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Pilot studies on management of ageing of nuclear power plant components

Results of Phase I

INTERNATIONAL ATOMIC ENERGY AGENCY

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PILOT STUDIES ON MANAGEMENT OF AGEING
OF NUCLEAR POWER PLANT COMPONENTS:
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FOREWORD

Ageing in nuclear power plants (NPPs) must be effectively managed to ensure that the required plant safety and reliability are maintained throughout plant service life. To this effect, many IAEA Member States have initiated specific projects aimed at a better understanding and managing of the ageing of various plant components, systems and structures. To facilitate co-operation between these Member States and thus to enhance the safety and reliability of operating nuclear plants the IAEA has initiated pilot studies on the management of ageing of four representative plant components: the primary nozzle of the reactor pressure vessel, a motor operated valve, the concrete containment building, and instrumentation and control cables.

Phase I of the pilot studies has been completed and its results are presented in this report. The report documents current understanding of ageing and methods for monitoring and mitigation of this ageing for the above NPP components, identifies existing knowledge and technology gaps and defines follow-up work to deal with these gaps. Together with a methodology for ageing management studies, presented in IAEA Technical Reports Series No. 338, Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety, this report provides the technical basis for the implementation of the Phase II studies under IAEA co-ordinated research programmes (CRPs) scheduled to start in 1992.

The present report will be of interest not only to the CRP participants, but also to other NPP technical staff, designers, researchers and regulators interested in the management of NPP ageing. It was drafted by technical specialists from 16 Member States and the CEC at Technical Committee Meetings in November 1990 and October 1991 under the direction of the Chairman A.R. DuCharme of the USA and the Scientific Secretary J. Pachner of the IAEA. The work of all participants is greatly appreciated. In particular, contributions by working group chairmen and an editorial team identified at the end of this publication are acknowledged.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Ageing[1] in NPPs must be effectively managed to ensure that required safety margins are maintained throughout plant service life, including any extended life. Many organizations in Member States are developing the necessary technical basis for managing nuclear plant ageing.

To facilitate the exchange of information and collaboration among these organizations the IAEA initiated in 1989 work on pilot studies on management of ageing of NPP components. In November 1989, a methodology for performing ageing management studies [1] was developed, four safety significant NPP components were chosen for the studies, and common technical issues to be addressed in the pilot studies were identified. The four NPP components representing different safety functions and materials that were selected on the basis of their safety significance and their susceptibility to different types of ageing degradation are the primary nozzle of a reactor pressure vessel; a motor operated isolating valve; the concrete containment building; and instrumentation and control cables within the containment.

1.2. COMMON TECHNICAL ISSUES

Common technical issues relating to the management of ageing of nuclear plant components and the pilot studies are as follows.

(i) What is the current understanding of relevant ageing phenomena and how are the research results and operating experience being fed back and used in operational plants?

(ii) What are the potential safety impacts of the ageing mechanisms identified if they are not adequately mitigated by maintenance, operating practices or replacement?

(iii) What and how effective are the existing techniques used to monitor and to mitigate component ageing degradation?

(iv) How effective are current procedures for predicting future component performance based on the evaluation and trending of data on component operation, maintenance, testing and inspection?

(v) At present, in-service monitoring often provides assurance only that the component will function under normal service conditions. What is being done to demonstrate (with confidence) that these components will function as required under abnormal and accident conditions?

(vi) What methods and criteria have been developed to enable the remaining service lives (including required safety margins) of these components to be predicted?

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[1] In this report ageing is used to mean the process by which the physical characteristics of a component, system or structure change with time or use; this process may proceed by a single ageing mechanism or by a combination of several ageing mechanisms.

1.3. OBJECTIVES OF THE PILOT STUDIES

The objectives of the pilot studies are, for each component, to identify dominant ageing mechanisms, and to identify or develop an effective strategy for managing ageing effects caused by the identified mechanisms.

Results of the pilot studies will have application in monitoring the degradation and in preventive maintenance of the selected components, including the development of criteria for decisions on the type and timing of preventive maintenance actions; in predictions of component performance and remaining service life under all expected service conditions, including postulated accident and post-accident conditions; in future designs and material selection; and in amendments of applicable codes, standards and regulatory requirements.

1.4. IMPLEMENTATION OF THE PILOT STUDIES

The pilot studies are being implemented in two phases using the methodology for ageing management studies documented in Ref. [1]. Phase I pilot studies (i.e. interim ageing studies) were completed through Technical Committee Meetings of November 1990 and October 1991, and Phase II studies (i.e. comprehensive ageing studies) will be implemented through IAEA co-ordinated research programmes (CRPs) scheduled to start in 1992.

Work performed in Phase I consisted of a review of current understanding of ageing and methods for monitoring and mitigation of this ageing for the four selected NPP components (see Sec. 1.1), an identification of relevant knowledge and technology gaps, and a formulation of recommendations for follow-up work in the form of a work statement for Phase II studies. Results of this work are presented in this report. Together with results to be obtained from Phase II studies they will provide the technical basis for managing ageing of the selected NPP components.

1.5. STRUCTURE OF THE REPORT

Sections 2–5 present status reports on current understanding, monitoring and mitigation of ageing of the selected NPP components, relevant gaps in knowledge and technology, and statements of work for the CRPs. Although these reports follow the same format, they may vary in style; this reflects the different backgrounds of the participants in the different working groups that drafted the reports. Status reports are presented in Sections 2.1–2.5, 3.1–3.6, 4.1–4.6 and 5.1–5.6, gaps in knowledge and technology in Sections 2.6, 3.7, 4.7 and 5.7, and CRP work statements in Sections 2.7, 3.8, 4.8 and 5.8.

2. PILOT STUDY ON THE MANAGEMENT OF AGEING OF REACTOR PRESSURE VESSEL PRIMARY NOZZLE

This pilot study deals with the management of ageing of reactor pressure vessel (RPV) primary nozzles. The following report first presents a summary of current understanding, monitoring and mitigation of RPV primary nozzle ageing; second, it identifies related knowledge and technology gaps; and third, it presents a statement of work for a co-ordinated research programme designed to address the identified gaps.

2.1. DESCRIPTION OF RPV PRIMARY NOZZLE

The description given below reflects current practice in design and manufacture of RPVs. Plants of older design generations, which are of particular interest from the point of view of life evaluation, may be different to that described in this report.

2.1.1. Scope

The pilot study considers:

- the RPV inlet and outlet nozzle region of a PWR
- the RPV feedwater nozzle region of a BWR.

The nozzle region considered includes areas of the vessel shell where stresses are significantly influenced by the nozzle penetration, and also includes sections of the nozzle up to and including the weld between the nozzle and the safe-end or the attached pipe, as appropriate.

2.1.2. Specifications, standards and regulations

2.1.2.1. Design and manufacture

The design and manufacture of the RPV is subject to the provisions of applicable national codes and standards for Class 1 components (pressure boundary components of NPP), for example American ASME Code Section III, German KTA 3201, Japanese MITI 301, or French RCC-M. In addition to fulfilling these provisions, the system supplier imposes specific requirements to ensure compatibility with other parts of the system.

Some national codes and standards had not been developed when the first generation plants were built. The design and manufacture of these plants were based on specifications established only by the system suppliers.

The design specification is of particular interest with respect to evaluating retrospectively the operational history of the nozzle, in as far as that it specifies the operating temperature and pressure, test conditions and a set of transients on which the design was based.

2.1.2.2. In-service (ageing)

ASME Code Sections III and XI and other national codes provide design rules to protect the nozzle against corrosion and fatigue crack initiation. The effects of thermal

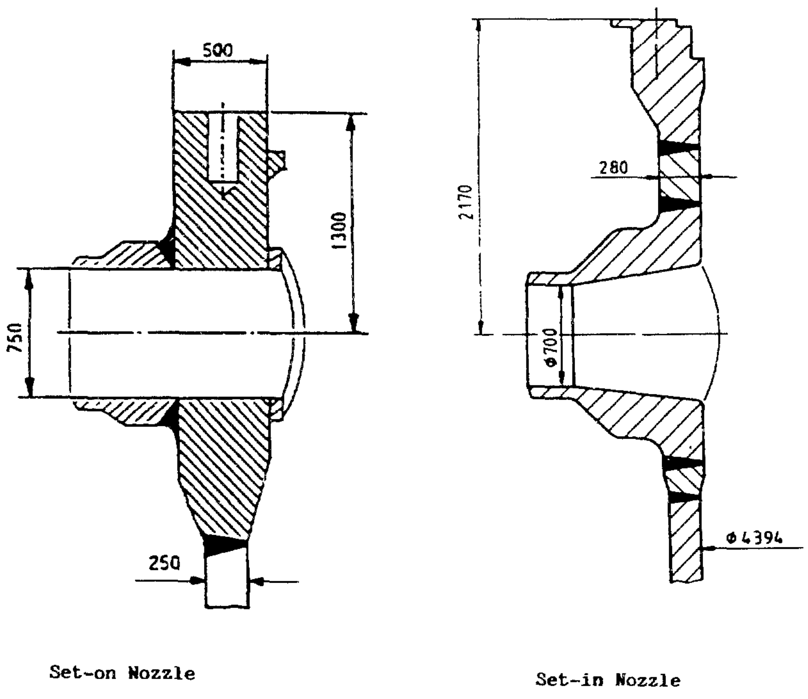


FIG. 1. Set-on and set-in nozzle designs.

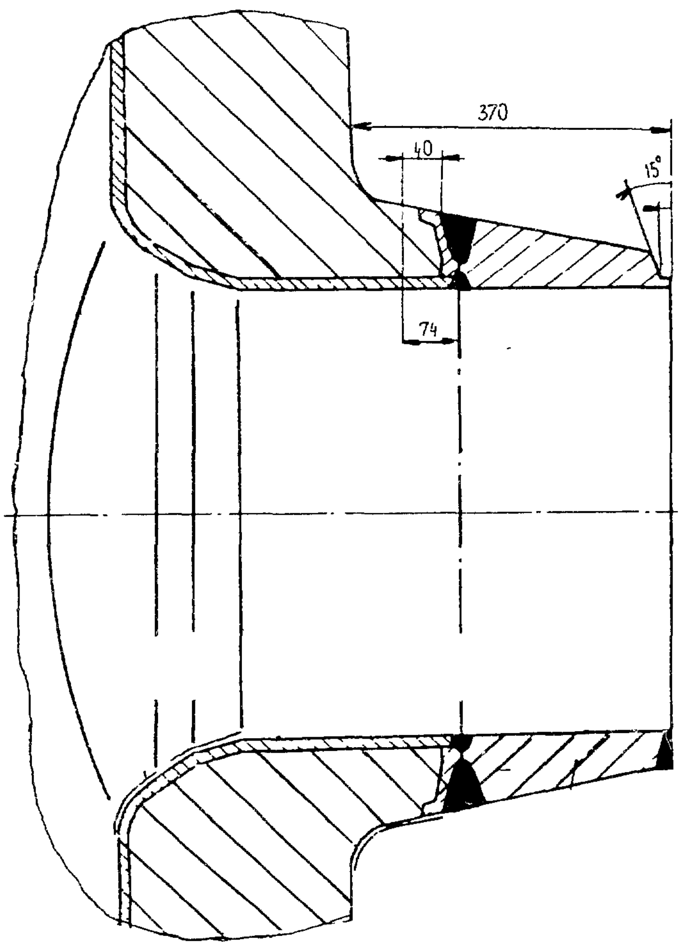


FIG. 2. Machined or extruded nozzle design of WWER RPV.

ageing and cyclic loading on the material, in as far as they affect toughness properties, are taken into account in Soviet codes. ASME XI and KTA contain requirements for in-service inspection (ISI) of nozzle welds. KTA 3201.4 requires the nozzle to be inspected not only in the weld areas but also in areas of parent material expected to be highly stressed (e.g. nozzle corner). ASME XI and other national codes (KTA 3201.2) contain rules for evaluating defects detected during ISI considering fatigue crack growth in a high temperature water environment.

Comprehensive standards or regulations for detecting, monitoring and evaluating ageing of pressure boundary component material, including RPV nozzles, apparently do not exist in any country.

2.1.3. Design

The main features of the design of the RPV primary nozzle are as follows:

- The portion of the shell of the vessel around the nozzle can be fabricated either from forgings or from longitudinally welded plates.

- The nozzle can be designed as set-on or set-into the shell of the vessel (Fig. 1).

- Set-on nozzles are either forged and welded to the vessel shell, or are machined or extruded out of the shell as is the case for nozzles of WWER-440 and WWER-1000 RPVs, designed by the former USSR (Fig. 2).

- The inside surfaces of both the flange and the nozzle are usually clad with multiple layers of austenitic stainless steel.

- If an austenitic type of material is selected for the main coolant pipe or the feedwater pipe, the nozzle is provided with an austenitic safe end transition section (Fig. 3). Although usually made of stainless steel, this section is made of Inconel in the case of WWER plants in Finland.

- BWR feedwater nozzles are provided with a thermal sleeve which prevents the low temperature feedwater coming into contact with the nozzle or vessel wall before mixing with warmer water in the downcomer. The sleeve can be a loose-fit, or interference-fit, or of welded design.

The detailed geometry of the nozzle depends on the type of reactor, on the thermal output, and on the type of materials used, etc.

The accessibility of the nozzle from the inside and outside of the vessel is important from the point of being able to undertake in-service inspection and monitoring. In case of PWRs, access to the nozzles from the inside of the vessel is in general possible using remote controlled devices. Due to the arrangement of the internal components of BWRs, access to the nozzles from the inside is very limited. Access from the outside depends on the specific configuration of the concrete biological shielding, the RPV support structure and other components.

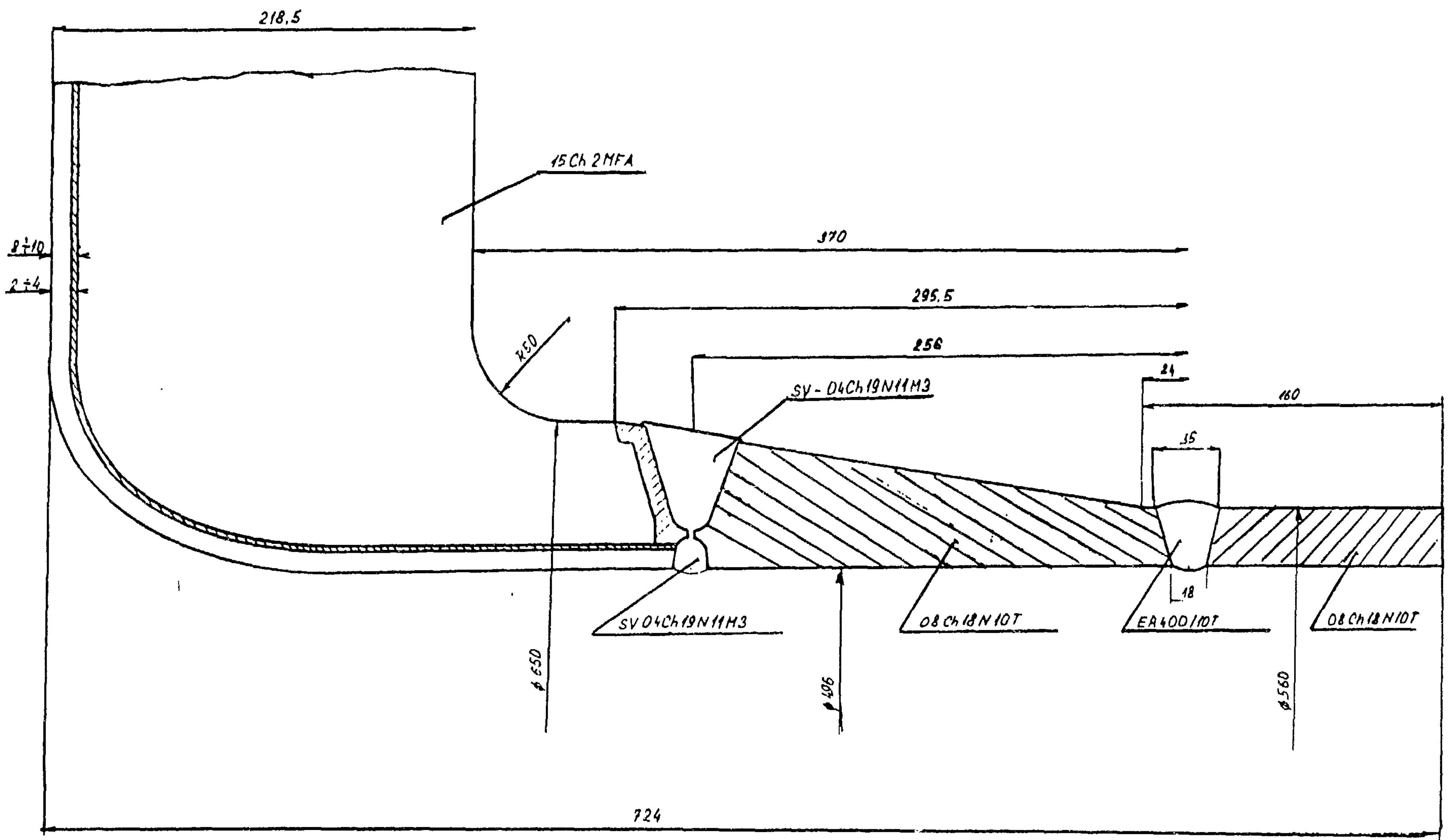


FIG. 3. Nozzle with safe-end (WWER-440 RPV).

TABLE I. TYPES OF MATERIALS USED FOR RPV-NOZZLES AND FLANGES

Materials	Specified analysis, %											Miscellaneous	Standards, Regulations
	C max.	Si	Mn	P max.	S max.	Ni	Cr	Co max.	Mo	Ti	Nb		
SA 508 Cl.2	0,27	0,15-0,4	0,5-1,0	0,025	0,025	0,5-1,0	0,25-0,45	-	0,55-0,7	-	-	V≤0,05	ASME
22NiMoCr37	0,16-0,21	0,15-0,35	0,5-1,0	0,02	0,015	0,5-0,9	0,2-0,5	0,03	0,45-0,7	-	-	Cu≤0,20 Al0,01-0,04	VdTÜV-WB366 (5.72)
SA 533 Type B	0,25	0,15-0,4	1,15-1,5	0,035	0,04	0,4-0,7	-	-	0,45-0,6	-	-	-	ASME
16 MND 5	0,2	0,10-0,30	1,15-1,55	0,015	0,012	0,5-0,8	≤0,25	-	0,45-0,5			V ≤ 0,03 Cu ≤ 0,20 Al≤ 0,04	RCC-M
20MnMoNi55	0,17-0,23	0,15-0,3	1,2-1,5	0,012	0,008	0,5-0,8	≤0,2	0,03	0,4-0,55	-	-	Cu≤0,12 V≤0,02	VdTÜV-WB 401
15Cr2MFA	0,13-0,18	0,17-0,37	0,3-0,6	0,02	0,02	0,4	2,5-3,0	-	0,6-0,8	-	-	V0,25-0,35 Cu0,20±0,05	former Soviet regulations
15Cr2NMFA	0,13–0,18	0,17–0,37	0,3–0,6	0,02	0.02	1,00–1,5	1,8–2,0	–	0,5–0,7	–	–	V≤0.10	former Soviet regulations

2.1.4. Materials

2.1.4.1. Material selection

RPV primary nozzles of Western design are manufactured from ferric fine grained quenched and tempered low alloy steels from the MnMoNi-type (e.g. SA 508 Cl. 3, SA 533 Gr. B, 20MnMoNi 55, 16 M N D5) or NiMoCr-type (SA 508 Cl 2, 22 NiMoCr 37). For reactors designed in the former Soviet Union, a 2.5Cr1MoV type or a NiCrMoV type of steel is used. Typical chemical compositions are given in Table I. Both austenitic stainless steel and ferritic steel are used for main coolant piping of PWRs or feedwater piping of BWRs depending on the system supplier. For cladding material, a 18Cr10Ni type of material is used either stabilized by sufficient Ti or Nb content or non-stabilized in the case of many vessels. The weld metal for the nozzle to safe-end or pipe weld is similar to the parent material of the safe-end or pipe (ferritic or austenitic).

2.1.4.2. Material properties

The absorbed energy in Charpy V-notch tests is widely used as a measure for evaluating the condition of aged RPV materials. The latest issues of the ASME, KTA or RCCM codes evaluate toughness behaviour with reference to RT_{NDT} which is derived on the basis of, among other requirements, a minimum charpy energy value of 68 J (transverse specimen) at the lowest service temperature and the nil ductility transition (NDT) temperature. There is a requirement to derive RT_{NDT} for all ferritic materials of the nozzle (base material, weld metal and heat effected zone (HAZ)).

According to current Soviet specifications, toughness requirements for ferritic materials are based on the T_{ko} concept. A minimum mean charpy energy of about 70 J (transverse specimens) is required at the lowest service temperature not requiring drop weight testing. In former Soviet specifications, lower values were permissible, and the amount of testing required did not always allow a proper evaluation of the toughness behaviour of nozzle materials. There are very limited data available concerning HAZ and weld metal of nozzle to safe-end welds of WWER-440 type reactors.

Data on fatigue behaviour and crack growth behaviour in both air and high temperature water conditions as well as data on corrosion resistance are available for most nozzle materials in the open literature and in codes and standards, e.g. ASME Sections III and XI, or KTA 3201.2.

2.1.5. Fabrication

2.1.5.1. Butt welded joints

Nozzle-to-vessel weldments are made by manual arc welding or submerged arc welding techniques. For safe-end weldments, the nozzle is buttered with Inconel or austenitic stainless steel weld metal. Stress relief heat treatment at temperatures relevant to the base material is performed at this stage. The austenitic safe-end is then welded by a circumferential butt weld to the buttered nozzle without the necessity of a further stress relief heat treatment. The safe-end to pipe weld is usually performed on-site.

Because of the former Soviet requirement for rail shipment of the RPV, only very short nozzles are fabricated on WWER vessels at the manufacturing facility. These nozzles are produced by pressing and machining out the vessel shell and by deposit welding using multiple weld heads. Cladded transition sections are welded between the nozzle and the main coolant pipe at the reactor site. Where a safe-end is required, as to the stainless steel piping in a WWER-440, this is also welded to the nozzle at the site. Special welding techniques (so-called "Cardo-techniques") are applied for the nozzle to pipe weld when stainless steel clad ferritic pipes are used if no counter weld cladding from inside is performed.

2.1.5.2. Cladding

Cladding on the cylindrical surfaces of the nozzle is laid by a submerged arc strip process. Usually two layers of cladding are used to achieve a sufficiently corrosion resistant chemical composition on the surface and to avoid underclad cracking of susceptible base material. Main coolant pipes in PWRs have one layer of cladding which is laid using a special technique. At the corner of the nozzle bore, cladding is laid using manual arc welding. Up to eight layers are necessary to achieve the specified thickness and properties.

2.1.5.3. Heat treatment

After welding of the nozzle to the vessel, stress relief heat treatment is undertaken at temperatures from 580°C to 620°C for MnMoNi and NiMoCr types of material, and at temperatures between 620°C to 660°C for Russian 2.5Cr1MoV type of steel.

2.1.5.4. Machining and surface finish

Machining and surface finishing of the radii of weld toes and the nozzle corner are important for the following reasons:

- incorrect application of fabrication methods (e.g. grinding) can produce residual stresses in the surface layer which may contribute to the risk of intergranual stress corrosion cracking (IGSCC) in unstabilized austenitic stainless steel cladding;

- the effectiveness and results of non-destructive examination (NDE) methods depend on the surface finish achieved.

If in-service NDE is by ultrasonic scanning from the cladding surface, the required surface finish is achieved by grinding. The surface finish specified depends on the sensitivity to roughness of the intended NDE method. Otherwise strip cladding can remain in the as-welded condition.

2.1.5.5. Pre-service pressure test

A hydrostatic pressure test is performed at the end of the manufacturing process. The test pressure is normally a factor of 1.3 to 1.5 higher than system design pressure and the test temperature is maintained higher than the corresponding brittle fracture transition temperature of the material.

2.1.5.6. Non-destructive examination

All surfaces of the nozzle in the final machined and heat-treated condition are examined using standard methods (e.g. visual, dye penetrant, magnetic particle). Welds are examined by radiographic and/or ultrasonic techniques. Special ultrasonic techniques are used to examine the interface between cladding and base material for lack of fusion and for underclad cracking (the latter not in all countries).

2.2. SERVICE CONDITIONS

2.2.1. Normal operating conditions

During normal operation, the bulk coolant temperature and pressure do not change significantly. The normal operating pressure is 12.5 - 15.5 MPa for PWRs and 7-8 MPa for BWRs. The maximum coolant temperature is about 330°C. Operating procedures set limits on the rate of temperature change permitted during normal operation (e.g. load following, normal operating transients, start-up and shut-down, etc.) to typically 10-20°C/h.

2.2.2. In-service leak and pressure tests

The system leakage test is performed at a test pressure that is not less than the specified system operating pressure corresponding to 100% reactor power. For the system pressure test, conditions are within the range specified for the pre-service pressure test.

2.2.3. Accident conditions

Accident conditions can be of different severity. Among the most important ones are those where the coolant temperature changes at a rate that is well above that normally permitted by operating procedures.

The most severe conditions for a PWR nozzle arise in the event of a pressurized thermal shock transient. The expected frequency of such events determined on the basis of current operating experience is of the order of 0.18 events per reactor-year. Cold water from the emergency cooling system enters the RPV through the inlet or outlet nozzles (or through a special nozzle) and causes thermal shock to the nozzles when the system is still at pressure. Most incidents with loss of coolant from the secondary system result in overcooling the reactor coolant system and cause a rate of temperature change much higher than for normal operation (e.g. for WWER-440 about 45°C/h).

2.2.4. Loads

RPV primary nozzles are subject to several different types of loads and the resulting stresses can be quite high. Stresses arise from the operating conditions of pressure and temperature, as residual stresses from fabrication, from external piping loads (due to suppressed thermal expansion), and from supporting the weight of the RPV. Additional thermal stresses can arise during normal plant transients and during accident conditions, and in some plants can also be caused by thermal stratification and leakage flows. For some plants, there is also a requirement to consider stresses arising from earthquake loading.

TABLE II. SPECIFICATION OF WATER CHEMISTRY

Parameter	Siemens-KWU (FRG)	EPRI (US)	Westinghouse (US)	VGB (FRG)	J-PWR (Japan)	EdF (France)	WWER 440/1000 (SU)	WWER 440 (Finland)
Lithium hydroxide	0.2 - 2*	0.2 - 2.2*	0.7 - 2.2*	0.2 - 2.2*	0.2 - 2.2*	0.6 - 2.2* 0.45 - 2.2**		
Potassium hydroxide	-	-	-	-			2 - 16.5#	2 - 22#
Ammonia	-	-	-	-			> 5	> 5
Hydrogen	2 - 4	2.2 - 4.5	2.2 - 4.4	1 - 4	2.2 - 3.15	2.2 - 4.4	2.7 - 4.5	2.2 - 4.5
Oxygen	< 0.005	< 0.01	< 0.005	< 0.005	< 0.005	< 0.1	< 0.01	< 0.01
Chloride	< 0.2	< 0.15	< 0.15	< 0.2	< 0.05	< 0.15	< 0.1	< 0.1
Fluoride	-	< 0.15	< 0.15	-	< 0.1	< 0.15	< 0.05	< 0.1
Conductivity (25°C)	< 30	*	*	-		1 - 40*	4 - 80*	-
pH (25°C)	5 - ≈ 8.5	*	4.2 - 10.5*	*	4.2 - 10.5	5.4 - 10.5	> 6	> 6
Dissolved iron	(< 0.05)	-	-	-	-	-	-	-
Total iron		-	-	(< 0.01)	-	-	< 0.2	
Sulphate	-	0.1	-	-	-	-	-	-
Silica	(< 0.5)	-	< 0.2	-	-	< 0.2	-	-
Suspended solids	(< 0.1)	0.35	< 1	-	< 0.5	< 1	-	
Aluminium	-	-	< 0.05	-	-	< 0.1	-	-
Calcium	-	-	< 0.05	-	-	< 0.1	-	-
Magnesium	-	-	< 0.05	-	-	< 0.1	-	-

() = normal operating value
\- = not applicable/specified
* = According to Li and B concentration
** = According to Li and B concentration, new treatment
\# = Calculated taking into account Σ K + Na + Li

Concentrations in mg/kg (ppm)
Conductivities in μS/cm (μmhos/cm)

2.2.5. Water chemistry

Limits are specified for water chemistry parameters during normal operation (see Table II as an example for PWR). However, at some plants, these limits cannot be kept all of the time if water chemistry control is not automatic or if sampling is of low frequency. Oxygen and chloride concentrations are of particular importance during start-up and test conditions .

2.3. POTENTIAL DEGRADATION MECHANISMS AND OPERATING EXPERIENCE

The following ageing mechanisms should be considered for the RPV nozzles:

- Intergranular stress corrosion cracking of cladding
- High cycle fatigue
- Low cycle fatigue, strain induced corrosion cracking
- Intergranular stress corrosion cracking of safe-end welds
- Thermal ageing
- General corrosion, pitting and erosion corrosion
- Irradiation embrittlement.

These mechanisms are already well known and only a brief description is given below. Detailed descriptions of the mechanisms are given in IAEA-TECDOC-540, Safety Aspects of Nuclear Power Plant Ageing.

All of these ageing mechanisms, except thermal ageing and irradiation embrittlement are sensitive to the environmental conditions of stress and water chemistry. However, the data about these environmental conditions normally available from standard tests and plant monitoring is frequently insufficient to determine or predict degradation. Each plant has its own characteristics of component geometry, water chemistry and operational control and general statements about degradation behaviour cannot be made. However, it seems that ageing mechanisms are more sensitive to environmental conditions in BWRs, whereas in PWRs they are more strongly influenced by the effects of mechanical loading.

2.3.1. Intergranular stress corrosion cracking (IGSCC) of cladding

This form of cracking has been most prevalent in BWRs. The necessary conditions required for IGSCC are sensitized austenitic material, a sufficiently high stress, and an aggressive water chemistry. Cracking initiates spontaneously, usually at surface pits or other imperfections, and can progress rapidly depending on the stress, temperature and water chemistry. If cracks penetrate through the cladding, attack of the underlying ferritic material is possible by another mechanism.

Although great care is usually taken to avoid sensitized cladding material, this can still sometimes occur. The most common ways in which material becomes sensitized are:

(a) During post-weld heat treatment.
(b) During service at elevated temperature. This is relatively rare as most plants are designed to operate at temperatures well below those at which sensitization occurs.
(c) During weld deposition when welding parameters are not sufficiently controlled.

Stresses in the cladding result mainly from differential thermal expansion between the stainless steel cladding and ferritic steel base. The nozzle geometry creates a complicated degree of constraint so stresses are unlikely to be uniform. Further stresses arise from service loads. The final stress distribution in the cladding is complex and difficult to determine. It can also be affected by grinding operations or by local repair work.

IGSCC occurs generally in oxygenated water chemistries above 0.2 ppm and is also effected by the presence of other ions such as sulphates and chlorides. Effective monitoring and control of water chemistry is essential in order to reduce periods of operation outside specified limits to a minimum.

IGSCC of non-stabilized 18Cr10Ni type of austenitic stainless steel piping is one of the major sources of operational disturbances in BWRs. Cracking has occurred especially in heat affected zones of weldments of recirculation piping which were sensitized during welding and/or during service. First cracks were detected in a by-pass line at Dresden Unit 2 in September 1974 and since then more than 1000 indications in welds have been detected. Non-stabilized cladding material is not immune to this type of cracking.

IGSCC cracking in the cladding of the RPV head of Quad Cities Unit 2 was detected during a scheduled outage in March 1990. Whilst it is believed that there have been no recorded instances of IGSCC affecting cladding in RPV nozzle regions, this type of cracking has to be considered a degradation mechanism for nozzles clad with non-stabilized material.

2.3.2. High cycle fatigue

BWR feedwater nozzles are usually fitted with a fixed or loose fitting thermal sleeve designed to protect the nozzle surfaces from rapid fluctuations of temperature. For nozzles with loose fitting thermal sleeves, variable leakage flows have been found to occur between the sleeve and the safe-end and produce rapid fluctuations in temperature. The cyclic thermal stresses generated by these fluctuations have caused fatigue cracks to initiate in the cladding of many BWR nozzles. In some cases, this cracking has penetrated to the base material.

Once cracks have grown away from the surface zone and out of the range of local surface stress perturbations, growth by high cycle fatigue usually ceases. However, further growth by another mechanism, such as low cycle fatigue or stress corrosion cracking can still occur. Poor water chemistry can increase the growth rate.

Most BWRs in operation in the USA today went into service originally with feedwater nozzles having loose-fitting thermal sleeves. The first cracks due to high cycle fatigue were revealed during a scheduled refueling outage at Millstone Unit 1 in August 1974. Since then high cycle fatigue has become a generic problem in these BWRs. Remedial measures that have been taken include:

- optimization of thermal sleeve design
- improved sparger design
- removal of cladding in the nozzle.

2.3.3. Low cycle fatigue and strain induced corrosion cracking

Low cycle fatigue is not only a mechanism that can cause crack initiation and growth, but in some instances it can also cause changes in material properties.

An important source of low cycle fatigue is from changes in load resulting from normal plant operations and behaviour (startup, shutdown, load following). Detailed geometric design is intended to limit fatigue damage from stress concentrations at the crotch corner and other locations and it is important to be able to determine the stress concentrations accurately. The ferritic base material of the nozzle is as vulnerable to low cycle fatigue as the cladding. Fatigue damage can occur in both PWR and BWR nozzles.

At the beginning of the 1980s, non-destructive examinations revealed circumferential cracking in the region of feedwater nozzles of RPVs in German BWRs. The major cracks had initiated from weld defects (root gaps, weld misalignment) at the 12 o'clock position in the horizontal piping and had grown to a significant depth through the wall thickness. Small cracks were found at the 6 o'clock position in the nozzle.

Temperature and strain measurements revealed that, during certain infrequent types of reactor operation, high bending stresses were present in the piping adjacent to the feedwater nozzles. These stresses were caused by thermal striping which produced flow stratification where hot tongues of water at the 6 o'clock position flowed counter to colder layers. The temperature differences around the pipe caused significant bending stresses at the welds from which cracks initiated at the 12 o'clock position. Environmentally assisted cracking (i.e. strain induced corrosion cracking) was subsequently able to occur and cause severe cracking. The problem was solved by:

- operational engineering measures to reduce the number and amplitude of stress cycles,
- optimization of sparger design, avoiding stratification,
- backfitting measures such as replacement of thin walled piping using improved welding techniques for the nozzle to pipe weld.

2.3.4. Intergranular stress corrosion cracking of safe-end welds

Intergranular stress corrosion cracking of the transition weld where the ferritic material of the nozzle is joined to the austenitic safe-end has been experienced in BWRs. This weld is of complex construction, having a layer of buttering on the ferritic side, and is usually made of Inconel 82 or 182 weld metal.

These welds have a complex stress distribution due to the effects of differential thermal expansion between the ferritic and austenitic components. Welding the thick sections produces residual stresses which may not be entirely relieved by post weld heat treatment. Determination of the stress field in safe-end welds remains a major problem.

Cracking usually commences at a surface geometric imperfection such as caused by misalignment or poor machining. The driving mechanism is related to the corrosion potential in the coolant and this is why this type of cracking seems to be restricted to BWRs. Cracking rates vary widely depending on the environmental conditions. Inspection of the weld is difficult because of its complex construction.

2.3.5. Thermal ageing

Thermal ageing of RPV nozzle regions is not generally considered to be a significant degradation mechanism. It requires sustained periods of operation at a high temperature where phosphorus and other elements in solution segregate at grain boundaries or form precipitations embrittling the material. Its effect is to shift the ductile-brittle transition temperature to higher values.

The susceptibility to ageing depends on the material microstructure. Heat affected zones are generally more susceptible than weld or base material. In the materials typically used for RPV nozzles, ageing does not begin to occur until temperatures of over 400°C are reached. As the rate of ageing depends on temperature, variations in temperature need to be taken into account. Whilst thermal ageing is not currently considered to be a problem, possible long term ageing effects will need to be taken into account when life extension is considered.

2.3.6. General corrosion, pitting and erosion-corrosion

General corrosion and pitting is only likely to occur in uncladded regions of nozzles during shut down periods. Erosion-corrosion can be a degradation mechanism for uncladded nozzle regions depending on water chemistry parameters (e.g. pH, oxygen content), the flow rate of the coolant, and the magnitude and frequency of local flow disturbances.

2.3.7. Irradiation embrittlement

In general, the material properties in the RPV nozzle region are not as significantly affected by neutron irradiation as those in the RPV core region. However, in vessels whose core region has been annealed, the core region may be no longer the leading location determining the residual life of the vessel. In these cases, the effect of changes in nozzle material properties on vessel integrity in emergency core cooling situations needs to be taken into account when life extension is considered.

2.4. MONITORING OF RPV NOZZLE AGEING

A range of techniques exist for continuous monitoring and surveillance of LWR primary circuit conditions and components including the primary nozzles. These techniques can provide information about ageing effects to be used as a basis for mitigating ageing.

2.4.1. Plant instrumentation and transient recording

A record of the conditions a component has experienced during its life provides essential information for evaluating ageing effects. The most important conditions to be recorded include the temperature, pressure, and water chemistry of the coolant including any transients, excursions or fault occurrences.

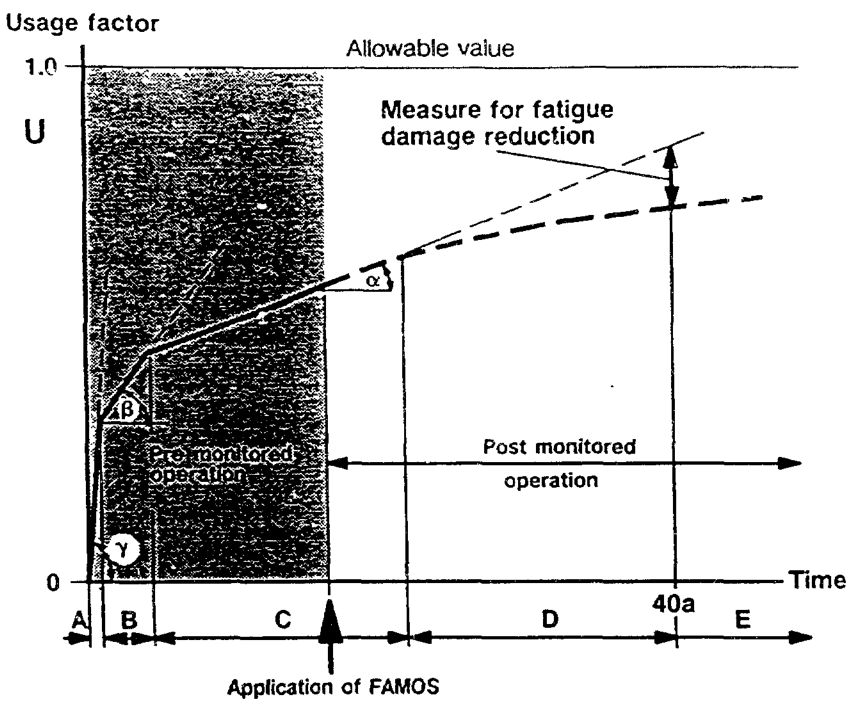
Considerable care is given to choosing the positions for installing measurement devices because it is recognized that, even under normal operating conditions, some parts are subjected to more severe conditions than others (e.g. bimetallic vessel to pipe welds).

For monitoring the average coolant temperature, several thermocouples of appropriate types are used. In addition, thermographic methods are sometimes used for temperature mapping and detecting local temperature variations.

It is imperative that conditions and transients during commissioning (i.e.the test period before normal operation commences) are recorded since these can contribute to ageing effects.

2.4.2. Fatigue monitoring systems

Fatigue monitoring systems offer the capability to monitor the actual fatigue usage of components caused by each operational event provided a detailed stress analysis is available. It is useful to install such a system to a plant even at its midlife since it can provide a more accurate assessment of fatigue usage in the premonitored time period than that based only on general plant operating records (see Fig. 4.). After installation of the monitoring system, an average rate of fatigue usage can be established for the loading conditions corresponding to different types of reactor operation. Using these usage rates, a fatigue usage can be calculated for the total operating life of a component making due allowance for commissioning and early reactor operation.



A Commissioning	(Fatigue accumulation rate γ)
B Initial operation period	(Fatigue accumulation rate β)
C Experienced operation	(Fatigue accumulation rate α)
D Operation considering fatigue damage reduction	(Reduced fatigue accumulation rate)
E Plant life extension	

FIG. 4. Fatigue monitoring system - mid-plant life application.

2.4.3. Surface inspection

Visual examination of the entire vessel surface is part of the usual periodic in-service inspection and is an effective method to reveal unexpected problems. Television cameras and optical devices taking either video or still photographs are commonly used in visual inspection and can provide a permanent record. Photothermal radiometric imaging techniques may be applied for mapping cracks or corrosion. Methods such as dye-penetrant are used to inspect the inner surfaces of the nozzle region and are applied, for example, to WWER-type reactors. Eddy-current examination is sometimes used for detecting defects in the cladding.

2.4.4. Ultrasonic examination

The use of ultrasonic examinations of the reactor pressure vessel and associated components at periodic intervals throughout the operation of the nuclear power plant is well established. Examinations need to be planned and undertaken with great care and thoroughness and provided with appropriate validation to have the necessary high levels of confidence in detection and sizing of flaws. The Programme for the Inspection of Steel Components (PISC) has demonstrated that ultrasonic examination can be effective in determining information about crack-like defects needed for structural integrity assessments.

As a result of PISC and other programmes investigating and developing the effectiveness of NDE, it appears that safe detection of defects with ultrasonic techniques requires scanning large areas with multi-angle probes including high angles of incidence or probes specialized for detection of near surface defects. Cut-off levels must be low. Sizing and characterization of defects require additional special techniques.

2.4.5. Acoustic emission monitoring

The rate of crack growth of known defects is an important factor when assessing residual life. Acoustic emission can be used for monitoring crack growth in-service, especially that associated with thermal cycling, corrosion-erosion and transient loading conditions. However, it is noted that at present there is no general relationship between each increment of crack growth and the corresponding acoustic emission signal. This limits the current usefulness of acoustic emission as a quantitative monitoring technique.

2.4.6. Defect assessment

Fracture mechanics is an effective tool to:

- predict residual life after a defect has been detected
- determine critical defect sizes to assure integrity and safety of the structure
- predict the size and stability of cracks which produce leakage corresponding to the capability of leak detection systems in order to justify leak-before-break.

Uncertainty of these predictions is allowed for by using conservative values of defect size, material properties and crack growth rates.

2.4.7. Water chemistry monitoring

An important factor influencing ageing degradation is the water chemistry of the coolant. The time when the water chemistry has been outside current operational limits and the extent of transgression can be used to indicate potential ageing degradation. On-line systems for directly measuring the corrosion potential and the oxygen content in the coolant are available for monitoring and controlling its water chemistry.

2.5. MITIGATION OF RPV NOZZLE AGEING

Once the causes of ageing degradation are identified (e.g. improper water chemistry or excessive thermal or mechanical load transients), engineering measures or procedural controls can be applied to mitigate ageing.

Such mitigating actions can be :

- changing water chemistry to minimize environmental effects on crack initiation and crack growth
- changing operating procedures or installing instrumentation and controls to avoid or reduce transients conditions
- backfitting and repair activities (changes in design, repair of defects).

2.6. KNOWLEDGE AND TECHNOLOGY GAPS RELATING TO THE UNDERSTANDING AND MANAGEMENT OF RPV NOZZLE AGEING

2.6.1. Understanding of environmental effects

Recent research has revealed that the oxygen content and flow rates of the coolant have a significant influence on crack initiation and crack growth in low alloy ferritic steels under both constant and cyclic load. In the case of cyclic loading, a significant influence on crack initiation was attributed to the load frequency. Further work to clarify and quantify synergistic effects controlling the rate of corrosion fatigue is necessary so that more reliable predictive models can be developed. In addition, there is a need to establish threshold values of K_{ISCC} for stress corrosion cracking of nozzle materials exposed to coolant with different oxygen contents under static load to quantify the safety margin against stress corrosion cracking under operating conditions.

2.6.2. Determination of fatigue damage by NDE

Various methods to measure fatigue damage in components directly by non-destructive tests are being developed in a number of research organizations. These include methods based on the following techniques:

- X ray diffraction
- Barkhausen effect technique
- Ultrasonics (velocity, attenuation, harmonic generation)
- Electrochemical methods
- Etching and replication
- Positron annihilation.

No single method has yet become established and validated, but methods based on determining microcrack initiation, changes in microstructure, and changes in electrical resistance are probably more developed than others. However, further work is necessary before there is a verified method that can be applied with confidence. Evaluation criteria, in terms of acceptable limits, will need to be established to interpret fatigue damage measurements.

2.6.3. Effectiveness and reliability of inspection techniques for crack detection

It is likely that inspection of the RPV nozzle region can be carried out effectively and reliably, but a definite conclusion will be probably only reached in a few years from now when some remaining inspection problems will have been solved and the inspection techniques more fully validated.

The PISC programme and others are assessing whether the inspection techniques currently used are actually capable of detecting defects in the nozzle region before failure. Particular problems are demonstrating and increasing the extent of coverage throughout the region and interpreting signals from the safe-end transition weld and austenitic materials.

Inspection techniques need to be able to provide reliable information for structural integrity assessment with regard to the location, size and characteristics of defects. Further development and validation of techniques to obtain this information for the nozzle region, (particularly the crotch corner and the safe-end weld) is required in order to have the necessary confidence in their use.

The reliability of the information depends very much on the way in which inspection techniques are validated and inspection personnel are qualified. Techniques for producing appropriate test pieces for validating the capability of inspection techniques and personnel still require further development. Such test pieces have to contain defects which simulate both commonly occurring situations and limiting cases that could be imagined during actual plant inspections. One solution, which has been used several times, is to use parts of structures known to contain real defects. An alternative approach is to introduce artificial defects in realistic assemblies which produce the same responses as those from conservative defects. This latter solution has many advantages.

An important way of achieving inspection reliability is by increased automation. New methods of automation are currently being developed in several countries.

2.6.4. Verification of assessment methods

Conventional stress analyses of the nozzle region can be refined to determine more accurately the magnitudes of stress concentrations in the RPV nozzle region. Complex areas such as the safe-end weld require particular attention.

There is a need to see if current conservatism believed to be inherent in design fatigue curves obtained using small scale specimens, both notched and unnotched, can justifiably be reduced by more extensive materials testing on specimens with different microstructures or by an agreed statistical treatment of the test data. It is also necessary to obtain a better understanding of the effects of stress concentration, fatigue strength reduction factors representing the effects on fatigue damage of peak stresses and different microstructures (e.g. plate, weld, HAZ, etc.). Application of different approaches to assessing fatigue life would illustrate the degree of variability in current predictions.

Residual stresses can arise following fabrication operations such as welding, cladding, grinding and heat treatment. These stresses may be local or more general to the nozzle region. There is a need to obtain a greater understanding of distribution and magnitude of residual stresses in the nozzle region by both measurement and calculation.

The calculated rate of growth and assessed limiting sizes of defects in nozzle regions such as the safe-end weld and the crotch corner are important factors determining residual life. In the past, if these calculations were undertaken, they were using a variety of simplified approaches and models. For life extension, there is a need to demonstrate the conservatism of these approaches by, for example, direct determinations of stress intensity factor using three dimensional finite element models containing cracks which can now be attempted with modern computing facilities. There is a need to apply and develop the methodology necessary to obtain a better understanding of the sizes of crack that can be tolerated in the nozzle region without compromising safety margins.

Where the methodology or data used in calculational methods is uncertain, there is a general requirement to verify the methods by testing scaled-down or full size components or structural features (e.g. welds, corners, penetrations, cracks, etc.). A particular example is the need to verify leak-before-break calculations for the safe-end weld by large scale testing. Uncertainties in the calculations include the fracture toughness data and the methodology used for calculating leakage and critical crack sizes under normal operating conditions and seismic loading.

2.6.5. Evaluation of operational experience

Investigations of ageing degradation in RPV nozzle regions (with special emphasis on SCC or fatigue, for example) have been performed by various organizations. There is a need to summarize and publish the results of these investigations in order to identify common problems and mitigation methods that may be applicable to other plants.

2.6.6. Probabilistic failure analysis

Probabilistic failure analysis provides a means for determining the effects of different ageing mechanisms on the probability of failure at different times during the life of the component. It applies probability theory in conjunction with a mechanistic model of failure to derive a probability of failure. The factors that determine failure, such as defect size, material properties, inspection accuracy and reliability, loading and testing, and degradation effects such as embrittlement and crack growth, can be incorporated in the model as deterministic values or statistical distributions. Probabilistic failure analysis can be used as a means to assess the relative likelihood of different failure modes (eg leakage or break), and can be used to optimize the effectiveness of different mitigation actions. It can provide data for use in probabilistic safety assessment. There is a need to apply probabilistic failure analysis to the nozzle region so as to quantify the contribution of the nozzle regions to the total probability of failure for the vessel and the likelihood of leak-before-break for probabilistic risk assessment. So far, analyses of this type are in an early stage of development.

An analysis of the probability of failure requires knowledge of the distributions of material properties and defect sizes. This knowledge is not currently available for safe-end welds and other locations in the nozzle region.

2.7. STATEMENT OF WORK FOR CO-ORDINATED RESEARCH PROGRAMME ON MANAGEMENT OF AGEING OF RPV PRIMARY NOZZLE

2.7.1. Objectives

Exchange of Information

(1) To present and compare methods used by the participating organizations for evaluating the actual condition of nozzles in service; topics of interest include stress distribution under operational loads, material properties and inspection techniques, etc.

(2) To present and compare methods used by the participating organizations for assessing structural integrity and predicting the remaining life of nozzles; topics of interest include the scatter in fatigue and crack growth data, the underlying mechanisms of ageing and their kinetics, and approaches to structural integrity assessment.

(3) To report on mitigating actions (engineering and procedural) which have been used and on their effectiveness in reducing the rate and effects of ageing.

Collaborative Case Study

(4) To compare methodologies for evaluating the current condition, for assessing structural integrity and predicting remaining life by applying them to a selected WWER-440/213 nozzle; data relating to the nozzle will be supplied to interested participants; for this comparison, it will be necessary:

(4.1) to identify significant degradation mechanisms for the particular nozzle selected;

(4.2) to identify the information and data needed to evaluate the effects caused by the degradation mechanisms identified;

(4.3) to collect and, if necessary, to generate the identified data;

(4.4) to assess the structural integrity and remaining service life using a range of recognised approaches and procedures; for example, stress analysis including residual stresses, fatigue usage calculation, evaluation of crack growth, assessment of critical defect sizes, deterministic and probabilistic approaches;

(4.5) to assess the effectiveness of different techniques for inspecting the selected nozzle;

(4.6) to evaluate ways of mitigating ageing degradation of the nozzle;

(4.7) to generalize the conclusions from the case study of the selected nozzle to other nozzles with similar design features.

2.7.2. Scope of work

The CRP scope includes:

- the RPV inlet and outlet nozzle region of a PWR
- the RPV feedwater nozzle region of a BWR.

The nozzle region considered includes areas of the vessel shell where stresses are significantly influenced by the nozzle penetration, and also includes sections of the nozzle up to and including the weld between the nozzle and the safe-end or the attached pipe, as appropriate.

- The methodology for ageing management studies documented in Ref. [1] will be used.

- Information relating to objectives (1) to (3) will be supplied by individual participants and made available to all CRP participants.

- The selected nozzle for the case study will be an RPV inlet nozzle of a WWER-440 Model 213 PWR from a specific nuclear power plant to be decided at the beginning of the study. This choice is made for the following reasons:

  * this type of nozzle has features which are representative of those of many other RPV nozzles (eg. safe-end, bimetallic safe-end weld, environmental conditions);

  * information and comparisons can be drawn from studies of WWER- 440 Model 230 PWR RPVs which are already being made;

  * reassessment of the structural integrity of WWER RPVs according to recognised international codes and procedures is of wide current interest.

2.7.3. Tasks to be performed

Exchange of Information

The following tasks apply to objectives (1), (2) and (3).

Task No.	Task description
1.	Compile and document relevant information
2.	Evaluate and compare information supplied in task 1
3.	Report results.

Collaborative Case Study

The following tasks address objective (4). Proposed work will be performed in two stages.

STAGE I

Task No.	Task description
1.	Select and describe the nozzle according to Section 2.1 of this report.
2.	Review operating experience and identify significant ageing mechanisms.
3.	Identify data required for evaluating current condition and for predicting remaining life.
4.	Collect and evaluate available data identified in task 3.
5.	Derive any missing data (e.g. by extra analysis, testing, inspection, comparisons with other relevant data).
6.	Define approaches for the assessment of structural integrity.
7.	Perform assessment of structural integrity using methods defined by task 6.
8.	Report Stage I results.

STAGE II

Task No.	Task description
9.	Recommend effective and practical mitigation actions based on Stage I findings.
10.	Generalize findings from the case study on WWER-440/213 nozzle to other nozzles with similar design features.
11.	Recommend appropriate applications of the case study results (e.g. in future designs or codes and standards).
12.	Recommend further work (e.g. evaluation of recommended mitigation actions or development of mitigating methods).
13.	Report Stage II results.

2.7.4. CRP network

Organizations from Bulgaria, Czechoslovakia, France, Germany, Hungary, Russia, Switzerland, the United Kingdom and the United States of America as well as the Commission of the European Communities - Institute of Advanced Materials/Joint Research Centre (IAM/JRC) have indicated their interest in this CRP. Other Member State organizations and Organisation for Economic Co-operation and Development/The Nuclear Energy Agency (OECD/NEA) will be invited to participate. Actual participants will be organized in a CRP network to facilitate co-operative work.

3. PILOT STUDY ON THE MANAGEMENT OF AGEING OF MOTOR OPERATED ISOLATING VALVE

Motor operated valves (MOVs) are extensively used in almost all plant fluid-mechanical systems. Operating experience indicates that the operational readiness of nuclear power plant safety related systems have been affected by MOV degradation and failures. The aim of the present pilot study is to improve the understanding and management of MOV ageing and thus to help assure required MOV performance.

3.1. DESCRIPTION OF A TYPICAL MOTOR OPERATED ISOLATING VALVE

3.1.1. General

Motor operated valves (Fig. 5) perform an essential safety function in safety systems, e.g. emergency core cooling and residual heat removal systems, of nuclear power plants under accident conditions. They have to open or close with a high degree of reliability so as to initiate core cooling or to isolate the primary circuit and containment.

Gate, butterfly and globe valves, normally classified in safety classes 1 and 2 and seismic categories I and II, are used to fulfill these functions.

3.1.2. Specifications, standards and regulations

A specification drawn up by the system supplier should define the design and functional requirements that the valve and its actuator should fulfill under normal, upset and accident plant conditions. This includes the number of pressure and temperature cycles, the number of opening and closing operations, materials to be used and other requirements. Furthermore, environmental parameters which may affect the valve performances should be provided (e.g. pressure, temperature and dose rates). These values with appropriate safety margins must be taken into account in the design of MOVs.

Safety related MOVs in modern plants have to meet national qualification standards which require that they are suitable for the environmental, electrical, mechanical and hydraulic conditions associated with normal service and accident conditions described in Section 3.2 below. The functionality of MOVs under these conditions must be demonstrated by testing and/or analytical methods.

In the older plants, important valve requirements were not always defined in the specifications that would be included today. Because of the knowledge gained from operational experience, measurement of fluid dynamic forces or better understanding of environmental conditions. If the original MOV design does not fulfill today's requirements, then the knowledge about the behaviour of the valve under actual service conditions is useful to judge the risk of not performing its intended function under accident conditions.

If current operating conditions of a MOV are found to deviate from those originally specified (for example, with respect to pressure, temperature, load cycles, environment, fluid and loads from the attached piping), the MOV should be reanalyzed to assure its capability to fulfill its intended function.

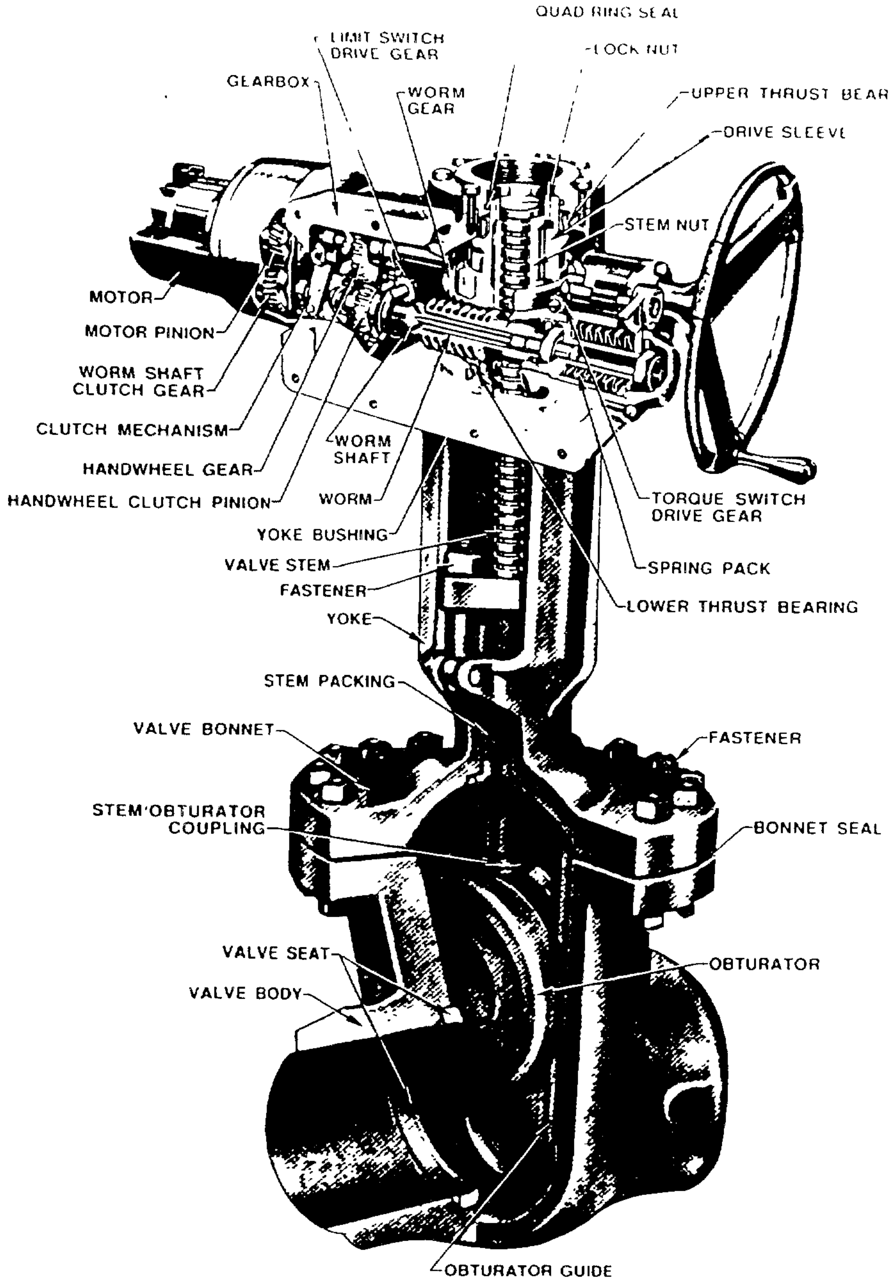


FIG. 5. A typical motor operated valve.

As the nuclear community matured, so have the specification requirements for the hardware. We know from operating experience that under normal operating loads the existing plant hardware is performing its intended function. The areas of current concern, however, are ageing and MOV performance under postulated accident conditions. These concerns have yet to be fully addressed.

3.1.3. Design

Valves found in older plants and in some newer plants were purchased in accordance with their respective national codes which consider primarily the pressure boundary qualification. Functional valve qualification, if performed at all, in many cases was based on analysis and extrapolation of smaller valve qualification results. In some Member States design specifications are being developed to extend the qualification by comparisons and extrapolation.

Many valves installed in older nuclear power plants were not specifically designed and manufactured for NPP application. Often, valves of safety classes 2 and 3 were used which were designed on the basis of standard system pressure and temperature ratings but without considering all expected NPP service conditions (see Section 3.2).

In many cases valves and actuators have been designed and analyzed for performance as separate units rather than as complete assemblies. Modern actuators have been qualified for maximum thrust and torque ratings with margins. An actuator, for example, is qualified for one stall cycle, but analyses still need to be conducted to determine if there is a weak link in the valve. The stem, valve seats, yoke and bolting must be analyzed to determine if the valve assembly can withstand normal operational loads and accidental loads as well as the actuator stall loadings.

Ideally, the actuator, motor, switch assembly, and power/control cables are qualified to national design and environmental standards for electrical equipment. These standards have matured over the years and now include pre-ageing and accident condition environmental qualification. For example, actuators must have enough power to operate the valve under accident conditions and to overcome friction forces acting on seat rings, guides and the stem in the packing area. The stem has to be capable of sustaining loads without buckling, even when a torque or limit switch malfunction disables the stopping of the actuator-motor. The cable sizing for high torque actuator motors, particularly in direct current (DC) applications, has not been well understood, and undersized plant cables have contributed to motor failures.

If analysis reveals that an existing valve cannot meet these design requirements, modification or replacement of the valve should be considered. Programmes for valve modifications based on the results of the qualification programmes are already being implemented by some countries.

3.1.4. Materials

The behaviour of MOV materials under expected environmental loading conditions needs to be assessed for ageing effects even after years of excellent operation because these effects are cumulative and may impair valve performance under accident conditions.

Recently, cracks have been found on cast austenitic stainless steel (SA-351cr CF8/CF8M) casings of valves which were initiated by ageing of cast material. Ageing of cast valve casings after 20 years of operation probably depends on the ferrite content within the austenitic material.

The insulation material of the actuator-motor and the materials of the sealing gasket and packing, lubricants, and torque spring deteriorate with time and have to be examined periodically and replaced if necessary. Relevant guidance may be found in the following reports:

- ELECTRIC POWER RESEARCH INSTITUTE, EPRI NP-5697, Valve Stem Packing Improvement, Final Report, Palo Alto, California (1988)

- ELECTRICITE DE FRANCE, EdF/DER, T. 22/77/10, Guide pratique pour le choix et la pose des garnitures de presse-etoupe pour la robinetterie des PWR, France.

It is important to collect information about MOV material degradation gained from operating experience and research in databases and to alert the plant personnel to the existence of a recognized risk. This information should be available to all utilities and manufacturers of MOVs and valve actuators in order to improve material specifications for new valves.

3.1.5. Fabrication

The fabrication techniques and conditions of safety related MOVs should be carefully controlled to assure that the performance of produced MOVs meets requirements determined in a qualification programme. This qualification programme for valves and actuators should consider all design basis conditions for a given MOV application.

3.1.6. Applicable standards and regulations

MOVs should be designed to meet specified functional requirements for the normal as well as the accident conditions. In particular, the valve body is designed to meet the applicable pressure vessel code rules. Individual utilities may have to use regulatory requirements given while their national industrial codes are being updated.

The actuator, motor, switch assembly and power/control cables are qualified to national design and environmental standards for electrical equipment under accident conditions. Environmental qualification of the actuator, motor, switch assembly and power/control cables may be conducted by pre-ageing and exposing the entire assembly to accident conditions specified by the national regulatory body.

Mechanical qualification of valves may be required to predict valve performance under accident conditions, for example, to ensure closure in the event of blowdown. In order to undertake mechanical qualification for accident conditions, some countries have recently constructed special blowdown loop testing facilities and have required that the performance of valves is demonstrated by testing.

Each country has its national in-service testing standards. These standards are also slowly evolving. Updated standards should emphasize the need for maintaining MOV qualification during its service life and the ability to detect ageing effects before failure. This is a very difficult task due to a wide range and level of original valve qualification and the limited capability of existing testing methods. Also, it should be recognized that accident conditions cannot be simulated during in-service testing.

3.2. SERVICE CONDITIONS

The age-related degradation of a valve and its actuator is dependent, among other things, on the level of its original qualifications and its service conditions, including operating environment and operating history. The service conditions to be considered in ageing management studies can be regarded as "stressors" because they can lead to degradation of the valve and actuator through various physical or chemical processes.

Examples of service conditions which have been identified and investigated to date include environmental and loading conditions such as:

- number and frequency of operations
- temperatures (internal and external, cycling)
- pressure (internal, cycling)
- radiation
- flow rates
- phase change
- system chemistry (corrosion)
- internal mechanical loading (due to friction and/or fluid dynamic loading)
- electrical loading (on motors, wiring)
- humidity (external)
- external loading from connected piping, self weight
- solid matter in the fluid (erosion)
- normal valve positions (torque spring set)
- lubrication breakdown
- vibration.

Ageing studies and equipment operability studies (qualification improvement) should address the ability of installed valves and actuators to operate (repeatedly) under normal operating and design basis event conditions. Some of these conditions listed below must be considered in combination for certain specific applications.

- high temperature (internal and external)
- high humidity
- flooding
- seismic events
- high pressure differential
- high flow rates
- actuator stall loadings
- electrical voltage fluctuation
- high radiation.

3.3. OPERATING EXPERIENCE

Current knowledge about MOV failures and problem areas has been obtained from plant operating histories and test programmes. In addition to plant failure reports, event and maintenance reporting systems and manufacturers' operating and maintenance manuals and notifications offer information on degradation mechanisms. These degradation mechanisms could lead to failures without adequate in-service test and maintenance programmes. The simulated full scale test programmes conducted by several Member States have shown valve problem areas that could only be identified in a real plant environment if a design basis event were to occur.

The failure data consist mostly of failures that have occurred in test situations (periodic, motion testing of in-service valves) but little knowledge is available from real demand abnormal or accident situations with more severe operational and environmental conditions (e.g. differential pressure, high temperature).

TABLE III. AGEING DEGRATION MECHANISMS OF BOTH THE VALVE AND ACTUATOR WHICH HAVE RESULTED IN THE FAILURE OF SAFETY RELATED MOVs TO PERFORM THEIR INTENDED FUNCTION

SITE	DEGRADATION DESCRIPTION
Gate valve assembly	Gate wear, corrosion Guide wear, corrosion Yoke bushing wear Valve stem wear, corrosion, distortion Fastener loosening Valve seat wear, corrosion Bonnet seal deterioration Stem packing wear, deterioration Valve body erosion
Gearbox assembly	Gear wear Shaft wear, distortion Fastener loosening Stem nut wear Stem lock nut loosening Spring pack response change Drive sleeve wear Clutch mechanism wear Seal wear, deterioration Bearing wear, corrosion Lubricant deterioration, hardening Stem nut lubrication degradation
Electric motor	Bearing wear, corrosion Insulation breakdown
Switches	Contact pitting, corrosion Gear/cam wear Insulation breakdown Parts breaking
	Fastener loosening Grease hardening

The major problems leading to critical valve failures have generally occurred in the valve actuator, e.g. torque switch malfunction. Among the mechanical valve parts, the stem is the main contributor to critical failures.

Identification of the root causes of failures is in many cases difficult and thus the age-dependency of failures cannot always be defined. As an example, torque switch malfunction may be due to wrong adjustment, drifting, grease hardening, spring loosening, etc. It should be noted that the detection of defects (degraded condition) prior to MOV failures allows better identification of root cause(s) related to ageing.

The list in Table III describes a variety of degradation mechanisms which can result in MOV failures.

3.4. CURRENT UNDERSTANDING OF MOV AGEING

The current knowledge on MOV ageing has been obtained primarily from operating experience and qualification programmes. Operating experience with motor operated valves shows that failure mechanisms can be divided into two basic categories:

(i) those failure mechanisms that result in excessive friction such that valve actuator/motor cannot drive the valve open or closed in the required time period

(ii) those failure mechanisms that prevent the valve from being operated over its full stroke.

In category (i) the failure is normally due to:

- lack of adequate lubrication causing wear and "jamming" of valve stem or wear of the bearing
- overtightening of valve packing
- corrosion of bearings, stem guides or valve stems
- deposition of "crud" on valve seat
- motor trip on overload.

In category (ii) the failure is normally due to an electrical fault or malfunction of an electrical switch, i.e.:

- failure of switchgear
- failure of limit switch in open or closed position
- faulty torque switch operation.

Another failure in this category might be the uncoupling of motor drive shaft from valve stem (before or after gearbox).

In a failure analysis performed in Finland, of forty-six critical failures occurring over the period considered in the study, about one third were category (i) failures and the remainder were category (ii). A main contributor to the electrical failures was the oxidization of contractors. In a French study, 25 % of failures of electric valves were due to the actuators.

Other age related failure causes identified from operating experience in other Member States are listed below.

- DC motor failure due to insulation breakdown caused by electromagnetic spikes
- non-disengagement of hand control device
- total blockage of kinetic transmission
- misadjustment or drift of switches (for example if the torque spring is loaded over a long period of time, the spring takes a permanent set and delivers less torque for a given torque switch setting)
- incorrect sizing of certain elements
- stem nut lubrication degradation.

The identification of age related failures from operating experience is often difficult because, due to the complexity of the valve-actuator system, the root causes attributable to ageing cannot always be identified with certainty. To achieve better understanding of MOV ageing, ageing mechanisms listed in Table III should be further evaluated. The investigation should focus on those parts of the MOV which form the pressure boundary and those parts which are important for valve performance and operability.

3.5. MONITORING OF MOV AGEING

In-service inspection and testing of MOV actuating systems are undertaken during refuelling and at prescribed intervals in accordance with national standards.

Present inspection and testing methods for verifying pressure boundary integrity are generally accepted as sufficient. However, valves in older plants need to be inspected more extensively and frequently for the existence of minimum wall thickness and for the existence of cracks and abrasion inside the casing than valves in newer plants. Also, provisions for inspecting aged valve bodies in other safety related fluid-mechanical systems must be made to avoid surprises.

Current methods to verify valve capability frequently use valve stroke time as a measure of valve performance but these are of limited value as it takes a large load change in motor output to produce a detectable change in the speed of a typical MOV high torque motor. There is now an effort to develop better in-service test procedures by the use of monitoring systems; current thinking is that the measurement of motor power would be a more useful test than stroke time test. The CEA and EDF are developing an electrical power monitoring system. MOV diagnostics using electrical current signature analysis has been developed in research sponsored by the USNRC at ORNL.

Electrical power and current signature analysis are both qualitative actuator measurements. None of these measurement systems can provide information on the conversion of torque to thrust that takes place in the stem nut, which is outside of the control of the torque switch. The conversion of torque to thrust is dependent on stem nut lubrication and on stem loading. Industry has referred to this phenomenon as a rate of loading effect. This phenomenon can invalidate any in-service test method that cannot load the valve to its design basis load during the in-service test.

The typical in-service test is performed with an unloaded valve (no pressure or flow). Therefore, the load on the actuator is minimal. For limit switch valve control there is no load on the actuating system other than the load due to valve internal friction. For torque switch controlled valves the disc hits the seat and the torque and thrust build up very rapidly (milliseconds). This transient is very difficult for most power measurement systems to follow because of the complex internal voltage, amperage, and wattage calculations. Systems that measure raw peak to peak current and voltage and then process the data through the software after the test may be the best in following the transient. The systems would require at least 1000 Hz sample rate to be of any value.

Developments in monitoring MOV ageing should be aimed at measuring torque and stem thrust along with motor power to perform a complete diagnosis of motor operated valve performance. To make this effort more meaningful, the test should be performed with the highest design pressure and flow load that can be practically obtained on the valve. This is particularly important for limit controlled valves where there is not a seat load. For these valves the analyst may wish to consider motor actuator dynamometer testing. For torque-controlled valves, the seat loads make the test more meaningful.

In-service tests are performed to verify limit switch and torque switch settings which assure valve closure and sufficient thrust. Tests are also performed to detect electrical degradation. Commercial diagnostic systems are now available from several companies which facilitate these tests.

3.6. MITIGATION OF MOV AGEING

Currently, the mitigation of MOV ageing is normally based on planned preventive and corrective maintenance programmes. The data for planning maintenance comes from knowledge of the degradation processes, from operating experience and periodic tests on a specific valve or type of valve. The maintenance programme must be specific for each valve or type of valve considering the operating history and experience.

A planned maintenance programme can be supported by condition monitoring techniques. These techniques, described in Section 3.5, can provide indications of the start of degradation of MOVs before valve failure.

To mitigate ageing problems, less durable components such as seals or gaskets are systematically replaced. However, guidelines should be developed for timely mitigation of MOV ageing. In addition, operating procedures can sometimes be improved to reduce ageing. The maintenance programme must be regularly updated according to accumulated operational experience.

3.7. KNOWLEDGE AND TECHNOLOGY GAPS RELATING TO THE UNDERSTANDING AND MANAGEMENT OF MOV AGEING

3.7.1. Understanding of MOV ageing

A database on the failure and malfunction of MOVs in plant operation should be developed, maintained, and made available to interested parties. This database would provide information for understanding MOV ageing including degradation and failures, their cause and correction and thus alert operators of nuclear plants to potential problems and potential maintenance needs. Databases of this type are being developed in several countries today specifically for their own operating nuclear plants. However, they lack the common format needed to understand age related degradation processes and to utilize them internationally. One of the first tasks of the pilot study should be the development of a common database format useful to understand MOV ageing.

3.7.2. Monitoring of MOV ageing

A system or device needs to be identified and/or developed that could detect ageing of MOVs, would enable early detection of degradation, and allow for maintenance or correction before failure occurred and thus enhance valve availability and reliability. Such system or device may be able to identify the probable cause of degradation so that appropriate maintenance action could be taken. Research has been undertaken on the use of motor current signature analysis and power measurement. The design of such system or device should be further developed to detect age related degradations in MOVs and should be demonstrated by in-plant experience. The device may have to measure more than the current or power, for example, the thrust on torque controlled valves. Monitoring of age related degradation in limit controlled valves may require technology not yet developed.

3.7.3. Risk and reliability assessment

Stochastic models have been developed for use in probabilistic risk assessments and assessments of valve reliability . Current stochastic models are also available for describing the time dependence of ageing processes accurately. However, more data are needed to assess the degraded MOV conditions and failure for use in probabilistic risk assessments. Maintenance schedules and the need for valve replacement and repair could then be predicted more accurately. Those valves which represent the largest contributor to risk could then be identified and given appropriate attention and maintenance.

3.7.4. Qualification methods

MOVs are currently qualified before installation for environmental, electrical and mechanical (thermal-hydraulic) conditions associated with accident conditions using both testing and analysis. Because the design of MOVs varies and specified accident conditions vary from country to country and in application, it would be difficult to conduct an international test programme. However, there is a need to exchange information on the techniques and results of qualification tests. Universal guidelines for MOV qualification considering ageing effects should be developed.

The degradation of valves in-service is simulated by pre-ageing treatments prior to testing under accident conditions. It would be desirable to remove few selected MOVs after actual in-plant ageing (say 20 years) and determine by a qualification test whether ageing degradation has affected the original qualification. Information could then be exchanged to assist in formulating future modifications in qualification test procedures.

Improved mechanical qualification methods would also enable utilities to specify motors and actuators more closely matched to the thrust force required during valve opening, closure or isolation under accident conditions.

3.7.5. Maintenance procedures

Guidelines should be developed so that maintenance procedures would adequately address known ageing degradation processes. These guidelines would be used by utilities and manufacturers of valves in developing detailed maintenance procedures.

Improved MOV diagnostic systems utilizing current, voltage, valve torque and stem thrust or strain, etc. are now available which can be used to accurately set the limit and torque switches of the valves. However, results of a recent validation programme in the USA revealed a wide range of MOV diagnostic equipment accuracy. Initial calibration of the thrust measurement devices was a problem for most of the systems. Users of test equipment should be aware of its capabilities and limitations.

Better maintenance procedures that alleviates ageing concerns coupled with advanced diagnostic tools to detect degraded condition would reduce the risk of valve failure and give greater assurance of safe operation under accident conditions.

3.8. STATEMENT OF WORK FOR CO-ORDINATED RESEARCH PROGRAMME ON MANAGEMENT OF AGEING OF MOTOR OPERATED ISOLATING VALVE

3.8.1. Objectives

(1) To improve understanding of MOV ageing mechanisms and effects, and thus help assure functionality of MOVs under both normal operating and accident conditions.

(2) To identify effective and practical methods for monitoring of MOV ageing capable of timely detection of MOV anomalies attributable to age related degradation.

(3) To develop guidelines for risk and reliability assessment of MOV ageing.

(4) To improve MOV qualification methods and formulate MOV qualification guidelines.

(5) To establish guidelines for effective MOV maintenance to alleviate ageing effects and concerns.

3.8.2. Scope of work

The CRP will include:

- The entire motor operated valve as defined in Section 3.1 and illustrated in Fig. 5. Upon completion of Stage I, recommendations will be developed for selecting one or more specific MOVs and for Stage II work.

- The methodology for ageing management studies documented in Ref. [1] will be used.

3.8.3. Tasks to be performed

All five CRP objectives will be addressed in parallel.

Phased approach described below will be used to meet the CRP objectives. Such an approach will facilitate achievement in Stage I of intermediate results useful for MOV ageing management and the initiation of appropriate follow-up work under Stage II.

STAGE I

Task No.	Task description	Duration (months)
	(1) Understanding of MOV ageing	
1.1	Specify a practical common format for collection of incident data and exchange of evaluated data on age related malfunction and degradation of MOVs (including a root cause and origin of incidents, when available). Also specify the means for information exchange to be used.	3
1.2	Exchange evaluated data using the format specified under 1.1.	6
1.3	Analyse compiled data and identify dominant ageing mechanisms for MOVs.	12
1.4	Prepare Stage I report and recommend follow up work on understanding of MOV ageing for Phase II CRP, if needed.	6
	(2) Monitoring of MOV ageing	
2.1	Specify a practical common format and means for the exchange of information on MOV monitoring and diagnostic methods and on their effectiveness in detecting MOV degradation before failure. (All ageing mechanisms listed in Table III should be considered).	3
2.2	Exchange information specified under 2.1.	6
2.3	Analyze compiled data and identify (a) effective and practical MOV monitoring methods, and (b) those significant ageing mechanisms for which effective monitoring methods are not available.	12
2.4	Prepare Stage I report and recommend follow-up work on monitoring MOV ageing for Stage II, if necessary.	6
	(3) Risk and reliability assessment of MOV ageing	
3.1	Specify a practical common format for the collection and exchange of operating experience data useful for risk and reliability assessment of MOV ageing. (These data should be	3

Task No.	Task description	Duration (months)
	useful for probabilistic safety assessment (PSA), failure modes and effects analyses (FMEA) and for ranking of MOVs in terms of their risk significance.) Also specify the means of information exchange to be used. Tasks 1.1 and 3.1 should be co-ordinated.	
3.2	Exchange data specified under 3.1.	6
3.3	Compare existing stochastic models describing MOV ageing using compiled data.	12
3.4	Prepare Stage I report and recommend follow-up work to develop improved stochastic models of MOV ageing in Stage II, if needed.	6
	(4) MOV qualification methods and guidelines	
4.1	Specify a practical common format and means for exchange of information on MOV qualification methods and results and, in particular, on techniques used to account for MOV ageing effects.	3
4.2	Exchange information specified under 4.1.	6
4.3	Compare compiled information and identify effective MOV qualification methods and techniques.	12
4.4	Prepare Stage I report and recommend follow-up work on MOV qualification for Stage II, including the development of qualification guidelines and improved qualification methods, if needed.	3
	(5) Guidelines for MOV maintenance	
5.1	Specify a practical common format and means for the exchange of information on MOV maintenance methods and their effectiveness in mitigating ageing effects. (For example, current reliability centred maintenance projects used for the optimization of maintenance programmes could provide useful information on the effectiveness of MOV maintenance.)	3
5.2	Exchange information specified under 5.1.	6
5.3	Compare and analyze compiled information, and identify effective MOV maintenance methods.	12
5.4	Prepare Stage I report and recommend follow-up work on MOV maintenance, including the development of maintenance guidelines that would address all significant ageing mechanisms identified in task 1.3.	3

STAGE II

Objectives and scope of Stage II will depend upon the findings and the overall progress made in Stage I. It will also depend upon additional needs of participating Member States, the knowledge and technology gaps existing at the end of the Stage I, and the availability of resources.

For example, Stage II study may be needed to narrow the knowledge and technology gaps relating to:

- qualification testing of naturally aged MOVs under accident conditions,
- selection and prioritization of MOVs based on their susceptibility to ageing and their risk significance, and
- built-in diagnostic and monitoring features useful for monitoring and predicting MOV performance.

3.8.4. CRP network

Organizations from Bulgaria, Czechoslovakia, Finland, France, Germany, the United Kingdom and the United States of America have indicated their interest to participate in the CRP. An invitation will be extended to other interested parties. Actual participants will be organized in a CRP network to facilitate co-operative work.

It is anticipated that the CRP will involve the participation of utilities, manufacturers, regulatory and research laboratory personnel, depending on the task. For example, participation of utilities will be needed to exchange information on valve failures since the utilities and sometimes manufacturers control this information. Studies of age related risk analyses are highly specialized and will probably involve personnel from scientific, consulting and national research laboratories. The development of safety guidelines is of specific interest to regulatory bodies but should take into account input from laboratories and utilities. The research on MOV monitoring should involve both the research laboratories and the utilities to provide for co-ordinated development and demonstration. The exchange of information on MOV qualification methods and test results will likely involve regulatory bodies, manufacturers and utilities.

It is recognized that it would be difficult to establish joint testing or laboratory research work because of the unique nature of the design of valves used in each country. However, the technology gaps can be more effectively filled if all countries with active research programmes on MOVs shared the results of their research and work together to develop generic guidelines.

4. PILOT STUDY ON THE MANAGEMENT OF AGEING OF CONCRETE CONTAINMENT BUILDING

The following review of current understanding, monitoring and mitigation of ageing of concrete containments recognizes and draws upon research from many countries. This review does not attempt to be exhaustive in its presentation of the discussion of age-related degradation mechanisms or their significance. Many other documents have done this (e.g. IAEA-TECDOC-540, NUMARC Industry Reports on Containments, etc.). This review summarizes and generalizes the status of ageing in concrete containments and focuses on those areas where co-ordinated research would be of most international benefit and use.

4.1. DESCRIPTION OF CONCRETE CONTAINMENT BUILDING

4.1.1. General description

Containments provide the ultimate barrier between the primary reactor systems and the outside environment. In addition to providing a physical barrier, containments are designed to withstand internal pressure resulting from design basis accidents. Concrete is widely used in containments design due to its durability and ease of construction. Joints in the concrete are typically sealed with water stops and other sealants. Piping and electrical connections to and from equipment inside containment pass through specially designed leak tight penetrations. Some containments are designed as prestressed/post-tensioned structures and are provided with post-tensioned tendons.

4.1.2. Applicable standards and regulations

There are separate national standards/requirements governing design, construction, operation, testing, etc. for each of the various components of the concrete containment such as the reinforced concrete, liner, reinforcement, tendons, etc. Because of these differences in standards, there is a corresponding variation from country to country of the susceptibility to ageing and significance to plant safety. The codes and standards in use tend to be conservative and depend upon types of materials available in each country and the climate of the country.

4.1.3. Design description

Concrete containment structures are designed to separate the reactor and other safety-significant systems and equipment from the outside environment. External and internal events are considered in the design. External events include severe weather conditions (e.g. floods and tornados) as well as potential missile impingement (airplane crashes and turbine blading). Critical internal events include loss of coolant accidents (LOCAs) and high-energy-line break. The containment is capable of withstanding the differential pressure resulting from the above postulated events and represents the final physical barrier to radioactive material before the outside environment. Appendix II provides examples of containment design evolution in different countries.

4.1.4. Materials

The design and construction of the containment depends on the reactor type. Typical subcomponents include:

- Concrete
- Rebar
- Structural steel
- Liner plate
- Prestressing system
- Penetration assemblies
- Waterstops.

4.2. SERVICE CONDITIONS

Differences in standards and environment from country to country produce variations in design and service conditions and in the postulated events to be considered. However, the following generalizations can be made.

Temperature

Inside the containment wall the temperature is normally between 20 and 65°C. Inside containment temperatures will vary over the structure as a result of stratification or from local interior heating sources. Outside the containment the temperature depends on the local weather and can be between 60 and -30°C. The concrete design also considers temperature gradients across the wall thickness, and the rate of temperature changes during operational and weather transients.

Pressure load

During normal operation, containments are typically maintained at atmospheric pressure. Containment systems are also designed to support an internal pressure load derived from LOCA conditions (about 0.4 MPa).

Relative humidity

While relative humidity outside the concrete containment can vary from near 0 to 100%, the inside environment is typically controlled to levels not greater than 70%.

Extreme external events

Tornados, earthquakes, tsunami, floods and other external events are typically considered design basis events, and are included in the design and design analysis.

Extreme internal events

LOCA condition inside the containment can create extreme conditions for a short duration (high temperature, radiation, pressure, and humidity) which can accelerate some ageing degradation processes. Hydrogen generation during severe accidents and the accompanying explosive potential are also considered.

4.3. OPERATING EXPERIENCE

The performance of concrete containment structures has, in general, been excellent. It should be noted that these structures have not been subjected to a design basis challenge (e.g. a large LOCA). Degradation reported to date has been limited in both its occurrence and extent.

For reinforced concrete, minor cases of cracking and corrosion of rebar have been noted. In addition, instances of local freeze/thaw deterioration in the dome regions of containment have been identified.

For prestressing systems, the degradation noted consists of minor corrosion of the prestressing tendons and isolated cases of loss of required prestress.

Penetrations in the liner, which form part of the containment pressure boundary, have developed leaks through local tearing and bulging of the liner. These leaks have been uncovered during inspections and tests that evaluate containment leak tightness.

In general, the limited degradation experience to date attests to the conservatism of the governing design codes and the inherent benign environment. Where degradation has been noted, design or construction errors have typically been identified as the prime cause, leaving the structure susceptible to premature damage by ageing factors or use.

4.4. CURRENT UNDERSTANDING OF AGEING OF CONCRETE CONTAINMENT BUILDING

Concrete containments are subject to a multitude of potential degradation factors acting to reduce their service life or performance. In the context of general building service life, a list of degradation factors is given by ISO 6241 and for general materials service life a list is given by ASTM 632. Degradation factors relevant to building performance and service life are summarized in Tables IV and V, respectively.

In the specific context of NPP concrete containments, the global consideration of degradation factors can be reduced to a much smaller subset of potentially significant factors. Taking the containment structural elements for consideration individually, these factors are summarized below.

4.4.1. Concrete degradation

4.4.1.1. Concrete cracking during early hydration

Cracking can occur when concrete is in the initial casting plastic state by drying shrinkage, plastic shrinkage and settlement processes. This in itself is not considered an ageing mechanism. However, the cracks resulting from these processes, as well as any thermal hydration cycle cracks, represent an initial unique event which should be minimized by standard recommended concreting procedures. Cracks formed during this process can amplify other ageing mechanisms.

TABLE IV. DEGRADATION FACTORS (AGENTS) RELEVANT TO BUILDING PERFORMANCE

Nature	Origin			
	External to the building		Internal to the building	
	Atmosphere	Ground	Occupancy	Design consequences
1. Mechanical agents				
1.1. Gravitation	Snow loads, rain water loads	Ground pressure, water pressure	Live loads	Dead loads
1.2. Forces and imposed or restrained deformations (4.01)	Ice formation pressure, thermal and moisture expansion	Subsidence, slip	Handling forces indentation	Shrinkage, creep, forces and imposed deformations
1.3. Kinetic energy	Wind, hail, external impacts	-	Internal impacts, wear	Water hammer
1.4. Vibrations and noises	Wind, thunder, airplanes, explosions, traffic and machinery noise	Earthquakes, traffic and machinery vibrations	Noise and vibration from music, dancers, domestic appliances	Services noises and vibrations
2. Electromagnetic agents				
2.1. Radiation	Solar radiation, radioactive radiation	-	Lamps, radioactive radiation	Radiating surface
2.2. Electricity	Lighting	Stray current	-	Static electricity, electrical supply
2.3 Magnetism	-	-	Magnetic fields	Magnetic fields
3. Thermal agents	Heat, frost, thermal shock	Ground heat, frost	User emitted heat, cigarette	Heating, fire
4 Chemical agents				
4.1. Water and solvents	Air humidity, condensation, precipitation	Surface water, ground water	Water sprays, condensation, detergents, alcohol	Water supply, water waste, seepage
4.2 Oxidizing agents	Oxygen, ozone, oxides of nitrogen	-	Disinfectant, hydrogen peroxide	Positive electro-chemical potentials
4.3. Reducing agents	-	Sulphides	Agents of combustion, ammonia	Agents of combustion, negative electrochemical potentials
4.4 Acids	Carbonic acid, bird droppings	Carbonic acid, humic acid	Vinegar, citric acid, carbonic acid	Sulphuric acid, carbonic acid
4.5. Bases	-	Lime	Sodium hydroxide, potassium hydroxide, ammonium hydroxide	Sodium hydroxide, cement
4.6 Salts	Salty fog	Nitrates, phosphates, chlorides, sulphates	Sodium chloride	Calcium chlorides, sulphates, plaster
4.7. Chemically neutral	Dust, soot	Limestone, silica	Fat, oil, ink, dust	Fat, oil, dust, soot
5. Biological agents				
5.1. Vegetable and microbial	Bacteria, seeds	Bacteria, moulds, fungi, roots	Bacteria, house plants	-
5.2. Animal	Insects, birds	Rodents, worms	Domestic animals	-

(Source: RILEM report - Prediction of service life of building materials and components, CIB W80/RILEM 71-PSL, 1987)

TABLE V. DEGRADATION FACTORS (AGENTS) AFFECTING THE SERVICE LIFE OF MATERIALS AND COMPONENTS

1. Weather factors	Radiation
	Solar
	Nuclear
	Thermal
	Temperature
	Elevated
	Depressed
	Cycles
	Water
	Solid (such as snow, ice)
	Liquid (such as rain, condensation, standing water)
	Vapour (such as high relative humidity)
	Normal air constituents
	Oxygen and ozone
	Carbon dioxide
	Air contaminants
	Gases (such as oxides of nitrogen and sulphur)
	Mists (such as aerosols, salt, acids and alkalies dissolved in water)
	Particulates (such as sand, dust, dirt)
	Freeze-thaw
	Wind
2. Biological factors	Microorganisms
	Fungi
	Bacteria
3. Stress factors	Stress, sustained
	Stress, periodic
	Stress, random
	Physical action of water, as rain, hail, sleet, and snow
	Physical action of wind
	Combination of physical action of water and wind
	Movement due to other factors, such as settlement or vehicles
4. Incompatibility factors	Chemical
	Physical
5. Use factors	Design of system
	Installation and maintenance procedures
	Normal wear and tear
	Abuse by the user

(Source: CIB W80/RILEM 71 PSL, 1987)

4.4.1.2. Freeze-thaw

Degradation of a saturated cement paste or porous aggregate can occur as a result of expansion of water during the freeze-thaw cycle which eventually, in extreme cases, causes spalling of the concrete surface and exposure of the reinforcement. Freeze-thaw activity is typically limited to above ground surfaces where water can collect. If not corrected, freeze-thaw damage has the potential to compromise the structural integrity of the concrete.

The problem is only likely to occur in geographic regions with a severe weathering index and can be accommodated in the design stage by incorporation of the following in the mix specification:

- Low water-cement ratio
- Air entrainment typically in the range of 3 to 6% for low water-cement ratio concrete and higher for high water-cement ratio concrete
- Aggregate with adequate durability for the expected exposure.

The onset of significant damage, indicated by cracked or spalled concrete, can be determined by visual inspection.

4.4.1.3. Concrete leaching

Water passing through cracks or joints in concrete can dissolve or leach the soluble calcium hydroxide that was produced primarily by the hydration of the cement, as well as leaching other alkalis of sodium and potassium. Long term leaching increases porosity and permeability of the concrete and reduces its strength. Leaching also reduces alkalinity locally thus lowering the pH level and increasing the potential for rebar corrosion.

As with freeze-thaw considerations, a good concrete mix should preclude significant leaching effects. The mechanism should be detectable by visual inspection: leached material tends to react with atmospheric carbon dioxide to form a white deposit. Leaching degradation is noticeable in above ground areas from the white deposit residue left on the surface.

4.4.1.4. Aggressive chemicals

Concrete is naturally alkaline (pH about 13) and even good quality concrete can be attacked by acids (pH about 5.5), predominantly by accelerated leaching of calcium hydroxide. In normal operation, the containment should not be exposed to any chemicals and any spills or leakages can be considered as special cases. Some containments will be exposed to industrial pollution from acid rain or salt spray at seaside locations. In below ground situations, sulphate solutions of sodium, potassium and magnesium sometimes found in groundwaters may attack concrete, often in combination with chlorides. Again the attack is initially on the calcium hydroxide component of the concrete though attack on the actual cement paste hydration product can occur, leading to disintegration of the concrete surface.

In above ground situations, these effects should be minimal on good quality concrete and detectable by routine visual inspection. Since below ground concrete is not subject to visual examination, groundwater characteristics should be monitored to determine if chemical attack is a potential degradation mechanism.

4.4.1.5. *Alkali-aggregate reactions (AAR)*

There are three kinds of reactions that are possible between particular aggregates and the alkaline pore solution within cement paste: AAR, cement-aggregate reaction and alkali-carbonate reaction. The reactions need specific regimes to occur. They can produce a water imbibing gel that, if given access to moisture, can create sufficient internal expansion to cause internal microcracking and possibly surface macrocracking. Upon exposure to air, the gel reacts with atmospheric carbon dioxide to form a white calcium carbonate deposit.

Despite extensive research and extensive experience in various countries, there is no definitive universal predictive test for AAR, although various standards do exist and recommendations are available to minimize the risk from AAR. Tests are available to positively detect the presence of AAR. From international experience the time period before AAR can be detectable can be very long, approximately 5 - 10 years. Compliance with these standards, coupled with routine visual inspections, should minimize the risk of AAR becoming a significant mechanism. However, more knowledge is required to understand the mechanism and to develop precise predictive reaction indicators.

4.4.1.6. *Carbonation*

The calcium hydroxide in the cement paste hydration products can react with the carbon dioxide in the atmosphere to reduce the natural alkalinity of the concrete. This occurs primarily at relative humidities of between 40% and 70%. The primary concern is the reduced alkalinity of the affected concrete: if the carbonation depth reaches the level of the reinforcing steel, then the passivity of the steel may be lost, creating potential corrosion especially in the case of chloride access coupled with the presence of oxygen and moisture. For good quality concrete, the depth of penetration should be extremely low, about 1 mm per year. Mechanical properties of the concrete are not affected significantly; there may be more shrinkage, but the strength increases slightly.

4.4.1.7. *Elevated temperature*

The compressive strength, tensile strength, and modulus of elasticity of concrete are reduced when it is subjected to prolonged exposure to elevated temperatures. Literature suggests that reductions in excess of 10% begin to occur in the range of 82 to 93°C. The effect of elevated temperature is highly dependent on the moisture content of the concrete.

Containment concrete generally does not experience temperatures higher than 65°C during normal operation. Locally, concrete containments may experience moderately higher temperatures. At temperatures in this range, concrete structures do not require any special consideration.

4.4.1.8. *Irradiation*

Concrete can undergo changes in properties if exposed to excessive neutron and/or gamma radiation. The following characterizations relevant to containment concrete have been made :

- Heat caused by radiation effects (nuclear heating) can cause a reduction in mechanical properties, loss of moisture, and volume change.

- The effects of neutron irradiation are measurable at levels as low as 10^{19} n/cm^2.
- Nuclear radiation seems to have little effect on the shielding properties of concrete beyond the moisture loss produced by heating.
- Radiation degradation of concrete is not readily observable.

Depending on the individual plant geometry, power level, and fuel type, the neutron flux at the containment shell will vary. For example, in PWRs substantial shielding reduces the neutron flux and energy reaching the PWR containment shell, resulting in levels of accumulated exposure during the course of normal operation that are far below the levels necessary to cause degradation. This shielding is provided by the water inside the reactor vessel, the reactor vessel itself, the biological shield wall (concrete) and a substantial air gap. Neutron fluence levels at the containment wall are typically less than 10^{14} n/cm^2. These levels may be exceeded at very localized areas but are no larger than 10^{17} n/cm^2. Both are well below the thresholds for radiation or radiation heating degradation. The maximum integrated gamma dose at the outside of the reactor pressure vessel corresponding to 80 years of operation is approximately 9.3 x 10^9 rad [2] which is well below the dose at which measurable degradation begins.

4.4.2. Degradation of reinforcing steel

4.4.2.1. Corrosion

Concrete's high alkalinity (pH > 12.5) provides an environment around the embedded steel which protects it from corrosion. However, when the pH of this environment is reduced (pH < 11.5) by either the leaching of alkaline products through cracks, entry of acidic materials or carbonation, corrosion can take place. Chlorides accelerate corrosion and could be present in constituent materials of the original concrete mix (i.e., cement, aggregates, admixtures and mixing water), or they may be introduced environmentally. The severity of corrosion is influenced by the properties and type of cement and aggregates, and the moisture level within the concrete.

Studies have also been conducted to determine the effects of stray electrical currents on reinforcing steel. In addition, lightning conductors exchange electrons with the atmosphere, and if connected to reinforcing steel, may cause accelerated corrosion.

The primary initiator for corrosion are chlorides in groundwater, especially when the level of the groundwater table fluctuates. In addition, the exterior and interior surfaces of the containment shell and dome may be susceptible when the appropriate moisture and oxygen conditions are present.

Corrosion products have a volume greater than that of the original metal. The development of these corrosion products subjects the concrete to tensile stress, eventually leading to hairline cracking, followed in time by rust staining, spalling and more severe cracking. These actions will expose more reinforcing steel to a potentially corrosive environment and the concrete to further deterioration, resulting in more extensive degradation of both.

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[2] 1 rad = 1.00 x 10^{-2} Gy.

The risk of corrosion damage is minimized by the use of good quality concrete and adequate cover over the reinforcing steel. Concrete used for containments is typically of high quality with relatively high strength (4000 psi)[3], low water-to-cement ratio (0.35 to 0.45), and air entrainment (3-6%). Aggregates are also typically well graded, which contributes significantly to low permeability. Accordingly, reinforcing steel is inherently resistent to corrosion degradation, with potential for corrosion only when exposed to fluctuating levels of aggressive groundwater.

Presence of reinforcement corrosion can be detected by visual inspection for cracking, spalling, and staining on the surface. Below ground, the potential for corrosion can be determined by monitoring groundwater chemistry as mentioned in Section 4.4.1.4.

4.4.2.2. Elevated temperature

Hot rolled reinforcing steel shows a reduction in yield strength and modulus of elasticity at elevated temperatures; at 375°C, the reduction is 15%. Above this temperature, the reductions become more pronounced, reaching about 50% at 590°C and more than 80% at 775°C. At temperatures up to 315°C, the bond strength between concrete and reinforcing steel is unaffected. Since normal operating temperatures within containments are much lower, 45°C to 65°C, elevated temperature effects on reinforcing steel will not cause significant degradation.

4.4.2.3. Irradiation

Steel degradation due to neutron irradiation is caused by the displacement of atoms from their normal lattice positions. The effect of this is to increase the yield strength, decrease the ultimate tensile ductility and increase the ductile to brittle transition temperature. These defects on a macroscopic level produce what is referred to as radiation induced embrittlement which is taken into account in the design and operation of reactor pressure vessels. Currently available data indicate that these effects on the mechanical properties of the steel are measurable at 10^{18} n/cm^2 (E > 1 MeV).

The doses experienced by containment reinforcing steel (typically less than 10^{14} n/cm^2 for PWRs) should be well below this threshold.

4.4.3. Degradation of prestressing system

4.4.3.1. Corrosion

Prestressing tendons are normally maintained in a passive environment by protective grout or grease. When corrosion of prestressing tendons occurs, it is generally in the form of localized corrosion. Most corrosion related failures of prestressing tendons have been attributed to pitting, stress corrosion, hydrogen embrittlement, or some combination of these. For corrosion to occur, the protective environment must be disturbed or somehow compromised.

Pitting is a highly localized form of corrosion. The primary parameter affecting its occurrence and rate is the environment surrounding the metal. Stress corrosion results

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[3] 1 psi = 6.895×10^3 Pa.

from the simultaneous presence of a conducive environment (typically consisting of hydrogen sulfide, ammonia, nitrate solutions or seawater), a susceptible material and tensile stress. Hydrogen embrittlement (technically not a form of corrosion) occurs when hydrogen atoms, produced by corrosion or excessive cathodic protection potential, enter the metal lattice. Hydrogen produced by corrosion is not usually sufficient to result in hydrogen embrittlement of carbon steel.

Corrosion of prestressing wires causes cracking or a reduction in wire cross-sectional area. In either case, the prestressing forces applied to the concrete are reduced. If the prestress forces are reduced below the design level, a reduction in design margin would result.

Corrosion of prestressing wire can be significantly mitigated by maintaining grease or other corrosion inhibitors or grout in the tendon ducts. Grease can be sampled and tested during periodic tendon surveillance for the presence of voids, free water, or changes in chemical properties.

4.4.3.2. Elevated temperature

Exposure of heat-treated and drawn prestressing wire to elevated temperatures is similar to the annealing process. There is a loss in tensile (yield and ultimate) strength, and an increase in relaxation and creep losses. These changes in material behaviour are due to alterations in the crystal structure of the metal and do not reverse upon cooling. Research indicates that exposure to temperatures up to 204°C reduces the tensile strength by approximately 10%. This same research indicates that exposure to a temperature of 60°C for 50 years causes a 300% increase in relaxation (percent of initial tension) over wire tested at room temperature (20°C) for the same period.

Typical containment temperatures are not high enough to cause significant loss of prestressing tendon tensile strength. Further, the effect of elevated temperatures on relaxation and creep properties of containment tendons are considered during design, in calculating the prestress losses. Therefore, elevated temperature effects on prestressing tendons should not be significant when the containment structures are designed and operated within the above limits.

4.4.3.3. Irradiation

Irradiation affects the mechanical properties of steel by dislodging atoms from the metal lattice, creating vacancies and interstitial atoms. This increases the tensile strength of the metal, but reduces the ductility. For prestressing wires and strands, radiation exposure will cause a decrease in the expected relaxation levels.

Irradiation can also affect the grease used in the tendon ducts. The effect of gamma radiation on grease is a loss of viscosity.

Studies have shown that exposure of prestressing wire to a neutron dose of 4×10^{16} n/cm^2 has a negligible effect on the mechanical properties of the wire. Tests of corrosion inhibitors specifically formulated to protect prestressing tendons indicated no changes in physical properties outside the original material specification range when irradiated to 10^{10} rad. Nominal exposure levels are well below the 10^{14} n/cm^2 level that is typical for the containment wall, and also well below the degradation threshold.

4.4.3.4. *Prestressing losses - relaxation*

After the prestressing tendons are tensioned during construction, there is a tendency for the resulting stress to reduce with time because of several factors:

- stress relaxation of the prestressing wire,
- shrinkage creep, or elastic deformation of the concrete,
- anchorage seating losses,
- tendon friction, and
- reduction in wire cross-section due to corrosion.

With the exception of the corrosion effect these losses are calculated and considered in the design process. If the losses were to exceed those considered in the design, the result would be a reduction in the design margin. Prestress losses are monitored, as part of the in-service inspection programme, by periodic liftoff tests.

4.4.4. Degradation of liners and penetrations

4.4.4.1. *Corrosion*

Concrete containment liners may be constructed of many materials but are most frequently built from steel plates welded together to provide a leaktight containment boundary. Penetrations are constructed primarily of steel parts connected by welding or mechanical tightening and occasionally employ elastomeric sealing elements. Corrosion and localized failure of the liner or penetrations could limit the leaktightness capabilities of the structure.

The same mechanism that causes reinforcement corrosion, namely electrochemical corrosion, can initiate corrosion of containment liners or penetrations. In addition, because of its accessibility to the outside environment, microbiologically induced corrosion (MIC) is also possible. Corrosion can occur in liners from either the exposed or unexposed (concrete) side of the plate. Corrosion from the exposed side would be more likely to occur at the floor locations, and anywhere inside the containment where fluids are likely to collect. As noted in the discussion on reinforcing steel (Section 4.4.2.1.), electrochemical corrosion requires aggressive ions and oxygen to be present for the reaction to take place. Much remains to be known about MIC-type corrosion; however, the potential is generally dependent upon the plant water systems, and direct contact with the liner. For example, the use of untreated water can result in microorganisms being imported with silt which settles on the containment floor, forming an environment conducive to anaerobic bacterial attack.

Normal surveillance inspections and tests performed to insure the pressure boundary capability provide an excellent means for identification of the existence of corrosion on the exposed side, prior to significant degradation taking place. As an example, ASME Code Section XI, Subsection IWE and Appendix J to Code of Federal Regulations 10 CFR Part 50 contain inspection requirements for liners and penetrations in accessible areas.

For corrosion on the unexposed side of the liner, additional surveillance and inspection programmes are needed. These additional measures should focus on those regions not normally accessible or inspected via established surveillance programmes.

Initial inspections should focus on whether the potential for aggressive groundwater exists at a level that could affect the liner or penetrations (see Section 4.4.1.4). If this is confirmed, further investigation as to the condition of the liner should be performed. The ageing management programme to be implemented will likely involve destructive examination, and should focus on inaccessible areas, in the most probable areas of degradation, particularly areas adjacent to the fluctuating groundwater levels. The exact programme would be specific to the site under consideration.

4.4.4.2. Elevated temperature

As noted in the discussion on reinforcing steel (Section 4.4.2.2.), when steels are subjected to elevated temperatures, reductions in yield strengths occur. The temperatures at which significant reductions in yield strength occur are above those associated with normal plant operation. The concrete to steel bond may be adversely affected by high temperature piping penetrations. Leaktightness inspections should include these locations.

4.4.4.3. Irradiation

As noted under the reinforcing steel section (Section 4.4.2.3.), steels can undergo changes in ductility and yield strength when subjected to irradiation. Concrete containment liner plates and penetrations are not expected to be exposed to cumulative levels of irradiation sufficient to initiate degradation. Cumulative fluence levels at the containment interior surface, considering 80 years of operation, are expected to be approximately 10^{14} n/cm^2, which is several orders of magnitude below levels at which degradation in steel is measurable.

4.4.5. Waterstop

Sealing elements in concrete joints make a significant contribution to the leaktightness of the containment boundary. While many sealing strategies are used, a common element in many of these strategies is the use of waterstops. Waterstops may be constructed of metal or elastomeric compounds; loss of bond or intimate contact between the waterstop and the concrete opens a path for increased leakage. Shrinkage of the concrete and deterioration of the waterstop contribute to this process. For waterstops, deterioration occurs by metallic corrosion discussed above and by shrinkage, loss of plasticizer, embrittlement and cracking of elastomeric materials. For example, PVC waterstop can shrink about 3% during the first 20 years of plant life.

Waterstop deterioration can be detected by a regular programme of leak rate testing. Deterioration can be mitigated by proper material specification and installation requirements. For example, proper joint preparation, proper splicing, proper location of the waterstop within the concrete element, proper concrete consolidation and the use of additional sealing barriers are effective means of mitigating this type of degradation.

4.4.6. Miscellaneous age related degradation

4.4.6.1. Fatigue

Fatigue can be a factor in the degradation of concrete and steel containments. It may be defined as progressive degradation produced by cyclic loadings which are less than the maximum allowable static loading. The occurrence of fatigue effects in concrete is highly dependent on the materials used, construction practices, thermal gradients, and the relative magnitude of stress change under load.

During static loading of steel components, stress concentrations are distributed in ductile materials by plastic flow. However, under repeated or cyclic loading, plastic deformation may occur on a microscopic scale and cause fatigue failure.

Cyclic loadings include:

- startup/shutdown cycles which cause thermal expansion stresses,
- leakage rate test pressurizations,
- seismic loads,
- crane loads,
- pipe reactions at penetrations.

Design standards provides fatigue curves which give the ratio of maximum stress to failure stress versus the number of cycles to failure. These standards indicate that, in terms of the ratio of applied stress to failure stress, the fatigue strength of plain concrete is essentially the same whether the mode of loading is in tension, compression or flexure.

Typically, design codes for concrete require that the applied stress will be less than 71% of the failure stress, assuming a load factor of 1.4 for dead loads. Based on the fatigue curves, 100 000 cycles of this stress are required to cause excessive cracking or spalling. Since the total number of actual cycles is far below this value, fatigue of concrete in the containment structure is not significant and would not cause cracking that could initiate other age related degradation mechanisms.

Fatigue of steel reinforcing bars is not a significant factor in reinforced concrete structures. Typically, stresses sustained by reinforcing steel are below 24 000 psi and are not cyclic. Thus, 24 000 psi represents a stress range, from zero to peak stress. Based on the fatigue curves for reinforcing bars and a stress range of 24 000 psi it can be estimated that fatigue of the reinforcement is of no concern until approximately one million load cycles have occurred.

Design of liners for concrete containments is not fatigue controlled since stress/strain cycles occur only a small number of times and produce only minor stress/strain fluctuations.

4.4.6.2. Settlement

All structures have a tendency to settle during construction and early life. The most pronounced settlement is sustained during the first months after construction. The amount of settlement depends on the physical properties of the foundation material, which may range from rock (with little or no settlement likely) to compacted soil (with some settlement expected).

Concrete and steel structural members can be affected by differential settlement between supporting foundations, within a building or between buildings. Severe settlement can cause misalignment of equipment, and could lead to overstress conditions within the structure.

TABLE VI. SUMMARY OF AGEING MECHANISMS OF CONCRETE CONTAINMENT STRUCTURES ACCORDING TO THEIR SIGNIFICANCE

NO SIGNIFICANT AGING DEGRADATION POTENTIAL	AGING MANAGED BY CURRENT EFFECTIVE SURVEILLANCE/TEST/INSPECTION PROGRAMS	ADDITIONAL MANAGEMENT TECHNIQUES REQUIRED
CONCRETE o Freeze-thaw o Leaching of calcium o Reactions with aggregates o Elevated temperature o Irradiation MISC. MECHANISM o Fatigue LINER o Elevated temperature o Irradiation REINFORCEMENT o Corrosion (above grade) o Elevated temperature o Irradiation PRESTRESSING SYSTEMS o Elevated temperature o Irradiation	MISCELLANEOUS MECHANISM o Settlement LINER o Corrosion (accessible areas only) PRESTRESSING SYSTEMS o Corrosion o Prestressing Losses	CONCRETE o Aggressive chemicals REINFORCEMENT o Corrosion (below grade, subject to aggressive groundwater LINER o Corrosion (inaccessible areas only) PRESTRESSING SYSTEMS

Settlement is allowed for in the initial design of the containment. Design considerations include whether the foundation is in a rock or compacted soil environment. For most plants, settlement is not expected to be significant, and the rate of settlement drops to a negligible level shortly following the construction of the structure. For these containments, settlement is not a significant ageing mechanism.

Certain containment sites (e.g. soft soil sites) are designed, acknowledging the potential for more significant settlement over time. For these sites, monitoring the progress of settlement over time is required. When this condition exists, monitoring programmes are put in place after the initial commissioning of the structure, providing positive feedback so that the design allowances are not exceeded. Such programmes are discontinued only after the settlement rates are well understood and known to fall within acceptable limits. Accordingly, although settlement can be a potentially significant degradation mechanism, the effects can be determined early in the life of the structure. The use of effective ongoing monitoring programmes can confirm that settlement is within acceptable limits.

4.4.7. Summary

Of the degradation or ageing mechanisms listed in Table IV, those potentially significant to concrete containment are listed in Table VI. Some can be mitigated through effective management; others are identified as requiring further management. These are listed in column 3 of the Table VI.

4.5. MONITORING OF CONCRETE CONTAINMENT BUILDING DEGRADATION

There is a variety of techniques for monitoring degradation of concrete containments. The extent to which these are used will vary from country to country. A summary of these techniques is provided below.

(i) Concrete

(a) Visual inspection for cracks, spalling, leaching and gel reaction products.

(b) Instrument aided inspections including built-in strain gauge detectors, and use of ultrasonic techniques to examine/probe any cracks observed visually.

(c) Test methods to determine presence of AAR if leaching or gel products are visible.

(ii) Liner

(a) Visual inspections for corrosion, attack by other chemicals, or cracks; (for steel liners) use of dye penetration testing if pressure test indicates leakage from the liner.

(b) Laboratory examination of cut-out samples from steel/metal liner.

(iii) Prestressing system

(a) Visual inspections, including checking anchors for damage, checking concrete under anchor for bearing failure, checking cables/tendons for corrosion, checking grease in tendon ducts.

(b) Measurement of tendon tension, checking of tendons for relaxation (measured value compared to design, as built, or last check values).

(iv) Building displacement and deformation

(a) Visual inspection identifies obvious gross settlement.

(b) Monitoring settlement by instrumentation includes the use of external strain gauges to check for continuing deformation/displacement or measurement of deformation using built-in targets.

(c) Checking of building expansion/deformation during pressure test against allowable design values is done by measuring strain of a wire anchored between two points.

(v) Penetration inspection/testing

(a) Visual inspections check for leaks, cracks, the physical condition and freedom of operation of bellows, and insulation/seals (for hardening, extrusion).

(b) Testing includes soap bubbles, pressure hold test and helium or Freon leak tests.

(vi) Overall containment pressure boundary

(a) Inspection, both visual and acoustic; check for hissing and cracking noises.

(b) Testing involves full fledged pressure hold test including measurements of pressure, temperature, humidity (moisture) both inside and outside the containment.

4.6. MITIGATION OF CONCRETE CONTAINMENT BUILDING DEGRADATION

The effects of ageing can be mitigated by fairly standard techniques which vary depending upon the nature of the problem. Some techniques are outlined below.

<u>Resin injection</u>

This method of structural repair is used for filling in and repairing cracks in concrete, and can be used for a whole range of cracks from the very small to fairly large sized ones. The objectives are to ensure that reinforcement steel is not exposed to the environment, to restore concrete to its original uncracked condition, and to preclude intrusion of water and potential leaching. The availability of air curing and very fast setting resins and glues of very high strength enables repair of a diversity of cracks in various environments. Resin can also be injected under high pressure.

<u>Coating on liners</u>

A whole range of coatings and paints are available. These may be either passive or active, and can retain plasticity/flexibility or form a thin tough shell, depending upon service requirement. They are particularly useful in filling small cracks while, at the same time, giving a cosmetic effect. They are also useful in protecting the containment from the internal or external environment.

<u>Crack sealing</u>

Where structural repair by epoxy injection is not necessary, a variety of techniques are available to seal cracks. These accomplish all the goals of epoxy injection except structural repair and allow subsequent movement of the crack.

<u>Repair of hatches/hatch seals</u>

Hatches are used routinely for moving equipment to/from containment. The rubber seals on the hatches are particularly susceptible to wear and tear and ageing. This can be mitigated by repair or replacement of the seals or even of the hatches. The use of fiberglass tape and fiber-glassing is particularly useful when the size of hatches or frequency of use makes repair uneconomical or infeasible.

<u>Retensionising of tendons</u>

Relaxation of tensioning cables can be repaired by retensioning. Cable ducts/holes should be greased. Individual cables (from a group) and damaged anchors can be replaced where permitted by the design.

Change of service conditions

Should circumstances so warrant, the (environmental) service conditions can often be changed or modified. This is particularly applicable to reducing the subsoil water table under foundations through use of continuous pumping, controlling humidity and/or temperature inside the containment. The use of coatings can be useful in changing (and maintaining) service conditions.

4.7. KNOWLEDGE AND TECHNOLOGY GAPS RELATING TO THE UNDERSTANDING AND MANAGEMENT OF CONCRETE CONTAINMENT AGEING

From the review provided above, the following areas requiring further work were identified as having the potential to produce practical results within the three year period of the planned co-ordinated research programme.

(1) Current experience and ageing management practices
(2) State-of-the-art repair techniques
(3) Crack mapping and depth measurements
(4) Condition indicators.

Further work addressing the above topics is described in Section 4.8 below.

4.8. STATEMENT OF WORK FOR CO-ORDINATED RESEARCH PROGRAMME ON MANAGEMENT OF AGEING OF CONCRETE CONTAINMENT BUILDING

4.8.1. Objectives

(1) To produce a summary of current ageing management practices and experiences for concrete containment structures, including:

- concrete constituent materials used
- in-service inspection and testing practices
- maintenance and minor repair practices
- major repair experience.

(2) To compile a state-of-the-art report on concrete repair techniques and materials specifically applicable to nuclear containment structures.

(3) To develop crack mapping and acceptance/repair guidelines applicable to nuclear containment structures, including:

- crack categorization based on width, depth, length, and type (structural or other)
- acceptance levels for cracks, considering exposure
- repair guidance, correlating crack category to repair method.

(4) To develop a set of practical condition indicators and associated guidelines for monitoring concrete containment ageing. The research will focus on current technology and programmes that can be implemented at operating nuclear plants.

4.8.2. Scope of work

This CRP will involve investigation of concrete containment buildings and their subcomponents: concrete, reinforcement, prestressing systems, structural steel and liner plate, penetration assemblies and waterstops. The work will consist of collection, categorization and evaluation of information relating to ageing management research and actual practices and experiences from Member States, and preparation of reports and guidelines.

The methodology for ageing management studies documented in Ref. [1] will be used by the CRP participants.

4.8.3. Tasks to be performed

The tasks to be performed are listed below for each of the objectives stated in Section 4.8.1. The work will be scheduled so that an evaluation of current experiences (objective (1)), is completed at an early stage to provide input to objectives (2), (3) and (4).

Task No.	Task description	Duration (months)
	(1) Current ageing management practices	
1.1	develop survey	3
1.2	solicit and collect information	4
1.3	evaluate survey information	3
1.4	produce draft documentary report	2
1.5	finalize and issue report	3
	(2) State-of-the-art repair techniques	
2.1	literature search	7
2.2	review information for applicability	3
2.3	develop guidelines document	6
2.4	finalize and issue report	4
	(3) Crack mapping and acceptance/repair guide	
3.1	develop survey	2
3.2	solicit and collect information	3
3.3	review and categorize information	5
3.4	develop criteria document	9
3.5	evaluate and review document	3
3.6	finalize and issue report	3
	(4) Condition indicators	
4.1	develop survey	2
4.2	solicit and collect information	3
4.3	review and evaluate information	5

Task No.	Task description	Duration (months)
4.4	develop condition indicator document	9
4.5	evaluate and review document	3
4.6	finalize and issue report	3

4.8.4. CRP network

Organizations from Canada, France, India, the United Kingdom and the USA have indicated their interest in this CRP. Other Member State organizations will be invited to participate. Actual participants will be organized in a CRP network to facilitate co-operative work.

5. PILOT STUDY ON THE MANAGEMENT OF AGEING OF IN-CONTAINMENT INSTRUMENTATION AND CONTROL CABLES

This pilot study considers the management of ageing of in-containment, low voltage (<1000 V) instrumentation and control (I&C) cables that must be qualified for a longterm service in harsh environments.

The following sections present a summary of current understanding, monitoring and mitigation of ageing of I&C cables as a basis for a co-ordinated research that would be of most international benefit and use.

5.1. DESCRIPTION OF TYPICAL I&C CABLES

5.1.1. Purpose and applications of I&C cables

Electrical cables are vital components of NPP instrumentation and control systems as they are the connecting elements between other system components (e.g. field transducers measuring important plant parameters and I&C equipment used to monitor and control these parameters) located both inside and outside containment. Safety related I&C cables must be qualified to perform their functions under specified service conditions, including design basis event (DBE) and post-DBE conditions.

There are many cable products made by different manufacturers for different applications and use in different ambient environment. According to application in-containment, low voltage (<1000 V) electrical cables can be divided into three groups:

- control cables used in low current circuits for applications such as position, pressure and level switches or valve actuators;
- instrumentation cables used in low current circuits for applications such as electronic transmitters, resistance temperature detectors or radiation monitors;
- low voltage power cables used for applications such as cooling fans, motor operated valves, solenoid operated valves and recombiners.

This pilot study is applicable also to low voltage power cables since they are essentially of the same construction as the control cables.

5.1.2. Types of cables

A wide variety of cables is used in operating nuclear power plants. Different types of cables include:

- single conductor
- multi-conductor
- shielded and unshielded
- coaxial or triaxial
- jacketed cable or armored cable
- cables manufactured with filter materials or without filter materials.

5.1.3. Types of insulating materials

Insulating materials in service today also cover a wide spectrum. In addition, cables often use different materials for insulation and jacket. Typical insulating materials are listed below:

- XLPE (cross-linked polyethylene)
- EVA (ethylene vinyl acetate)
- EPR (ethylene propylene rubbers)
- SiR (silicone rubber)
- PPO (polyphenylene oxide)
- ETFE (ethylene tetrafluoroethylene)
- PEEK (polyether-ether ketone)
- PVC (polyvinyl chloride)
- CSPE (chlorosulfonated polyethylene).

These materials are widely used as the basic material, but their ageing behaviour may differ significantly due to minor changes in chemical additives and compounds used in the production process. These changes, although present in quantities of the order of several ppm, may have a strong impact on the cable behaviour.

5.1.4. Functional requirements

There are three key functional requirements of a typical cable system:

- Voltage withstand capability under normal operating, test, accident, and post-accident conditions.
- Current carrying capacity as required for normal operating, test, or accident conditions.
- Must not interfere with the operations and functions of other devices in the vicinity.

Depending on specific cable application, its functional requirements may have to be satisfied for a period of a few hours up to a year after the onset of an accident such as a LOCA.

5.1.5. Replacement of cables

Replacement of the cabling system or its parts is a complicated task that would require a long outage. Safety aspects of a large scale cable replacement programme include the requirement to maintain vital safety functions during the replacement programme. Selection of suitable cable materials, routing of cables through non-critical areas and an adequate replacement strategy for cables in critical areas are therefore important tasks for design engineers. These tasks should be carried out in collaboration with plant operation, maintenance and construction staff.

5.2. SERVICE CONDITIONS

Service conditions are important factors that influence ageing of cables. Significant service conditions include:

- ambient temperature
- dose rate and total dose of ionizing radiation (gamma and neutron radiations)
- air or inert gas
- moisture, etc.

It should be recognized that the real service conditions involve synergistic effect of one or more of the above environments. The aforementioned conditions and operating history data should be collected, recorded and used as a basis for the management of cable ageing [2].

Comprehensive description of NPP service conditions relevant to ageing of cables is given in Ref. [3].

5.3. APPLICABLE STANDARDS AND REGULATIONS

Many standards exist which address some of the ageing concerns (e.g. IEEE 323, IEC 780, IEC 544-3, IEC 216). However, no existing standard or regulation provides comprehensive guidance on the management of age-related degradation of electrical cables. Such a document would be a valuable addition to existing procedures or guidelines.

5.4. POTENTIALLY SIGNIFICANT AGEING MECHANISMS

The function of I&C cables is to transmit low voltage power or signals. Ageing degradation of cables may reduce their capability to supply power to position, pressure and level switches and actuators or to transmit signals from sensors with sufficient accuracy. Reduced insulation resistance, high dissipation or loss factor, open circuits or short circuits can all result in a loss of cable function.

The cable function is primarily dependent on the stability of its polymeric components. It is therefore important to understand ageing degradation of these materials. Of the degradation mechanisms affecting the polymers used in cables in NPP operational environments, the most significant ones are thermal and radiation ageing. They are described in detail in Ref. [3]. It should be noted that although electrical characteristics (e.g. insulation resistance, power factor or loss factor) may change by several orders of magnitude, these changes may not be significant in terms of the function of the cable in the circuit, provided that the minimum margin required for performance in adverse environments is maintained.

Both radiation and thermal degradation can change the molecular structure of the polymers used in cables and induce changes in both mechanical and electrical properties of jacket and insulation. Dose rate and activation energy are therefore the dominating parameters for the evaluation of cable degradation. Experience shows that the dose rate effects and activation energy are strongly dependent upon the materials used in the cable. Moreover, the presence of oxygen is one of the main causes of the observed dose rate effect. More detailed explanation of this effect is given in Ref. [4].

5.5. COMPONENT FAILURE HISTORY

Evaluation of existing databases has shown the difficulty of identifying age related failures [5]. However, deterioration of cables in the containment building reported in Ref. [6] clearly shows the impact of combined radiation and temperature effects.

5.6. AGEING MANAGEMENT TECHNIQUES

Due to the lack of practical non-destructive monitoring techniques, preventive maintenance involving replacement of cables or parts of them exposed to severe environments is currently the most common ageing management technique. Potentially useful monitoring methods based on electrical, mechanical and chemical techniques are currently being evaluated. Alternative ways of ageing management have been elaborated and tested in several countries (see Appendix III, D.2). These strategies for the evaluation of long term resistance of cables in harsh environments involve testing of:

- cable deposits (used in newer NPPs), and
- cable specimens removed from real positions in an NPP.

Other ageing management techniques have been used and should be considered in the design of new facilities. These include the routing of cables through areas of less severe environment and placing shielding between cables and radiation sources.

5.7. KNOWLEDGE AND TECHNOLOGY GAPS RELATING TO THE UNDERSTANDING AND MANAGEMENT OF I&C CABLES AGEING

Knowledge and technology gaps relating to significant service conditions that can influence cable ageing (see Section 5.2) are identified below:

(i) The effects of temperature on cables are well understood and exhaustive literature on the subject exists. Care must be taken with the use of activation energies which can vary significantly within a cable. Practical in-situ monitoring methods for thermal degradation are not currently available.

(ii) Cables specifically designed for radiation environment that face less than 30 kGy (3 Mrad) of total dose during the expected lifetime are not subject to large changes in electrical properties (see Appendix III, D.2). Lifetime assessment is necessary with higher integrated doses.

(iii) Air or inert gas: The presence of oxygen in a radiation environment is known to have a significant effect on the degradation of most polymers. Dose rate effects are normally larger in air than in an inert atmosphere, however, these effects are not fully understood and quantified.

(iv) Moisture: The effects of moisture on cable ageing has been investigated and found not to significantly affect the degradation mechanisms, see Refs [5], [7] and [8].

The planned co-ordinated research programme should address the following issues: Synergistic effects of temperature and radiation are recognized as being an important factor in the degradation of cables. It is important to determine the degradation that occurs in real plant environment and compare this with the predictions and models developed from accelerated tests.

Once a theoretical model to predict residual life is validated through the use of real time plant data, a means of assessing the status of predictive parameters in-situ should be developed.

Note regarding end-point-of-life criterion:

As far as international practice is concerned, 50% elongation at break figures are used as a comparative criterion to measure the specific changes in properties. This criterion is used in two ways:

(a) 50% elongation at break as a relative measure
(b) 50% elongation at break as an absolute measure.

For the consideration as a criterion for end-point-of-life, experience shows that the real end-point-of-life can be found between 20 and 30% of the absolute elongation at break depending on cable materials. On this basis an acceptable margin is incorporated in the procedures.

5.8. STATEMENT OF WORK FOR CO-ORDINATED RESEARCH PROGRAMME ON MANAGEMENT OF AGEING OF IN-CONTAINMENT I&C CABLES

5.8.1. Objectives

(1) To validate predictive cable ageing models accounting for synergistic effects that take place when radiation and thermal ageing of cables occurs over a long time-span in real plant in-containment environment.

(2) To provide practical guidelines and procedures for assessing and managing ageing of I&C cables in a real plant in-containment environment.

5.8.2. Scope of work

This CRP will investigate in-containment, low voltage (i.e. less than 1000 V) single and multiple conductor electrical cables used in instrumentation and control circuits of systems and equipment important to safety; because of their similarity low voltage power cables may be also included.

The materials to be investigated are limited, for practical reasons, to three types of typical polymers: XLPE, EPR and EVA.

The scope of work includes cable to cable splices but excludes terminations.

The work programme is based on collaborative work with the distinct tasks distributed between the participants.

5.8.3. Tasks to be performed

Task No.	Task Description
	(1) Predictive cable ageing models
1.1	Obtain real time naturally aged cables from NPPs and measure their mechanical and electrical properties.
1.2	Obtain reference cables of the same material which have not been subjected to the ageing environment. Measure their mechanical and electrical properties.
1.3	Obtain data on actual environmental conditions which pertain to the aged cable samples received, i.e. temperature, dose rate, total dose, humidity, etc.
1.4	Carry out sequential and combined environment tests on the reference cable materials in appropriate environments to supplement available data and for comparison with the naturally aged samples.
1.5	Assess all of the data from tasks 1.1 and 1.4.
1.6	Attempt to validate the available predictive models using collected data.

Note: Tasks 1.1 and 1.3 are expected to be completed in the first year, and tasks 1.4 and 1.5 in the second year of the CRP.

Task No.	Task Description
	(2) Guidelines for assessing and managing ageing of I&C cable
2.1	Collect published data on three materials identified in Section 5.8.2.
2.2	Solicit, through the IAEA, unpublished data and operating experiences on these materials from all Member States.
2.3	Assess the collected data with respect to models validated in the task 1.6.
2.4	Develop guidelines on the assessment and management of ageing in low voltage cables using the experience gained in the CRP.

Note: Tasks 2.1 and 2.2 are expected to be completed in the first year, and task 2.3 in the second year of the CRP. Task 2.4 will involve all of the participants with the target completion in the third year of the CRP.

5.8.4. CRP network

Organizations from France, Germany, the Republic of Korea, the United Kingdom, and the USA have indicated their interest to participate in the CRP. An invitation will be extended to other interested parties. Actual participants will be organized in a CRP network to facilitate co-operative work.

APPENDIX I

ACTIVITIES IN PARTICIPANTS' COUNTRIES RELATING TO MANAGEMENT OF AGEING OF MOTOR OPERATED ISOLATING VALVES

This appendix outlines some of the existing and planned activities and areas of interest relative to knowledge and technology gaps identified in Section 3.7. Presented information and the views expressed are based on the personal knowledge and understanding of the participants in the Working Group 2 on the MOV pilot study, and do not necessarily reflect those of their governments or parent organizations.

I.1. PAKISTAN

Activities relative to knowledge and technology gaps

The Karachi Nuclear Power Plant (KANUPP), one of the earliest CANDU type plants, has a total capacity of 137 MW(e). The plant went into operation in early 1972 and is of the same age group as that of the Douglas Point NPP in Canada (decommissioned 1984).

The operating and maintenance practices at KANUPP find their origin largely in the old Canadian practices and it has not been possible to review these practices and bring them up-to-date. However, the operating and maintenance practices are by and large serving their purpose notwithstanding the room for further services and improvement. Recently, a Technical Information Management (TIM) system has been initiated giving a summary of the various failures and malfunctions observed on particular equipment having safety implications, including MOVs. It also gives a record of the maintenance on each particular equipment and provides a guideline to ageing predictors.

Areas of interest relative to technology gaps

A surveillance and monitoring system which could detect ageing degradation or malfunction of MOVs and other safety equipment would be of particular interest. Such a system would possibly allow early detection of the malfunction and a need for necessary corrective actions before the actual breakdown of the system/equipment.

Another area of particular interest would be the preparation of guidelines for improvement in maintenance procedures for better ageing management. We are in the process of reviewing existing maintenance procedures and developing new ones where none have been available. A guideline would be of immense help to use in such a review and development of improved maintenance procedures.

I.2. CANADA

Activities relative to knowledge and technology gaps

(1) Database

- Candu Owners Group (COG) conducts a review function of all significant event reports generated by Candu stations in Canada and other countries.

- INPO's database of significant event reports is reviewed by COG and the Atomic Energy Control Board's (AECB) Event Analysis Group to identify events of interest to Candu owners for possible action.

- No specific review is performed of events that involve safety significant MOVs failure or malfunction.

(2) Surveillance and monitoring

- Testing of MOVs that perform a safety function is carried out at all Candu plants at pre determined intervals in order to confirm that system reliability targets are met. This testing involve leak tightness checks and stroke timing, and is carried out "on-line".

- The use of current measurement for signature analysis of MOVs is under development as an element of improved maintenance programmes currently under development.

(3) Time dependent models for risk and reliability assessment

As part of the Operational Safety Management Programme, Canadian utilities conduct on-going reliability based assessment of performance of safety related systems and components (including MOVs). Technical groups at the stations monitor test records, control room logs, work and deficiency reports to maintain comprehensive safety records. This information is used to update component failure rate data and to predict future safety system unavailability. The participant is not aware of any work that may be underway to develop mathematical models for reliability assessment of ageing MOVs.

(4) Qualifications/requalifications programme

Canadian plants that have been in operation for more than 10 years had no 'formal' component qualification programme. More modern plants have such a programme which covers the full spectrum of normal and accident environmental conditions. Ontario Hydro has recently prepared a comprehensive environmental qualifications programme for older NPPs which will be used in the upgrading and backfitting of safety related components inside containment. However, this programme is specifically designed for 'harsh environment' qualification and does not cover mechanical/operational qualification parameters.

(5) Maintenance procedures

Canadian nuclear utilities are developing predictive maintenance programmes that use diagnostic techniques and component reliability assessments as planning tools. At some of Ontario Hydro plants, MOV diagnostic tools are being used to assess the condition of and to 'set-up' safety related valves as part of the scheduled maintenance programme.

Areas of interest relative to technology gaps

It is thought that the AECB and Ontario Hydro are interested in co-operation with other interested parties in all 5 areas, particularly the areas 2, 4 and 5. Initially, however, such co-operation would likely be limited to information exchange.

The AECB has recently written to NPP licensees in Canada to request information on "ageing management" in four following areas:

- the assessment of the continued validity of steady state and dynamic analysis of the plant;

- the scope of review of degradation mechanism that could impact significantly on safety;

- assessment of the continued validity of reliability assessments of special safety systems, safety support and safety related systems; and

- the adequacy of the planned maintenance programme with respect to age related reliability.

I.3. FINLAND

Activities relative to knowledge and technology gaps

(1) Work done and planned at Technical Research Centre of Finland

At the Technical Research Centre of Finland, there is an ongoing research programme on structural integrity of NPPs, which is funded by the ministry of trade and industry. In this programme, the project "Reliability assessment of maintenance in nuclear power plants" is focused on developing models and methods for ageing assessment of components and piping, and on the development of maintenance and inspection strategies using reliability engineering methods. The interest area of Technical Research Centre is thus primarily in risk and reliability assessments. As the modeling of component behavior requires data of high quality, the development of data bases for MOV failures is also of interest.

During 1989-1990, an analysis of MOV operating experiences in Finnish BWR plants was performed in order to identify major contributors to failures and to reveal possible time-dependencies in failure occurrence. The results of the study can be further used to evaluate the suitability and effectiveness of maintenance and testing programmes.

(2) Valve backfitting project at Loviisa PWR plant

The utility has started in 1988 a five year project to qualify the electric actuators as accident classified. Tailor-made qualification programs are developed at Loviisa nuclear power plant because both the accident and post-accident conditions of the power plant differ from those at other PWR plants. The utility is also developing a system of how to be aware of the state of the LOCA-qualified valve and actuator during the operational period (30 years).

I.4. FRANCE

Activities relative to knowledge and technology gaps

Comments on the proposed programme to fill the knowledge and technology gaps by Mr. Campan (Commisariat à l'énergie atomique (CEA)/Cadarache).

(1) Several databases exist but are owned by the utilities. They can be available for specific identified use such as MOV ageing data. Nevertheless, it is important to establish an international format in order to allow international cross-checking.

(2) Monitoring techniques are very important to predict valve problems. Utilities will probably use them if a benefit is demonstrated in maintenance improvement. Work is performed in France using mainly the signal of the actuator power.

(3) PRA is a very adequate methodology to evaluate changes in safety level associated with evolution of component performance. As far as the data are existing, it could be a good way to evaluate the impact of MOV ageing on the general safety of the system.

(4) A national regulation for electric material qualification is existing (RCC-E). For mechanical qualification a big test programme is under way in France, by Electricité de France (EDF), and a regulation will be written later on.

(5) EDF has a maintenance procedure permanently updated according to the results obtained from all the plants of the same standard. Up to now ageing is not very much involved.

For activities mentioned above, there is an interest in France for information exchanges if utilities and manufacturers agree to participate.

Comments on the proposed programme to fill the knowledge and technology gaps by Mr. Delêtre of Commisariat à l'énergie atomique, Institut de protection et de Sûreté nucléaire (CEA/IPSN).

(1) Database

IPSN does not have access to the overall EDF database, but only to safety significant events. It is therefore possible to access data only with EDF authorization. The EDF data base is not specifically oriented to ageing problem detection and measurement. Therefore, it is important to develop a common format for data records which contains useful information for ageing management.

(2) Surveillance and monitoring system

The studies in this framework are not managed by IPSN.

(3) Failure models for risk and reliability assessment.

In this field we have developed a failure modes and effects analysis (FMEA) study on discharge valves (safety valves SEBIM) with EDF authorization and constructor information, with the objective of defining critical components and anticipated ageing phenomena. We hope to be able to use a PRA study which has been developed by IPSN for the 900 MW(e) PWR in order to evaluate the impact of equipment ageing on the reliability of safety systems.

(4) Mechanical and electrical qualification

We have specification for electrical components in the French national code RCC-E, and at present, EDF is developing a general specification for mechanical equipment qualification. Qualification involves tests and analysis. For tests, general and particular specifications exist. In the field of qualification by comparison, specifications are still in preparation. We would be interested in a guideline for MOVs qualification.

(5) Maintenance procedures

Generally maintenance procedures are good but not systematically oriented to ageing problem detection. In this way we think that a guideline could be developed to be sure that procedures take into account ageing phenomena.

I.5. RUSSIA

Proposal for the programme on motor operated isolating valves

The research programme on ageing process of NPP components (valves) should be based, on the one hand, on studying and predicting physical processes that determine ageing of NPP components, and on the other hand, on operating experience relevant to those components. In this respect, a special surveillance programme is proposed to obtain information needed to calculate quantitative probabilistic indicators of life-time of an NPP component, based on operating experience.

This surveillance programme should be determined by a mathematical model used in the calculations (developments of the aforementioned models should be taken into consideration in the surveillance programme). Such mathematical models should be probabilistic because this reflects the nature of ageing processes. Also these models should take into account accepted (proven) practice of maintenance and repairs and the real possibility of obtaining required information from NPPs.

Russia, particularly the Nuclear Information Center of the Research Institute for NPP Operation, could develop the probabilistic models (outlined in V. S. Emeljanov's paper) and relevant computer programmes (software) to define reliability indicators for NPP hardware. Moreover, Russia would be able to present relevant operating experience data obtained from its NPPs.

With regards to MOV pilot study, it would be useful to choose one of the primary circuit valves, whose lifetime characteristics can significantly influence reactor safety (e.g. T33-main closing valve 9).

I.6. SWITZERLAND

Activities relative to knowledge and technology gaps

Switzerland is in the specific situation to have reactors designed and built by three different manufacturers. Therefore, a pragmatic way was chosen to fill knowledge and technology gaps developing during the ageing period of nuclear power plants and components.

(1) Database on failures and malfunctions of MOVs

Switzerland shares the data base developed and maintained by VGB Germany, which is distributed to UNIPEDE, INPO and WANO. These databanks are not specific enough to provide details about MOV failures and malfunctions to use the information, for instance, as an input to PRA evaluation.

(2) Surveillance and monitoring systems

No surveillance and monitoring systems have been installed or are in use for the time being to provide information about the current status of MOVs ageing.

(3) MOV models for risk and reliability assessment

No MOV models have been developed for risk and reliability assessments of individual valves. Such models are in use to perform sensitivity analyses only. The data base to perform PRA evaluations to support decisions on risk or reliability of valve operations is considered not detailed enough, especially for operation under accident conditions.

(4) Mechanical and electrical qualification methods

Electrical qualification methods have been adopted from the requirements of the countries of origin of the reactor plus specific requirements defined by the authorities and the purchaser of the MOVs. The electrical qualification requirements may be revised, if better knowledge, for instance, about ageing under radiation and temperature is available. A mechanical qualification for safety relief valves was applied. To a certain extent, MOVs have been mechanically qualified by demonstrating their ability to open or close under maximum loads. For aged MOVs no requirements exist to perform mechanical requalifications after a certain time of operation.

(5) Maintenance procedures

MOVs and especially motor operators follow a written maintenance procedure which was agreed upon by the authorities and the utilities. Maximum inspection intervals have been defined therein. A tendency exists to check the correct setting of torque and limit switches and the actual motor power or torque in a more frequent manner and to compare these data to get early information about ageing deficiencies, if any. Plant licensees are required to demonstrate reliability of safety related valves to open or close without damage or sticking under maximum load conditions developed by the installed actuator.

I.7. UNITED KINGDOM

Activities relative to knowledge and technology gaps

(1) MOV database

Current: Little reliability data available specifically on safety-related nuclear MOVs. Systematic collection of MOV reliability data by NPP operators has recently commenced. SRS databank likely to contain data on reliability of MOVs in other applications. Data have been contributed to EUREDATA pilot study on a reliability database for mechanical equipment which focussed on valves.

Future: Possibility of a research programme involving collection and analysis of nuclear MOV reliability data, dependent on funding and access to data.

(2) MOV Monitoring

Current: No continuous monitoring of MOV performance other than whether they are shut or open. Periodic tests of gas circuit valve operation are undertaken.

Future: Possibility of a research programme involving development of equipment for measuring aspects of MOV operation, and its use in tests of selected valves in-service, dependent on perceived benefits.

(3) Time dependent PRA

Current: PRA is undertaken which includes MOV reliability data, but the PRA is not time dependent - it is assumed that the reliability data covers all failure modes including those that are age-related. Effects of time dependencies with regard to maintenance intervals have been evaluated.

Future: Possibility of a research programme to establish the feasibility of obtaining age-related MOV reliability data, dependent on funding and access to data.

(4) Equipment qualification (EQ)

Current: Limited EQ of MOVs fitted to older plants was undertaken before operation. Comprehensive EQ and accelerated life testing of MOVs for the new Sizewell B PWR is being undertaken.

Future: Possibility of EQ tests on aged MOVs removed from operating or shutdown plants, dependent on funding and perceived benefits.

(5) Maintenance

Current: Maintenance procedures are reviewed to ensure that MOV reliability remains adequate.

Future: The results of any research carried out under items (1), (2) or (3) above would be considered with regard to the need for changes to maintenance procedures.

I.8. UNITED STATES OF AMERICA

Activities relative to knowledge and technology gaps

(1) Database

There are three databases maintained in the USA on component/system failures: NPRDS plant database which collects extensive plant component failure data from the utilities; LER (Licenses Event Report) database which is limited to safety significant failure data; and NPE (Nuclear Power Experience) reports available from a US consulting firm. There are limitations and some problems with all three databases, e.g., incomplete and insufficient descriptions of failure, cause and effects analyses and corrective actions.

(2) Monitoring system

Oak Ridge National Laboratory under the sponsorship of the Nuclear Regulatory Commission (NRC), has recorded MOV electric currents and analyzed the time traces to show various types of valve degradations. Diagnostic test systems have come a long way in the past few years. The NRC recently co-operated with the MOV User's Group (MUG) in diagnostic system validation testing. Each diagnostic vendor tested his system against a standard. None of the systems came away from the tests without problems (as identified in the MUG preliminary report). To a degree all of them depend on indirect measurements to determine the performance of a valve operator assembly. None of the systems make enough measurements to determine whether the indirect measurements are proportional to each other. Those making actuator only measurements (on the actuator) are often mislead by the stem nut performance.

(3) Ageing models for PRA risk analysis

The NRC has an active programme with Bill Vessely of SAIC to develop mathematical models of ageing components and determine the risk significance.

(4) Mechanical qualification

Mechanical qualification tests of motor operated gate valves simulating closure accident conditions were conducted by NRC in 1988 in the Wyle Laboratories pressurized water loop and repeated in 1989 for a BWR steam line accident blown down in the KWU Karlstein loop. In both instances the valves failed to close fully (or isolate) and would have required a stem thrust and motor greater than analysis would have predicted. On disassembly of the test valves, the seats were found to be badly galled. The NRC also removed several valves from the Shippingport facility, one of which was tested at the Kernforschungszentrum Karlsruhe's (KfK) Heissdampfreaktor (HDR) facility near Frankfurt. The work is reported in NUREG/CR-4877. The NRC qualification process for MOVs has allowed the use of manufacturer flow loop tests and analysis to predict performance on blowdown. Recently, the ASME standards group responsible for valve qualification formed a new working group to bring the results of NRC and other work into the valve qualification standard. Mechanical qualification of MOVs must now be demonstrated by the utilities to assure closure.

(5) Maintenance

The NRC inspections have reported numerous instances of motor burnout and MOV failures due to mis-adjusted limit and torque switches. Licensees have been requested to make appropriate improvements in maintenance.

Areas of interest relative to technology gaps

The NRC has active research programmes in all five technology gap areas. The information and research results and future developments could probably be shared in a co-operative programme with the qualifier that the NPRDS database is maintained by the US utilities and their permission and agreement would be necessary. Participation in an IAEA co-operative programme would be dependent upon available funding and approval.

APPENDIX II

DESIGN EVOLUTION OF CONCRETE CONTAINMENT BUILDINGS

This appendix provides information on the concrete containment design evolution in the USA, France and Canada.

II.1. USA - PWR CONTAINMENT STRUCTURES

The containment consists of those elements required to maintain the integrity of the pressure boundary following a postulated design basis event.

Prior to 1965, containments for plants between 50 MW(e) and 400 MW(e) consisted of steel vessels, either free-standing steel cylinders with hemispherical domes and elliptical or flat bottoms (Figs 6 and 7), or steel spheres (Fig. 8).

As plant size increased to around 800 MW(e), the increased metal thickness and the need for post-weld heat treatment began to influence the design of steel containments. Thus, in the mid-1960s, some designs changed to composite, steel lined, reinforced concrete containments. These typically consisted of 10 ft thick base mats, 4 ft-6 in[4]. thick cylindrical walls and 2 ft-6 in. to 3 ft-6 in. thick hemispherical domes, with pressure retaining capability provided by a steel liner of varying thickness (1/4 to 1/2 in.). Concrete design strength varied between 3000 and 5000 psi with reinforcing steel having yield strengths of 40 000, 50 000 and 60 000 psi. (See Figs 9 and 10.)

Prestressed containments came into being in the late 1960s. Two of these (Ginna and Robinson 2) were prestressed in only the vertical direction. Fully prestressed containments developed in three phases; these are depicted in Fig. 11.

II.2. FRANCE - PWR CONTAINMENT STRUCTURES

II.2.1. EDF-900 MW(e) nuclear power Plants

The reactor building (Fig. 12) is designed:

- to contain the radioactive fluids and fission products which may result from a postulated internal accident, and

- to withstand internal pressurization from high energy pipe breaks within it.

It consists of an external structure, the containment, and internal structures.

[4] 1 ft = 3.048×10^{-1} m;
1 in = 2.54×10^{1} mm.

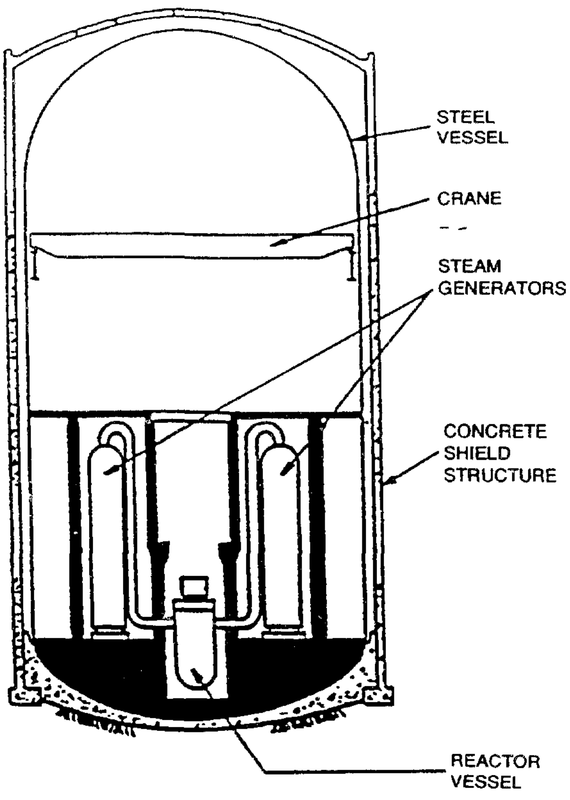

FIG. 6. Free-standing PWR steel containment with elliptical bottom.

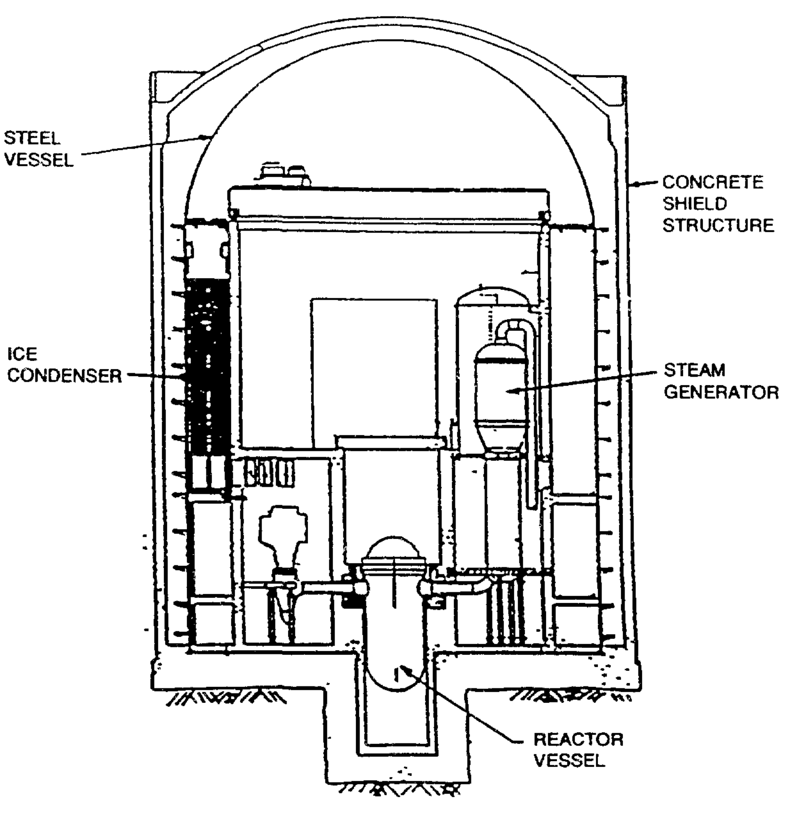

FIG. 7. Free-standing PWR steel containment with flat bottom and an ice condenser.

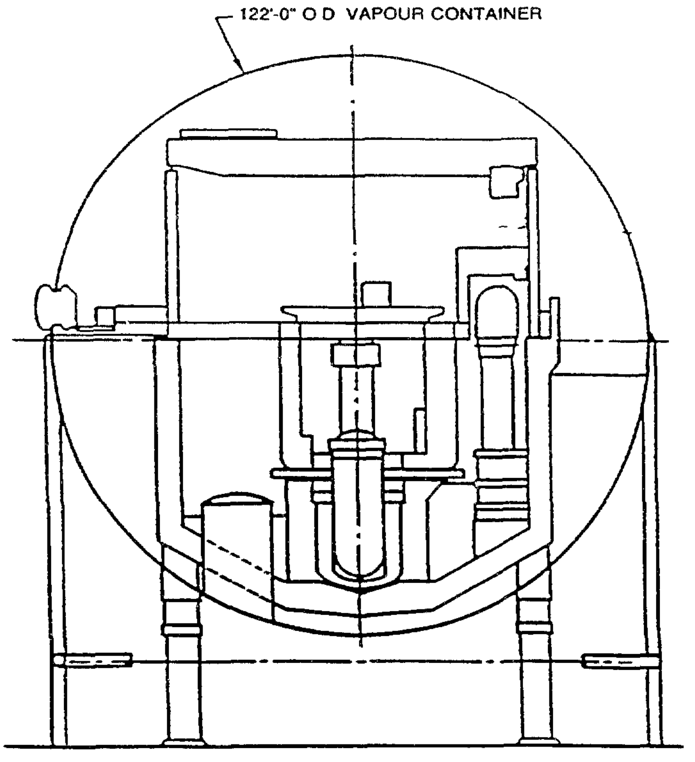

FIG. 8. Spherical free-standing PWR steel containment.

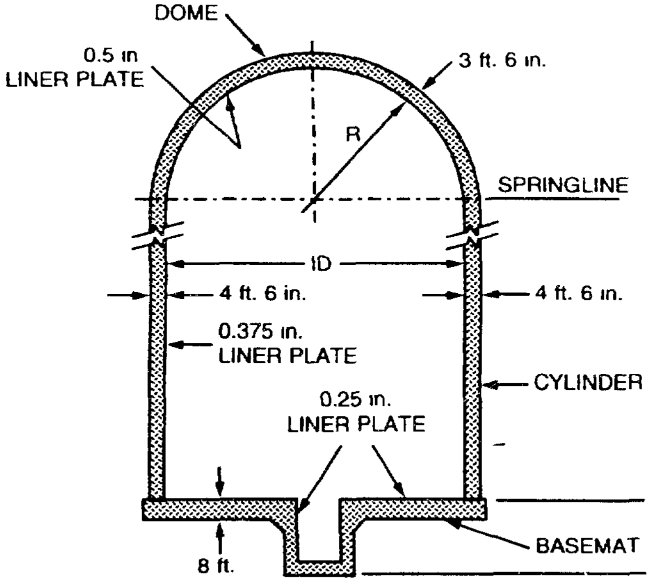

FIG. 9. Typical steel lined reinforced concrete containment.

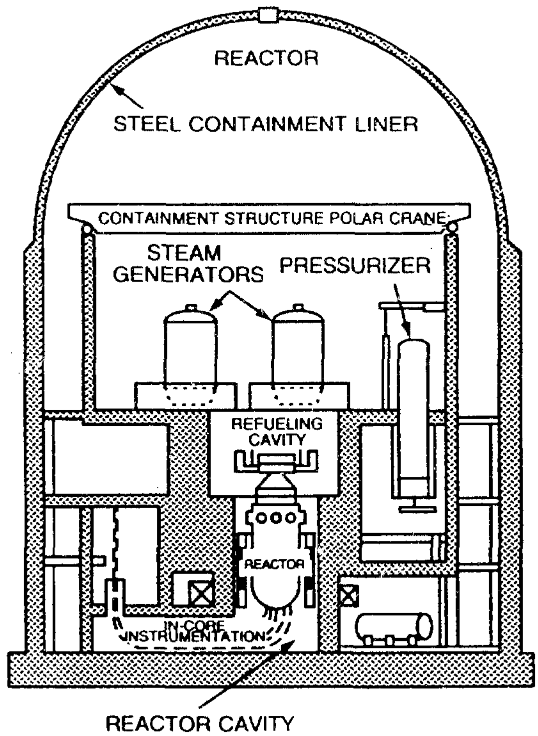

FIG. 10. Typical reinforced concrete containment.

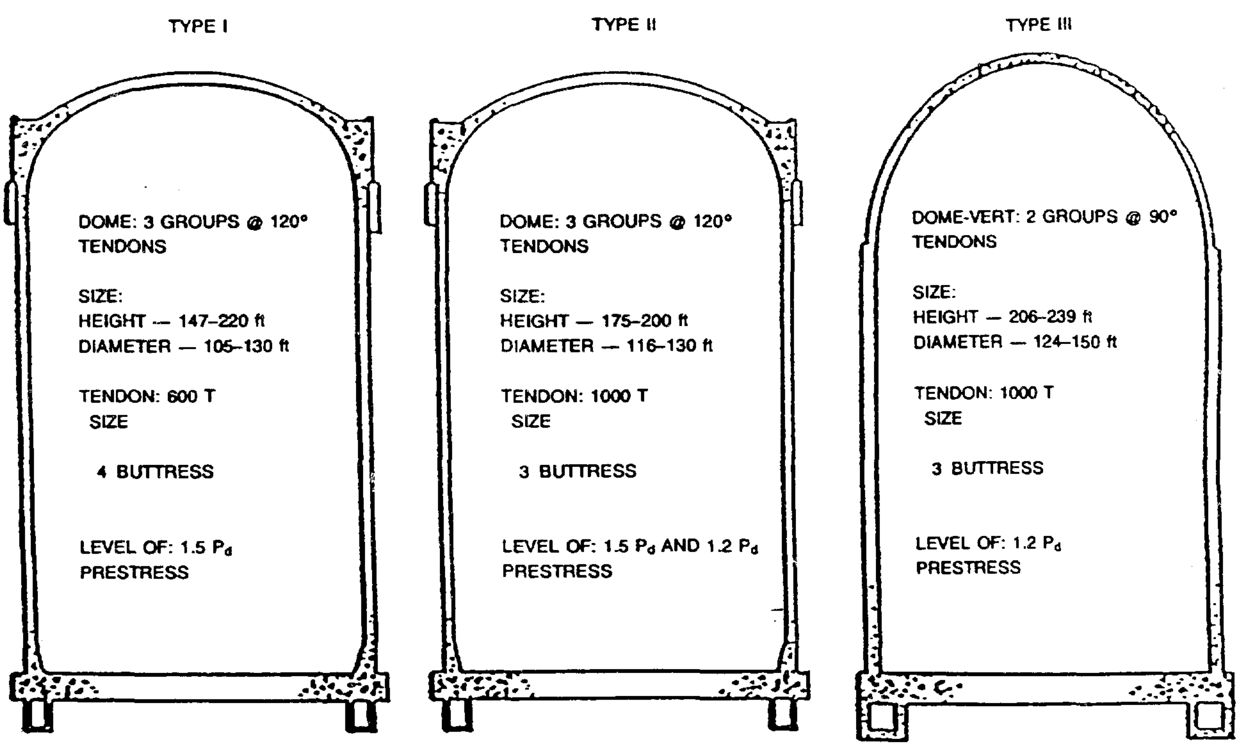

FIG. 11. Generic types of prestressed concrete containments.

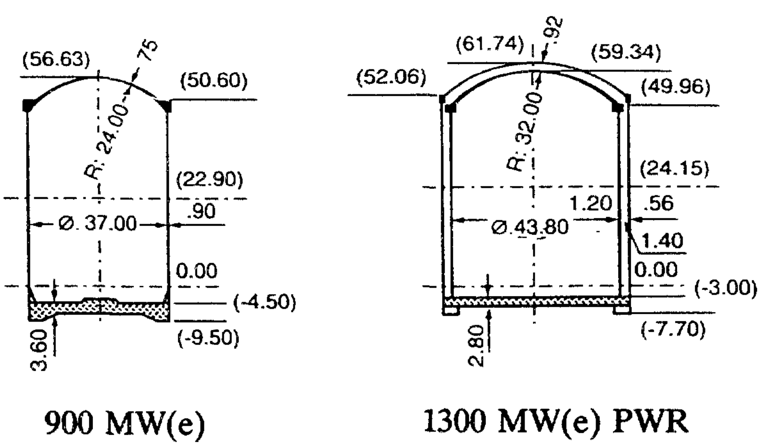

FIG. 12. Standard EDF concrete containments.

Containment

The containment is a cylindrical wall closed at the top by a spherical shaped dome and at the bottom by a slab (independent with classical foundation, part of the common nuclear island bottom slab in case of a seismic bearing pads). The inside height at the center line is approximately 60 m and the inside diameter 37 m; the prestressed concrete thickness is 0.9 m for the wall and 0.8 m for the dome.

The leak tightness is ensured by a 6 mm steel liner fixed to the concrete wall by means of mild steel anchors.

As the containment is important to safety, tests are performed on the materials (reinforcement, concrete, liner) and on the technique (prestress test). Before commissioning, a test is performed at 1.15 times the design pressure (5.6 bar)[5] and then periodic tests are performed at the design pressure (5 bar) during the first shutdown for refueling and every 10 years. For these tests, the leak rate must be less than 0.3% per day of the total mass of gas inside the containment. Furthermore, the containment stresses and temperatures are continuously monitored by audioindicators and thermocouples inside the containment.

II.2.2. EDF-1300 MW(e) nuclear power plants

The containment of a PWR 1300 MW(e) unit is a double wall structure (Fig. 12).

The inner containment, in prestressed concrete, is designed to withstand pressure and to ensure leak tightness. It consists of a cylindrical wall of about 50 m in height, the inside diameter being 43.8 m, topped with a semi-spherical shaped dome with ring girder. The thickness is 1.20 m for the wall and 0.92 m for the dome. The inside height at the center line is 61.1 m and the total internal volume is approximately 71 500 m^3 (giving a free internal volume of 68 000 m^3). At level +42.25, a concrete corbel supports the polar crane (with a trolley of 205 t and 35 t plus one more 195 t trolley during the construction period).

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[5] 1 bar = 1.00 x 10^5 Pa.

The outer containment, of reinforced concrete, is designed to resist external impacts. It consists of a cylindrical wall of about 51 m in height, the inside diameter being 49.8 m, topped with a semi-spherical shaped dome. The thickness is 0.55 m for the wall and 0.40 m for the dome.

The space between these containments is kept slightly below atmospheric pressure (15 mbar) and any leakage through the inner wall is then collected and monitored before release. The maximum leak rate of the inner containment under loss of coolant accident (LOCA) conditions is 1.5% per day of the total mass of gas inside this containment.

The containment walls are closed at the bottom by a prestressed concrete raft, of constant thickness (about 3 m), with a perimeter gallery for tensioning the vertical cables of the inner containment wall. It is provided with a system of orthogonal drains which run into the gallery, the leakages being collected as radioactive effluents.

As the containment is important to safety, numerous tests have been carried out for R & D purposes and to prove that such a concrete structure is tight enough and that it is possible, with appropriate devices, to detect and repair on-site the defects during construction. The concrete must be carefully worked up so as to reduce as much as possible hydraulic or thermic shrinkage, desiccation and heterogeneity. Before commissioning (at 1.15 times the design pressure: 5.6 bar) during the first shutdown for refueling and then every ten years (at the design pressure = 5 bar), mechanical strength tests of the inner containment are performed. Furthermore, instrumentation is provided to monitor its behavior during operation (12 pendulums, 4 invar wires, some 30 extensometers and about 25 thermocouples inside the concrete). Finally, overall and partial leak tightness tests are periodically performed for the inner and outer containments and for penetrations.

II.2.3. Characteristics of the N4 containment

The N4 containment is a double wall structure built on a solid monolithic raft.

The inner containment is made of prestressed concrete without liner. It is designed to withstand pressure (5.3 bar abs.) and to ensure leaktightness. It consists of a cylindrical wall topped with a spherical dome with ring girder. The inside diameter is 43.8 m, the wall thickness is 1.20 m for the wall and 0.90 m for the dome. The lower level of the containment is 2.40 m under the ground level, the upper part of the dome 60.78 m above the ground level.

The inside height at the center line is 62 m and the total internal volume is approximately 87 300 m^3, giving a free volume of about 72 000 m^3.

Four families of tendons prestress the concrete:

- two in the cylinder (1 horizontal, 1 vertical)
- two in the dome (with a 90° shift).

The outer containment is designed to resist external impacts and to collect the leaks from the inner containment. It is a reinforced concrete structure with a cylindrical wall and a spherical dome. The inner diameter is 49.8 m, the wall thickness 0.55 m for the wall and 0.40 m for the dome.

The space between these containments, 1.80 m wide, is kept slightly below atmospheric pressure (-15 mbar) and any leakage through the inner wall is then collected, filtered and monitored before release. The maximum leak rate of the inner containment under LOCA conditions is 1.5% per day of the total mass of gas inside this containment.

The containment walls are closed at the bottom by a concrete raft, of constant thickness (about 3 m), with a perimeter gallery for tensioning the vertical cables of the inner containment wall. It is provided with a system of drains which run into the gallery.

Before commissioning, after one year of operation, and every ten years, a full pressure strength test of the inner containment is performed. The test pressure is 1.15 time the design pressure for the first test, the design pressure for the later tests.

The internal structures are reinforced works inside the inner containment whose functions are:

- to support the NSSS equipment under normal and accidental conditions,
- to protect the inner containment from fluid jets and internal missiles,
- to protect personal against radiations,
- to make a partition between the reactor coolant loops and between some equipment.

These structures are connected to the inner containment by the floor supporting the heavy components at level +6.56 m. They consist of:

- The reactor pit: this cylindrical work (thickness of about 2 m) supports the reactor vessel, limits the amplitude of its motion in the event of an accident and ensures biological protection,

- the skirt: this cylindrical shell (thickness 1 m), together with other walls and floors forms compartments for the main NSSS equipment supporting and protection,

- the reactor cavity: lined with stainless steel, it allows good personal protection during refueling operations.

Table VII provides main characteristics evolution in the French PWR containment design.

TABLE VII. FRENCH PWR CONCRETE CONTAINMENT DESIGN MAIN CHARACTERISTICS EVOLUTION

	TIHANGE	CP1/CP2	P 4	N4
Power MW(e)	870	900	1300	1450
Construction Decision (Year)	1969	1974-1977	1979	1983
Design	Double wall with steel liner	Single wall with steel liner	Double wall (concrete)	Double wall (concrete)
Design Pressure (bar)	4.1	5	5.2	5.3
Test pressure (bar) 1st and 10 years outages	4.1	5	5.2	5.3
Wall thickness (cm)	inner wall 70	90	inner wall 120	inner wall 120

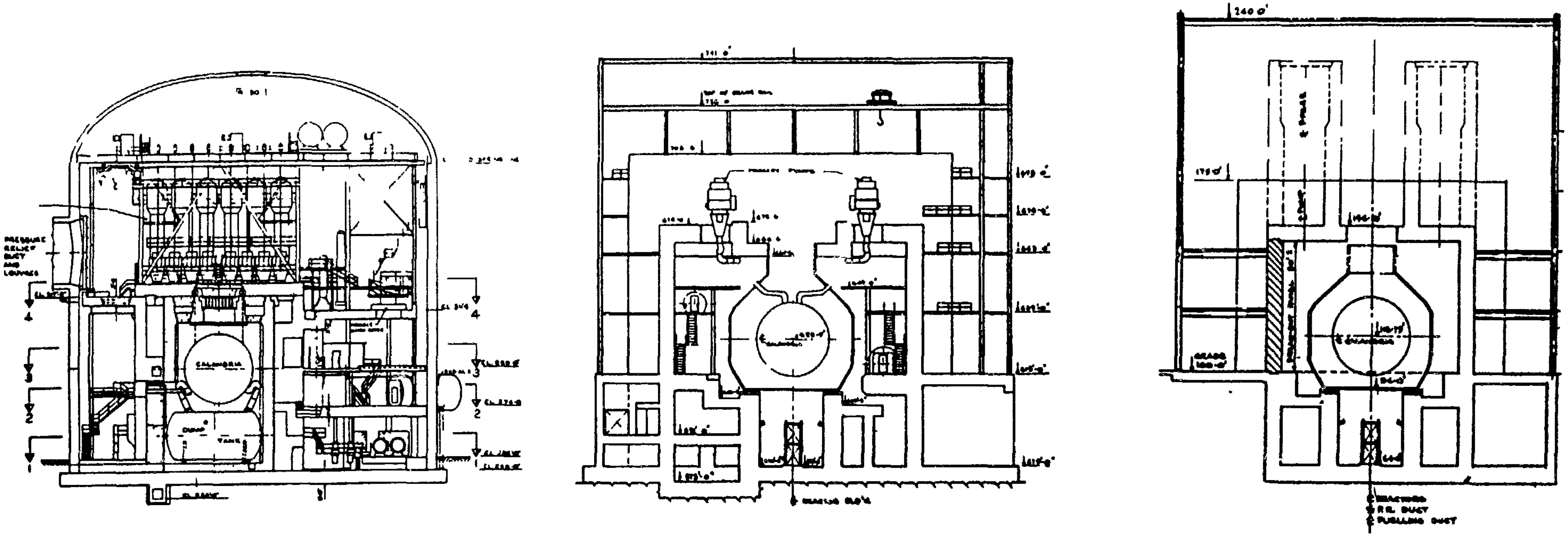
TABLE VIII. ONTARIO HYDRO MULTI-UNIT STATIONS

Station	First Unit In Service	Net Output MW(e)	Containment Volumes(m^3)		Reactor Building Design Pressure kPa(g)
			Reactor Building	Vacuum Building	
Pickering A	1971	4 x 515	51 000 each	82 000 for all 8 units	41
Pickering B	1983	4 x 516	51 000 each		41
Bruce A	1977	4 x 769	92 500 *	62 000	69
Bruce B	1985	4 x 860	95 000 *	62 000	83
Darlington	1990	4 x 880	140 000 *	95,000	96

* Shared by four reactor vaults and interconnecting ducts.

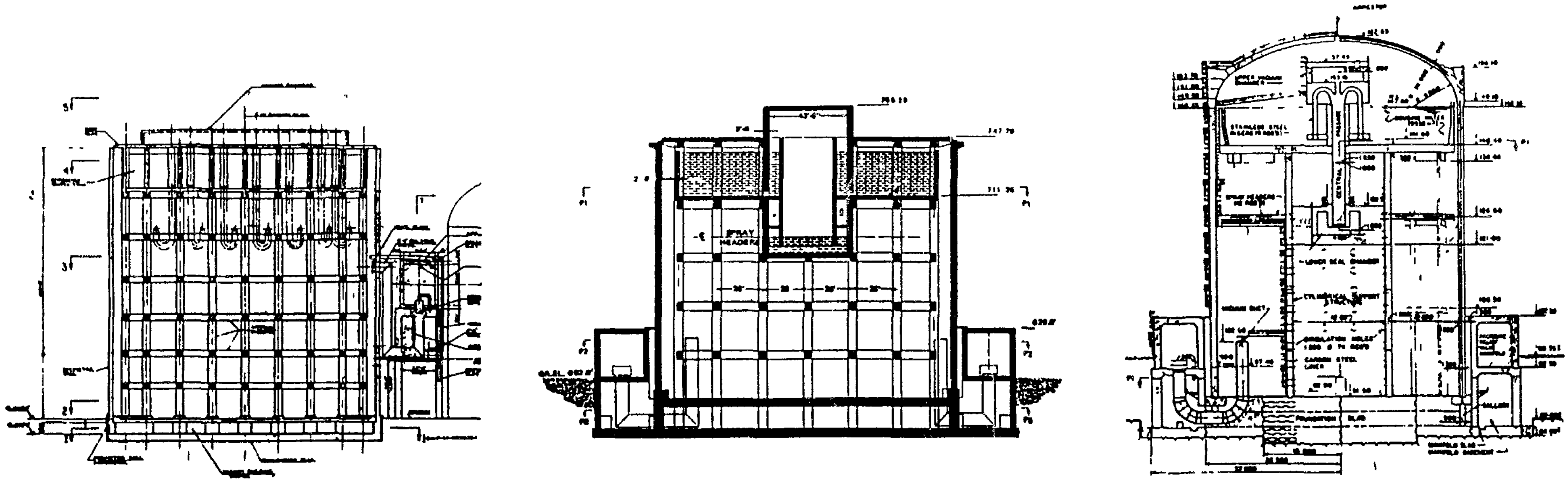
TABLE IX. REACTOR BUILDING DIMENSIONS (m)

	Pickering	Bruce	Darlington
	circular:	rectangular:	rectangular:
Base area	dias = 42.5	50.0 x 35.4	25.0 x 55.5
Wall height	35.6	49.4	58.2
Wall thickness	1.2	1.8	1.8
Dome height	10.8	N/A	N/A
Roof thickness	0.5	1.8	1.8



Pickering Bruce Darlington

FIG. 13. Cross-sections of Ontario Hydro reactor buildings.



Pickering Bruce Darlington

FIG. 14. Cross-sections of Ontario Hydro vacuum buildings.

TABLE X. VACUUM BUILDING DIMENSIONS (m)

	Pickering	Bruce	Darlington
Diameter of base area	50.3	48.8	48.0
Wall height	50.8	45.4	56.6
Wall thickness	0.9	1.1	1.3
Dome height	N/A	N/A	14.5
Roof thickness	0.6	0.9	1.0

II.3 CANADIAN MULTI-UNIT CONTAINMENT STRUCTURES

All Canadian multi-unit NPPs, owned by Ontario Hydro, have a negative pressure containment system designed to relieve accident pressures and confine accident gases. To provide this negative pressure capability all generating stations are equipped with a vacuum building as part of the containment system. One vacuum building normally services four reactor buildings.

These structures have evolved in form and concept since the original design in the mid-1960s. Containment has changed from separated reactor buildings (e.g. Pickering) to interconnected reactor buildings. Older reactor buildings are domed cylinders in shape, while newer buildings tend to be buttressed cube shapes. There has been a steady increase in design pressure and a reduction in contained volume as reactor power has increased and less equipment has been housed inside the containment (see Table VIII). Table IX gives the dimensions of representative reactor buildings shown in Fig. 13.

Vacuum buildings have evolved from a normally reinforced cylinder to a heavily post-tensioned domed cylinder construction (see Fig. 14). Table X shows typical vacuum building dimensions. During this evolution the concrete specified for these buildings has increased in strength from 28 MPa at 28 days to 35 MPa at 28 days. Post-tensioning tendons have typically been grouped in place except for the most recent vacuum building which employs greased tendons.

APPENDIX III

LIST OF PAPERS PRESENTED AT THE TECHNICAL COMMITTEE MEETINGS OF NOVEMBER 1990 AND OCTOBER 1991

A. Papers on RPV primary nozzle

A.1 Ageing Degradation of RPV Primary Nozzle - Mechanism and Countermeasures by M. Erve and M. Miksch (Siemens AG, Power Generation KWU).

A.2 Some results of the primary nozzle stress analysis in CSFR as the basis for the ageing calculations by S. Novak and M. Hrazsky, Nuclear Power Plant Research institute, Trnava, CSFR.

A.3 Evaluation Leak and Failure Probability of Nuclear Plant Equipment Element and Piping by V.V. Tkachev and V.G. Vasilyev, Scientific and Engineering Centre for Safety in Industry and Nuclear Power, Moscow, USSR, and by A.A. Tutnov, Kurchatov Institute of Atomic Energy, Moscow, USSR.

A.4 Selection of CEC Activities in the Area of Ageing Degradation in Nuclear Power Plants by H.A. Maurer, Commission of the European Communities.

A.5 Vessel Nozzles for French 900 MW(e) PWR Units by J.P. Combes and R. Capel, EDF-SPT, France.

A.6 Inspection Performance with a View to Pressure Vessel Life Management by S. Crutzen and P. Jehenson, IAM-JRC Ispra, Italy.

A.7 Ageing of VVER 440 RPV Nozzle and Some Results of its Evaluation Carried out in CSFR by M. Hrazsky and I. Cineura, NPP Research Institute, Trnava, CSFR.

A.8 A Recommendation for Visual Examination of Primary Nozzles of VVER 440 and VVER 1000 reactors by K. Varga, Hut-Hungary Kft, Hungary.

A.9 Primary Nozzle of a Reactor Pressure Vessel - Experience and Interests of Skoda Concern by M. Brumovsky, Skoda Concern, CSFR.

B. Papers on motor operated isolating valve

B.1 Motor Operated Valve Ageing, Reliability and Operation Readiness by D.M. Eissenberg (ORNL) and W.S. Farmer NRC (USA).

B.2 The Application of Continuous Markov's Processes for the Assessment of Nuclear Power Plant Hardware by V.S. Emelyanov, Russia.

B.3 Pilot Studies on Motor-Operated Valves as Part of the Emergency Core Cooling System by H. Boyne, Switzerland.

B.4 Analysis of MOV Operating Experiences in Finish BWR Plants by K. Simola and K. Laokso, Finland.

B.5 Automatic Diagnostics by J. Campan, France.

B.6 Qualification for Accident Conditions of Equipment other than Electrical Equipment by G. Delêtre, France.

B.7 Use of Motor Current Signature Analysis at the EPRI M&D Center by H. Haynes, R. Kryter and B. Stewart, ORNL, USA.

B.8 Monitoring Motor Operated Valve Performance by R. Steele, Jr., K.G. DeWall and J.C. Watkins, U.S.A.

B.9 Motor Operated Valves: French Experience Feedback, Incidents and Corrective Actions, Testing Systems by M. Uhart and R. Capel, France.

B.10 LOCA-Qualification of Electrical Equipment at the Loviisa PWRs (IVO) by J. Snellman, Finland.

C. Papers on concrete containment building

C.1 An overview of Concrete Containment Aging Degradation Mechanisms and Management Techniques by J.A. Statton, Bechtel Corporation, USA.

C.2 UK Experience of In-Service Surveillance of Prestressed Concrete Reactor Pressure Vessels by B.N. Grainger, Nuclear Electric plc, UK.

C.3 Experience and Initiatives for Life Assurance of Civil Structures at Ontario Hydro, by C.B.H. Cragg, Ontario Hydro, Canada.

C.4 Investigations Relating to the Containment Building as Part of the French Lifetime Project by J.P. Combes and R. Capel, EDF-SPT, France.

D. Papers on instrumentation and control cables

D.1 The Understanding and Management of the Ageing Degradation of Instrumentation and Control Electrical Cabling by J.Y. Henry, CEA/IPSN, France.

D.2 Longterm LOCA and Radiation Resistance of Cables by A. Bleier (Dept. E141) and W. Michel (Dept. E471).

D.3 Effect of I&C Cable Ageing on Nuclear Reactor Safety by V.V. Kondratiev and V.K. Prozorov, USSR.

D.4 Summary of AEA Technology Experience in Lifetime Prediction and Monitoring Techniques for Polymeric Cable Components, by S.G. Burney and J. Dawson, AEA Technology, Harwell, U.K.

D.5 Investigations Relating to Electric Cables as Part of the French Lifetime Project by J.P. Combes and R. Capel, EDF-SPT, France.

E. General papers on ageing of NPP components

E.1 Ageing Research in the CEA by J.L. Campan, France.

E.2 UK Regulatory Requirements Relating to the Ageing of NPP Components by E.M. Pape, Nuclear Installations Inspectorate, UK.

E.3 Application of Continuous Markov's Processes for the Assessment of Nuclear Power Plant Hardware by V.S. Emelyanov, USSR.

E.4 French Lifetime Project its Progress and its prospects by J.P. Combes and R. Capel, EDF - SPT, France.

E.5 Some observations of Nuclear Research Reactors Ageing by K. Varga, Isotope Institute of Hungarian Academy of Sciences, Hungary.

E.6 Ageing research - Programme of Studies for some Components and Equipment Used in Indian PHWRs by V. Venkat Raj, A.K. Babar, A.K. Ghosh, J.S. Bora and A. Kakodkar, India.

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