safety. As neutrons are particles without electrical charge, it is not possible to detected them directly. To detect neutrons, sensors must have:

— A converter material, which is the source of ionizing particles electrically charged.
— A medium in which these particles may give their energy.
— A system able to detect and indicate this energy.

A wide variety of sensors are used for this purpose. This includes:

- Gas ionization sensors.
- Boron 10 (uncompensated, compensated), boron trifluoride and helium 3 proportional counters.
- Boron and fission ionization chambers.
- Intrinsic semiconductors.
- Scintillation detectors.
- Self powered neutron detectors (SPNDs).
- Fission couples, etc.

In industrial reactors, for ex-core neutron flux measurements, the source range level is typically measured using boron 10 or boron trifluoride (BF_3) sensors, while the intermediate range detectors are typically boron reaction, gamma compensated ionization chambers, and the power range detectors are boron reaction, non-compensated ionization chambers. Ex-core detectors typically operate based on the ionization principle. In-core detectors are quite often made of fission chambers containing uranium-plated electrodes.

In order to comply with regulatory requirements for nuclear safety, it is the normal practice to perform calibration on detectors periodically.

This may be more essential in sensors made of fission counters/fission chambers that are used for the measurement of in-core/ex-core flux. The sensitivity of these chambers should change in its output due to its burn-up of fissionable material leading to discrimination functions (neutron from gamma). In-core detectors are used on many occasions during startup, or for flux mapping of core during power operation at various core sections.

4.1.2.1. Flux detector ageing mechanisms

The problems observed in the field are related to manufacturing practices resulting in degradation of insulation properties at the chamber and seals resulting in leakage and erroneous readings in flux measurement. Because this kind of sensor is not mass-produced, the precise manufacturing techniques will have an important influence on the ageing mechanisms. Precise details of the manufacturing process may remain a closely guarded secret. This to say that other ageing mechanisms, than those described here, may be found in practice and that it may be very difficult to make a link between an ageing mechanism and the real cause.

Typically neutron flux sensors have a lifetime shorter than the reactor. They are consumable parts, which need to be changed often. The ageing mechanism is mainly related to the type of sensor.

Proportional counters have a main characteristic that is the gas multiplication factor. This factor is excessively sensitive to gas quality: presence of impurities, oxygen traces or humidity may create negative heavy ions that stop multiplication. This decreases the multiplication factor and then changes the characteristics of the sensor. In this case the neutron flux sensor is no longer serviceable. The impurities may be introduced during sensor fabrication. Oxygen or humidity may be a result of chamber leakage. Ionization chambers such as boron trifluoride (BF_3) flux detectors may be vulnerable to Ionic attack of the central wire. The BF_3 ex-core detectors often have a noisy signal and suffer loss of gamma discrimination. The degradation of insulation resistance in fission couples, fission chambers, or Self Powered Neutron Detectors (SPNDs), has a similar effect in modifying the sensitivity.

Practically, degradation of proportional counters is seen on the discrimination curve of the sensor. The degradation effect is less sensitive on the curve, counting rate versus high voltage, for this kind of sensor. Typically these sensors may have an operational life of 5 or 6 years as source range detectors.

In ionization chambers there is no multiplication phenomena, therefore they are less sensitive on the gas quality. Degradation is mainly related, for these sensors, to degradation of the sensitive lining, boron for example. The sensitive lining is a consumable element but, in general, it is not this phenomenon that limits the service life.

For these sensors, degradation is related to physical-chemical reactions at the interfaces between metal/deposit/gas. These reactions induce loss of conductivity and then a change in sensor characteristics seen by saturation curve. An increase of conductivity induces a change at the beginning of saturation curve, the slope decreases. After this first phase, there is a change of the plateau of the curve, the slope is no longer negligible. Typically, these detectors may have an operational life of 10 to 20 years as intermediate or power range detectors.

In addition to the sensing element, a neutron flux detector consists of a connector and a mineral insulated cable. Ageing mechanism of connectors and cables are described later (see Section 4.4). The output signal of a neutron flux sensor is very low, thus this signal is very sensitive to electromagnetic disturbances and to any significant decrease of insulating resistance.

The biggest problem in fission chamber detectors is often from mechanical abuse. They are delicate and their tolerances are tight; therefore any dent can cause a problem with the detector and may result in sensor failure. Ex-core detectors do not have many problems except for water that collects in their well. The water has boron and it corrodes the detector.

Any slow performance deterioration due to ingress of moisture/radiation induced degradation in electronics that are sometimes located within reactor containment for better performance against noise may be unnoticed.

4.1.2.2. Effect of ageing on neutron flux detectors

Ionic attack on BF_3 detectors can lead to catastrophic failure. The central wire is cut and the detector is unable to produce any signal. In the examples given before, each type of flux detector has a normal response curve function of the sensor but also of external parameters; high voltage power supply value for example. The ageing effects typically change the response curve of the sensor. For example the linear, with a low slope, response of the ionization detector may vary progressively with ageing. Typical changes are: first a progressive increase of the slope followed in a second period by the fact that there will no more be a linear portion in the response curve.

Ageing effects on the performance of neutron flux detectors can manifest themselves in calibration drift and response time degradation. As such, periodic calibration verification and response time testing may be performed and the results trended to identify ageing effects and estimate the residual life of the detector. Neutron flux detectors are also prone to cable and connector problems due to the harsh environment (thermal, radiation and other effects) and the low levels of signals. These cable and connector problems are usually more widespread than are encountered on other sensors such as thermocouples and RTDs, however there are no generally accepted practices for managing the cable and connector problem.

4.1.2.3. Ageing management of neutron flux detectors

Different strategies are practicable: systematic, predictive and preventive maintenance, conditional preventive maintenance, or accidental case maintenance. Each strategy has advantages and disadvantages.

Systematic preventive maintenance, for example as in Japan, where ex-core neutron detectors are changed every outage (every 18 months) is very straightforward. The sensors, normally, never see ageing effects, but this strategy is very expensive.

Preventive conditional maintenance: the conditions are defined by criteria in relation to the response curve of the sensor. Therefore maintenance staff have to verify the response curve in operation and measure and/or calculate the various parameters. These parameters are compared to the criteria and maintenance is performed according to the results. This maintenance is done during an outage before failure, so this strategy is intended to optimize technical and economical aspects.

Predictive maintenance of nuclear instrumentation systems involve in situ response time testing and trending of calibration drift and testing the neutron sensor cable and connectors. These tests are discussed below. Breakdown maintenance waits for the sensor failure, which may require a reactor outage to change the sensor and therefore has a detrimental effect on reactor availability.

As discussed above, an important predictive maintenance procedure that is practiced in NPPs involves periodic calibration verification and response time measurements performed on the neutron sensors. During the last decade, on-line techniques have been developed and validated for tracking the ageing of neutron flux detectors. These techniques are described in the Annex. Furthermore, three reports listed in the bibliography (NUREG/CR-6343, NUREG/CR-5851, NUREG/CR-5501 [1]) include detailed information on these techniques. These techniques are also described in the USA country report (see Annex).

Ageing management techniques for cables and connectors include time domain reflectometry (TDR), measurement of AC and DC impedances, and cable and insulation resistance tests. These techniques are also described in the body of this report and in the Annex.

It should be noted that most of the techniques discussed in this section require baseline data to establish acceptance criteria. This baseline data is best acquired when the detectors, cables and connectors are first installed in a plant. In lieu of this baseline data, the results may be compared with other redundant sensors or with measurements taken from other similar plants.

4.2. ELECTRONICS

High temperature and temperature cycling are the dominant causes of ageing in electronic components and circuits. Manufacturers use these effects to accelerate ageing to force the infant failures of such items to be removed prior to shipment The widely used bathtub curve model for failure rates of electronic components (see Fig. 1) is used to convey the concept of three phases of a component's operating life:

— Infant mortality ('burn-in').
— Normal use.
— End of life ('wear-out').

The initial phase is often used by manufacturers during work testing, to ensure delivery of reliable components. Otherwise these failures are revealed during initial commissioning or early operation. The latter two phases of operation are of direct concern to ageing as addressed in this report. There are accepted models and parameters for electronic component reliability during normal operation. However, there are no comparable accepted models for the end of life phase. In deed, as lifetimes are known to vary dramatically between identical components in similar applications, such a model is likely to be application specific. Empirical models to estimate the end of life may be developed, provided there is sufficient historical data for the performance and operating conditions of the specific equipment under consideration.

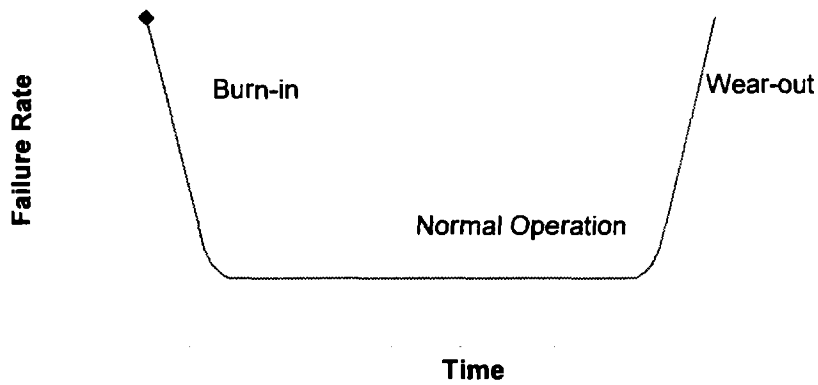


FIG. 1. Bathtub curve model for failure rates of electronic components.

There are also a number of specific mechanisms relating to electronic components which should be considered when developing an ageing management strategy:

— Overvoltage.
— Number of starts/power-ups.
— Electrostatic discharge.

The following sections consider the ageing and management of ageing of specific electronic components. However before the individual items are considered, it is important to appreciate that a poor initial design can have an enormous effect on the ageing of a component. Examples of design faults include:

— Incorrect choice of contact materials for rotary switches, which are operated infrequently - contacts may oxidise and cease to function correctly.
— Incorrect choice of contact materials for relays where low current may cause a build up of oxide on contacts leading to increased resistance and possible failure.
— Inadequate specification of power rating for passive or active components.
— Poor ventilation or cooling of equipment enclosures.

Another specific problem in the field of electronic components (mainly digital components) is obsolescence. Electronic components have a life cycle which may be represent by the typical curve shown in Fig. 2.

This curve gives number of components sold versus time is given in arbitrary units for the two axes. Each component has such a curve, but the life-cycles are becoming shorter (in some cases as low as 14 months). This 'ageing' effect is more pronounced for the latest and complex IC components. In this case, obsolescence becomes a more important problem than ageing. Some electronic components present real specific ageing mechanisms. The following are some the well known examples:

— easy to detect and common to various systems: electrolytic capacitors;
— not easy to detect in operation: MOVs;
— not known or documented: fuses;
— common to various technologies: semiconductors.

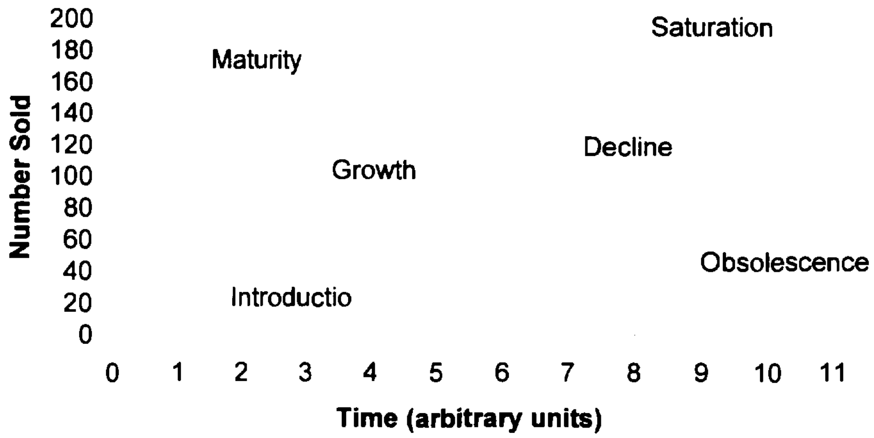


FIG. 2. Life cycle of electronic components.

4.2.1. Electrolytic capacitors

There is much operational experience and a large amount of research has been carried out on the ageing of electrolytic capacitors, but few published reports exist. The Nuclear Maintenance Assistance Center (NMAC) of the Electrical Power research Institute (EPRI) is presently funding a project to investigate the failures of electrolytic capacitors. This report is expected to be published by early 1999.

The dominant ageing mechanism for capacitors with liquid electrolyte is loss of electrolyte through the end cap seals. This is a particular problem with rubber seals where degradation (perishing) of the rubber provides a leakage path for the electrolyte. At a temperature of 20C, this process could take 10 years on a typical electrolytic capacitor, but it is accelerated by increasing temperature. The increasing use of Teflon seals has reduced the extent of this problem, but many older components remain in service and are still subject to this type of failure.

The dominant mechanism for solid electrolyte is evolution of the electrolyte pH.

Special precautions are also necessary to prevent the loss of oxide at the electrodes, due to chemical attack, on stored components (see below).

4.2.1.1. Effect of ageing on capacitor performance

Loss of electrolyte increases the equivalent series resistance (ESR) and decreases capacitance. Eventually, the capacitor will fail either open or short circuit. The failure may be catastrophic and consequences will depend on how the capacitor is employed in the circuit. There is some evidence that electrolytic capacitors used in power supply circuits tend to exhibit increased noise at the end of their life. ESR increases, and is increased by, internal temperature leading to a possibility of thermal runaway and the ultimate destruction of the component.

4.2.1.2. Ageing management of electrolytic capacitors

Once a capacitor has lost its electrolyte, it must be replaced. A variety of measures may be taken to guard against the consequence of loss of electrolyte:

— Periodic replacement.
— Replacement of all similar components when the first failure is detected.

— Use devices rated for a higher temperature than required.
— Periodic testing/monitoring of components and spare modules; leakage current, capacitance value, ESR, power factor. This may include endurance tests on sample components at maximum rated temperature and voltage.
— Temperature measurement of component.
— Power supply ripple current measurements.

The loss of oxide during storage may be recovered by a controlled application of voltage to the item by either:

— A slow and progressive increase of voltage up to the nominal voltage for individual components. This technique is suitable for components rated greater than 150V or:
— A sudden application of the nominal voltage for components rated up to 150V. This method may not always work and can destroy the capacitor.

In view of the risks associated with storing electrolytic capacitors, it is recommended that new devices be used wherever possible.

4.2.2. Metal oxide varistors (MOVs)

MOVs are used to protect electronic modules against overvoltage. The devices has an internal resistance that varies inversely to the voltage applied (e.g. an overvoltage) thus protecting other devices in the circuit by short circuiting the physical path.

4.2.2.1. MOV ageing mechanisms

Each time an MOV is subject to an overvoltage, a part of the component is destroyed, creating a memory effect, this is the ageing mechanism. This process can continue until the MOV itself is destroyed, failing to a short circuit condition and thus creating a functional failure of the circuits that depend on it.

4.2.2.2. Effect of ageing on MOV performance

Prior to failure, the ageing does not significantly affect the performance of the component or the overvoltage trigger level. Consequently it is difficult to assess the condition of the component.

4.2.2.3. Management of MOV ageing

Periodical replacement is the only protection against ageing. There are no known methods for determining residual life, although complete failure is detected by functional failure of the electronic module.

4.2.3. Fuses

Fuses are widely used as protective devices against over current in electronic or electrical circuits. Some operational problems have been encountered although research into ageing failures has been limited.

4.2.3.1. Fuse ageing mechanisms

The initial transient current when power is applied to a circuit may be 3 to 4 times the nominal current. Slow blow fuses will not operate (blow) for such short transients but there may be a loss of fuse material through vaporization.

4.2.3.2. Effect of ageing on fuse performance

Progressive loss of fuse material will reduce the effective rating of the fuse and may lead to spurious failures later in life.

4.2.3.3. Management of fuse ageing

As the lifetime of the component is related to the number of starts the only effective management technique is preventive maintenance. This preventive maintenance may be conditional; replacement of all fuses on a set of equipment when the first fuse failure is encountered.

A common error is to increase the rating of a fuse when a random failure occurs. The failure may in fact be age related and increasing the fuse rating will reduce the protection offered by the fuse.

4.2.4. Components containing simiconductors

Semiconductors in general are known to be sensitive to gamma radiation and electrostatic discharge (ESD). A study on radiation vulnerability of semiconductor components was performed in the early 1990 in Germany. It revealed that MOS technology components are especially sensitive. For cases exceeding 10 Gray, total replacement by different technology or by "hardened" (shielded) components was recommended. A standard exists for the hardening of semiconductor qualification for space applications (MIL-STD-801).

Table VI gives an overview on the qualitative radiation sensitivities of different semiconductor components with low values standing for increased radiation sensitivity.

The application of electronics within radiation prone areas should therefore be kept to a minimum. Nevertheless, some components like area radiation monitors require in situ preamplifiers (pulse channels during startup), which are thus exposed to considerable dose levels under normal operating conditions and especially in case of LOCA conditions, when they are of special importance. The bulk of electronic equipment is normally located as much as possible outside of the containment, enabling easy maintenance in practice.

The imminent change in characteristics of electronics over longer time periods either needs periodic calibration or replacement. The use of radiation hardened components in such applications mitigates this problem.

4.3. RELAYS

The extensive use of relays in I&C equipment provides a wealth of operational and research data. The reader is referred to the bibliography for details (see NUREG/CR-4715).

4.3.1. Relay ageing mechanisms and effect on performance

There are three sub-components of a standard electromagnetic relays which may be vulnerable to ageing:

— The relay coil;
— The relay contacts;
— Ancillary components such as contact spacers and plugs and sockets.

As with any electro-mechanical device the relay is particularly sensitive to the use of unsuitable components or inadequate qality asurance during manufacture.

TABLE VI. RELATIVE RADIATION SENSITIVITY OF REPRESENTATIVE SEMICONDUCTOR COMPONENTS (low values stand for increased radiation sensitivity)

Device Family	Device Type	Displacement (Note 1)	Total dose (Note 2)	Transients (Note 3)	Single event upset (Note 4)
Diodes	Rectifier	3	3	1	4
	Switching diodes	3	3	2	4
	Voltage reference	3	4	3	4
Transistors	LF	1	1	2	4
	HF	3	3	3	4
	Microwave	3	3	3	4
	Power	1	1	1	2
	JFET (MESFET)	3	3	3	4
	MOSFET pMOS	4	2	4	3-4
	MOSFET nMOS	4	1	4	3
Digital integrated circuits	TTL	3	3	3	4
	LSTTL	3	3	4	4
	CMOS	4	1	4	2
	CMOS (Rad. Hard.)	4	3	4	2
	HCMOS	4	1	4	2
	SOS (Rad. Hard.)	4	3	4	4
	ECL	3	3	3	4
	IIL	2	2	3	3
	GaAs	4	4	3	3
Linear integrated Circuits	Operational Amplifier	1	1	2	2
	Comparator	2	1	2	2
	Ref. Source	2	2	2	3
Microprocessors	HMOS	4	1	1	1
	CMOS	4	1	2	1
	EEPROM	3	1	2	1
	TTL	3	3	2	3
	ECL	3	4	2	3
	IIL	2	1	2	2
Optoelectronics	LED	2	4	4	4
	Phototransistor	1	1	1	2
	Optocoupier	1	1	1	2
	Solar Cell	2	4	1	4
	Optical Waveguides	3	1	1	4
	CCD	3	1	1	1

Notes:

(1) "Displacement" marks the individual sensitivities against displacement damages to material crystal structure.

(2) Vulnerability in terms of cumulative radiation is estimated under "total dose".

(3) Vulnerability in terms of single exposures is estimated under "transients".

(4) "Single event upset" refers to local malfunctions like radiation induced bit-errors in memory components.

Ageing of relay coils is primarily a problem on relays which are continually energized. Excessive heat may be generated, by the coil or associated components, which may cause the coil to burn out or adversely effect other components within the relay or nearby (e.g. chemical breakdown of varnishes causing contact contamination, or changes in component dimensions). In pneumatic time delay relays heat may cause embrittlement of the diaphragms causing set-point drift, or setpoint drift on undervoltage relays.

Relay contacts may age due to three main mechanisms:

— Contact oxidation on Normally Open (NO) contacts, or contacts where the material is inadequately specified for the actual duty current. This can be a problem for both low and high currents;
— Contact welding or pitting due to excessive current (possibly caused by switching of inductive loads);
— Chemical attack e.g. due to the use of high sulphur content rubber components within the relay. Internal ancillary components may deform due to temperature or chemical attack.

Variations in ground potential may cause problems and should be considered as a design issue.

4.3.2. Management of relay ageing

The importance of a good initial design cannot be overstated. This should include adequate environmental control for relay systems: A system with a large number of normally energized relays will generate a lot of heat, which should be removed to prevent excessive temperatures.

Inspection and test of relays on a batch basis is also advised to ensure poor manufacturing is detected prior to components being put into service. When in service a periodic visual inspection may be helpful in identifying any chemical breakdown or degradation of components or contacts. Regular cleaning of relay contacts may also be beneficial in specific circumstances.

Other procedures do exist for in situ testing of relays, however these are expensive and not universally applicable. There have been proposals to develop methods for early detection of relay ageing using thermal signature analysis, contact resistance measurement or evaluation of time behavior (e.g. response time). In all cases a suitable replacement strategy was found to be more cost effective. Various national studies have been carried out on relay failure rates but the wide variety of results implies that the life of relays is application and design dependent.

Most relays are rated for a certain number of operations and life will depend on how the relay is used. Relays which are repeatedly exercised (e.g. reed relay analogue multiplexers) may need periodic replacement.

4.4. CABLES AND CONNECTORS

The ageing of cables has long been recognized as a potential threat to safe operation of NPP, especially towards the end of life, and a large body of research exists into ageing mechanisms. Also the IAEA has organized a co-ordinated research programme on management of ageing of I&C cables [3].

A complete survey of this work is beyond the scope of this report and the reader is referred to the bibliography at the end of this report for further information. However there are certain common factors and principles associated with cable ageing which are summarized in the following sections which also include information on ageing of connectors.

Operating experience has shown that the problems arising from the ageing of cables are relatively small in comparison with those associated with other I&C components. This has been confirmed by German and French experience. Under normal ambient conditions cables can be expected to out-live the rest of the plant. However the potential threat posed by common mode failure of cables has been considered serious enough to justify a large amount of work in many countries. The most severe scenario is the requirement for cables to continue to function normally during a loss of coolant accident (LOCA) at the end of plant life.

These concerns are mainly about I&C cables as opposed to power cables. Research and dvelopment (R&D) work has been conducted to characterize the ageing mechanisms and to develop testing and monitoring techniques for use in nuclear power plants. The R&D has produced a diverse set of techniques for assessment of cable health and condition. This includes chemical testing of insulation composition, mechanical testing of insulation ductility and electrical measurements performed on both cable conductors and insulation. The main ageing stressors for cables are:

— elevated ambient temperature or humidity;
— cyclic mechanical stress;
— exposure to radiation.

For all of these stressors the tensile strain of the insulation has proved to be the limiting factor in every case. Table VII identifies the behavior of various cable types when subject to such stresses.

The replacement of analogue systems with digital technology can reduce the total length of cabling and the number of connectors employed with consequential reduction of operating and maintenance (O&M) costs.

New, additional cabling becomes necessary when additional systems are added to the existing plant layout, e.g. loose parts monitoring systems or the implementation of an emergency control room. In the latter case special attention has to be drawn on the proper grounding, because differences in ground potential between the two buildings can result in unacceptable voltages at branching relays. Optical coupling can be a solution in all cases of grounding potential differences and wherever electromagnetic interference may be a concern. The ongoing reconstruction of the safety relevant I&C in German plants, Unterweser and Neckarwestheim I, rely totally on optical connections between the cabinets for this reason.

As far as flux detector cables are concerned, the following are noteworthy:

(1) The requirements in respect of insulation resistance and screening are much more demanding for flux detector cables than for sensors such as thermocouples or RTDs.

(2) The most common failure mechanism of mineral insulated metal sheathed cables in moisture ingress as a result of mechanical damage or corrosion. The simplest test for monitoring this is insulation resistance. It should be noted that insulation resistance measurements on mineral insulated cables should not be made using high voltage; typically 100V should be the maximum on cold cables.

(3) Connectors on mineral insulated cables are particularly vulnerable to damage because they are fragile in themselves and because they provide a seal on the cable.

(4) Disturbing in-line connectors should be avoided, the cable seals may be damaged and it may be difficult to re-establish a hermetic seal. This must be balanced against the advantages offered by routine cable tests in predictive maintenance.

(5) Mineral insulated detector cables may have an insulation covering to protect against earth loops. If this becomes damaged, interference levels could be increased.

TABLE VII. QUALITATIVE AGEING BEHAVIOR OF CABLE INSULATION MATERIALS

Type	Elevated temp.	Radiation	Remarks
PVC Polyvinyl chloride	sensitive	Insensitive	Embrittlement at elevated temperatures, limited stretchability
XLPE Cross-linked polyethylene	sensitive	Insensitive	Embrittlement at high temperatures and under radiation, limited stretchability
HRPVC Heat resistant polyvinyl chloride	sensitive		Embrittlement over 105°C
PP/PE Polypropylene/ Polyethylene	sensitive	sensitive	Flexibility is lost at elevated temperatures/under radiation
PTFE Polytetrafluoroethylene	insensitive	sensitive	Cracks due to embrittlement
FRLS PVC Fire retardant low smoke PVC	insensitive	insensitive	Good performance
EPR Ethylene propylene rubber			No degradation
VPE Cellular polyethene			No degradation
SIR Silicon rubber			Insulation failure possible over 25 years
PTFE Polytetrafluoroethylene			Insulation failure possible over 25 years

4.4.1. Life-assessment technique for NPP cables

The condition of polymer-based cable material can be best characterized by measuring elongation at break of its insulation materials. However, it is not often possible to take sufficiently large samples for measurement with a tensile testing machine. The problem has been conveniently solved by utilizing a differential scanning calorimetry technique. From the tested cable, several microsamples are taken and the oxidation induction time (OIT) is determined. For each cable the correlation of OIT with elongation at break and the correlation of elongation at break with the cable lifetime can be performed. A reliable assessment of the cable lifetime depends on the accuracy of these correlations. Consequently, synergistic such as; dose rate effects and effects resulting from the different sequence of applying radiation and elevated temperature should be taken into account.

When cables approach their end of life, they start cracking on bending or may even become brittle. It has been observed that cable terminations are most susceptible to ageing effects by virtue of their chemical reaction with oxygen.

Specific mention should also be made of the use of neoprene rubber sleeves used for insulating cable terminations. There have been occurrences where chemical attack or degradation has caused neoprene to suffer a total loss of insulation resistance and become conductive.

The following points should also be noted:

— Abrating of electrical plant can result in 100% duty cables running close to or above their design rating. In such cases relatively small increases in conductor temperature due to increased electrical load can become life threatening;
— A change from analogue to digital technology may require less cables;
— Changing standards for signal segregation may require extra cabling.

4.4.2. Effect of ageing on cable performance

There are three ways in which ageing may affect cable performance:

— Mechanical failure;
— Loss of insulation resistance;
— Failure or degradation of cable conductor.

Currently there is only limited information on the ageing of fibber optic cables. However, a report entitled "Assessment of Fiber Optic Pressure Sensors" is available with fundamental information about the principle of operation of fiber optic sensors and their potential for nuclear power plants. This report is listed in the bibliography as NUREG/CR-6312. A more detailed report on fiber optic sensors is NUREG/CR-5501 [1], which is also listed in the bibliography.

4.4.3. Management of cable ageing

There are two main methods:

— Accelerated life testing. This may also include the installation of pre-aged samples on operating plant to allow their subsequent removal and testing later in life.
— Testing of existing cables in situ.

There are many methods for testing cables and a full exposition is beyond the scope of this report, however a brief summary is given in the following list:

— Visual examination of insulation and measurement of cracks or crack growth.
— Hardness testing of insulation. This may only be done on specific sections of cable and there may be hot spots elsewhere.
— Chemical analysis of insulation. This requires a sample to be removed.
— Electrical insulation tests.
— Measurement of tensile strength.
— Low frequency or swept frequency dielectric loss measurements.
— Time domain reflectometry (TDR) testing of conductors and connectors.
— AC and DC resistance tests of conductors.

Most of these measurements require baseline data for comparison and interpretation. It is common practice for plants to develop a database of cable characteristics and repeat tests periodically to identify any significant change from the baseline. In lieu of baseline data, the cable characteristics from similar installations may be used.

4.4.4. Ageing mechanisms for connectors

The dominant ageing mechanisms for connectors are mechanical wear and oxidation of contacts. Mechanical wear is caused whenever a connector is disturbed. This may be influenced by:

- Poor reliability of the module resulting in frequent replacement;
- Need for frequent calibration or modification of parameters.

Contact oxidation may be a result of inappropriate choice of contact materials for the duty of the circuit, use of incompatible materials for male and female parts, or operation in an aggressive environment.

4.4.5. Effect of ageing on connector performance

Mechanical wear and oxidation both lead to an increase in contact impedance which may vary from a few ohms to a complete open-circuit. The consequences of this will depend on how the connector is employed in the circuit. In a switching circuit a small increase in resistance may be tolerable but in a sensitive analogue circuit (e.g. processing very low signal levels) such a small increase in resistance may have a major effect.

In some cases the faults may appear intermittently which will generally cause great difficulty to maintenance staff in diagnosis and correction of the fault.

4.4.6. Management of connector ageing

Connectors should be left undisturbed wherever possible. Repeated breaking and making of connections may lead to mechanical wear. This is especially important for printed circuit board (pcb) edge connectors.

Heat drying of connectors before installation can help eliminate failures due to moisture, and storage of spare parts in an inert atmosphere (nitrogen) is also recommended. Thermographic scanning of connectors whilst in service can give an indication of high resistance points which may give early warning of failure.

A note of caution: care should be taken when upgrading I&C systems which use new technologies with higher input impedances than existing systems (e.g. analogue to digital or relay to digital). The low current drain can lead to contact oxidation problems if existing connectors are retained.

Experience with the use of TDR or LCSR techniques in testing RTDs, thermocouples, and neutron detectors has shown that these techniques can reveal connector problems, especially if baseline data is available for comparison.

4.5. OTHER COMPONENTS

The sections above have considered some of the main physical I&C components which may be vulnerable to ageing. However there are other items, which are potentially vulnerable, which can affect the function of I&C equipment, and it often falls to the I&C engineer to manage their performance. Examples of such items are considered in more depth in the following sections.

4.5.1. Equipment enclosures and mountings

All safety or safety related I&C equipment must be protected against hazards which could render the system inoperable when a demand is placed upon it under an accident or fault condition. The main hazards with a possible relation to enclosure ageing aspects are:

- water (liquid or steam);
- seismic event;
- extreme temperature conditions (hot or cold);
- electromagnetic compatibility and radio frequency interference (EMC and RFI);
- chemical spillage;
- mechanical shock due to impact (e.g. vehicle collision).

In most cases the defence against the hazard is the physical quality of the equipment enclosure which can be determined by regular inspection.

Water or chemical: The state of the door seals will affect the Ingress Protection (IP) rating of an enclosure and particular attention should be paid to older equipment which may use rubber compound seals which can perish or crack.

Seismic: There is a known problem with the ageing of elastomers used for anti-vibration mountings which is currently under investigation by EdF. Replacement of such items is difficult and rarely designed for. Therefore plans or contingency measures are required as soon as possible to deal with future events.

Extremes of temperature: A more indirect ageing effect may be caused by deteriorating performance of Heating and Ventilation systems or equipment cooling systems. Situations have been observed where equipment has been operated with enclosure doors open to maintain acceptable operating temperature for the equipment. However this will seriously undermine the protection against various hazards especially EMC, RFI and Steam or water release.

EMC/RFI: A specific aspect of "enclosure ageing" are functional modifications. These modifications may needed to add cables, electronic modules, etc. All such modifications may affect the immunity or emission of the electronic system inside the enclosure and the enclosure itself if new holes are made. Another case is specific EMC/RFI enclosures, which use conductive seals for the doors. These conductive seals, may become with time resulting in poor electrical contact which in turn will reduce immunity. Periodic maintenance is required to manage this problem.

4.5.2. Equipment standards

Development of standards and equipment in NPP I&C is so rapid that it is impracticable and uneconomic to keep all equipment as state of the art. This may be considered as a potential ageing of the equipment which may accelerate obsolescence, with manufacturers unwilling or unable to supply non-compliant components, or impose restrictions on the operation of existing equipment (e.g. European EMC/RFI standards, [31, 32].

The most obvious defence against this threat is constant vigilance and awareness of developing standards. It is, however, impracticable to review the specification of all I&C equipment whenever standards change and a periodic review should be sufficient to identify any shortfalls before any adverse impact is suffered. More direct action is recommended when considering the effect of changing standards on obsolescence. A forward looking spares management policy is necessary to ensure that adequate spares are held or can be procured to support a system until the end of its operating life.

4.5.3. Storage media

All nuclear power plants have a statutory duty to keep records (archives) for the operating life of the plant. Implicit in this requirement is a need to be able to read the archived data, which may only be performed infrequently. It should also be remembered that historic data is an asset which cannot be recreated once lost. Obsolescence of storage media may cause problems in this area and it is suggested that retrieval routes are exercised on a regular basis to ensure that they remain viable. Retrieval equipment should also be covered by a spares management programme as for other I&C equipment. Maintenance and design documentation may also be vulnerable to this threat.

There are known problems with storage of data on magnetic tapes (print-through) which may be defended against by regular (e.g. 5 yearly) respooling of the tape. Early compact discs (CDs) were also known to have a finite lifetime (~10 years). Experience with current devices such as digital audio tape (DAT), write once and rewritable optical disks is currently limited but it should be assumed that these devices do have a finite lifetime and appropriate defences installed.

4.5.4. Ageing of programmable I&C systems

The hardware of programmable systems is built using similar technologies to that used in many analogue systems (semiconductors, resistors, capacitors, cables and connectors, etc.).

The short product life-cycles of programmable systems means that obsolescence becomes a much more serious problem than ageing of the hardware. In this case the issue of spare parts management and forward planning becomes of vital importance.

Computer software does not age in the classical sense, however loss or expertise or inadequate documentation can lead to a decrease in maintainability. This will also cause difficulties when attempting to replace the systems and early measures for management of ageing are vitally important for this class of systems.

5. MANAGEMENT OF AGEING

The purpose of this section is to give guidance on the development of a strategy to manage I&C ageing. The objectives and constraints are first considered and a general approach for developing an ageing management strategy is then suggested. Subsequent sections give details of some of the tools and methods which may be used in the implementation.

5.1. OBJECTIVES AND CONSTRAINTS

An ageing management strategy must meet the needs of a utility within an environment, which will be specific to the utility and subject to many constraints. It is suggested that the objectives and constraints are considered prior to the adoption of any strategy to ensure it is both feasible and beneficial. A suggested outline for each of these issues is given in the following schemes.

5.1.1. Objectives

The objective of an ageing management strategy is to provide for timely detection and mitigation of ageing effect in I&C systems important to safe and economic plant operation. In particular to:

- manage and minimize the risk to nuclear safety and business targets;
- be cost effective;
- prolong equipment and system life where practicable and desirable.

5.1.2. Constraints

(1) An ageing management programme will require money and human resources. Justification of expenditure, supported by cost benefit and risk analysis, should comply with utility business planning process to ensure approval. (Note: This statement assumes that should an ageing management programme be required to support nuclear safety then this will be consistent with the targets of the business, however the justification for expenditure will still probably have to be presented in accordance with the planning process).

(2) It is unlikely, although not impossible, that the business organization will be significantly changed to meet the needs of an I&C ageing management strategy. Where possible, the process should use established procedures and lines of communication. Careful judgements are required when any new process is proposed. Motivation of staff is vital and they must be convinced of the benefit of a new approach to maximize its chance of succeeding. It is to be emphasized that this applies to all staff involved in the process, from first line maintenance technicians to company managers.

(3) In many cases the strategy will require the prediction of future performance based on no historic data. There may be established company procedures for such judgements, usually in the form of risk assessments, but in many situations this will not be the case.

(4) Ageing issues are usually plant specific and input from experienced staff is vital. However such staff may only be available for short, unpredictable periods of time. It is vital that the use of these staff is controlled to gain maximum benefit. Ideally staff will be taken out of line, but if this is not possible time must still be set aside to ensure that their views, opinions and experience are captured.

(5) The particular ageing management strategy selected must be appropriate for the safety relevance of the system concerned. IAEA Safety Series No. 50-SG-D1 [33] defines the categories of safety relevance of plant systems. This is further interpreted for I&C systems by IAEA Safety Series No. 50-SG-D3 [34].

(6) The regulator will require safety margins to be preserved and that the programme is being under taken in accordance with a systematic methodology.

(7) Options for the support of existing equipment will be constrained by what happens to the equipment suppliers. The extent of support for and knowledge of older ranges of equipment will generally reduce as product ranges develop. In some cases companies will cease to trade altogether causing a loss of design knowledge and support. This may have severe consequences for some plant systems (e.g. reactor protection systems).

5.2. SUGGESTED PRACTICE

The IAEA Code on the Safety of Nuclear Power Plants for Operation (50-C-O) [35] and the associated Safety Guide on Management of NPPs (50-SG-O9) [36] require that appropriate documented management programmes are established in order to achieve the objectives and responsibilities of the operating organization with respect to safe operation of NPPs. The preceding sections of this report deal with some of the ageing management techniques for typical nuclear power plant I&C equipment. These would be the key elements of an overall I&C equipment important to safety Ageing Management Programme (AMP), whose objective should be to maintain the fitness-for-service of I&C equipment throughout their service life. Such an AMP should be implemented in accordance with guidance prepared by an interdisciplinary I&C equipment ageing management team organized on corporate or owner's group level. This could be worked out through a policy document approved by responsible plant management. For general guidance on the organizational aspects of a plant AMP and interdisciplinary ageing management teams, refer to the IAEA safety report [35].

When developing an ageing management policy, the plant should assign priorities to activities proposed for the AMP. Other important issues to be considered in the policy document are organizational aspects, resources management and division of responsibilities.

As noted in Ref. [35] an effective ageing management programme of I&C equipment (components) can be best accomplished under a systematic umbrella-type programme that integrates already established programmes relevant to managing of ageing.

Experience shows, that most of the components of an AMP already exists at most plants. A survey of existing programmes that could contribute to managing of ageing of I&C equipment important to safety is recommended as a first step for establishment of an AMP. Examples programmes, which should fall under the survey, are listed below:

— existing component specific programmes for I&C equipment important to safety;
— equipment qualification programmes;
— maintenance programmes, including surveillance, in-service inspection, testing and monitoring programmes;
— data collection and records management programmes;
— operating experience feedback programmes including analysis of the significant events;
— spare parts management programmes;
— personnel qualification and training programmes;
— external programmes such as R&D.

In addition, all relevant documentation should be included in the survey, such as:

— operational procedures;
— limits and conditions for safe operations;
— deterministic safety analysis results;
— PSA/PRA results or claims made on equipment reliability;
— regulators requirements;
— applicable standards.

The ageing management process itself consists basically of following three phases:

— selection of components important to safety for which ageing should be evaluated;
— understanding dominant ageing mechanisms in the selected components and identifying or developing effective and practical methods for testing, monitoring and mitigating ageing;
— managing the ageing degradation in the selected components by effective practices and initiatives in surveillance, testing, maintenance and operations.

For first two phases of such a process refer to IAEA Technical Reports Series No. 338, Methodology for the Management of Ageing of NPP Components Important to Safety [2]. Another relevant IAEA technical publication is Safety Safety Series No. 50-P-3, Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing [1]. Discussion on some issues related to the third phase of an ageing management process is provided in following subsections in relation to plant information management, human aspects management and maintenance management.

Ageing management programmes should be subject of periodic assessment for optimization, possible improvements and, mainly, for the effectiveness. The Safety Guide on periodic safety Implementation and Review of Nuclear Power Plant Ageing Management Programme [37] provides guidance on a systematic assessment of effectiveness of such programmes.

5.3. MAINTENANCE, TESTING AND SURVEILLANCE STRATEGIES

The ageing of I&C components can be in principle managed with the following means:

— replacement of equipment;
— optimal testing and maintenance routines;
— control of environmental conditions.

Periodic testing combined with corrective maintenance continues to be the basis for maintaining the function of I&C equipment in most utilities. Increasingly condition monitoring techniques are also applied to enable predictive maintenance actions. The maintenance needs of I&C equipment may change with time, and accumulated experience can be used to adjust maintenance or testing activities. Periodic Safety reviews may also be used to ensure that the maintenance plan is adequate to ensure the system will continue to meet its safety duty.

The next section reviews common maintenance practices for I&C equipment and then considers some principles which can be used to develop these practices into an ageing management strategy.

5.3.1. Replacement of equipment

Equipment replacement schedules and practices are very different within the nuclear power industry and guidelines are not typically available as to when a particular type of equipment must be replaced. Obviously, malfunctioning equipment that cannot be repaired must be replaced as necessary. The following replacement policies can be identified:

— equipment is replaced on scheduled basis, whether it has failed or not;
— there is no testing nor maintenance action on the equipment, and it is replaced on failure;
— equipment is maintained or tested on scheduled basis and replaced when found faulty;
— operation of degraded components is continued in order to postpone the replacement until a convenient moment, e.g. plant overhaul or outage.

The replacement policy is related to the equipment safety significance, accessibility and expected effects of failure on plant availability. With an on-line monitoring system, equipment condition can be verified and repair and replacement schedules can be established ahead of time before equipment fails completely. In ageing management with component replacement, spares availability and obsolescence are of specific concern. These issues are briefly described below.

Spare parts management

Management of spares stockholding is vital to ensure availability of electronic systems. First line maintenance on site is usually by module replacement and several factors determine the number of spares which should be held:

(1) Number of modules of the same type;
(2) Maximum allowed time to repair;
(3) Reliability of the module in service;
(4) Time taken to procure spares from a supplier;
(5) Obsolescence of spare parts.

The first two factors are easily determined, however the calculation of reliability is less straightforward and depends on the amount of operational experience.

For new systems, the theoretical reliability of individual components may be used to predict the reliability of a module. For an established system the observed reliability may be used. Where modules are returned to a central repair facility (e.g. to the manufacturer) improved statistics should be available from a larger population of modules. However care must be taken to ensure that suppliers' reliability figures correctly account for the use of modules held as spares. Spare modules will not be in operation and may distort reliability figures which are based on the number of modules sold against those returned for repair.

Experience has shown that theoretical predictions of reliability are pessimistic. Where this is not the case it generally indicates a design fault. However it is also important to make an allowance for possible end of life effects. This is difficult to predict as most reliability calculations assume a fixed failure rate (e.g. exponential failure law).

Effective management of spares requires a database of module reliability that should include:

(1) Module reliability;
(2) Cause(s) of failure. Information supplied by module repairer;
(3) A Facility to detect long term trend of failures to allow detection of ageing effects. This is considered in more detail later in the report;
(4) The importance of such a database cannot be overstated.

Component obsolescence

Obsolescence of electronic components has been discussed in Section 4 of this report and its implications and management is considered further here.

A component becomes obsolete when the last manufacturer in the world decides to stop production. However obsolescence for a component which never fails is not a problem. The problem arises when obsolescence strikes a component with finite reliability. Once in this situation repair of the module is impossible if the failure is due to the obsolete component and no spares are stockpiled.

Various methods exist to manage such a situation and the choice depends on local parameters and the maintenance strategy of the utility. Among the parameters one may cite are:

— economic weight of the supplier versus manufacturers of electronic components;
— existence of a contract between utility and supplier;
— existence of a contract between suppliers and manufacturers.

However transcending all of these aspects is the problem of information and knowledge of the status of a component. Unless specific action is taken otherwise, component obsolescence is generally discovered when a module is sent for repair or an attempt is made to purchase the component only to discover that it is no longer available.

To avoid this situation it is necessary to anticipate this phenomenon. The simplest way to anticipate obsolescence is to arrange strategic storage of all components by conducting an obsolescence study on the system. Table VIII gives an indication of the types of components and the need for stockpiling.

Such strategic storage is not expensive, provided it is done while the components are still available on the market. An EdF study on a specific item of digital equipment gives the following results:

Number of cubicles:	2 000 items
Initial Cost of cubicles:	1 000 MFF
Number of active components:	9 500 000
Strategic storage sufficient for 20 years:	92 000 components
Cost of the stock:	500 000 FF

TABLE VIII. STOCKPILING REQUIREMENTS FOR ELECTRONIC COMPONENTS

Component	Need for stockpiling
Discrete semiconductors (diodes, transistors, thyristors) .	Essential for specialist devices including high tolerance and special to project devices especially where only a single supplier exists. More common transistors and diodes may be multisourced and less of a problem.
Integrated circuits and microprocessors.	Depends on technology employed. Multisourced components are less at risk but Microprocessor chips are especially vulnerable to obsolescence and stockpiling is recommended. Equivalents are unlikely to be available in this case.
Passive components (resistors and capacitors)	Stockpiling is not normally required but may be necessary for high tolerance devices such as ultra high stability resistors or non standard value devices. (Note: Stockpiling of electrolytic capacitors is not recommended as discussed in Section 4.)
Potentiometers (including specialized motors and gearboxes used in servo-motor driven potentiometers)	Stockpiling is not normally required but can be essential for custom built devices (e.g. characterized multitapped potentiometers).
Reed relays	Essential for specific components and where pcb mounted. (Component 'footprint' or packageing may change.)
Connectors	Stockpiling essential. Little standardization exists especially for early ranges of components and special manufacture can be extremely expensive. Lack of connectors can render 'form and fit' manufacture of systems modules impracticable.
Cables	Stockpiling is essential for special purpose cables (e.g. super screened cable often used in flux detector systems).
Switches and lamps	Not usually a problem unless special to project.

Note that the volume of the storage is calculated on a basis of reliability data of modules and components given by the database of spare parts.

In this example the cost of the strategic stock is less than 0.1% of the initial cost of the equipment. This is a small premium to ensure that the modules can be repaired over the next 20 years.

Such a stock must be stored in good conditions to preserve the ability to solder the component and avoid some problems such as electromigration in silicon of components. A solution patented by EdF is to store components in a glass tube filled with nitrogen.

Apart from physical ageing of I&C components, the unavailability of spare parts is becoming of increasing concern for some NPP operators. This applies to cases, where the acceleration of technological advances in I&C technology supplies limited markets. As stated above, German I&C manufacturers have announced that spare parts for nearly all current system families will not be produced after 2005. Any refurbishment project and all new installations will therefore be required to use digital technology. However , large markets like the US do not seem to face this problem at the moment, and in fact, due to licensing issues, software V&V concerns, and the high cost, US plants are generally reluctant to implement digital I&C systems in safety related applications. It is advisable that the lifetime management of a system and its spares management policy be considered as soon as it is procured rather than wait until it has been operating for several years. This is especially true for digital electronics and computer based systems and is considered further below.

5.3.2. Periodic testing

Most electrical and I&C equipment is tested at specific intervals to verify proper performance, detect ageing degradation, and determine when the equipment must be replaced. There are no definitive guidelines for test frequencies of specific equipment or systems and most test intervals have been established through custom and practice or developed on a case by case basis. Sometimes, the same type of equipment in the same type of nuclear plant within the same utility is tested with different procedures and different testing intervals. There are also plants that only test their equipment on demand or only when there is an obvious problem. The only consensus that can be found is that a majority of plants perform testing based on their operating or testing cycles that range from about one month to about two years. Equipment, especially I&C systems, are tested (calibrated and response time tested) once every fuel cycle either on-line or during a refueling outage. Calibration verification and response time testing of reactor protection systems (e.g. trip channels) is usually carried out on a more frequent basis. Staggered testing of multiple redundant channel systems is often employed to minimize peak unreliability.

Ideally, testing frequencies must be established based on objective ageing data and performance histories. However, traditionally, the tests are pre-scheduled, based on manufacturers' guidelines, specifications by plant designers, utility requirements, etc. In many cases, test intervals are inherited and remain unchanged throughout the life of the equipment. The IAEA Safety Guide SG-50-08 [38] gives general guidance on parameters which should be considered in determination of test frequencies and suggests a periodic re-evaluation to account for the effects of ageing.

5.3.3. Condition monitoring

Condition monitoring has become a very popular activity in many industries including the nuclear power industry. Recent preventive maintenance technologies have provided economical tools such as PC-based data acquisition and analysis systems to help monitor the performance of equipment on a periodic or continuous basis while the plant is operating. This can help justify running the equipment without periodic hands-on verification tests until a malfunction is detected or the equipment degradation has exceeded a threshold. For example, vibration analysis or noise analysis may be performed on plant equipment such as pumps and other rotating machinery to verify that the normal and historical signatures are not changing as the equipment ages in the plant. In addition activities such as on-line thermal imaging and similar techniques are becoming popular in condition monitoring programs.

Passive measurement of the AC or DC output of existing sensors or instrumentation channels can reveal vital information about performance characteristics and the health of sensors and associated instrumentation, as well as the plant itself. For example, the AC (noise) output of sensors can be analyzed in frequency domain or time domain to monitor the response time of the sensor and in the mean time, identify and track the process. Problems such as loose parts, flow anomalies, excessive vibration or wearing of reactor internals, reactor stability, and other factors can be monitored and accounted for by on-line monitoring of existing signals in nuclear power plants.

5.3.4. Channel checks

Channel checks are carried out by comparing readings between multiple redundant channels of I&C and adjusting instruments whose readings exceed a predetermined deviation limit. Channel checks are carried out once or twice per shift by operations or maintenance staff on safety or safety related equipment. With on-line drift monitoring using PC-based data acquisition and analysis system, or the plant computer, the need for channel checks may eventually be reduced or eliminated.

More complex methods using models or plant structure information to compare measurements of different types (e.g. pressure, temperature and flow readings) have been developed. However experience is limited and few published reports exist to demonstrate the long term robustness of such methods.

5.3.5. Optimization of maintenance practices

In recent years, condition-based monitoring and risk-based analysis concepts have lead to efforts to identify test frequencies and replacement schedules based on analysis of historical performance. Furthermore, probabilistic risk analysis (PRA)/probabilistic safety analysis (PSA) techniques are providing the foundation for identification of equipment important to safety and which should therefore be attended to more often. The effectiveness of maintenance programmes can be evaluated using historical data, PRA/PSA, and other means to ensure that important equipment is not failing, and guidelines, including testing and evaluation procedures and frequencies, can be developed to avoid problems.

Economic pressures have stimulated the nuclear industry to optimize testing schedules to reduce maintenance costs. As a result, efforts are now underway to define objective basis for the frequency of tests performed on the plant equipment. To accomplish this, plants are using equipment performance histories, generic information from databases, and the results of ageing research programmes such as the ones documented in the reports and publications listed in the bibliography section of this TECDOC. For example, a review of databases as well as data from representative plants has shown that there is insufficient degradation of pressure transmitters in nuclear power plants to justify full calibrations at every refueling outage. As a result, the nuclear power industry is discussing the issue with the regulators in an attempt to extend the instrument calibration intervals. These efforts are getting support from new test equipment that has been developed to record the steady-state output of pressure instrumentation channels in nuclear power plants and identify the instruments that are drifting.

There are also activities underway in several countries to move toward condition based calibration. A majority of hands-on calibration efforts in nuclear power plants have been found to be unnecessary and sometimes counter-productive. An on-line monitoring system can identify the items of equipment that drift and segregate them for manual calibrations during refueling outages. With this approach, equipment that does not drift is not calibrated, saving cost, radiation exposure, and risk of calibration-induced damage to the equipment and the plant. The implementation of an on-line monitoring system for calibration verification not only provides a means to optimize calibration frequencies, but also will help identify problems as they occur.

A systematic methodology such as reliability centered maintenance (RCM), could be applied to evaluate and optimize the current maintenance programmes. However, the RCM methodology should be adapted to the analysis of an existing maintenance programme, and the analysis work effort should be reduced to make it more useful in practice. RCM methodology, which is fully applicable for mechanical and electromechanical systems (sensors for example), is less efficient for pure electronic systems. Indeed, it is very difficult to make a complete FMEA and FMECA for electronic components.

Listed below are some key ingredients needed for the development of maintenance strategy of ageing equipment (the order has no importance):

(1) *Failure histories*

Purpose: At first, a rough quantitative analysis to rank components according to detected failures followed by a detailed analyses to provide failure modes and description of effects.

Limitations: The problem areas of plant information systems are discussed in Section 5.5. For rough (first impression) quantitative analyses the failure database may often be satisfactory. Overall the number of failures may be very low limiting the benefit of the analysis.

(2) *Maintenance information*

Purpose: At first, to identify maintenance costs and current maintenance strategy. For detailed analyses review the current maintenance activities, including testing.
Limitations/problems: This requires the existence of maintenance databases with information on maintenance operation, type of maintenance (preventive/corrective), and costs of plant personnel or external companies, spare parts, etc.

(3) *Effect on plant availability*

Purpose: To identify the costs of failures. Some information can usually be obtained from the failure records.
Limitations/problems: In this field it is often not so easy to identify the cost. Due to redundant systems, I&C failures rarely result in the unavailability of the plant. If I&C failure reduces plant availability, so the cost is the value of lost generation. The French experience shows, that the unavailability caused by I&C failures is of the order of 0.1% of the total plant unavailability.

(4) *Safety significance for safety related systems and equipment*

Purpose: Provide information on component safety significance. It is a limiting factor, if there are other indicators which would suggest maintenance reduction. Component safety classification or indicators from probabilistic safety analyses (risk importance measures) can be used for this purpose.

It is of benefit to prioritize the components in order to concentrate the efforts on most important items. A way of setting priorities is to class the different systems of a plant by answering the following questions:

- What are the systems important for safety?
- What are the systems important for availability?
- What are the systems whose maintenance cost is high?

A first conclusion is that a system which does not belong to any of the three lists, needs only corrective maintenance. For the other systems a simplified procedure is presented in Fig. 3. The words many, high, low, almost no, have no specific meaning and should naturally be defined in a real application.

If further analyses are needed, the observed and potential failure modes should be studied in detail, accounting for environmental and operational conditions, and safety and availability aspects.

In an ideal situation, the production of "top ten" is straightforward, and thus the lists would be continuously updated. This would require information systems where databases on maintenance, failures, manufacturer's information, etc. are linked together.

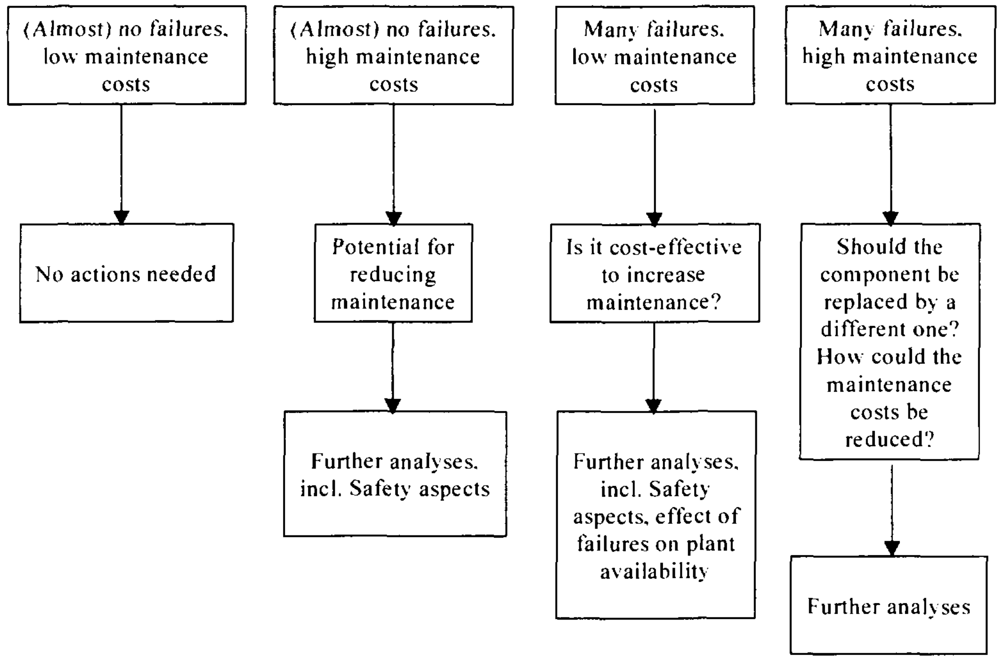


FIG. 3. A simplified procedure for identification of needs for changes in maintenance strategy.

As an example of development of maintenance strategy, the following points summarize the phases in the history of maintenance planning at EdF:

Phase 1: In the very early stages, maintenance is based on documentation supplied by the manufacturer.

Phase 2: Plant personnel develop a local maintenance programme and increase maintenance by using accumulated operating experience and time from Phase 1. This second strategy induces high costs because maintenance people try to reach technical perfection on systems.

Phase 3: Define a "basic preventive maintenance programme" which is the synthesis of the accumulated experience of all the plants. This national programme replaces the old local programmes. The costs stay high or decrease slowly.

Phase 4: Use RCM to optimize Phase 3. The optimization decreases the costs without detriment to safety or availability of systems.

Final Phase: (Current EdF position): Introduce where it is possible, conditional maintenance in place of preventive maintenance.

Each phase is a possible strategy but does not achieve the same result in terms of cost, technical results, availability, etc.

The scope for development of maintenance strategies naturally depends on the national regulations. In some countries, the margin for alternative maintenance strategies is very limited, because the legally required periodic testing and the manufacturer's specifications determine the time and scope of component replacements. The NPP operators try to prove that reduced testing frequencies are sufficient and that more testing can be performed during normal operation, utilizing the existing redundancies with the aim of reducing outage times.

EPRI has produced a family of I&C life-cycle management methodologies which provide a structured approach for the consideration and management of many of the issues referred to above. In particular the System Maintenance Plan methodology describes a method to optimize the maintenance and management of an existing I&C system.

5.3.6. Periodic safety reviews

Previous sections have considered ageing management strategies based on continuous and frequent monitoring of equipment performance and events. An alternative approach may be based on a periodic review of systems and equipment to ensure that design targets are being met and that the ageing is not likely to be of specific concern for the next period of operation.

Periodic reviews consider the operational and maintenance history of the equipment to detect the onset of ageing problems. Assessment of operating environment will also assist in these judgments. The periodic review also provides an opportunity to consider information from other sources such as research programs and other utilities.

The periodic review allows the engineer to stand back from day to day plant problems and take an overview of performance. However the review requires plant historical data of the same quality and depth used in other ageing management strategies and if this data is not available, the review programme will require a large effort to retrieve it.

5.3.7. Environmental monitoring

Monitoring the temperature, radiation, humidity, and other conditions to which an I&C component is exposed has become a popular means for life extension and ageing management. The useful life of equipment is typically specified by manufacturers based on the expected conditions to which the equipment may be exposed during normal operations. On the other hand if the equipment is used in a milder environment, then its expected lifetime is typically longer than the life specified by the manufacturer. If the equipment is used in a harsh environment, its lifetime may be shortened depending on the intensity of its environmental conditions. In a nuclear power plant, the control room temperature is monitored in a number of locations near instrument cabinets as a way of determining the useful lifetime of the equipment in the instrument cabinets. For this application, RTDs are often used for the environmental temperature measurements. These RTDs may be periodically cross-calibrated to verify their accuracy because even one-degree difference in the environment temperature can make a significant difference in how long the equipment may survive. Evidence from such monitoring can be used to persuade regulators to extend the permitting operating period of the I&C equipment concerned.

Environmental monitoring is also used for life extension of large plant components such as pressurizers. For example, the environmental temperature outside the pressurizer nozzle is measured in some plants as a measure of its ageing status and for life extension purposes.

5.4. EXAMPLE METHOD FOR EQUIPMENT LIFE EXTENSION

As described elsewhere in this TECDOC extension of equipment operating life is a possible ageing management strategy. This is often possible as I&C systems are generally designed and developed to meet high reliability targets. The importance of conservative design, specifying components to operate well within their normal ratings rather than just to operate within specified limits, has already been discussed but is restated here. Equipment life extension will also be greatly assisted by careful design of maintenance, again highlighting the need for ageing management strategies to be in place at the start of system life.

The following sections outline a formal life extension strategy, which has been successfully used on RBMK power station in the Russian Federation.

5.4.1. Method

The method is generally applicable where systems have been licensed for a defined life, and therefore formal relicencing is required. The essence of the method is a structured review of the current state of the equipment and an assessment of its operational history paying particular regard to:

- Significant increase in failure rates or deviations from normal performance;
- Exhaustion and/or obsolescence of spare parts.

The method is broken down into four distinct stages.

5.4.2. Stage 1 – review

Key steps are:
- Analyze system data and performance from the previous period of operation, paying particular attention to failures and deviations from normal performance or condition.
- Survey the condition of the equipment looking for physical signs of ageing. Measure parameters such as insulation resistances and power supply output voltages, degradation of which could indicate early signs of ageing.
- Survey the inventory of spare parts.

Stage 1 is then formally reported describing the results of the review and identifying the requirements for:

- Spare parts procurement.
- Module or sub-system replacements.
- Repair work.

5.4.3. Stage 2 – remedial actions

Stage 2 involves taking action on the recommendations made in stage 1 and then formally reporting the outcome.

5.4.4. Stage 3 – assessment

Stage 3 is a formal assessment of the system against its technical specification requirements and would include:
- Power supply unit performance.
- Function of all alarm elements in the system.
- Failure modes and mechanisms.
- Functional capability.

This stage if formally reported to identify any deviations or shortfalls and to suggest any possible solutions for correction.

5.4.5. Stage 4 – justification for extension

Stage 4 should only proceed subject to a successful outcome of the previous stages. It is a formal proposal and justification for extending the operating life of the system and as a minimum should include:

— Additional operating period of the system.
— General review of the previous period of operation.
— Identification of any changes in system characteristics.

5.5. PLANT INFORMATION MANAGEMENT

At every nuclear power plant, design, operating and maintenance information is collected and stored in order to provide information for various decision making situations. This plant specific operating experience is also of utmost importance in ageing follow-up and decision making related to mitigation of the effects of ageing.

Often, the existing plant information systems do not meet all the needs of an efficient ageing management of I&C components. Some problem areas are listed below:

— The collection of information is not "centralized" (e.g. maintenance personnel may have a separate database for calibration data, etc.).
— Information on repairs done outside the plant (e.g. failed electronic cards sent to manufacturer or to a research department) is not included.
— Failure and component coding and classification may not be suitable for I&C systems.
— Lots of old records are not stored in computer system, or they are on a hardware basis which has become incompatible to today's systems (e.g. 8"-disks).
— The systems do not provide means for displaying graphically component failure histories and for trending.

There may be a need, for reasons other than ageing management, to upgrade or replace the plant data collection system. At this time, it is beneficial to account for the ageing management aspects. Modern plant-wide, decentralized operation management PC network systems combine trending modules and spare parts management functions.

Detailed descriptions of good practices for data collection and record keeping are given in IAEA Safety Series No. 50-P-3 (1991), Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing [1]. Some of the key aspects of data collection practices for ageing management are listed here:

— The information system should provide comprehensive and accurate information about components, including baseline information and operation and maintenance data.
— Data should be entered by maintenance and operations personnel directly in a machine readable form, and the data entry should include an appropriate quality control mechanism.
— Databases distributed throughout the plant should have common organization and format.
— The system should provide adequate tools for data analysis, graphical display and production of reports. This could include a detailed classification of age-related "keywords" for data retrieval, and trend analysis tools.

The coding of I&C components should be reviewed. As an example, it may often be the case, that the failures of control electronics are listed under the code of the mechanical component (pump, valve). All the components of measuring instrumentation, i.e. transmitters, converters, limit signal units, etc., have the same code referring only to the measurement. A more detailed coding would help the data retrieval and analyses.

The accumulated information on component failures can be used to identify possible problem areas and effects of ageing. The analyses of failure histories can be roughly divided into quantitative and qualitative studies.

In the simplest quantitative analyses, the number of failures of components (or component groups) are identified. This information can be used to focus further analyses to problem areas. Trend analyses are aimed at identifying time dependence of failure occurrence. Increasing trends are alerting signals of possible age-related problems. In some cases, trend analyses may be used to predict the remaining lifetime of components, but in general there are no reliable methodologies for this purpose. Decisions for in-time replacement will therefore have to be taken on the ground of statistical failure data and on the results of pre-ageing tests, especially those for LOCA resistance, and a large amount of judgment.

Qualitative analyses are used to identify age-related failures from other random failures. The most elementary form of a qualitative analysis is reading through the failure descriptions. Even a single failure description may reveal an ageing problem. On the other hand, a qualitative study of a sudden peak in failure occurrence may reveal a common cause maintenance error and not an age-related phenomenon. Thus, it is important that conclusions are not drawn solely on the basis of quantitative analyses. More detailed analyses can be done e.g. by applying failure modes and effects analyses (FMEA) methodology on plant operating experience. Analyses of root causes of failures are sometimes difficult or impossible to do because of the sparse or missing information in failure reports, but the maintenance personnel can often provide further details for the analyst on a specific event.

Besides the plant specific information, the operational feedback from other plants is an important source of information. Organizations such as INPO, WANO provide an international focus for this and failure data is also collected on a national level: VGB in Germany, EdF in France, and both commercial nuclear power utilities in the UK as examples.

5.6. SKILLED PERSONNEL TO MAINTAIN THE EQUIPMENT

Much of the information on older I&C equipment resides with individuals who will become unavailable at some stage through retirement or other reasons. Loss of this corporate memory is inevitable and it is unrealistic to assume that such information can be transcribed from the individual prior to departure. The following actions can be taken in such instances. Interviewing staff to ascertain:

— Current I&C equipment problems and possible root causes.
— Anticipated equipment performance or reliability problems.
— Historical problems of a one-off nature which were costly to rectify.

Methodologies have been developed to extract such information from staff. A specific example is the Electrical Power Research Institute (EPRI) System Maintenance Plan (SMP) methodology EPRI-TR-106029 which defines a structured series of questions for plant staff to assist in determination of future maintenance plans for the equipment. It is worth noting that such interviews should not be restricted to maintenance staff. Operations and engineering staff may also possess valuable opinions and information.

The plant personnel maintaining the I&C equipment and implementing ageing management tests must have adequate basic training as well as specific training to conduct the ageing management tests. The training of these personnel must be documented. A number of sources may be used for the training of personnel to perform ageing management tests. This includes on-the-job training, formal classroom training provided by plant training departments or the vendor of the test equipment, etc.

Most I&C engineers and technicians in nuclear power plants have good training on conventional test and equipment performance verification such as calibration work on sensors and instrument channels, measurement of insulation resistance, loop resistance, and other electrical parameters to perform I&C troubleshooting, etc. However, in using new and automated test equipment and implementing new ageing management tests such as noise analysis, in situ response time measurements and cable diagnostics, some advanced training is required. In particular, computer training is important although computer programming abilities are not always needed. The plant maintenance personnel need computer training not in the programming area, but in using various computer equipment and software packages.

New techniques used in analysis of test data such as artificial intelligence, neural networks, and export system concepts are among new areas on which there is very little knowledge, experience, and training among average nuclear power plant personnel. These new areas are often the driving force behind most of the modern signal analysis techniques for on-line testing of plant equipment. Thus, future development of training programs and training activities in nuclear power plants should take these areas into consideration.

5.7. AGEING MANAGEMENT OF PROGRAMMABLE SYSTEMS

The short product life cycles of programmable system hardware requires that an ageing management strategy be in place as soon as a system is installed. This requires management of documentation for hardware and especially software to ensure that future replacement is practicable and straightforward. Documentation should follow the accepted best practice for the utility but should specifically include the following documents to assist in ageing management:

Software:

— Functional specifications abstracted to a high enough level to allow implementation using a different software language or operating system.

— Descriptions of software modules and interfaces in plain language adequate to define functionality.

— Information on configuration and other special-to-project aspects of the software or operating system.

— Regular backups of amendable or alternating data with special attention paid to systems where redundant or backup processing exists.

Hardware:

— Definition of interfaces, data exchange protocols and standards.

— Detailed specification and description of each module sufficient to allow individual items to be reconstructed or emulated. This is especially important where overall system maintainability is compromised by problems with a small number of modules (this is usually the case). Where no such information exists then a 'root cause of problem' analysis should be carried out to identify key areas for development of such documents. Comprehensive redrafting of such documents to cover an entire system is likely to be prohibitively expensive and is not justified for modules which are proven to be reliable and adequate spares exist.

Replacement of programmable systems is considered in more detail in IAEA-TECDOC-1016, “Modernization of Instrumentation and Control in Nuclear Power Plants” [5]. Discussion of the practicality and benefits of upgrading analogue or digital systems to programmable systems is strictly beyond the scope of this report although it is worth noting the advantages which may accrue (e.g. increased functionality, reducing manning levels and maintenance) may be more that offset by the cost of qualifying the software.

6. CONCLUSIONS

The objective of an I&C ageing management strategy is to provide for timely detection and mitigation of ageing effects in I&C systems in order to manage and minimize the risks to safety and business targets. In this report, ageing management of I&C equipment is addressed from two directions. First, the report provides information on important ageing mechanisms and their effects on the performance of typical nuclear power plant I&C equipment, as well as a means to manage the ageing at equipment level. Second, generic guidelines for developing an ageing management strategy for I&C equipment at NPPs are discussed. Current ageing research activities are reviewed, and also covered in country reports annexed to this publication. Furthermore, standards related to I&C ageing are listed. Although there is no specific standard on ageing of NPP I&C equipment, ageing test methods for electrical and I&C equipment have been addressed in a number of international standards.

The ageing management process consists basically of three phases, namely: selection of components for which ageing should be evaluated, understanding the ageing mechanisms, and development and implementation of test methods for monitoring and mitigating ageing.

With respect to ageing management, the choices are:

(1) Periodic measurement and testing as needed to verify the performance of the equipment and ensure that ageing has not resulted in unacceptable degradation. In this report, techniques such as on-line calibration verification, loop current step response testing, and noise analysis were discussed as new means for verifying the performance of sensors and related instrumentation while the process is operating.
(2) Replacement of components.
(3) Controlling and slowing down the ageing process by either optimizing the maintenance procedures, or changing the operating or environmental conditions.

Besides the above-mentioned technical aspects of ageing management, the human and organizational aspects are of importance. Action should be taken to transfer the knowledge of experienced plant personnel to the new generation in order to avoid loss of information, e.g., due to retirement of individuals. This can be managed to some extent by interviewing staff. The maintenance and operations personnel must also have adequate training to use new and automated test equipment. Another important question is the organization of the ageing management programme — guidance for this is given in Ref. [1].

NEXT PAGE(S)
left BLANK

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Data collection and Record Keeping for the Management of Nuclear Power Plant Ageing: A Safety Practice, Safety Series No. 50-P-3, IAEA, Vienna (1991).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for the Management of ageing of Nuclear Power Plant Components Important to Safety, Technical Reports Series No. 338, IAEA, Vienna (1992).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Pilot Study on the Management of Ageing of Instrumentation and Control Cables, IAEA-TECDOC-932, IAEA, Vienna (1997).

[4] IAEA Specialists Meeting on Experience in Ageing, Maintenance, and Modernization of Instrumentation and Control Systems for Improving Nuclear Power Plant Availability, Rep. NUREG/CP-0134, Rockville, Maryland, 1993.

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Modernization of Instrumentation and Control in Nuclear Power Plants, IAEA-TECDOC-1016, IAEA, Vienna (1997).

[6] UNITED STATES NUCLEAR REGULATORY COMMISSION, "Guidance on Cross Calibration of Protection System Resistance Temperature Detectors", Standard Review Plan, Rep. NUREG-0800, Washington, DC (1997).

[7] INTERNATIONAL ELECTROTECHNICAL COMMISSION, IEC Standard "Nuclear Reactors Response Time in Resistance Temperature Detectors (RTDs) — In-Situ Measurements" Rep. CEI/IEC-1224 (1993).

[8] INTERNATIONAL SOCIETY FOR MEASUREMENT AND CONTROL, Response Time Testing of Nuclear Safety-Related Instrumentation Channels in Nuclear Power Plants, ISA Standard S 67.06 (1984).

[9] INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS, Standard Installation, Inspection, and Testing Requirements for Power, Instrumentation and Control Equipment at Nuclear Facilities, Standard 336 (1995).

[10] INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS, Standard Criteria for Periodic Testing of Nuclear Power Generating Station Safety Systems, Standard 338 (1987).

[11] INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations", Standard 323 (1983).

[12] INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS, "Standard Calibration for Post Accident Monitoring Instrumentation for Nuclear Power Generating Stations", Standard 497 (1981).

[13] INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS, "Standard for Design Qualification of Safety System Equipment Used in Nuclear Power Generating Stations", Standard 627 (1980).

[14] German Nuclear Safety Standard, KTA Rule 3706, "Repeating Proof of the Coolant Loss-Breakdown Resistance of Electrical and Instrumentation and Control Components of the Safety Systems", draft safety standard, June 1994.

[15] UNITED STATES NUCLEAR REGULATORY COMMISSION, Proceedings of the Workshop on Nuclear Power Plant Ageing, Rep. NUREG/CP-0036, Bethesda, Maryland (1982).

[16] UNITED STATES NUCLEAR REGULATORY COMMISSION, Nuclear Plant Ageing Research (NPAR) Program Plan, Rep. NUREG-1144, US NRC, Washington, DC (1987).

[17] PHILLIPS, P.J., Fingerprinting the Thermal History of Polymeric Materials, Electric Power Research Institute, Rep. EPRI TR101205 (1992).

[18] GARDNER, J.B., SHOOK, T.A., "Status and prospective application of methodologies from an EPRI sponsored indenter test project", Proceedings on Power Plant Cable Condition Monitoring, Rep. EPRI EL/NP/CS-5914-SR, Electric Power Research Institute, (1988).

[19] OECD NUCLEAR ENERGY AGENCY, Nuclear Safety Research in OECD Countries, Capabilities and Facilities, OECD-NEA, Paris (1997).

[20] Proc. PLIM/PLEX 97 Conf., Nuclear Engineering International, United Kingdom (1997).

[21] HAYNES, H.D., KRYTER, R.C., How to monitor motor-driven machinery by analyzing motor current, Power Engineering (1989).

[22] HAYNES, H.D., "Ageing and service wear of electric motor-operated valves used in engineered safety-feature system of nuclear power plants", Ageing Assessments and Monitoring Method Evaluation, Rep. NUREG/CR-4234, US Nuclear Regulatory Commission, Washington, DC (1989).

[23] HASHEMIAN, H.M., et al., On-Line Testing of Calibration of Process Instrumentation Channels in Nuclear Power Plants, Rep. NUREG/CR-6343, US Nuclear Regulatory Commission (1995).

[24] UNITED STATES NUCLEAR REGULATORY COMMISSION, Periodic Testing of Electric Power and Protection Systems, Regulatory Guide 1.118, Rev. 2, US NRC (1978).

[25] Reactor Pressure Vessel Internals, German Nuclear Safety Standard, KTA3204, Draft Safety Standard Revision, June 1997.

[26] STEGEMANN, D., RUNKEL, J., "Experience with vibration monitoring in German PWRs", (Proc. SMORN VII Symposium, Avignon, 1995).

[27] WACH, D., BASTEL, W., "On-line condition monitoring of large rotating machinery in NPPs", Nuclear Power Plant Instrumentation, Control, and Human-Machine Interface Technologies (Proceedings conf. 1996), Pennsylvania State University (1996).

[28] UNITED STATES NUCLEAR REGULATORY COMMISSION, Failure of Rosemont Models 1153 and 1154 Transmitters, NRC Information Notice 89-42, US NRC, Washington, DC (1985).

[29] HASHEMIAN, H.M., PETERSEN, K.M., "Loop current step response method for in-place measurement of response time of installed RTDs and thermocouples", (Proc. 7th Int. Symp. on Temperature), Volume Six, American Institute of Physics, Toronto, (1992) 1151–1156.

[30] AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Methods for Testing Industrial Resistance Thermometers, Standard E644, ASTM (1978).

[31] "Electromagnetic compatibility", Generic Emission, Euro EMC Standard EN 50081-2: Part 2: Industrial Environment.

[32] "Electromagnetic compatibility", Generic Immunity, Euro EMC standard EN 50082-2: Part 2: Industrial environment.

[33] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Functions and Component Classification for BWR, PWR, and PTR: A Safety Guide, Safety Series No. 50-SG-D1, IAEA, Vienna (1979).

[34] INTERNATIONAL ATOMIC ENERGY AGENCY, Protection System and Related Features in Nuclear Power Plants, A Safety Guide, Sefety Series No. 50-SG-D3, IAEA, Vienna (1983).

[35] INTERNATIONAL ATOMIC ENERGY AGENCY, Code on the Safety of Nuclear Power Plants: Operation, Safety Series No. 50-C-O (Rev.1), IAEA, Vienna (1988).

[36] INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Nuclear Power Plants for Safe Operation: A Safety Guide, Safety Series No. 50-SG-O9, IAEA, Vienna (1984).

[37] INTERNATIONAL ATOMIC ENERGY AGENCY, Implementation and Review of Nuclear Power Plant Ageing Management Programme. A Safety Report, Safety Series No. 15, IAEA, Vienna (1999).

[38] INTERNATIONAL ATOMIC ENERGY AGENCY, Surveillance of Items Important to Safety in Nuclear Power Plants: A Safety Guide, Safety Series No. 50-SG-O8, IAEA, Vienna (1990).

BIBLIOGRAPHY

This section provides a listing of reports, papers, and publications that include ageing information and data on electrical and I&C equipment in nuclear power plants.

Acoustic Monitoring Systems for Loose Parts Detection in Characteristics, Design Criteria, and Operational Procedures, IEC Standard CEI/IEC 988 (1990).

ALT, FUCHS, KRAPF, PETER, SCHALK, SEEVERS, WENK, Alterungsmanagement in deutschen Kernkraftwerken, VGB Technical Association of Large Power Plant Operators, Essen (1997).

AMERICAN NUCLEAR SOCIETY, Nuclear Plant Instrumentation, Control and Human-Machine Interface Technologies (Proc. 1996 American Nuclear Society Int. Top. Mtg, "NPIC&HMIT'96, Vol. 1 and 2, Pittsburgh, PA (1996).

AMERICAN SOCIETY OF MECHANICAL ENGINEERS, In-Service Monitoring of Core Support Barrel Axial Preload in Pressurized Water Reactors, ANSI/ASME Standard OM-5 (1981).

AMERICAN SOCIETY FOR TESTING AND MATERIALS, Standard Methods for Testing Industrial Resistance Thermometers, ASTM Standard E66 (1978).

AMERICAN SOCIETY OF METALS, Nuclear Power Plant Ageing, Availability Factor and Reliability Analysis (Proc. Int. Conf. San Diego, 1985)(GOEL, Ed.) American Society of Metals (1985).

ARLOTTO, A., "Understanding ageing: A key to ensuring safety" (Proc. Int. Conf. on Nuclear Power Plant Ageing, Availability Factor and Reliability Analysis, San Diego,1985) (GOEL, V.S., ed.), Nucl. Safety **28** 1 (1987) 7-10.

BEHBAHANI, A., MILLER, D.W., Experimental and mathematical simulation techniques for determining an in-situ response time testing method for neutron sensors used in reactor power plant protection systems, Nuclear Technol. **67** 14 (1984).

BROOKHAVEN NATIONAL LABORATORY, Second ANS Workshop on the Safety of Soviet-Designed Nuclear Power Plants, Summary Rep. BNL-52457, New York (1996).

CARFAGNO, GIBSON, R.J., A Review of Equipment Ageing Theory and Technology, Rep. EPRI NP-1558, Electric Power Research Institute (1989).

CHIU, C., et al., Root Cause Analysis for ITT Barton Pressure Transmitter Failures at the Trojan Nuclear Power Plant, Rep. 88-011, Rev. 1, Failure Prevention, Inc. (1988).

CONGRESS, OFFICE OF TECHNOLOGY ASSESSMENT, Ageing Nuclear Power Plants: Managing Plant Life and Decommissioning.

DINSEL, M.R., et al., In Situ Testing of the Shippingport Atomic Power Station Electrical Circuits, Rep. NUREG/CR-3956, Nuclear Regulatory Commission (1987).

ELECTRIC POWER RESEARCH INSTITUTE, A Review of Equipment Ageing Theory and Technology, Rep. NP-1558, Electric Power Research Institute, Palo Alto, California, 1980.

ELECTRIC POWER RESEARCH INSTITUTE, Instrumentation and Control Life Cycle Management Plant Methodology, Rep. EPRI TR-1055555, Vols 1 and 2.

ELECTRIC POWER RESEARCH INSTITUTE, Plant Communications and Computing Architecture Plant Methodology, Rep. EPRI TR-104129 Vols 1 and 2.

ELECTRIC POWER RESEARCH INSTITUTE, Instrumentation and Control Upgrade Evaluation Methodology, Rep. EPRI TR-104963.

ELECTRIC POWER RESEARCH INSTITUTE, Instrumentation and Control System Maintenance Planning Methodology, Rep. EPRI TR-106029 Vols 1 and 2.

ELECTRIC POWER RESEARCH INSTITUTE, Seismic Ruggedness of Aged Components, Rep. EPRI NP-5024 (1987).

ELECTRIC POWER RESEARCH INSTITUTE, Correlation Between Ageing and Seismic Qualification for Nuclear Plant Electrical Components, Rep. EPRI NP-3326 (1983).UNITED STATES NUCLEAR REGULATORY COMMISSION, Proceedings of the Workshop on Nuclear Power Plant Ageing, Bethesda, MD,Rep. NUREG/CP-0136 (1982).

ELECTRIC POWER RESEARCH INSTITUTE, Remote Calibration of Resistance Temperature Devices (RTDs), Rep. NP-5537, Palo Alto (1988).

ELECTRIC POWER RESEARCH INSTITUTE, Maintenance and Applications Guide for Control Relays and Times, Rep. EPRI TN-102067, Palo Alto, CA.

ELECTRIC POWER RESEARCH INSTITUTE, Fingerprinting the Thermal History of Polymeric Materials, Rep. EPRI TR-101205, University of Tennessee, Knoxville (1992).

ELECTRIC POWER RESEARCH INSTITUTE, Guideline on Licensing Digital Upgrades, Rep. EPRI TR-102348, Palo Alto, CA (1993).

ELECTRIC POWER RESEARCH INSTITUTE, The Value of a Power Plant's Remaining Life: A Case Study With Baltimore Gas & Electric Co., Rep. EPRI EA-4347, Palo Alto, CA (1985).

ELECTRIC POWER RESEARCH INSTITUTE, Nuclear Power Plant Common Ageing Terminology, Rep. EPRI TR-100844, Palo Alto, CA (1992).

ELECTRIC POWER RESEARCH INSTITUTE, Natural Versus Artificial Ageing of Electric Components, Rep. EPRI TR-106845, Palo Alto, CA (1997).

Erneuerungsstrateglen fur Lelttechniksysteme in Kernkraftwerken. VGB Arbeitskreis Leittechnik in Kernkraftwerken, VGB Technical Association of Large Power Plant Operators, Essen (1997).

FOSTER, C.G., et al., Sensor Response Time Verification, Rep. NP-267, Electric Power Research Institute, Palo Alto, CA (1976).

FOSTER, C.G., Sensor Response Time Verification, Rep. EPRI NP-267, Electric Power Research Institute (1976).

FURGAL, C.M. CRAFT, E.A. SALZAR, Assessment of Class 1E Pressure Transmitter Response When Subjected to Harsh Environment Screening Tests, Rep. NUREG/CR-3863, US Nuclear Regulatory Commission, Rep. SAND 84-1264, Sandia National Laboratories (1985).

GEHL, A.C., HAGEN, E.W., Ageing Assessment of Reactor Instrumentation and Protection System Components, Age Related Operating Experiences, Rep. NUREG/CR-5700, Oak Ridge National Laboratory (ORNL)/US Nuclear Regulatory Commission, Washington, DC

HASHEMIAN, H.M., Ageing of Nuclear Plant Resistance Temperature Detectors, Rep. NUREG/CR-5560, Analysis and Measurement Services Corporation/US Nuclear Regulatory Commission, Washington, DC (1990).

HASHEMIAN, H.M., Effect of Ageing on Response Time of Nuclear Plant Pressure Transmitters, Rep. NUREG/CR-5383, Analysis and Measurement Services Corporation/US Nuclear Regulatory Commission, Washington, DC (1989).

HASHEMIAN, H.M., Long Term Performance and Ageing Characteristics of Nuclear Power Plant Pressure Transmitters, Rep. NUREG/CR-5851, Analysis Measurement Services Corporation/US Nuclear Regulatory Commission, Washington, DC (1994).

HASHEMIAN, H.M., On-Line Testing of Calibration of Process Instrumentation Channels in Nuclear Power Plants, Rep. NUREG/CR-6343, Analysis and Measurement Services Corporation/US Nuclear Regulatory Commission, Washington, DC (1995).

HEINBUCH, IRBECK, BASTL, Ageing Diagnosis, Prediction and Substitute Strategics for I&C, Bayernwerk Munchen und Institut fur Sicherheitstechnologie GmbH Garching (1995).

Industrial Platinum Resistance Thermometer Sensors, IEC Standard CEF/IEC751 (1983).

INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Aspects of the Ageing and Maintenance of Nuclear Power Plants (Proc. Int. Symp. Vienna, 1987) IAEA, Vienna (1988).

INTERNATIONAL ATOMIC ENERGY AGENCY, Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing in a Safety Practice, Safety Series No. 50-P-3, IAEA, Vienna (1991).

INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety, Technical Reports Series No. 338, IAEA, Vienna (1992).

INTERNATIONAL ATOMIC ENERGY AGENCY, Advanced Control Systems to Improve Nuclear Power Plant Reliability and Efficiency, IAEA-TECDOC-952, Vienna (1997).

INTERNATIONAL SOCIETY FOR MEASUREMENT AND CONTROL, Response Time Testing of Nuclear Safety in Related Instrument Channels in Nuclear Power Plants, ISA Standard S 67.06 (1984).

INTERNATIONAL SOCIETY FOR MEASUREMENT AND CONTROL, Dynamic Response Testing of Process Control Instrumentation, ISA Standard S26 (1975).

KEENAN, M.R., Moisture Permeation Into Nuclear Reactor Pressure Transmitters, Rep. SAND-83-2165, Sandia National Laboratories (1984).

MANGUN, B.W., Stability of small industrial platinum resistance thermometers", J. Research of the National Bureau of Standards. **89** 4 (July-August 1984).

MEALE, B.M, SCATFERWHITE, D.G., An Ageing Failure Survey of Light Water Reactor Safety Systems and Components, Rep. NUREG/CR-4747, Idaho National Laboratory/US Nuclear Regulatory Commission, Washington, DC (1987).

MULLENS, J.A., Experience in Degradation of Pressure Sensor/Sensing Line Systems, Interim Rep. NRC FIN No. B0481, Oak Ridge National Laboratory, Instrumentation and Controls Division (1982).

NATIONAL INSTITUTE OF STANDARDS AND TECHNOLOGY, Detection of Incipient Defects in Cables by Partial Discharge Signal Analysis, Rep. NISTIR 4487, Gaithersburg, MD (1992).

NATIONAL RESEARCH COUNCIL, NATIONAL ACADEMY OF SCIENCES, Digital Instrumentation and Control Systems in Nuclear Power Plants (1997).

NUCLEAR ENGINEERING INTERNATIONAL, Proc. PLEX 93 Zürich International Conference and Exhibition, Zürich, 1993.

NUCLEAR OPERATIONS GROUP, ROSEMOUNT ENGINEERING COMPANY, Status Report on Oil Loss Rosemount Model 1153 Series B/D and 1154 Transmitters, Rosemount Rep. D9200129, Rev. A, Minneapolis, Minnesota (1992).

Nuclear Reactors — Response Time in Resistance Temperature Detectors (RTDs) — In-Situ Measurements, IEC Standard CEI/IEC 1224 (1993).

NUCLEAR REGULATORY COMMISSION, Inspection, Surveillance and Monitoring of Electrical Equipment in Nuclear Power Plants, Vol. 2, Pressure Transmitters, Rep.; NUREG/CR-4257 (1986).

NUCLEAR REGULATORY COMMISSION, Nuclear Plant Ageing Research (NPAR) Program Plan, Rep. NUREG-1144 (1987).

NUCLEAR REGULATORY COMMISSION, Insights Gained From Ageing Research, Rep. NUREG/CR-5643 (1992).

OFFICE OF TECHNOLOGY ASSESSMENT, US CONGRESS, Ageing Nuclear Power Plants: Managing Plant Life and Decommissioning, OTA-E-575, Washington, DC (1993).

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, A Symposium on Nuclear Research Surveillance and Diagnostics (Proc. SMORN VII Symp.), Vol. 1&2, OECD, Paris (1996).

Periodic Tests and Monitoring of the Protection System of Nuclear Reactors, IEC Standard 671 (1980).

Power Plant Cable Condition Monitoring (Proc. workshop San Francisco, 1988) (Del Valle, ed.), Electric Power Research Institute (1988).

Proc. Joint DOE/EPRI Int. Conf. on Cost Effective I&C Technology Upgrade for NPPs, Nashville, 1995, Rep. EPRI/TR-105148, EPRI, Palo Alto, California.

Proc. NRC Regulatory Information Conference, Rockville, Maryland (1995).

Proc. of Topical Mtg on Nuclear Plant Instrumentation and Control and Man-Machine Interface Technologies, Oak Ridge, Tennessee (1993).

Proc. Int. Topical Mtg on WWER Instrumentation and Control, Prague, 1997.

Qualification of Electrical Items of the Safety System for Nuclear Power Generating Stations, IEC Standard CEI/IEC 780 (1984).

REBING, et al., Untersuchung zur Alterung bzw. Der Lebensdauer von elektrischen Einrichtungen des Sicherheitssystems und der Storfallinstrumentierung in kerntechnischen Anlagen unter betrieblichen Einflussen (Alster). TUV Norddcutschland e.V. Abschlubbericht zum BMU – Forschungsvorhaben SR441 (1992).

SANDIA NATIONAL LABORATORIES, Ageing Management Guideline for Commercial Nuclear Power Plants — Electrical Cable and Terminations", Rep. SAND96-0344, Albuquerque, NM (1996).

SCHOHL, G.A., et al., Detection of Air in Sensing Lines from Standing Wave Frequencies, Trans. American Nuclear Society, 1987 Annual Winter Meeting, Los Angeles, CA (1987).

STRAHM, R.C., YANCEY, M.E., TMI-2 Pressure Transmitter Examination Program Year-End Report: Examination and Evaluation of Pressure Transmitters CF-1-PT3 and CF-2-LT3, Rep. GEND-INF-029, EG&G Idaho, Inc., Idaho Falls, Idaho (1983).

TENNESSEE VALLEY AUTHORITY, Response Time of Sensing Lines, DNE Calculation Number SQNRAJ9886, Rev. 5, Sequoyah Nuclear Power Plant (1988).

THIE, Surveillance of Instrumentation Channels at Nuclear Power Plants, Vol. 2: An Approach to Classifying Problems and Solutions, Rep. EPRI NP-6067, Electric Power Research Institute (1989).

TOMAN, , G.J., BACANSKAS, V.P. SHOOK, LODLOW T.A., An Ageing Assessment of Relay and Circuit Breakers and System Interactions, Rep. CR-4715, BNL-NUREG-2017, Brookhaven National Laboratory, Franklin Research Center, Philadelphia, PA, United States Nuclear Regulatory Commission (1987).

UNITED STATES NUCLEAR REGULATORY COMMISSION, Proceedings of the IAEA's Specialists Meeting on Experience in Ageing, Maintenance, and Modernization of I&C Systems for Improving NPP Availability, Rep. NUREG/CP-0134, Rockville, Maryland (1993).

UNITED STATES NUCLEAR REGULATORY COMMISSION, Proceedings of the International Nuclear Power Plant Ageing Symposium, Bethesda, MD Rep. NUREG/CP-0100 (1988).

UNITED STATES NUCLEAR REGULATORY COMMISSION, Proceedings of the Ageing Research Information Conference, Rockville, Maryland, Rep. NUREG/CP-0122 (1992).

UNITED STATES NUCLEAR REGULATORY COMMISSION, Proceedings of the US Nuclear Regulatory Commission, Eighteenth Water Reactor Safety Information Meeting, Rockville, MD, Rep. NUREG/CP-0114, Vol. 3 (1990).

UNITED STATES NUCLEAR REGULATORY COMMISSION, Proceedings of the Twenty-First Water Reactor Safety Information Meeting, Bethesda, MD Rep. NUREG/CP-0133 (1993).

UNITED STATES NUCLEAR REGULATORY COMMISSION, Proceedings of the Twenty-Third Water Reactor Safety Information Meeting, Bethesda, MD Rep. NUREG/CP-0149 (1995).

VORA, NRC Research Programme on Plant Ageing: Listing and Summaries of Report Issued Through September 1993, NUREG-1377, Rev. 4, US Nuclear Regulatory Commission, Washington, DC.

NEXT PAGE(S) left BLANK

ABBREVIATIONS

AMP	ageing management programme
DAT	digital audio tape
EMC	electromagnetic compatibility
EMI	electromagnetic interference
ESR	equivalent series resistance
FMEA	failure modes and effects analysis
FMECA	failure modes effects and criticality analysis
I&C	instrumentation and control
IP	ingress protection
KTA	Kerntechnischer Ausschuss (German nuclear regulatory standard)
LCSR	loop current step response test (a method for in situ response time testing of RTDs)
LER	licensee event report
LOCA	loss of cooling accident
MCSA	motor current signature analysis
MOV	motor operated valve metal oxide varistor
NMAC	nuclear maintenance assistance center
NPAR	nuclear plant ageing research
NPRDS	nuclear plant reliability data systems
O&M	operating and maintenance
OIT	oxidation induction time
PI	power interrupt (a method for testing the in situ response time of force-balance pressure transmitters)
PLIM&PLEX	plant life management and extension
PRA	probabilistic risk assessment
PSA	probabilistic safety analysis
PWR	pressurized water reactor
R&D	research and development
RBMK	light-water-cooled, graphite-moderated reactor (LWGR)
RCM	reliability centered maintenance
RFI	radio frequency interference
RTD	resistance temperature detector
SSCs	structures, systems, and components
SMP	system maintenance plan
SPNDs	self power neutron detectors
T/C	thermocouple
TDR	time domain reflectometry
TUV	technical service organization
WWER	vodo-vodyannoy energeticheskiy reactor (Russian-made PWR)
V&V	verification and validation

Annex

COUNTRY REPORTS

NEXT PAGE(S)
left BLANK

XA0055079

EXPERIENCE BASED AGEING ANALYSIS OF NPP PROTECTION AUTOMATION IN FINLAND

K. SIMOLA
VTT Automation,
Espoo, Finland

Abstract

This paper describes three successive studies on ageing of protection automation of nuclear power plants. These studies were aimed at developing a methodology for an experience based ageing analysis, and applying it to identify the most critical components from ageing and safety points of view. The analyses resulted also to suggestions for improvement of data collection systems for the purpose of further ageing analyses.

1. INTRODUCTION

Nuclear power plant (NPP) ageing analyses are aimed at identifying the important ageing mechanisms and at developing proper ageing mitigation methods in order to ensure the safe operation of the plant to the end of its planned lifetime [1]. Although safety significant equipment have to fulfill strict quality requirements and they have well defined maintenance programmes, the follow-up of ageing is important. For instance changes in operating conditions affect the lifetime of components.

This paper describes Finnish research studies aimed at evaluating the current state of reactor protection systems and identifying the most critical components from ageing and safety points of view. The research work was initiated by a preliminary study of relay failures and cable ageing [2]. In the second study, a methodology for analyzing the ageing of an automation system was developed [3]. The method was applied to selected automation chains of a reactor protection system. In the third study, a similar approach was used but more detailed failure modes and effects analyses were applied to selected items [4].

2. PRELIMINARY STUDY ON AGEING OF I&C EQUIPMENT

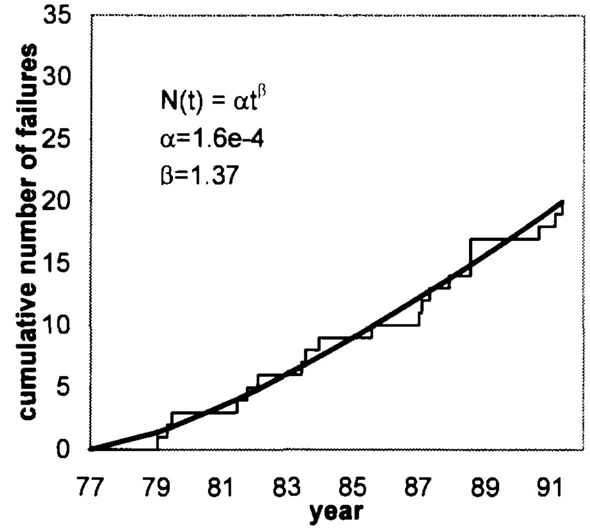
2.1. Analysis of reactor protection system relay failures

The preliminary study was focused on relay failures in the Reactor Protection System (RPS) of Loviisa PWR plant. This study was limited to the analysis of operating experiences collected from work orders at the power plant. One objective of the study was to assess the possibility of using accumulated plant operating experience in ageing analyses.

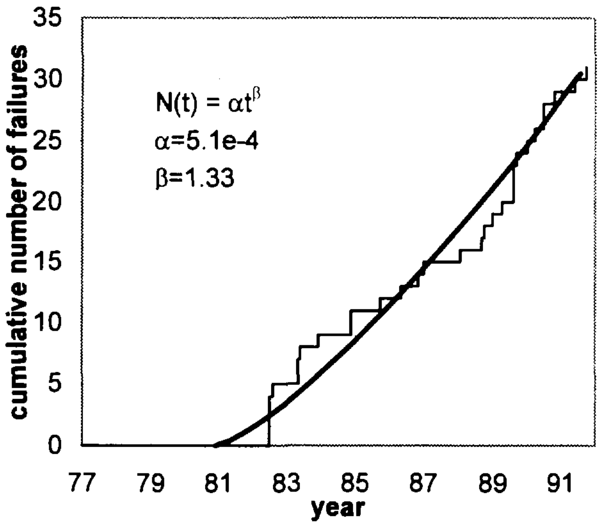
In the study, failure histories of 340 (170 per unit) continuously energized relays in the protection logic of the RPS of the two plant units were collected. The observation period in the study was 15 (11) years at plant unit 1 (unit 2). The numbers of reported failures during these observation periods were 20 at unit 1 and 31 at unit 2. After the data collection, a trend analysis of reported failures was performed and a slightly increasing trend in failure reporting was identified. The trend analysis was done by estimating the parameters of Weibull process from data. The trends are presented in Fig. 1.

The dominating failure mode according to the failure reports was the coil burn out. A more detailed investigation revealed that, at unit 2, most of the failures occurred in relays of two cabinets and especially in relays connected in series with resistors.

Temperatures and voltages were measured in relay cabinets. The voltage measurements showed over voltage due to varying resistance values of the resistors. It was noticed that the ambient temperatures were higher in cabinets of relays connected in series with resistors. Furthermore, it was observed that the temperatures were higher in unit 2 which could explain the higher failure rate in this unit.



a)



b)

FIG. 1. Failure trends of relays. a) unit 1 b) unit 2.

It was concluded that the probable reasons for the coil burn-out failures are the over voltage due to varying resistance values of the resistors and the elevated cabinet temperatures caused by the heat produced by the resistors. It was recognized that the failure rate of some relays could possibly be decreased by reducing the environmental stresses caused by temperature and voltage. The ventilation in cabinets has been improved.

Recommendations were given also concerning the data collection practices. Although failure descriptions were often missing causing difficulties in the identification of critical failures, and the classification of plant instrumentation components is not very detailed, the analysis showed that failure data was useful in indicating problem areas and trends.

2.2. Cable ageing

In the study of cable ageing, a literature review was made and the ageing surveillance programme of in-containment cables at Loviisa PWR plant were evaluated. In the present surveillance programme, cable samples are taken for measurements with a five year's interval in order to follow the environmental effects on cable materials. However, due to cable replacements, the ageing follow-up from the beginning of the plant operation is possible for only one of the present cable types in safety significant installations. Additional samples were taken for possible further measurements. The results obtained in the surveillance programme could be complemented with accident testing of naturally aged cables. A testing programme to evaluate the possible dose-rate effect is introduced.

3. AGEING STUDY OF REACTOR PROTECTION SYSTEM OF OLKILUOTO BWR

After the preliminary study on relay failures, a more extensive ageing study on reactor protection automation was initiated. One objective of the study was to present an ageing analysis approach and apply it to the automation chains of a reactor protection system of Olkiluoto BWR plant. The second objective was to evaluate the possible ageing effects of equipment and their safety significance.

3.1. Analysis approach

As the aim of the study was to evaluate the effects of ageing on the system safety, the collection and analysis of all relevant plant specific information related to the system was considered important. However, due to limited resources, the study had to be focused on the most important topics. The selected approach is primarily based on the analysis of operating experience, e.g. failure reports and maintenance histories.

The analysis of the system structure is important in order to identify the critical paths where failures could prevent the propagation of the protection signal from measuring devices to the intended valve or pump actuation. Existing PSAs can be used or fault trees can be constructed to indicate the safety importance of components. The analysis of occurred failures is essential for the identification of recurrent failures and trends, and the information on the structure of equipment is needed to understand their failure modes. Qualitative methods, such as failure modes and effects analysis (FMEA), can be applied to analyze both the occurred and other possible failure modes. The environmental and operating conditions, and the maintenance practices are reviewed in order to identify their influence on age-degradation. The steps of the study are presented in Figure 2.

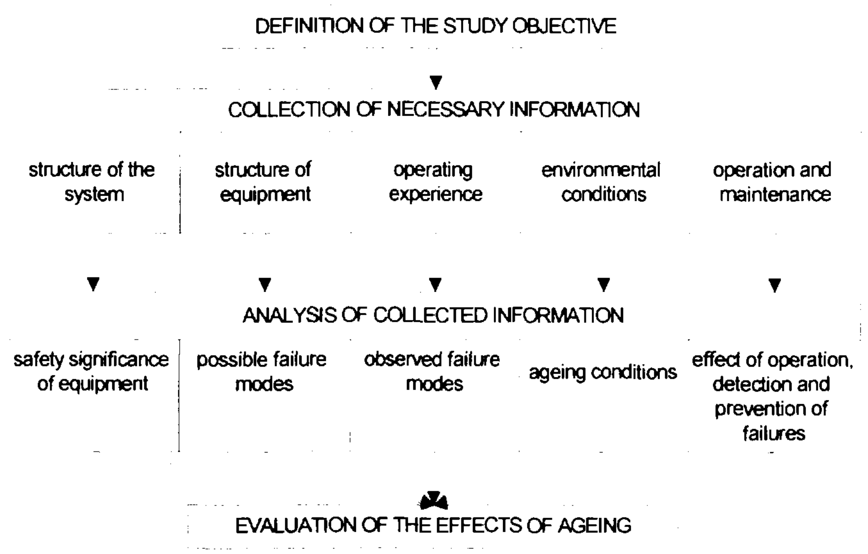


FIG. 2. Steps of the ageing study of reactor protection system.

3.2. Definition of the boundaries of the study and analysis of operating experience

The first step in the study was to select a representative part of the reactor protection system of the plant for the ageing analysis. The reactor trip train components from measurement devices to control electronics were selected as the study objective. The principal scheme of the protection chains is presented in Appendix 1 of this national report. The reactor trip train can be divided into three parts: measurement instrumentation, protection logic, and control electronics. The measurement instrumentation includes sensors, transmitters, neutron flux detectors, room temperature, pressure and float level switches, and other control electronics. The protection logic is based on relays.

Another limitation was needed in regard to the safety functions to be included. The control electronics of three safety functions were included:

- closing of isolation valves of main steam lines;
- opening of valves in the relief system;
- starting of pumps in auxiliary feed-water system.

The operating experience was obtained from the plant failure database. Furthermore, calibration histories of pressure and level transmitters, limit signal units and I/U-converters were obtained from a maintenance database. Considering the number of equipment and the time period investigated, the overall number of failures was low. Especially, the failure rate of the relays was very low.

An increase in failure reporting during last years could be identified in room temperature and pressure measurement instrumentation. The failure modes and effects of room control switches were analyzed in order to identify possible ageing-related failure modes that could prevent the progression of protection signal. A simplified FMEA with specific attention to age-related failure modes was applied for this purpose. An example of such an analysis is shown in Table I.

TABLE I. FAILURE MODES, EFFECTS AND AGEING ANALYSIS OF A ROOM TEMPERATURE SWITCH

Component	effect	failure mode	failure cause	age related
Room temperature switch	no signal	setpoint too high	drift, e.g. due to grease hardening	yes
			wrong setting	no
		contact does not open	capillary tube or membrane damaged	no
			switch failed	yes
	unnecessary signal	setpoint too low	drift, e-g- spring has lost elasticity	yes
			wrong setting	no

3.3. Reliability studies of the protection automation

The effect of protection automation failures on plant reliability was studied with a living-PSA code [5]. It was of interest to identify how multiple failures of the system would increase the estimate of the core damage frequency.

The level of detail in PSA models was varying: the reactor measurement circuits were modeled in detail but e.g. room control chains were not modeled with same precision. Failure rates of control electronics were not considered separately but included in failure rates of pumps and valves. Accident sequences were selected for the studies by the following criteria:

— minimal cut sets for which probability exceeds a certain limit;
— sequence includes components of protection automation.

PSA models were used to study the impact of the increased failure rates of protection automation components on accident sequence frequencies. In this connection also the effect of the multiple failures of the system on the core melt frequency was identified. In the sensitivity analyses, the following aspects were considered:

— severity of consequences;
— probability of the accident sequence;
— sensitivity of the accident sequence probability to failures of protection automation.

It was observed that the frequency of the accident sequences, most sensitive to the increase of failure rates of protection automation, does not significantly contribute to the core melt frequency. Investigated cases with simultaneous failures in all four redundant channels were not realistic as result of ageing considering the testing and maintenance of equipment.

3.4. Conclusions

The number of failures was low, and the sensitivity analyses with the living-PSA tool showed that the impact of increase in failure rates of RPS components on plant safety was small. In reactor measurements, typical transmitter failure modes are calibration shifts and response time degradation. A major failure of transmitter or converter is detected immediately. For relays, only few failures had occurred, and critical failures are very unlikely at low voltages. Failures in control electronics may prevent operation of valve or pump but the evaluation of effects of various failures would require a more detailed study. The possible failure mechanisms of room control switches were evaluated with maintenance staff. Measuring switches have been replaced by a different type.

4. AGEING STUDY OF THE ESFAS OF LOVIISA PWR PLANT

In the study of Engineered Safety Features Actuation System (ESFAS) of Loviisa PWR plant, the approach used was similar to the one described above. However, in this study, a more detailed analysis was performed for some items selected on the basis of operating experience and a fault tree analysis. For these selected electronic devices, failure modes and effects analyses were performed. Table II shows the phases on the study and methods used.

TABLE II. PHASES OF THE STUDY AND METHODS IN THE AGEING ANALYSIS OF ESFAS

PHASE	METHOD
evaluation of failure histories	analysis of operating experience
evaluation of equipment safety significance	fault tree analysis
selection of components for more detailed studies	selection based on the results of phases 1 and 2
evaluation of ageing and failure modes for selected components	failure modes and effects analysis

The operating experience was collected from various sources. The failure data was collected mainly from work orders 1977-1990 and from the plant information system since 1989. Furthermore, information was obtained from reports of scheduled inspections and tests, and e.g. laboratory testing reports.

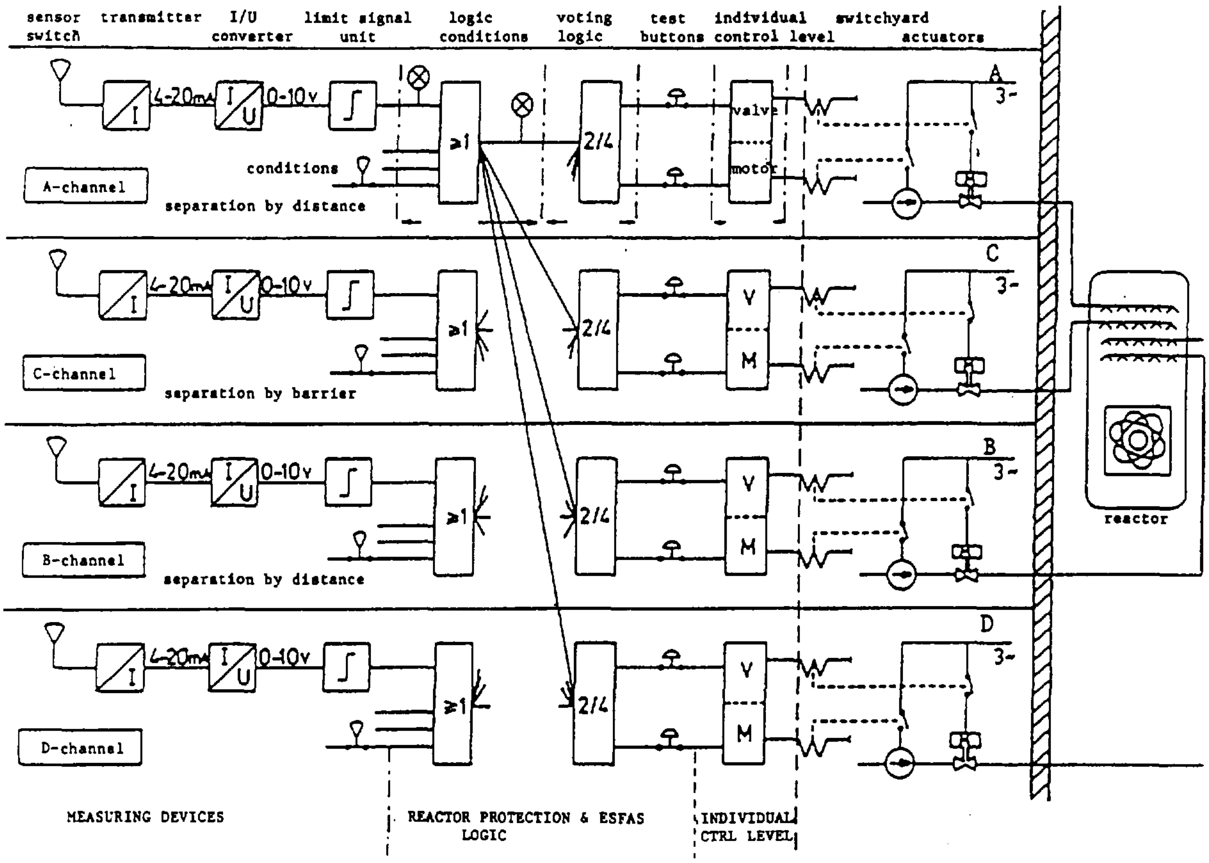


FIG. 3. Principal scheme of the protection chains of the Olkiluoto RPS.

A fault tree analysis was carried out for the safety function: "Starting of emergency feed water pumps if a steam generator level decreases below -140 mm". This was a qualitative analysis with not any failure rates included. The analysis shows which parts of the system are most important for the propagation of the signal. Based on the minimal cut sets, electronic cards were selected for further analyses.

Based on the analysis of operating experience, the limit signal unit and the priority unit were selected for detailed studies. Based on the fault tree analysis, a priority unit, an individual control unit, and a pulse/DC converter were selected for further investigations. An FMEA was performed for these selected cards, at least for those parts related to the paths of the protection signal.

The study showed, that the number of occurred failures is low, and not any clearly increasing trends could be identified. In most cases the failures have caused a false alarm, and only few failures had prohibited the propagation of the safety signal in one channel. These failures were identified mainly in individual control units (relay failures) and in limit signal units. The power supply was not included in the study but it was recommended for further ageing analysis activities.

5. CONCLUSIONS

A methodology for ageing analyses based on operating experience and reliability techniques was developed and applied to protection automation systems. In the cases studied, the amount of failures occurred during the plant operating time was low, and increasing trends in failure occurrence could be observed only for few components. However, as a degradation phenomenon may occur suddenly, it is advisable to inspect regularly failure and maintenance records, e.g. calibrations. The safety importance of age-degradation may be evaluated with PSA models, on conditions that sensitivity analyses can be carried out and that the models are detailed.

Regarding the development of data collection systems, some recommendations were given in order to enable better the utilization of plant experience in ageing studies. The information is often spread in various databases or paper archives, which hinders the efficient use of these records. Integration of various sources of information, such as failure reports and testing, calibration and other maintenance related data, is highly recommended. The system should also provide tools for trend analyses and graphical presentation and, e.g., age-related keywords for data retrieval. The quality of the data may be improved by motivating the plant personnel in filling up more precisely the failure reports.

REFERENCES

[1] Methodology for the management of ageing of nuclear power plant components important to safety. IAEA TS 338. Vienna, 1992.
[2] Simola K. Ageing of electrical and automation equipment - a preliminary study. STUK-YTO-TR 33, Helsinki, 1991 (In Finnish).
[3] Simola K, Hänninen S. Ageing study of protection automation components of Olkiluoto NPP. STUK-YTO-TR 58, Helsinki, 1993 (In Finnish).
[4] Simola K, Maskuniitty M. Ageing study of the ESFAS of the Loviisa NPP. STUK-YTO-TR 86, Helsinki, 1995 (In Finnish).
[5] Niemelä I. STUK living PSA code (SPSA). Use of Probabilistic Safety Assessment for Operational Safety. In: Proceedings of PSA'91 Symposium. Vienna, 1991, 782-783.

MANAGEMENT OF I&C AGEING

L. MOHRBACH
VGB Technical Association,
Technical Association of Large Power Plant Operators,
Essen, Germany

Abstract

This country report is based on data collected within working committees of VGB Technical Association of Large Power Plant Operators, Essen, Germany, namely the report „Ageing Diagnosis, Prediction and Substitute Strategies for I&C" by R. Heinbuch, J. Irlbeck (Bayernwerk Kernenergie AG München) and W. Bastl (Institut für Sicherheitstechnologie GmbH Garching). It was compiled in its current form by L. Mohrbach, VGB Headquarters, in his capacity as member of the IAEA Advisory Group on Nuclear Power Plant I&C Equipment: Periodical Testing, Evaluation and Maintenance Strategy.

1. INTRODUCTION

The operators of nuclear power plants in Germany follow different strategies for the maintenance of instrumentation and control (I&C) technology in their plants, including replacement. Basic tasks and goals of the management of ageing are the diagnosis of inadmissible changes of system characteristics and the provision of the required spare parts and replacement deliveries, taking into consideration operating experience and innovative progress. Key topics in this field are qualification for accident conditions, change-over to computer- based digital I&C or alternative solutions like e.g. replacement by application specific integrated circuits (ASICs).

The nineteen nuclear power plants that are in operation today (1998) in Germany were commissioned between 1968 and 1989. Since then, several I&C generations were developed by the industry, all of which are used in the power plants today. In the first generations, fixed- wired electronic systems in discrete semiconductor technology were applied, supplemented by process computer based information systems. Today, I&C for conventional and for nuclear power plants, be it new construction or replacement, is principally being realized in digital technology. This applies also for the I&C systems with safety relevance of nuclear power plants, both domestically (replacement) and for export projects (replacement and new construction). The manufacturing industry has announced that this technology will be favored in future, leaving the analog I&C systems with increasingly less supplier support.

The proof of functioning of the operational and safety relevant I&C is ensured by extensive repetitive tests. With these tests it was assumed until now that the proof for resistance to accidents, namely the loss-of-coolant accident (LOCA), attained at the plant erection within the frame of the type tests with timelapsing ageing simulation, wouldn't have to be repeated for the remaining life-span of the components.

In the last years extensive discussions and investigations have taken place about the period of confidence for the original accident resistance proof test. The results are laid down in KTA Safety Standard draft 3706 (June 13, 1994). The paper describes the requirements for the repetitive proof of the LOCA resistance of safety system I&C components.

The experience gained from the renewal of these systems is described by concrete examples. The important results are an improvement of the man- machine interface and of system and overall plant availability.

Furthermore, running old I&C technology reliably and securing the delivery of suitable replacement systems are defined as the basic tasks of the management of ageing.

2. PRESENT SITUATION

In an increasing number of cases the original I&C components are not produced any more. The supply of replacement parts, if not taken from stock, can today only be assured by expensive special production runs. These again live from a stock of discrete components (transistors, diodes, etc.) which is not refilled anymore. Only in some cases pin- compatible new generation components are on the market. The availability of spare parts is therefore in principle endangered and the arising costs thereof are not predictable anymore.

Ageing effects can be observed in the older plants (drift phenomena, electrolyte capacitors). The failure rates still lie within the normal range. Only until recently the maintenance measures needed for the securing of function could still be performed with reasonable expenditure in the old technology.

The exchange of the I&C systems will be necessary up to three times during the life-span of a power plant. Replacement actions are currently performed and their number is increasing. For I&C expansions in the operational and in safety relevant areas, computer based digital I&C systems are basically being engaged today. The market for suitable new I&C systems is expected to grow significantly, thus enabling further safe and reliable operation of the power plants.

3. PROOF OF FUNCTION, PERIODICAL TESTING

Qualification, quality assurance and proof of the required functional capabilities of electrical and I&C components is performed during manufacture, assembly and commissioning. The preservation of these characteristics is being proven by periodical testing, which is carried out either during operation or shutdown of the plant.

These tests require a relatively large personnel effort, if testing has to be performed manually. So, as far as possible, automatic testing equipment, namely computerized test devices are used, which fulfill high quality requirements. Of course, special attention is paid to the software.

4. PROOF OF ACCIDENT RESISTANCE

Safety relevant I&C components must be qualified for accident situations, notwithstanding ageing. Complete repetitive test for the preservation of performance under accident conditions on the built- in components are in general not possible for obvious reasons. Therefore the following measures had to be taken in order to achieve the long-term accident resistance of electrical and I&C components within the frame of the plant specifications:

- selection of suitable materials with regard to the expected operational temperature and radiological dose rate at the installation sites;
- design reserves in terms of mechanical stability and humidity protection;
- type and suitability tests on artificially pre-aged specimens.

An essential part of the type tests is the experimental reproduction of the stress under accident conditions on specimens. Test curves have been defined on the basis of LOCA simulations as realistic as possible.

Especially for plastics it has not been possible to achieve theoretical ageing predictions. Pre-ageing for the simulation of 40 years of reactor operation had to be applied. Acceleration factors of 40 for thermal and radiological pre-ageing have even proven to be too large for the prediction of the age-dependent decomposition of synthetics, mainly because the oxidation process cannot be simulated realistically enough in this short period.

For all these reasons the continued resistance to accident conditions of the I&C components must be secured during operation with other means. A basic accident proof design in the sense of the above mentioned measures remains as prerequisite.

5. PRESERVATION OF ACCIDENT RESISTANCE

In order to evaluate the accident resistance of I&C systems, investigations concentrated on components regarded as most susceptible to ageing. In general LOCA resistance is secured by a three-step approach, starting with theoretical investigations, tests (sometimes as part of the prescribed component revisions), and - where necessary - component exchange. Some examples are given below:

1. Inspections were recently performed for actuators of older plants, especially for valve positioners and magnetic positioners in general, including their cabling, plugs, terminals and containment penetrations. Recurrent testing is obligatory for these components every eight years. Initial findings revealed a need for extra qualification of older components. For economic reasons all components in question were exchanged with improved designs, dismissing the need for further investigations.

Under this revision also the housings of accident proof switches were tested on-site for leaks. On detection of undue leaks, cable connections including grouting were changed according to component specification. Long-term operation will now be possible, as long as the availability of the positioners and subparts can be secured under reasonable conditions.

2. For electric motors in the annulus outside of the containment, theoretical investigations have shown that no undue shortening of life of the windings had to be expected under the anticipated temperature rise in the annulus during a LOCA.
3. A third example dealt with the transducers of the reactor safety instrumentation. Investigations have shown that replacement is here the preferable option. The advanced age of the mechanical gauges and electrolytic capacitors was the driving reason for this exchange.
4. For a particular cable type, a R&D programme of the manufacturer has revealed a noticeable, exposure dependent radiation dose effect. From special tests on pre-aged I&C cable specimens, adapted to simulate a representative ageing profile in the vicinity of the primary circuit, a LOCA limit dose rate function was established by the manufacturer. Subsequently all cables which were reaching their permissible operational radiation dose limit for LOCA resistance in locations near primary loops have been exchanged.
5. Tests on selected 61- pole resin cable penetrations carried out using realistic conditions for an anticipated accident course for a BWR gave a satisfactory seal tightness. Further tests with 4-pole resin penetrations are planned.

With the consecutive application of this three- step approach considerable LOCA resistance confidence periods can be guaranteed for all component generations. Proof for the preservation of accident stability must be given in any case in time before individual confidence periods expire.

Correct measurement of operational temperature and radiation exposure at the location of the individual component is important in order to allow an ageing evaluation. This can be performed through on-line dosimeters and maximum temperature recorders over a typical reactor cycle and subsequent extrapolation over past and future time spans.

6. RESEARCH AND DEVELOPMENT PROGRAMMES, NATIONAL STANDARDS

The above mentioned three- step LOCA resistance evaluation procedure, but also other reasons like ageing effects and shortcomings in the type tests have in practice led to the exchange of most of the original accident proof I&C components. This experience, however, does not contradict that certain electrical and I&C components can remain in use for periods which lie in the range of the life expectancy of the whole plant. An exchange of all accident proof cables or large components like motors is regularly not necessary because of the small accident stress.

For this reason, for equipment which is basically qualified as accident proof it seems appropriate to develop on- line monitoring procedures. A VGB technical committee is working in this field since 1988.

In parallel to this utility working team, a national standard KTA 3706 "Recurrent Proof of the Accident Resistance of Electrical and I&C Components of the Safety System" was completed. Besides of the already mentioned replacement option, two further paths are kept open in this KTA standard for proving the preservation of accident resistance, i.e. substitute tests and special tests.

In the course of the substitute tests, components for which LOCA resistance is required may also be tested locally by means of simulation procedures like spray- testing, dive-pressure tests, and others. Especially spray- tests on terminals were performed by the manufacturer on order of the VGB committee. The associated test procedure (VESPA) was assessed by TÜV Rheinland, acting as independent expert. Further substitute tests, according to KTA, are sealing and heating tests.

In the special tests, the relevant steps of type testing are repeated. Typically three or six specimens have to be taken from the plant to form a large enough sample.

In both procedures, care has to be taken on how to assure a large enough load precursor, either by testing in a shorter time span (substitute testing) or by application of additional artificial ageing.

According to KTA 3706, the point in time for the start of the recurring LOCA resistance tests has to be fixed for each nuclear power plant individually. Test instructions must be set up before the test begins and they must have been approved by independent experts. The testing strategies, relating to component groups which should be put down in the test instructions, are currently being specified by a VGB working group.

In another study TÜV Nord investigated the ageing phenomena and lifetime expectancy of electrical equipment of the safety system and of the accident instrumentation in nuclear facilities under operational conditions. Apart from exceptions no ageing relevant changes in the failure behavior of the investigated nuclear power plant components were identified. In conclusion the study pointed out that electronic components usually have an innovation cycle which is shorter than the residence time in the plant. It is planned to continue this project with the co-operation of the utilities.

7. STRATEGIES FOR MAINTENANCE AND RENEWAL OF I&C

Preservation of function of older I&C systems required a reasonable effort for all NPP operators in Germany. Taking decreasing spare part availability for the older I&C technology into account, foresight strategies had to be developed.

From a current view the following options can be identified in principle:

— keeping large spare part stocks and/ or special production runs for old I&C systems (current practice);
— substitution of old I&C by means of functionally equivalent („pin compatible") components of modern micro electronics technology (ASICs) (current practice where appropriate);
— substitution of analogue I&C systems with new digital systems for operational (=not safety relevant) systems (partially practised), and
— total replacement with digital I&C, i.e. including safety systems (digital systems for safety relevant I&C is not yet certified in the USA and other countries).

8. SPARE PART KEEPING AND SPECIAL PRODUCTION OF OLD I&C

This method is currently being practiced especially in the area of safety relevant I&C. Quality assurance of the manufacturers and record- keeping for failure statistics of the old I&C are intensively supported by common actions of the utilities within the framework of the VGB working groups "Qualification of Electrical and I&C Components" and "I&C Failure Statistics".
In spite of the fact that failure behavior is currently still normal, and that the delivery of spares is still assured to a large extent, most utilities regard the renewal of these systems in the old plants necessary in the medium term.

9. DIGITAL SAFETY SYSTEMS

Digital I&C for application in safety systems has been developed or is being developed in several countries. In Germany, Siemens/ Power Generation Group KWU has developed TELEPERM XS with the support of the German utilities, for the long-term replacement of analog safety relevant I&C systems, and TELEPERM XP for operational systems.

TABLE I. REFERENCE LIST FOR SIEMENS TELEPERM XP (FOR OPERATIONAL SYSTEMS) IN NPPs

Plant Name	**Type**	**Country**	**System**	**Commissioning**
Grohnde	PWR	Germany		1996
Neckar-1	PWR	Germany	Water Clearing	1996
Muehleberg	BWR	Switzerland		1996
Borssele	PWR	Netherlands	Comprehensive I&C Retrofit	1998
St. M. de Garona	BWR	Spain	Cooling Water Clearing Condensate Filter System	1998
FRM-II Garching	Research	Germany		2000
Mochovce-1 and -2	WWER	Slovakia	Complete I&C	1998 - 1999
Lianyungang-1, -2	WWER	China	Complete I&C	2004 - 2005

An assessment of the TELEPERM XS system for safety relevant I&C performed by Gesellschaft für Anlagen- und Reaktorsicherheit GRS has confirmed its suitability for use in nuclear facilities under highest safety demands.

For the PWR plants Neckarwestheim-1 and Unterweser, reactor limitation systems, reactor power control systems and control rod I&C systems have been replaced in 1997 and 1998. The technical specifications for these systems have formed the basis for the licensing procedure. TELEPERM XS will also be applied for the reactor trip system and neutron flux measurement for the new high flux research reactor FRM-2 at Garching.

Furthermore, TELEPERM XS has or will be installed for various safety systems in 19 other reactors in eight other countries.

Type-testing of the system components and the software of TELEPERM XS has been completed in 1995. For these particular tests, the basic requirements for traditional systems (as laid down in the KTA rules) were adapted. These rules require theoretical checks, e.g. in terms of functional description, specification, load, and practical checks e.g. in terms of functionality, electromagnetic compatibility and environmental conditions.

TABLE II. REFERENCE LIST FOR SIEMENS TELEPERM XS (FOR SAFETY RELEVANT SYSTEMS)

Plant Name	**Type**	**Country**	**System**	**Commissioning**
Muehleberg	BWR	Switzerland	Level Monitoring System	1995
St. M. de Garona	BWR	Spain	Rod Control System Rod Position Measurement	1996
Borssele	PWR	Netherlands	Comprehensive I&C Retrofit	1998
Unterweser	PWR	Germany	Reactor Limitation System Reactor Power Control Rod Control System	1997
Neckar-1	PWR	Germany	Reactor Limitation System Reactor Power Control Rod Control System	1998
Oskarshamn-1	BWR	Sweden	Neutron Flux Measurement	1998
Forsmark-3	BWR	Sweden	Rod Control system	1998
Mochovce-1 and -2	WWER	Slovakia	Complete I&C	1998 - 1999
Bohunice-1 and -2	WWER	Slovakia	Reactor Trip system Reactor Limitation System Reactor Power Control Neutron Flux Measurement Engineered Safety Feature Actuation System	1998 - 1999
Paks-1 to -4	WWER	Hungary	Reactor Trip system Reactor Limitation System Neutron Flux Measurement Engineered Safety Feature Actuation System	1999 - 2002
Rovno-4	WWER	Ukraine	Reactor Trip system Reactor Limitation System Neutron Flux Measurement	2000
Chmelnitzki-1	WWER	Ukraine	Reactor Trip system Reactor Limitation System Reactor Power Control Neutron Flux Measurement	
Beznau-1 and -2	PWR	Switzerland	Reactor Trip system Reactor Power Control Emergency Feedwater System	1999 - 2001
FRM-II Garching	Re-search	Germany	Reactor Trip System Neutron Flux Measurement	2001
Lianyungang-1, -2	WWER	China	Complete I&C	2004 - 2005
Atucha-1	HPWR	Argentina	Second Heat Sink	Study
Forsmark-3	BWR	Sweden	Comprehensive I&C Retrofit	Study
Philippsburg-1	BWR	Germany	Reactor Trip System Rod Sequencing Calculation Local Core Area Monitoring	Study
Biblis-A and -B	PWR	Germany	Reactor Limitation System Reactor Power Control Rod Control System	Study

Likewise, type- testing has been applied to the new digital system. This was possible because:

- the system (regardless of its specific application) consists of modular hard- and software components;
- the system utilizes standardized software components, configurable to plant specific needs, and
- hard- and software interfaces can be described precisely and completely.

For software components, the following analyses have been performed:

- assessment of the development process and its measures for quality assurance;
- development from terms of reference over software specification, design, implementation to the code;
- verification of software interfaces;
- conformity with standards, mainly IEC 880;
- conformity with project internal guidelines, and
- assessment of failure and error analysis.

Finally, a system type test is performed upon commissioning to check the typical system properties:

- function;
- operational behaviour;
- no feedback of functions with respect to other functions;
- dynamic behaviour;
- reproducibility;
- maintenance and diagnostic properties;
- tolerable stress and failure behaviour;
- failure and error tolerance, and
- independence of the I&C system from the technical process (resistance to input data overload).

10. REPLACEMENT OF I&C COMPONENTS BY FUNCTION EQUIVALENT CIRCUITS (ASICs)

The development of micro- electronics has led to revolutionary changes in the production of electronic systems. Today a large amount of functions can be concentrated monolithically on an integrated circuit, in contrast to old I&C, where discrete parts (transistors, diodes, resistors, capacitors) had to be put onto printed circuit boards. The use of these pin- compatible Application Specific Integrated Circuits (ASICs) can be another solution for the replacement of old I&C.

The application of ASICs offers the following basic perspectives:

in terms of safety:

- the proven fixed structures remain;
- parallel signal processing;
- optimal protection against unintentional changes;

- high degree of integration (therefore fewer contacts);
- design and testing tools proven in mass fabrication;
- reduced software problems because only basic procedures of automation theory are needed during design;

in terms of functionality:

- all required analogue and binary signal functions can be realized;
- direct interface to digital processing possible;
- the test logic can be integrated directly onto the chip;
- high function density and consequently a small volume;

in terms of compatibility/ flexibility:

- pin-compatibility;
- flexibility for the protection function at planning stage can be ensured by design tools.

The VGB Working Group "I&C in Nuclear Power Plants" has initiated pilot tests of ASICs to replace old instrumentation and control systems. It is expected that qualification requires less effort in relation to software solutions.

11. IMPROVEMENT OF PERFORMANCE THROUGH NEW I&C

A system can be obsolete in spite of still fulfilling the originally specified properties. This is especially true for areas where much more powerful systems have become available and an improvement of plant safety or plant availability can be achieved.

The following measures (already taken in several German NPPS) can serve as examples for such improvements:

- plant process computers have been exchanged and extended in most plants, improving the man-machine interface, event diagnosis and expanding the extent and comprehensiveness of information;
- refuelling platform I&C systems have been refurbished, improving the functionality and the man- machine interface, thus contributing to minimize fuel outages;
- turbine I&C has been updated with the aim of improving the availability and the man- machine interface.

12. SUMMARY

The main goals for the management of ageing of electrical and I&C technology in nuclear power plants lie in the long-term preservation of function, taking into account technical development, operating experience and availability of spare parts in the future.

Long- term function is proven by repetitive testing. Safety relevant I&C components have to fulfill accident resistance criteria additionally. In extended research and development programs, solutions have been achieved which have been taken up in the KTA rule 3706.

TABLE III. LIST OF RELEVANT KTA RULES

Safety Standard	Title	Status (Date of Issue)	English Transl.	Reaff.
KTA 1202	Requirements for the Testing manual	R (8/84)	av.	1994
KTA1404	Documentation During the Construction and Operation of Nuclear Power Plants	R (6/89)	av.	1994
KTA 1501	Stationary System for Monitoring Area Dose Rates within Nuclear Power Plants	R (6/91)		1996
KTA 1502.1	Monitoring Radioactivity in the Inner Atmosphere of Nuclear Power Plants; Part 1: Nuclear Power Plants with Light Water Reactors	R (6/86)	av.	1996
KTA 3403	Cable Penetrations through the Reactor Containment Vessel	R (10/80)	av.	1996
KTA 3501	Reactor Protection System and Monitoring Equipment of the Safety System	R (5/85)	av.	1995
KTA 3502	Incident Instrumentation	R (11/84)	av.	1989
KTA 3503	Type Testing of Electrical Modules for the Reactor Protection System	R (11/86)		1997
KTA 3504	Electrical Drives of the Safety System in Nuclear Power Plants	R (9/88)		1993
KTA 3505	Type Testing of measuring Transmitters and Transducers of the Reactor Protection System	R (11/94)	av.	1997
KTA 3506	Tests and Inspections of the Instrumentation and Control Equipment of the Safety System of Nuclear Power Plants	R (11/84)	av.	1997
KTA 3507	Factory Tests, Post-Repair Tests and Demonstration of Successful Service for the Instrumentation and Control Equipment of the Safety System	R (11/86)	av.	1996
KTA 3701	General Requirements for the Electrical Power Supply in Nuclear Power Plants	R (6/97)		
KTA 3703	Emergency Power Generating Facilities with Batteries and Rectifier Units in Nuclear Power Plants	R (6/86)	av.	1992
KTA 3704	Emergency Power Facilities with Rotary Converters and Static Inverters in Nuclear Power Plants	R (6/84)	av.	1994
KTA 3705	Switchgear Facilities, Transformers and Distribution Networks for the Electrical Power Supply of the Safety System in Nuclear Power Plants	R 9/88	av.	1993
KTA 3706	Repeating Proof of Coolant Loss-Breakdown Resistance of Electrical and Instrumentation and Control Components of the Safety Systems	R (8/94)		
KTA 3901	Communication Devices for Nuclear Power Plants	R (3/81)	av.	1996
KTA 3904	Control Room, Emergency Control Room and Local Control Stations in Nuclear Power Plants	R (9/98)	av.	1993

Mainly for reasons of spare parts availability, strategies for the replacement of old I&C systems are required. The tendency is to replace old I&C technology in the area of both operational and safety relevant I&C systems with digital systems.

The new digital safety I&C technology requires, like any computer based system, a high degree of formality to describe the process relevant demands. When defining these demands, experience and latest developments can be used for improvements.

On the other hand formal descriptions and their qualification require a high effort. ASICs offer an alternative here.

Today the new I&C technology contributes essentially to the improvement of plant performance, e.g. with regard to the man- machine interface and the diagnostic capabilities.

Table III is a list of German KTA standards which are relevant to the ageing management of I&C systems:

PRACTICES OF PROLONGATION OF THE I&C EQUIPMENT LIFETIME

O.B. SAMOYLOV, V.S. PRONIN, I.D. SAVELOV,
V.A. BOGOMAZOV, V.V. DRUMOV
OKB Mechanical Engineering,
Nizhny Novgorod, Russian Federation

A.G. CHUDIN
Russian Federation Ministry for Atomic Energy,
Moscow, Russian Federation

Abstract

The lifetime of nuclear power plants (NPP) always exceeds the operational time of I&C systems. Ageing of I&C equipment in NPPs have many aspects. Research of these aspects is being performed in OKB Mechanical Engineering. Under condition of fast development of I&C systems and applying more stringent safety requirements, modernization of the equipment irrespective of its operational condition is getting important. However, an equipment of I&C systems operated in Russia was designed and manufactured applying highest requirements for a reliability of their work during its whole operational time. Therefore, in many cases it is not necessary to replace them in spite of expiration of their specified lifetime. During operation this equipment is maintained in a proper operation condition by a special service procedures stipulated by its development. When the equipment lifetime approaches to its end, lifetime extension for the certain period should be considered.

1. GENERAL CONSIDERATIONS

In Russia the questions of ageing of the NPPs I&C equipment are solved, first of all, together with questions of the substantiation of serviceability of the equipment during its required operation time. The significant part of the work in this direction is connected to careful experimental researches of behavior of the developed equipment in normal and extreme conditions, that is usual practice for development and design work. The system of the standards of various levels, programmes-technique of realization of tests concerning revealing influence of such effects, as temperature, humidity, vibration, etc. are used. The results of tests, in particular, enable to make the conclusion about mechanisms of ageing and about its influence on main operating performances with the course of time. These data, in turn, are used for assignment of service life of the appropriate system.

The special researches of influence of ageing for the NPP's equipment at Center of Accelerated Tests of the Electronic and Electrical Equipment and Cables (Research Institute of Scientific Instruments of the Russian Federation Ministry for Atomic Energy) are carried out. Major goals of the Center are:

- qualification tests and operating life evaluation of electronics, electrical engineering articles and cables for NPP;
- study of processes of thermoradiation ageing in articles and assembly materials;
- control of technical condition and determination of residual operating life of cables.

The Center's equipment provides for simultaneous or separate simulation of radiation, thermal and environmental effects in articles under test and for measurement of articles' and materials' properties. Methods have been developed for qualitative and quantitative analysis of defects detection, determination of the mechanism and kinetics of insulating materials' ageing, testing electrical engineering equipment on its resistance to degradation factors of nuclear plants.

Recently scale of special researches on ageing has decreased, as well as in other countries. The program of researches with reference to cables is now carried out most effectively. During the operation of I&C systems equipment regular researches of its availability index of product according to the operational documentation are carried out. So, for systems, important for safety, the various kinds of operations in this part are carried out with frequency from an once in change up to an once per one year (for shipment of fuel). The researches include check of quality of operation for direct purpose in all modes, and direct check of parameters connected to ageing - resistance of isolation, state of contacts of the relay, physical state of elements of the equipment, etc.

Such work is executed by the specialist of organizations providing maintenance, by developers and equipment's manufacturers, the representatives of other interested organizations, including supervision bodies, participate in researches.

On the basis of received results terms of replacement of the equipment of I&C systems have determined, the problems of development of the new equipment instead of served are accordingly solved. The main problems are connected to replacement of the served equipment. Usually till the time of required replacement of the system the elements, used in it, aren't being already manufactured any more and the development of blocks and devices or whole systems with new elements is required. It requires the certain financial costs, and also search of the new developers and/or manufacturers, if former have replaced a sphere of activity. In this connection prolongation of systems operation time has great importance. An experience of work in this direction is considered below.

Nowadays the term "ageing" is not completely determined for specific application. In a context of questions considered here we understand this term as change of the equipment's state during the time with increase of frequency of deviations from a normal state and refusals with loss any functions of this equipment. The causes of such changes may be physic-chemical processes in materials of elements, influence of a humidity, irradiation, etc. while in operation.

When equipment of I&C system is being developed, an opportunity of refusals is not only taken into account, but in a part of safety is even postulated. There are stipulated some means and methods of definition of refusals in systems, the procedures of definition of the refused elements, their replacement and the fast restoration of functions. There is provided a necessary duplication of elements and channels, whole complete sets of protection, etc. Therefore, as a rule, any individual refusal in a system can not cause the essential damage and in a part of ageing the considered question is not to prevent refusals in general, but to exclude an opportunity of operation of the system in the following conditions:

- with sharp increase of frequency of deviations from a normal condition and refusals;
- with impossibility of restoration of functions of system caused by fast exhaustion of spare parts;
- with impossibility of updating of spare parts owing to the termination of issue of suitable components for replacement and repair of blocks.

It is necessary to emphasize that during the development of the system all measures are undertaken all to postpone occurrence of such condition as far as possibly. In particular, for increase of reliability when the systems are being designed the components are put in such conditions of operation, with which the actual loadings are much lower than not only limits, but also nominal. It is supposed, that the ageing in such situation will occur slower. The most reliable and durable technical means are applied, the special acceptance of components will be carried out, etc. Procedures of service, check of serviceability, preventive maintenance, repair, work with spare parts, etc. had been carefully worked.

There are tests, including resource tests, with imitation of the various influencing factors of operation. On the basis of the data received by development and tests, in view of the data on reliability of elements, etc. there is being defined the nominated operation term of the equipment a I&C systems. The accepted measures to achieve the reliability have allowed to obtain devices with resource that exceeds a resource of components in some times. The practice has shown also, that real term of operation of the equipment of the I&C systems before occurrence of a condition of ageing, as we have determined above, can considerably exceed determined by the design documentation.

Therefore, works on prolongation of the lifetime of the equipment of the I&C systems has become the rather usual events. Thus it is important to fulfill the procedures, determined in the special documents, to achieve expected effect with required reliability, i.e. to prevent attributes of ageing, defined above, during time of prolongation of the permitted term of operation.

2. APPROACHES TO THE DEFINITION OF THE CONDITIONS OF THE I&C EQUIPMENT ALLOWING FOR A LIFETIME EXTENSION

During the exploitation of equipment of I&C systems maintenance instructions and regulation documents determine procedures of check of its serviceability, readiness for performance of the functions with occurrence of the certain conditions, etc. In particular, when reactor stops there will be fulfilled the check of circuits of formation and passage of protection commands. However these procedures are intended, first of all, for the checking of serviceability of system to the main purpose.

On the other hand, in the task of definition of a prolongation opportunity of the operation term it is necessary, besides, to estimate in a possible degree tendency of change of an equipment's condition, i.e. to predict change of a condition of system during the certain time and to reveal possible increase of frequency of deviations from normal work and refusals up to a unacceptable degree. For this the methods and means of diagnostics are used which allow to make required conclusion after realization of appropriate set of researches.

The works in this field nowadays are in a stage of development. The means and methods essentially depend on a kind of the equipment, with which the appropriate work are being carried out. It is enough difficulty to choose and to carry out the analysis of a condition of system just on those parameters, which are really in the maximal degree allow to carry out the required forecast. There are some experiences and practices considered below for a specific kind of the equipment of I&C systems - to the system for indication of a reactor's control roods position.

3. TECHNIQUE FOR MONITORING THE CONDITION OF A REACTOR CONTROL RODS POSITION INDICATION SYSTEM

The experience has shown that an indication system, developed by OKBM, has significant resistance to ageing. Taking into account this circumstance, the significant number of systems has passed procedures of prolongation of equipment's operation term. The received experience has allowed to generate now-in-use program - technique of auditing of system. The auditing of the system is fulfilled by four stages.

At the first stage a degree of system's deterioration for the previous operation phases and volume of required repair were defined. At the second stage a repair work is made, necessity of which is revealed at the first stage. Besides, if necessary, the replacement of separate units, and also updating of spare parts are made. At the third stage the functional checks of system and check of conformity of its basic characteristics to the characteristics specified in the engineering specifications were fulfilled. At the fourth stage on the basis of the results of the previous stages the opportunity of prolongation of system's lifetime are examined. The auditing are made when reactor is shutdown.

Before the beginning of the work the analysis of the data on a previous operation period is fulfilled. On this stage the important factor is the absence of growth of frequency of deviations from normal work of system, refusals in this period of time. The external survey is included into a structure of works on the first stage. It allows, in particular, to define a beginning of physical changes of an equipment's condition, which characterize ageing and increase of probability of refusals in a near future. There is fulfilled the measurement of electrical resistance to isolation. Decreasing of it also rather authentically characterizes process of ageing. There is fulfilled the measurement of output voltage of system's internal power supplies, etc. There is fulfilled the check of completeness a spare parts.

On the base of results of the first stage of auditing there is made the Act of survey of a technical condition of the system. This document contains results of the fulfilled checks, list of necessary repair work, list of elements & blocks, etc., necessary for updating a spare parts and realization of replacements. At the second stage of auditing there are made all works determined in the Act of survey. There is made the necessary repair, replacement of devices, updating of spare parts. An Act of performance of works on the second stage of auditing is made. At the third stage the serviceability of system on direct purpose and conformity of its characteristics to the technical specifications is actually checked. Check of serviceability of power supply lines, alarm elements, devices of formation of malfunction signals, trailer switches, on check of errors, etc. are included. When the third stage of auditing is finished an Act of realization of functional checks containing results of all fulfilled checks is made. In case of a deviation of any characteristic from engineering specifications it is offered to reveal the causes of this deviation and remove them. If they cannot be removed in terms, allotted for auditing, this question should be reflected in the Act for consideration at the fourth stage of auditing.

At the fourth stage the analysis of the received data is fulfilled and in case of the positive results of the fulfilled auditing a technical decision on prolongation of operational time of the system is made. This document should contains the following information:

- an operating time of the system;
- a generalisation of the results of the previous operation phase;
- results of the fulfilled auditing, confirmed an opportunity of prolongation of operation term of the system;
- and also results of the previous auditing for observation of dynamics of change in time of the basic characteristics of system.

Besides, a technique contains all security measures which should be undertaken during auditing. As the experience shows, the applying of the described technique allows to effectively solve the tasks of prolongation of operational time of the system.

4. ADDITIONAL RESEARCH FOR A SUBSTANTIATION OF A PROLONGATION OF OPERATIONAL TIME OF THE I&C SYSTEM

For the reactor control roods position indication system the additional research was fulfilled. That research was fulfilled on the completely efficient specimen allotted for it when its nominated operation time expired. The research was fulfilled for the separate elements of the system, such as transistors, diodes and capacitors removed from the system. The purpose of this research is a revealing of possible deviations of characteristics of these elements from the limits established by technical specifications. It was concluded that a majority of characteristics remain within the limits established by the documentation. The obvious attributes of ageing, on which it is possible to make the conclusion about decrease of reliability of system, are not found. On the basis of the data about deviations from normal work and refusals received during operation, including systems which operation lifetime has been already prolonged, the analytical research was also fulfilled. The received results confirm legitimacy of the used approaches.

5. CONCLUSION

The experience of I&C operation at NPPs shows a possibility of long-term trouble-free operation of the equipment of I&C systems with appropriate performance of the accepted rules of service. It has became the usual phenomenon a complete exhaustion of a design lifetime of I&C systems, and then its prolongation after research of the factors of ageing. The procedures of prolongation of operation lifetime are fulfilled under the established documents, with issue of Acts and Technical decisions. The procedures include tests of system for functioning on direct purpose, check of characteristics, and also research of diagnostic parameters, on which it is possible to predict growth of frequency of deviations from normal operation and refusals during time by which it is supposed to prolong operation lifetime of system.

MANAGEMENT OF I&C AGEING

D. CARON
Nuclear Electric, Ltd,
Gloucester, United Kingdom

Abstract

This report presents a brief perspective on the management of I&C ageing in UK NPPs. It does not address the issues of refurbishment strategies, the use of digital I&C or the technical details of advanced monitoring techniques. These items are discussed in other country reports and elsewhere in this TECDOC.

1. INTRODUCTION

The UK has 35 Nuclear Reactors connected to the National Grid. Twenty of these reactors are Gas Cooled reactors with Magnox fuel, 14 are Advanced Gas Cooled Reactors (AGRs) and there is one PWR - the most recent station Sizewell B. A whole range of I&C equipment is installed on these reactors; from electromechanical devices such as cam-timer circuits, through analogue electronic and relay equipment to the latest generation of computer based SCADA systems.

2. AGEING MANAGEMENT OF I&C SYSTEMS

The UK nuclear industry, and its regulator the Nuclear Installations Inspectorate, have always recognized that ageing presents a threat to plant systems of all types, including I&C. Until recently no specific measures have been taken for I&C systems although the practices of equipment selection, qualification and regular maintenance have always included features to guard against the effects of ageing. These areas are now considered in more detail.

3. EQUIPMENT CLASSIFICATION AND TYPE APPROVAL

All UK NPP I&C equipment is classified according to its safety significance and business impact and this classification is also applied to other activities carried out on the equipment such as maintenance, modification and storage of spare parts. All equipment important to nuclear safety is Type Approved before entering service. For established items specific Quality Assurance requirements are imposed on the manufacturing process which will include procedures for testing and component traceability, etc. New items are type-tested which includes a full range of environmental testing at extremes of temperature, humidity and vibration for extended periods to identify design failures which would render the safety function vulnerable to ageing mechanisms. Safety equipment is also tested to ensure it can withstand the environment or hazard which it is designed to protect against. Equipment Qualification for the UK's most recent plant, Sizewell B, included qualification of aged components against design basis hazards. For example a system which is intended to shut down the plant after an earthquake will be aged and seismically tested to ensure it can survive the earthquake.

In addition to type-testing, new equipment is also analyzed to predict its in-service reliability, and the reliability figures are then used to allow a maintenance regime to be derived which ensures the equipment continues to meet the safety claims based upon it. This analysis will include an allowance for ageing. A conservative system design philosophy is also used to ensure that a single random failure, such as could be caused by the onset of ageing, will be tolerated without detriment to safety.

4. MAINTENANCE PRACTICES

Traditionally I&C equipment has been maintained by a combination of scheduled and breakdown maintenance. Surveillance requirements for all systems relevant to nuclear safety are formally included in station documentation and in some cases are subject to approval by the regulator. On gas cooled reactors there is a maintenance schedule which covers all such items and specifies the type and frequency of maintenance. At Sizewell 'B' (PWR) the technical specifications for each system include the surveillance requirements.

The frequency for all scheduled maintenance is mainly determined from consideration of aspects such as calibration drift and an assessment of the probability (either numerical or qualitative) of an unrevealed failure during normal operation. The latter aspect is determined from a combination of operational history of the system or similar equipment but also includes an allowance for early detection of ageing effects to allow remedial action before the effects become more widespread. At Sizewell 'B' equipment performing a safety function is replaced at the end of the qualified life determined in the qualification process.

In recent years the established approach to maintenance has been examined to ensure it best serves the nuclear safety and business needs of the operators. This has led to the introduction of reliability centred maintenance (RCM) and review of effective maintenance (REM) techniques, whilst ensuring that there is adequate defense against low probability/high consequence events and this evaluation is continuing. A more extensive program of condition monitoring of I&C equipment is being investigated at Sizewell B.

5. CALIBRATION FREQUENCIES

Calibration frequencies are specified according to the duty and technical performance of the equipment concerned. In multiple channel redundant systems routine calibration of individual channels is usually staggered to ensure that under normal conditions only one channel is unavailable due to maintenance and each channel is always at a different stage of its maintenance cycle to minimize peak system unreliability. However staggered maintenance is often not possible on in-core/containment devices as access may be restricted to outages. Historically gas cooled reactor safety circuit equipment was calibrated every three months, but in several instances a re-evaluation of the process and equipment history has allowed the intervals to be extended, although in some cases extension has not been justifiable. For most routine calibration activities the entire channel is tested including the primary sensor and all items downstream. Primary in-core/containment sensors are specifically checked during major overhauls although continual plant surveillance causes anomalous readings to be checked as soon as they are detected and techniques are being developed to perform on-load checks of some primary sensors. On-line condition monitoring techniques such as noise analysis and channel checks are being investigated at Sizewell 'B' and moves are in hand to consider the application of these techniques to the gas cooled reactors. Excessive calibration drift of safety circuit I&C equipment may be classified as a dangerous failure mode which would then be formally investigated.

6. REGULATORY REQUIREMENTS

A triennial safety system review is carried out on all systems with an identified safety duty which includes a substantiation that the equipment continues to meet the safety claims placed upon it.

Site license conditions also require a periodic review of plant safety every 10 years. The aim of the periodic safety review is to assess the whole plant to ensure the safety case remains valid and supportable for the period of operation until the next review. For plant systems, including I&C equipment, this includes an assessment of ageing and degradation effects. The assessment includes a review of operating environment, maintenance processes, operational history and wider consideration of the implications of information available on ageing and degradation from other plants. It is beyond the scope of this report to describe the other important aspects of periodic safety review and how it applies to I&C equipment although it is discussed in other IAEA documentation.

Periodic safety reviews have recently been completed for all of the AGR plant. The assessment of ageing effects has included a comprehensive review of operational experience and a review of information available in the public domain and to date no significant generic effects have been identified other than those already mentioned in this TECDOC.

7. OPERATIONAL FEEDBACK (OEF)

Whilst UK nuclear power generators are legally obliged to inform the regulator of abnormal events which may impact safety they have also established an operational feedback (OEF) system to ensure that the lessons from all significant events are learned wherever applicable.

The objective of OEF "is to effect improvements in the design, operation and maintenance of UK nuclear power plants in order to minimize the risk of events, to enhance safety and improve availability". Five key activities support this objective:

1. Collection, recording and analysis of nuclear safety related events at UK nuclear stations such that information is available in a readily retrievable and usable manner.
2. Review of event information for generic issues and make recommendations for implementation on station to minimize the risk of events.
3. Collection, analysis and recording of information on good practices.
4. To facilitate the effective dissemination of event and good practice information to relevant staff.
5. To interface with other reporting organizations to both transmit and receive OEF information.

Site events are reported in accordance with a formal procedure which provides guidance on reporting, recording and notification of events. It defines the types of events to be reported, and to what extent they should be reported. The types of events that are included are incidents, abnormal occurrences, minor events and near misses.

The central feedback unit (CFU) are a central, co-ordinating team for OEF activities for all UK nuclear stations, and the interface between international organizations and corresponding databases. It receives event reports (where the event classification requires it to be reported off-site) from all UK Nuclear stations and screens them on a weekly basis for clarity and adequacy, and takes the decision on the most appropriate action to implement a solution. Human Factors and Engineering Departments also provide technical assessment of the reports. Only events that are required to be reported off-site, or where the OEF engineer deems the report to be beneficial to other stations are reported to the CFU. The event report is then made available to all NUPER users. The CFU also routinely screen information from international sources, with any significant events being made available on the NUPER database via an International Event Report. They will also screen UK reports for reporting to either World Association of Nuclear Operators (WANO) or the Incident Reporting System (IRS).

When an event has significant implications for safety, a mandatory assessment (MA) can be issued. The MA details are put onto NUPER, and will require a response from those stations and HQ department to which it is applicable. The purpose of a mandatory assessment is to determine the action taken at stations in response to an event or issue, and to consolidate good practices that are of benefit to the stations. In addition, any plant event that has been subject to an external panel of inquiry will result in an MA being issued to other stations. Stations may also respond to an event even if they are not required to by an MA, if they feel that other stations would benefit. This is known as 'deferred assessment'.

8. AGEING RESEARCH

The wide diversity of I&C equipment employed on UK NPPs has meant it has been difficult to justify research into the ageing of specific items unless there is actual operational experience of ageing phenomena. However electrical cables have been acknowledged as an area worthy of generic research due to their key role in almost all aspects of I&C and the potential effect of ageing on system integrity. Conclusions reached by UK sponsored research are generally consistent with those worldwide: the dominant mechanisms are heat and radiation which affect the physical and electrical properties of cable insulation.

There has been research and investigation into specific components where ageing related failures have been encountered but these have not revealed any general ageing trends or failure modes.

The UK nuclear industry also funds a comprehensive research programme for all aspects of nuclear power and the issue of ageing and obsolescence of I&C systems has recently been acknowledged as an area deserving attention.

9. MANAGEMENT OF AGEING

With the exception of SZB, the UK's most modern nuclear station, all UK NPPs have experienced an on-going program of replacement or refurbishment of I&C systems. Some of this work has arisen to combat the threat of ageing equipment although a greater amount has been due to obsolescence or a need for improved performance, including meeting more onerous nuclear safety requirements. The ageing mechanisms include all those mentioned in the main body of this TECDOC and are not dominated by any single cause. On all power stations, including the older magnox stations, much of the original I&C equipment remains. To date no large scale replacement of cables has been found necessary although it has been found necessary to replace some Vulcanised Rubber Insulated cables on a particular site.

The recent acceleration of I&C equipment development and the consequent shortening of product life-cycles has brought the combined problems of ageing and obsolescence into sharp focus. Whilst the management strategies of the past were essentially reactive, it is now recognized that a more forward-looking approach is required. The periodic safety reviews discussed above have considered the problems from a nuclear safety perspective and work has begun using I&C life-cycle management methodologies developed by EPRI to manage the commercial threat to businesses. At present this work has mainly involved the production of system maintenance plans for key systems, which address the issues of reliability, maintainability and obsolescence. Long term management plans for on-line data processing systems are also seen as a key activity. This work is being carried out with a regular review and a rolling horizon of several years (10 years is often chosen).

NEW TECHNOLOGY FOR OPTIMIZED I&C MAINTENANCE AND MANAGEMENT OF AGEING OF CRITICAL EQUIPMENT IN NUCLEAR POWER PLANTS

H.M. HASHEMIAN
Analysis and Measurement Services Corporation,
Knoxville, Tennessee, United States of America

Abstract

Advanced sensors and new testing and maintenance technologies have become available over the last ten years for nuclear power plants (NPPs) to replace outdated, obsolete, and troublesome instruments, provide for management of ageing of critical plant equipment, optimize maintenance activities, reduce maintenance costs and personnel radiation exposure, and at the same time, improve plant safety and availability. These new developments are reviewed in this TECDOC. The material covered here has been summarized from NUREG/CR-5501, a 1998 report written by H.M. Hashemian and his co-authors for the US Nuclear Regulatory Commission.

1. INTRODUCTION

New equipment and techniques have been developed over the last ten years to optimize the maintenance of NPPs, provide for management of ageing of critical plant equipment, and facilitate equipment calibration and other tests that must be performed in compliance with technical specifications and regulatory requirements. This includes automated or computer-aided measurements and on-line tests (Table I). In light of recent deregulation/liberalization of the electric power industries around the world, there is increased pressure on the nuclear power industry to become more competitive. As such, the nuclear industry has concentrated on reducing operations and maintenance (O&M) costs. O&M costs in the USA are presently responsible for about 70 percent of the overall cost of nuclear energy generation, with fuel being responsible for the remaining 30 percent. Therefore, reducing O&M costs can help make the nuclear industry more competitive. This is not as easy in coal and gas power generation where about 80% of the cost of electricity generation is spent for the purchase of the fuel.

In an effort to reduce O&M costs, while maintaining and improving safety, the nuclear industry has implemented risk-based maintenance, and is also depending on new technologies such as computer-aided test and measurements and on-line calibration and on-line maintenance to reduce manpower requirements and human errors, increase accuracy and reliability, and trend the maintenance data to account for ageing effects on the performance of the plant equipment. Some of these new technologies are reviewed in this report beginning with new methods for on-line verification of calibration of pressure instrumentation channels.

With digital test equipment, trending of data can be performed automatically and conveniently to identify the onset of problems and monitor for any performance degradation due to ageing or other effects. New analytical tools such as neural networks, artificial intelligence, and pattern recognition can now be implemented on PC-based test equipment to analyze the data and interpret the results to identify even small changes in the performance of plant equipment and alert the plant personnel of any significant problem or incipient failure.

2. ON-LINE CALIBRATION VERIFICATION

According to present procedures, hundreds of instruments are manually calibrated, typically at least once every fuel cycle, in almost all NPPs. The results of these calibrations over more than 20 years have shown that a majority of the instruments do not fall out-of-tolerance between refueling outages and therefore do not need a calibration. This has motivated the nuclear industry to try to extend the instrument calibration intervals through on-line drift monitoring. This work involves recording and analyzing the steady-state output of instruments during plant operation to identify drift

TABLE I. EXAMPLES OF NEW EQUIPMENT AND MAINTENANCE TECHNOLOGIES FOR IMPROVED PLANT EFFICIENCY AND SAFETY AND MANAGEMENT OF AGEING OF CRITICAL COMPONENTS

1	On-line verification of the calibration of process instrumentation channels and incipient failure detection of I&C channels
2	On-line detection of venturi clogging
3	In situ response time testing of pressure transmitters
4	On-line detection of blockages and voids in pressure sensing lines
5	In situ calibration of primary coolant RTDs and core exit thermocouples
6	In situ response time testing of RTDs and thermocouples
7	In situ testing to verify the installation of thermocouples in WWER reactors
8	New methods for remote testing the attachment/adhesion/embedment of temperature sensors and strain gages to solid materials
9	LCSR test for sensor and circuit diagnostics
10	In situ testing of cables and connectors
11	On-line measurement of Moderator Temperature Coefficient (MTC)
12	Predictive maintenance and management of ageing of reactor internals using existing plant instrumentation
13	On-line measurement of stability margins (decay ratios) in BWRs
14	Automated measurement of drop times of multiple control and shutdown rods and automated analysis to identify rod drop time, rod speed, and detection of sticking and sluggish rods
15	Automated measurement of timing and sequencing of CRDMs
16	Measurement of core flow and detection of core flow anomalies
17	On-line reactor diagnostics and root cause analysis by passive techniques

and other abnormal problems in instrument outputs. For redundant instruments, this is accomplished by comparing the readings of the redundant instruments to distinguish between process drift and instrument drift. For non-redundant instruments, process empirical modeling using neural networks and pattern recognition principles, or other techniques as well as physical modeling are used to estimate the process. This estimate is updated frequently and compared with the output of the corresponding instruments to detect any drift in the instrument output. Process modeling is also used with redundant instruments to provide added confidence in the results and account for common mode drift. The details of the modeling techniques, are presented in [1].

Another important benefit of process modeling is in verifying the calibration of instruments over their entire range. More specifically, the models can provide the reference for verifying the calibration of instruments during plant startup and shutdown periods when the instruments are exposed to inputs that cover a wide range.

The models can use existing plant operating data from diverse sensors to estimate and track any given process parameters. For example, the reactor coolant flow may be estimated independently using data from existing temperature, pressure, and flux sensors and a model that is previously trained at steady state and transient plant conditions to track flow as a function of these other input parameters.

3. ON-LINE DETECTION OF VENTURI FOULING

In addition to on-line verification of calibration of process instrumentation channels, process empirical modeling, pattern recognition, and neural network techniques can provide an effective tool for on-line detection of performance problems in individual instruments or the plant. For example, venturi flow elements can become clogged and result in erroneous flow indication. This has both safety and economical implications. Until recently, there has been no effective way to monitor for venturi fouling. In some plants, new ultrasonic sensors are installed to monitor the flow independently and track the deviation of the venturi sensors and the ultrasonic sensors as a means of detecting venturi fouling. Although the cost of the ultrasonic sensors can be as high as one million dollars, many plants have already installed these sensors because of the importance of accurate flow measurements. Another way to monitor for venturi fouling is to use modeling techniques [1] to track the flow and compare the results with the venturi flow indication to identify venturi fouling.

4. IN SITU RESPONSE TIME TESTING OF PRESSURE TRANSMITTERS

Accuracy and response time are two of the most important indicators of performance of pressure transmitters. As such, on-line methods have been developed to monitor the calibration and response time of pressure transmitters in NPPs. The on-line calibration technology was mentioned above. For on-line measurement of response time of pressure transmitters, the noise analysis technique is used as described in [2]. This method is based on recording the random noise which exists naturally at the output of most process sensors while the plant is operating. The noise can be analyzed in the frequency domain and/or time domain to give the response time of the transmitter. As documented in [2], this method has been validated for response time testing of pressure, level, and flow transmitters in NPPs.

For in situ response time testing of force balance pressure transmitters, in addition to noise analysis technique, a method called the Power Interrupt (PI) test is also available which has been validated for use in nuclear power plants. The details are covered in [2].

Both the noise analysis and PI methods are used in many nuclear power plants around the world for response time testing of pressure transmitters. The tests are performed remotely from the control room area while the plant is operating. These tests do not interfere with plant operation and can be performed on several transmitters at a time.

5. ON-LINE DETECTION OF CLOGGING IN IMPULSE LINES

Impulse lines are the small tubes which bring the pressure signal from the process to the sensor. Typically, the length of the impulse lines are 30 to 300 meters, depending on the service in the plant, and there are often isolation valves, root valves, snubbers, or other components on a typical impulse line. The malfunction in any valve or other component of the impulse line can cause partial or total blockage of the line. In addition, and more importantly, impulse lines can become clogged due to sludge and deposits that often exist in the reactor coolant system. The clogging of sensing lines can cause a delay in sensing a change in the process pressure, level, or flow. In some plants, sensing line clogging due to sludge or valve problems has caused the response time of pressure sensing systems to increase from 0.1 seconds to 5 seconds. This problem can be identified while the plant is on-line using the analysis technique as described in [3]. Basically, if the response time of the pressure, level, or flow transmitter is measured with the noise analysis technique, the results will include any delay due to the sensing line length, any blockages, voids, and other restrictions.

6. RTD AND THERMOCOUPLE CROSS CALIBRATION

Redundant RTDs and thermocouples in NPPs can be in situ calibrated at isothermal conditions using the cross-calibration technique. This involves a multichannel data acquisition system to quickly record the temperature indications of the redundant RTDs and thermocouples. These temperatures are then averaged and the deviation of each RTD or thermocouple from the average of all RTDs (excluding any outliers) is calculated. Once the outlier RTDs are identified, they are excluded from the data and the data is corrected for plant temperature fluctuations and any temperature differences between the loops or between the hot legs and cold legs. After these corrections are implemented, a new average temperature is identified for the RTDs and the deviation of each RTD and thermocouple from this new average is calculated. The details of the cross-calibration technique is presented in [4].

The cross-calibration tests are often performed at several temperatures during plant startup or shutdown periods. With this approach, if any RTD is out-of-tolerance, a new calibration table can be developed for the RTD using the cross-calibration data taken at three or more temperatures. Also if large deviations for thermocouples are identified, they can be adjusted to bring the thermocouples in line with each other and with the RTDs.

7. RESPONSE TIME TESTING OF RTDs AND THERMOCOUPLES

The response time of RTDs and thermocouples can change with ageing of the sensor. Many factors can contribute to this ageing degradation. For example, vibration can cause RTDs and thermocouples to move out of their thermowell and result in an increase in response time. Even a very small movement can cause a large change in response time. Temperatures can also cause changes in response time. For example, inherent voids in sensor insulation materials can expand or contract and cause the response time to change. For these and other reasons, response time of RTDs and thermocouples are measured periodically in NPPs. The measurement is made using the Loop Current Step Response (LCSR) method as described in [5].

The LCSR test is performed remotely from the control room area while the plant is operating. It provides the in-service response time of RTDs and accounts for all installation and process condition effects on response time. If the RTD is used in a thermowell, the response time that is obtained from the LCSR test includes the dynamic response of the RTD and the thermowell combined. Therefore, any gap in the RTD/thermowell interface is also accounted for in the LCSR test.

To perform the LCSR test, a Wheatstone Bridge is used along with a current switching network and signal conditioning equipment. The RTD is connected to one arm of the bridge and the bridge current is switched from about 1 Milliampere (ma) to about 40 ma. The current produces Joule heating (I^2R) and results in a temperature transient in the RTD sensing element. This increases the RTD resistance gradually and results in a voltage transient at the output of the bridge. This transient is recorded and analyzed to provide the response time of the RTD. The analysis is based on a detailed heat transfer model of the RTD. With this method, although the sensor is heated internally, the response time that results from analysis of the LCSR data is equivalent to the response time that would be obtained for the RTD if the process temperature around the RTD experienced a step change. The conversion of data from internal heating to provide the response to an external change in temperature has been proven both experimentally and mathematically.

The LCSR method can also be used to measure the response time of thermocouples as described in [6]. However, this requires higher heating currents and a different test procedure than RTDs. Therefore, in some NPPs, the response time of thermocouples are sometimes tested using the noise analysis technique as was described earlier for pressure transmitters. This is because the high heating currents (about 0.2 to 0.6 amp) that are needed for LCSR testing of thermocouples may be too high in these NPPs. Unlike RTDs, thermocouples are not subject to very stringent response time requirements. Nevertheless, response time testing using the LCSR and noise analysis techniques are performed on thermocouples in NPPs as a means of verifying the health of thermocouples and providing for ageing management.

8. VERIFYING THE INSTALLATION OF THERMOCOUPLES IN WWERs

In some WWER reactors, long thermocouples are used in long thermowells to measure temperature in different regions within the core. Typically, the thermocouples are force-fit into the thermowells, and it is important for the tip of each thermocouple to reach to the end of its thermowell so that it can provide the temperature of the intended point in the core. Therefore, the LCSR and noise methods have been used in WWER reactors to measure thermocouple response time as a means of verifying that thermocouples are at the bottom of their thermowell and are therefore measuring the correct temperature. The LCSR test is used during the installation of the WWER thermocouples to identify and resolve installation problems, and the noise analysis technique is used during plant operation to verify again that the thermocouples are properly installed.

9. LCSR TEST TO DETECT BONDING DEGRADATION

Temperature sensors such as thermocouples, thin-film RTDs, strain gages, and other resistive devices are used in NPPs and other applications for measurement of surface conditions on pipes, vessels, and other components. In these applications, the sensors are bonded to a solid material. For example, strap-on or cemented RTDs are sometimes used to measure surface temperature of pipes such as sensing lines in some NPP applications.

As a result of the long-term exposure to heat, humidity, vibration, and other process conditions, the bonding of sensor can deteriorate and cause the sensor to become detached from the solid surface and float in the air resulting in an erroneous indication. As such, new methods mostly based on LCSR testing have been developed to characterize the quality of bonding between sensors such as RTDs, thermocouples, and strain gages and a solid material. [7] describes how the LCSR is used for these applications. As discussed in [7], the LCSR has been successfully used in aerospace applications for verifying the bonding of thermocouples in solid fuels in rocket engines, and for defecting bonding problems with thin-film RTDs on the fuel lines of aerospace vehicles. For example, the LCSR was used to test the bonding of RTDs in a Space Shuttle application where thin-film RTDs are used for timely measurement of temperature changes as a means of detecting fuel leaks.

10. LCSR TESTING FOR SENSOR AND CIRCUIT DIAGNOSTICS

In addition to sensor response time testing and detection of sensor-to-solid bonding, the LCSR method can be used for diagnostics of wiring and circuit problems in instrumentation systems. For example, in the early 1980s, the LCSR method was being used for thermocouple response time testing on experimental nuclear fuel assemblies at the Argon National Laboratory. In addition to providing dynamic response information, LCSR tests identified a number of reverse-connected thermocouples. The problem was manifested in unusual LCSR transients that were obtained during the response time measurements. In another instance, in a NASA project, rocket nozzle thermocouples were found by the LCSR test to be reverse connected; an event similar to that of ANL. In the same NASA project, the LCSR method identified thermocouples whose measuring junctions were not at the normal point at the tip of the thermocouple, or had developed secondary junctions. These problems would have resulted in erroneous temperature measurements in a very important application [7].

Thermocouple inhomogeneity can also be detected using the LCSR method as described in [8]. Normally, the LCSR test is performed on a thermocouple by applying an AC or DC current to heat the thermocouples measuring junction. The current is applied for a few seconds and then switched off. This allows the thermocouple to return to the ambient temperature. In the LCSR test, the thermocouple output, as it returns to the ambient temperature, is recorded and analyzed to obtain its response time. If the thermocouple contains any significant inhomogeneity along its wire, the inhomogeneity will act as a secondary junction. In this case, the LCSR transient for the thermocouple will be abnormal. With adequate experience, one can correlate the abnormal LCSR transient to the presence of inhomogeneity in the thermocouple wire. The advantage of the LCSR test for this application is that it can be performed remotely on an installed thermocouple in an operating process.

In NPPs, RTDs have been found with sensing elements that open and close randomly causing erratic behavior. More specifically, the RTD would indicate an open circuit for a period of time, and then act normally. In other instances, RTDs with damaged sensing elements have drifted up for a while, then down, and eventually began to act normal again. These problems can be diagnosed and isolated using the LCSR test.

11. TESTING OF CABLES AND CONNECTORS

The condition of nuclear power plant cables, especially I&C cables, is tested in some plants for a number of reasons such as troubleshooting to identify or describe problems, and baseline measurements for predictive maintenance and ageing management. There are electrical tests, mechanical tests, and chemical tests that can be used to monitor or determine the condition of cables. The electrical tests have the advantage of providing the capability to perform the tests in situ, often with no disturbance to the plant operation.

The electrical tests involve impedance measurements and Time Domain Reflectometry (TDR) tests. The TDR test is popular in NPPs for identifying the location of a problem along a cable. Particularly, it is often crucial to determine if a cable problem is in the containment or outside the containment. For example, RTD circuits that have shown erratic behavior have been successfully tested by the TDR method to give the maintenance crew proper directions as to the location of the problem. The TDR technique has also been helpful in troubleshooting motor and transformer windings, pressurizer heater coils, nuclear instrumentation cables, thermocouples, Motor Operated Valve (MOV) cables, etc.

To determine the condition of cable insulation or jacket material, in addition to TDR, electrical parameters such as insulation resistance, DC resistance, AC impedance, and series capacitance are measured. It should be pointed out that determining the condition of cable insulation materials is a very challenging task. The lack of a suitable ground plane for making reliable electrical measurements hampers the success of the tests.

In mechanical testing of cables, the ductility of the cable insulation or jacket material is measured to determine if the material has become dry, brittle, or prone to crack. The test equipment is referred to as a Cable Indentor. Basically, the device is used to squeeze the cable and measure its relative hardness [9].

In chemical testing of cables, a small piece of the cable insulation material is peeled off for chemical analysis in a laboratory.

12. MANAGEMENT OF AGEING OF REACTOR INTERNALS

In the last ten years, predictive maintenance through vibration analysis has become one of the most prevalent practices in industrial processes. Using accelerometers and similar sensors, the vibration of operating machinery is measured passively while the process is on-line. The results of these tests are then trended to identify deviations from expected, normal, or historical behavior. This practice has proven to successfully identify the onset of many problems with industrial equipment, especially rotating machinery, and is believed to save billions of dollars every year by preventing equipment failures and plant downtime. In some cases, the vibration test equipment is installed in the plant permanently and data is collected and analyzed continuously, and in other cases, portable vibration equipment is used to test the plant periodically.

In NPPs, measurement of vibration of reactor internals can be performed very effectively and sensitively using the existing neutron detectors. For example, the ex-core neutron detectors in PWRs can measure the vibration of the reactor vessel and the reactor vessel internals to better than one mil resolution (1 mil = 1/1000 of an inch or 0.025 millimeters). Furthermore, through cross-correlation of neutron signals and other existing sensors such as the core exit thermocouples or the reactor vessel level sensors, the flow through the reactor can be characterized to detect flow anomalies, flow shifting, flow blockages, and other problems.

TABLE II: PREVENTIVE MAINTENANCE AND AGEING MANAGEMENT MEASURES FOR NPPs.

- VIBRATION MEASUREMENTS USING NEUTRON DETECTORS
  - Core barrel vibration measurements in PWRs
  - Thermal shield vibration measurements in PWRs
  - Instrument tube vibration measurements in BWRs
  - Measurement of pump vibration
  - Fuel assembly vibration measurements
  - Measurement of reactor vessel vibration

- LOOSE PARTS MONITORING AND VIBRATION MEASUREMENTS USING ACCELEROMETERS
  - Loose parts monitoring
  - Vibration measurements to detect shaft crack in BWR recirculation pumps

- ON-LINE TESTING OF DYNAMIC RESPONSE OF SENSORS AND ASSOCIATED COMPONENTS
  - Response time testing of pressure, level, and flow transmitters
  - Response time testing of thermocouples and RTDs
  - Blockage and void detection in pressure sensing lines
  - Oil loss detection in Rosemount and other transmitters
  - Management of ageing of process instrumentation systems
  - Detection of leakage in pressure sensing lines

- FLOW MEASUREMENT
  - Cross-correlation flow measurement using existing sensors
  - Core flow measurements in BWRs using neutron detectors
  - N-16 flow measurements by cross-correlation
  - Determination of flow transmission path

- THERMAL HYDRAULIC TESTS
  - Detection of flow anomalies and flow shifting in the reactor core and the plant primary system
  - Detection of standing waves in plant piping and their consequences
  - By-pass boiling detection in BWRs
  - Measurement of stability margin (decay ratio) in BWRs

- ON-LINE MEASUREMENT OF MODERATOR TEMPERATURE COEFFICIENT (MTC) OF REACTIVITY

Accelerometers are also used in NPPs for vibration measurements and loose parts monitoring. However, for the measurement of vibration of reactor vessel and its internals, neutron detectors have better resolution and accuracy than accelerometers.

TABLE II presents a listing of some of the predictive maintenance applications of the noise analysis technique in NPPs. Most of the techniques are discussed in [1].

An essential prerequisite for a successful preventive maintenance program is a comprehensive set of baseline data that must be obtained when the plant and its equipment are either new or in good working condition. NPPs will naturally develop problems as they age and it is therefore crucial to have a library of objective information on the normal behavior of the plant to be used to track problems, identify the root cause, and develop solutions.

13. LOOSE PARTS MONITORING

Loose parts monitoring shall be performed in NPPs on a continuous basis. This work involves accelerometers that are installed in several locations in the plant such as the reactor vessel, steam generators, reactor coolant pumps, etc.

Both audio signals and noise data records are used in performing loose parts monitoring. The audio signals are used to produce alarms if there are any significantly loose parts in the system. The alarm set points are selected depending on the plant and the sensitivity of the loose parts monitoring equipment. If a loose parts alarm is activated, then accelerometer output data are analyzed to confirm the loose part and identify its size and location. The size of a loose part is estimated using baseline measurements that are made with known masses and calibrated hammers. These hammers are used to intentionally hit the plant piping and vessel from the outside to calibrate the loose parts monitoring system. The noise signatures from the baseline measurements in terms of PSDs are compared with measured PSDs after a loose parts alarm is detected. This comparison along with other interpretation steps can help determine the size of the loose part that has caused the alarm.

As for location of a loose part, signals from accelerometers in various locations are cross-correlated to identify signal transmission times. This information is then used in such techniques as triangulation to locate the loose part. These efforts together with listening to audio signals from accelerometers, can often provide good estimates as to the presence, size, and location of any significant loose parts, provided that the loose part is not trapped or lodged in a conspicuous location.

In NUREG/CR-5501 [1], AMS has introduced a method for on-line analysis of loose parts noise data that helps identify any significant loose parts quickly and with high reliability. The method involves calculating and tracking the modified amplitude probability density (MAPD) of the noise signal. This parameter has a given baseline value for when the system contains no loose parts. Any significant deviation from this reference value can be tracked to determine if there is a loose part in the plant.

14. NOISE ANALYSIS APPLICATIONS IN BWR PLANTS

The existing neutron noise signals from average power range monitors (APRMs) and local power range monitors (LPRMs) in BWRs may be used to perform reactor diagnostics and to estimate the flow through the core. The APRM and LPRM signals are also used to measure the stability margin for the core in terms of a decay ratio.

Other applications of noise analysis in BWRs include response time testing of pressure, level, and flow transmitters, testing for sensing line blockages and voids, detection of cracks in the recirculation pump shaft, instrument tube vibration measurements, by-pass boiling detection, two phase flow estimation, oil loss detection in Rosemount pressure transmitters, etc.

15. PC-BASED SYSTEMS FOR AUTOMATED TESTING IN NPPs

Computer-aided testing and PC-based measurement and test equipment are quickly finding their way into NPPs. Today, PC-based equipment are used more than ever before to test, trend, document, and produce automated reports of maintenance work. A number of PC-based test equipment are presently in use in nuclear power plants. For example, instead of using strip chart recorders and timers, computer-based systems are now used in many NPPs to measure the drop times of control and shutdown rods. The new rod drop testing technology employs a PC-based data acquisition and real time data analysis system to measure the drop time of any number of rods which are dropped simultaneously and provide additional diagnostics. In particular, the system provides rod drop time results, rod recoil data, rod speed, and other rod movement parameters. This type of system is very useful to PWR and WWER plants. Also, an automated rod drop test system is particularly useful to Russian-made RBMK reactors in which more than 200 rods are tested frequently to verify proper rod movement at appropriate speeds [10]. Accurate and reliable rod drop time measurements and rod movement diagnostics are important especially in Chernobyl type RBMK reactors. In face, the interest in rod drop time measurements in RBMK reactors have increased in the aftermath of the Chernobyl accident in the Ukraine.

With automated test equipment, the drop times of multiple control and shutdown rods are now measured accurately and precisely in typically less than one hour for 50 rods compared to more than twelve hours that it often took using conventional equipment and technologies. Furthermore, the results of the tests can be analyzed and trended to identify rod movement problems and detect sluggish or sticking rods.

A PC-based test system is also in use in NPPs to verify the timing and sequencing of Control Rod Drive Mechanisms (CRDMs). Recent events involving rods that have moved contrary to what was expected have given rise to more routine testing of CRDMs as indicated in NRC Generic Letter 93-04 entitled “A Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies.” With the new PC-based CRDM test equipment, the analysis of the data and interpretation of results can be done in a few minutes compared to hours that it took to perform the tests manually using strip chart traces.

16. CONCLUSIONS

New instrumentation and maintenance technologies have emerged over the last ten years with great potential to benefit the safety and economy of NPPs, facilitate plant life extension, and aid in the management of ageing of critical plant equipment. Some of these new developments have been summarized in this country report. The report also reviewed testing and predictive maintenance technologies. In particular, in situ methods for measurement of performance (calibration and response time) of process sensors were discussed. More specifically, the loop current step response technique for in situ response time testing of RTDs and thermocouples as installed in operating processes was described, including the ways that this method can be used for diagnostics of problems in RTDs and thermocouple circuits. Also, on-line methods that can be used to extend instrument calibration intervals in NPPs were reviewed.

The fundamentals of the noise analysis technique and its application for a variety of predictive maintenance activities in NPPs were reviewed. This includes the use of the noise analysis technique for on-line detection of voids and clogging of pressure sensing lines, on-line measurement of response time of pressure sensors, measurement of vibration of reactor vessel, core barrel, thermal shield, fuel assemblies, and other components of the reactor system, loose parts monitoring, diagnostics of flow anomalies, and determination of root cause of unusual problems in the reactor system.

REFERENCES

[1] Hashemian, H.M., et al., "Advanced Instrumentation and Maintenance Technologies for Nuclear Power Plants." US Nuclear Regulatory Commission, Report Number NUREG/CR-5501 (September 1998).

[2] Hashemian, H.M., "Long Term Performance and Ageing Characteristics of Nuclear Plant Pressure Transmitters." US Nuclear Regulatory Commission, NUREG/CR-5851 (March 1993).

[3] Hashemian, H.M., et al., Effects of Ageing on Response Time of Nuclear Plant Pressure Sensors." US Nuclear Regulatory Commission, NUREG/CR-5383 (June 1989).

[4] Hashemian, H.M., et al., "Ageing of Nuclear Plant Resistance Temperature Detectors." US Nuclear Regulatory Commission, Report Number NUREG/CR-5560 (June 1990).

[5] Hashemian, H.M., and K.M. Petersen, "Loop Current Step Response Method for In-Place Measurement of Response Time of Installed RTDs and Thermocouples", (American Institute of Physics) Seventh International Symposium on Temperature, 6:1151-1156, Toronto, Canada, May 1992.

[6] Hashemian, H.M., "New Technology for Remote Testing of Response Time of Installed Thermocouples", United States Air Force, Arnold Engineering Development Center, Report Number AEDC-TR-91-26, Volume 1 - Background and General Details, January 1992.

[7] Hashemian, H.M., C.S. Shell, and C.N. Jones, "New Instrumentation Technologies for Testing the Bonding of Sensors to Solid Materials", National Aeronautics and Space Administration, Marshall Space Flight Center, NASA/CR-4744, May 1996.

[8] Hashemian, H.M., and K.M. Petersen, "Measurement of Performance of Installed Thermocouples", Proceedings of the Aerospace Industries and Test Measurement Divisions of the Instrument Society of America, 37th International Instrumentation Symposium, pp. 913-926, ISA Paper #91-113, San Diego, California, May 1991.

[9] DOE/EPRI Report, "Ageing Management Guideline for Commercial Nuclear Power Plants - Electrical Cables and Terminations", SAND96-0344, September 1996.

[10] Hashemian, H.M., Fain, R.E., "Automated Rod Drop Time Testing and Diagnostics in Soviet Designed RBMK and WWER Reactors", Proceedings of the 1996 Nuclear Power Plant Instrumentation and Control and Human Machine Interface Technologies, Penn State University, Vol. 2, pp. 1461-1468, May 1996.

CONTRIBUTORS TO DRAFTING AND REVIEW

Carron, D.A.	Nuclear Electric Ltd, United Kingdom
Chudin, A.G.	Ministry of the Russian Federation on Atomic Energy, Russian Federation
Droumov, V.V.	OKB Mechanical Engineering, Russian Federation
Govindarajan, G.	Bhabha Atomic Research Centre, India
Guillery, P.	EDF Production Transport EPN/DM, France
Hashemian, H.M.	Analysis and Measurement Services Corp., United States of America
Kossilov, A.	International Atomic Energy Agency
Krs, P.	State Office for Nuclear Safety, Czech Republic
Lukac, M.	VUJE Trnava a.s., Slovakia
Mohrbach, L.	VGB, Technical Association of Large Power Plant Operators, Germany
Neboyan, V.	International Atomic Energy Agency
Simola, K.	VTT Automation, Finland

Advisory Group Meeting

Vienna, Austria: 20–24 April 1998

Consultants Meetings

Vienna, Austria: 8–12 September 1997, 28 September–2 October 1998