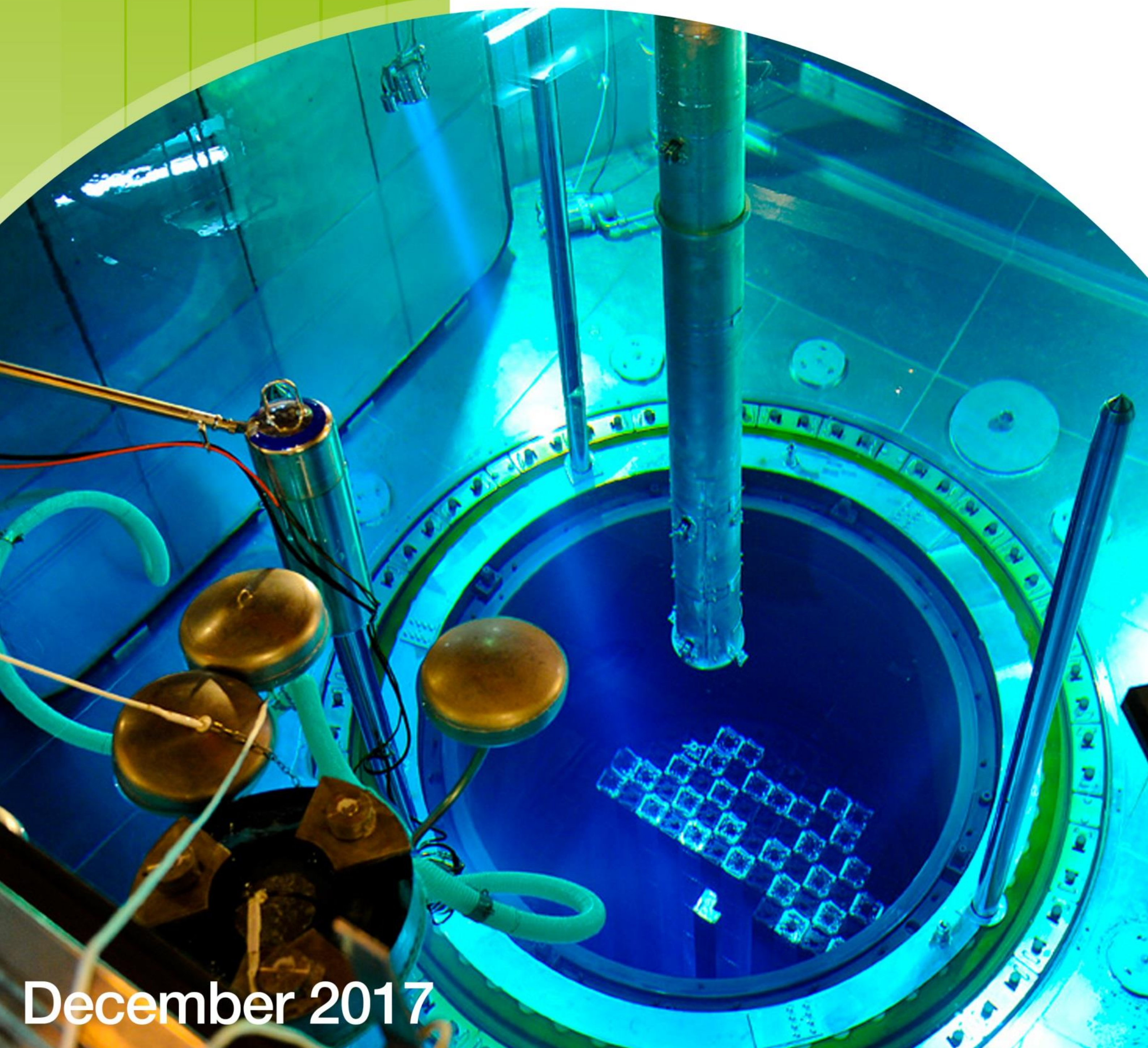
ENSREG TOPICAL PEER REVIEW 2017

FANC
federal agency for nuclear control

Ageing Management

Belgian national report



December 2017

EU Topical Peer Review

Ageing Management of

nuclear power plants and research reactors

Belgian National Assessment Report

Federal Agency for Nuclear Control

2017

Content

Executive summary

In 2014, the European Union Council adopted directive 2014/87/EURATOM to incorporate lessons learned following the accident at the Fukushima Daiichi nuclear power plant in 2011. Recognising the importance of peer review in delivering continuous improvement to nuclear safety, the revised Nuclear Safety Directive introduces a European system of topical peer review (TPR) which will commence in 2017 and will be repeated every six years thereafter.

The first topical peer review covers the **Ageing Management** of the nuclear power plants (NPPs) and the research reactors such as BR2, still in operation. The assessment of ageing management focuses on the overall ageing management program as well as on its implementation on specific Structures Systems and Components (SSCs) important to nuclear safety (electric cables, concealed pipework, reactor pressure vessels and concrete containment structures).

The need for structuring all activities relevant to Ageing Management in an Overall Ageing Management Program in order to ensure a complete program covering all SSCs is a relatively recent development in Belgium for both nuclear power and research reactors. Since the beginning of 2000s, a continuously growing attention is focused on Ageing Management by both the licensees and the Safety Authority.

For **NPPs**, a structured ageing management program was first introduced in 2004, and further enhanced with the possible LTO for the first NPP units since 2009. The present national assessment report on ageing management highlights that the ageing management program by ENGIE Electrabel is now in line with the international standards and should ensure an adequate management of the ageing of the safety-related SSCs during the rest of the lifetime of the NPPs.

The Safety Authority considers that recent and still ongoing reinforcement of this ageing management program in the framework of the LTO program for the first three units is a significant achievement. This new ageing management program for the LTO units is considered as complete and based on a systematic and comprehensive approach. The currently planned extension of this program to the ageing management program of the more recent units based on the program installed for the LTO units, and using the most recent international standards and guidance, is a positive initiative which will be a challenge for the forthcoming years.

No additional action or improvement has been identified by the Safety Authority for the overall ageing management program. The Safety Authority considers that on this topic the ongoing action plans set up in the framework of the last Periodic Safety Reviews (2012-2015) or in the framework of the LTO for the first units, in addition to the actions already performed in 2017 by the Licensee arising from its self-assessment in the frame of the TPR, are sufficient to achieve a complete ageing management program.

Nevertheless some past and recent events still highlight ageing issues, an inadequate monitoring of the ageing of some components and an insufficiency of remedial actions in the Belgian Nuclear Power Plants. Ongoing investigations must show if these recent findings are symptomatic of a structural or particular deficiency, in order to adapt in consequence the ageing management programs.

It can be concluded that the overall ageing management program by ENGIE Electrabel is in line with international standards but that its effective implementation on site can be further improved. The Safety Authority considers that the licensee has taken some necessary steps in this direction. Some tools have been recently developed and their effectiveness will be judged on the long term.

For the **BR2 research reactor**, the development of a complete ageing management program was part of the Periodic Safety Review (PSR) that was completed 2016. In that framework SCK•CEN is required to improve and implement the ageing management program of the BR2 to reach a level that is commensurable with the best international standards and practices for research reactors, before June 2019. Although the development of the ageing management program is still in progress, some gaps have been identified throughout the TPR assessment. The Safety Authority therefore formulates several recommendations in this report in order to fill these gaps.

The **Belgian regulatory oversight** on Ageing Management increased in the last decade with the focus put on the safety factor Ageing in the framework of the PSR and on the LTO for the first NPP units but also for the BR2. The regulatory requirements on ageing management have been clearly defined in the Belgian regulations for nuclear facilities. The strategic notes on LTO by the FANC and the PSR process to be performed accordingly to IAEA standards specifically focuses on the Ageing Management Program, while the Royal Decree of 30 November 2011 on the Safety Requirements for Nuclear Installations specifically requests an up-to-date and complete Ageing Management Program.

Although ageing management has been thoroughly followed-up by the Safety Authority in the recent years (review of LTO and PSR projects, regular meetings with the licensee and specific inspections), the Safety Authority is aware of the increasing importance of an adequate ageing management, to ensure further safe operation of the nuclear power plants and research reactors in Belgium. In consequence, the Safety Authority plans to continuously enhance the follow-up on this topic until the final shut-down and decommissioning of the nuclear facilities.

The first stage of the peer review process is the production of this national assessment report which describes the overall ageing management program and identifies its main strengths and weaknesses. The second stage of the peer review process, that will begin in 2018, is the peer-review itself between EU countries in order to share operating experience, identify good practices, common issues and follow-up actions to address the challenges posed by ageing management of nuclear facilities.

Samenvatting

De Raad van de Europese Unie heeft in 2014 de richtlijn 2014/87/EURATOM aangenomen om de geleerde lessen van het ongeval in 2011 met de kerncentrale van Fukushima Daiichi te implementeren. De herziene Richtlijn Nucleaire Veiligheid erkent het belang van zogenaamde peer reviews om een continue verbetering van de nucleaire veiligheid te verzekeren, en stelt om deze reden een Europees systeem voor van topical peer reviews (TPR) die van start gaat in 2017 en daarna iedere 6 jaar herhaald wordt.

De eerste topical peer review omvat het thema **Verouderingsbeheer** voor nog operationele kerncentrales en onderzoeksreactoren zoals BR2. De analyse van het verouderingsbeheer focust op het algemene verouderingsbeheerprogramma maar ook op diens implementatie op de structuren systemen en componenten (SSCs) die belangrijk zijn voor nucleaire veiligheid zoals elektrische kabels, ingebouwde leidingen, reactordrukvaten en containment structuren opgetrokken uit beton.

Het is een noodzaak om alle activiteiten met betrekking tot verouderingsbeheer te structureren in een overkoepelend verouderingsbeheerprogramma zodat het programma alle SSCs bevat. Deze aanpak is een relatief recente ontwikkeling in België voor zowel vermogensreactoren als onderzoeksreactoren. Sinds het begin van de jaren 2000 hebben de veiligheidsautoriteit en de exploitant dan ook een groeiende aandacht voor verouderingsbeheer.

Een gestructureerd programma voor verouderingsbeheer werd op de **kerncentrales** initieel geïntroduceerd in 2004, en aangevuld vanaf 2009 met het oog op een mogelijke LTO-uitbating voor de eerste reactoreenhedens. Dit nationale assessment report omtrent verouderingsbeheer toont aan dat het verouderingsbeheerprogramma van ENGIE Electrabel in lijn is met de internationale standaarden en dat het een adequate beheer verzekert van de veroudering van de veiligheid gerelateerde SSCs voor de resterende levensduur van de kerncentrales.

De veiligheidsautoriteit beschouwt de recentelijke en nog lopende versterkingen van dit verouderingsbeheerprogramma in het kader van de LTO-projecten voor de drie eerste eenheden als een significante prestatie. Dit nieuwe verouderingsbeheerprogramma voor de LTO-eenheden wordt als compleet aanzien, terwijl er ook vastgesteld wordt dat het gebaseerd is op een systematische en alomvattende aanpak. De geplande uitbreiding van dit programma naar de meest recente eenheden, gebaseerd op het programma uitgerold voor de LTO-eenheden en waarbij de meest recente standaarden en richtlijnen gebruikten worden, is een positief initiatief dat een uitdaging zal vormen voor de komende jaren.

De veiligheidsautoriteit heeft geen actie of verbetering geïdentificeerd voor het overkoepelende verouderingsbeheerprogramma. Het actieplan opgesteld in het kader van de laatste periodieke veiligheidsherzieningen (2012-2015) of in het kader van het LTO project voor de eerste eenheden, samen met de door de exploitant reeds in 2017 uitgevoerde acties die volgen uit de zelfanalyse in het kader van deze topical peer review, wordt door de veiligheidsautoriteit als voldoende beschouwd om een volledig verouderingsbeheerprogramma te bekomen.

Niettegenstaande zijn er recente en gebeurtenissen uit het verleden die de aandacht trekken op verouderingsproblemen, een inadequate opvolging van de veroudering bij bepaalde componenten en de ontoereikendheid van corrigerende maatregelen in de Belgische kerncentrales. Lopende onderzoeken moeten nog aantonen of deze observaties symptomatisch zijn voor een structureel of gedeeltelijk tekort zodat het verouderingsbeheerprogramma consequent kan aangepast worden.

Er kan besloten worden dat het overkoepelende verouderingsbeheerprogramma van ENGIE in regel is met de internationale standaarden, maar dat de effectieve implementatie op site verder verbeterd kan worden. De veiligheidsautoriteit stelt dat de exploitant de nodige stappen ondernomen heeft in deze richting. Sommige tools in dit kader werden onlangs ontwikkeld en hun efficiëntie zal op de lange termijn beoordeeld worden.

Het ontwikkelen van een volledig verouderingsbeheerprogramma was deel van de Periodieke Veiligheidsherziening van de **BR2 onderzoeksreactor** die beëindigd werd in 2016. In het kader van een dergelijke veiligheidsherziening wordt het SCK•CEN gevraagd het verouderingsbeheerprogramma van de BR2 te verbeteren en te implementeren voor juni 2019 zodat een niveau bereikt wordt dat vergelijkbaar is met de beste internationale standaarden en praktijken voor onderzoeksreactoren. Hoewel het verouderingsbeheerprogramma nog in ontwikkeling is, zijn enkele gaps gedefinieerd doormiddel van de TPR analyse. De veiligheidsautoriteit heeft om deze reden enkele aanbevelingen geformuleerd in dit rapport om deze gaps aan te pakken.

Conclusies

Het toezicht van de Belgische regulator op het verouderingsbeheer is de laatste tien jaar gestegen via de focus op de veiligheidsfactor veroudering in het kader van de Periodieke Veiligheidsherziening en in het kader van de LTO van de eerste eenheden maar ook BR2. De reglementaire eisen omtrent verouderingsbeheer zijn duidelijk vastgelegd in de Belgische reglementaire teksten voor nucleaire inrichtingen. De strategische nota's van het FANC, omtrent LTO en het Periodieke Veiligheidsherziening (PSR) proces dat uitgevoerd dient te worden volgens de IAEA standaarden, focussen specifiek op het verouderingsbeheerprogramma, terwijl het Koninklijk Besluit van 30 November 2011 houdende veiligheidsvoorschriften voor kerninstallaties specifiek een volledig en geactualiseerd verouderingsbeheerprogramma vraagt.

Hoewel verouderingsbeheer de recente jaren van dichtbij opgevolgd wordt door de veiligheidsautoriteit (review van LTO en PSR projecten, regelmatige meetings met de uitbater en specifieke inspecties), is de veiligheidsautoriteit bewust van het stijgende belang van een adequaat verouderingsbeheer om een verdere veilige uitbating van de kerncentrales en onderzoeksreactoren te verzekeren. Hierdoor plant de veiligheidsautoriteit om de opvolging van dit onderwerp continu te versterken tot de uiteindelijke sluiting en ontmanteling van de nucleaire inrichtingen.

De eerste stap in het peer review proces is de publicatie van dit nationaal assessment report dat het overkoepelende verouderingsbeheerprogramma beschrijft en diens sterktes en zwaktes definieert. De tweede stap van het peer review proces, dat in 2018 van start gaat, is de peer-review binnen de EU-lidstaten om uitbatingservaring te delen, goede praktijken en algemene problemen te identificeren samen met opvolgacties om de uitdagingen aan te pakken die gesteld worden door het verouderingsbeheer van nucleaire inrichtingen.

Résumé

En 2014, le Conseil de l'Union européenne a adopté la directive 2014/87/Euratom afin de prendre en compte des enseignements tirés de l'accident survenu à la centrale nucléaire de Fukushima Daiichi en 2011. Reconnaissant l'importance des « Peer Review » dans le processus d'amélioration continue de la sûreté nucléaire, la nouvelle directive sur la sûreté nucléaire instaure à partir de 2017 un tel système au niveau européen (Topical Peer Review). Il sera répété tous les six ans par la suite.

Le premier « Topical Peer Review » couvre la gestion du vieillissement des centrales nucléaires et des réacteurs de recherche encore en opération. L'évaluation de cette gestion se concentre sur le programme global de gestion du vieillissement ainsi que sur sa mise en œuvre pour certains SSCs (structure, système et composants) importants pour la sûreté nucléaire (câbles électriques, tuyauteries enfouies, cuves des réacteurs et structures de confinement en béton).

La nécessité de structurer toutes les activités pertinentes en terme de gestion du vieillissement dans un programme global afin de couvrir l'ensemble des SSCs est un développement relativement récent en Belgique tant pour les réacteurs nucléaires que pour les réacteurs de recherche. Ainsi, depuis le début des années 2000, une attention toujours croissante a été apportée à la gestion du vieillissement aussi bien par les exploitants que par l'Autorité de Sûreté.

Pour les centrales nucléaires, un programme structuré de gestion du vieillissement a d'abord été introduit en 2004, et étendu à partir de 2009 avec la possibilité d'un projet LTO (exploitation à long terme) pour les premières unités des centrales nucléaires. Au terme de la présente évaluation nationale, l'Autorité de sûreté conclut que le programme de gestion du vieillissement d'ENGIE Electrabel est aujourd'hui en accord avec les standards internationaux et devrait permettre d'assurer une gestion adéquate du vieillissement des SSCs liés à la sûreté pour la durée de vie restante des centrales nucléaires.

L'Autorité de Sûreté estime que le renforcement récent de ce programme de gestion du vieillissement dans le cadre du projet LTO pour les trois unités les plus anciennes est une réalisation significative. Ce nouveau projet pour ces unités est considéré comme complet et basé sur une approche systématique et globale. L'extension actuellement prévue de ce projet au programme de gestion de vieillissement des unités les plus récentes, en utilisant les standards et guidances internationaux les plus modernes, est une initiative positive et un défi pour les années à venir.

Aucune action ou amélioration supplémentaire n'a été identifiée par l'Autorité de Sûreté pour le programme de gestion de vieillissement global. L'Autorité de Sûreté considère que les plans d'action en cours mis en place dans le cadre des dernières révisions périodiques de sûreté (2012-2015) ou dans le cadre des LTO pour les plus anciennes unités, en plus des actions déjà accomplies en 2017 par l'exploitant découlant de se propre évaluation dans le cadre du « Topical Peer Review », sont suffisants pour atteindre un programme complet de gestion du vieillissement.

Néanmoins quelques événements passés et récents mettent en avant des problèmes de vieillissement, un suivi inadéquat du vieillissement et un manque d'action correctrice sur certains composants dans les centrales nucléaires belges. Les investigations en cours doivent montrer si ces manquements constatés récemment sont symptomatiques d'une déficience structurelle ou spécifique, en vue d'adapter les programmes de gestion du vieillissement.

En conclusion, le programme global de gestion du vieillissement par ENGIE Electrabel est en accord avec les standards internationaux mais sa mise en œuvre effective sur site doit encore être améliorée. L'Autorité de Sûreté estime que l'exploitant a pris certaines mesures nécessaires à cet effet. Certains outils ont en effet été récemment développés. Leur efficacité sera jugée sur le long terme.

Pour le réacteur de recherche BR2 du SCK•CEN, le développement d'un programme complet de gestion du vieillissement faisait partie de la révision périodique de sûreté, achevée en 2016. Dans le plan d'actions associé, le SCK•CEN doit améliorer et implémenter ce programme afin d'atteindre un niveau comparable aux meilleurs pratiques et standards internationaux pour les réacteurs de recherche, et ce avant juin 2019. Bien que le développement du programme de gestion du vieillissement soit toujours en cours, certains manquements ont été constatés lors de l'évaluation du « Topical Peer Review ». L'Autorité de sûreté formule dans ce rapport plusieurs recommandations pour y remédier.

La surveillance par l'Autorité de sûreté belge de la gestion du vieillissement s'est accrue ces dix dernières années en se concentrant, dans le cadre des révisions périodiques de sûreté, sur le facteur de sûreté « vieillissement » et sur le projet LTO pour les unités les plus anciennes des centrales, de même que pour le BR2. Ainsi les exigences réglementaires sur la gestion du vieillissement ont été clairement définies dans la réglementation belge pour les installations nucléaires. Les notes stratégiques de l'AFCN concernant l'approche à suivre pour un LTO et le processus de révision périodique de sûreté à mener conformément aux standards de l'AIEA adressent spécifiquement le programme de gestion du vieillissement, et l'arrêté royal du 30 novembre 2011 portant sur les prescriptions de sûreté des installations nucléaires demande explicitement un programme de gestion du vieillissement à jour et complet.

Bien que la gestion du vieillissement soit effectivement suivie par l'Autorité de Sûreté ces dernières années (Suivi et évaluation des projets LTO et des révisions périodiques de sûreté, réunions régulières avec l'exploitant et inspections spécifiques), l'Autorité de Sûreté est consciente de l'importance grandissante d'une gestion adéquate du vieillissement afin d'assurer une opération encore plus sûre des centrales nucléaires et des réacteurs de recherche belges. En conséquence, l'Autorité de sûreté compte renforcer le suivi de cette thématique jusqu'à l'arrêt final et au démantèlement des installations nucléaires.

La première étape du processus de « Topical Peer Review » est la publication de ce rapport national d'évaluation qui décrit le programme global de gestion du vieillissement et qui identifie ses points forts et points faibles principaux. La seconde étape, qui commencera en 2018, est le « peer review » lui-même entre pays de l'Union européenne qui permettra de partager l'expérience, d'identifier les bonnes pratiques, les problèmes courants et les actions de suivi dans le but de relever les défis posés par la gestion du vieillissement des installations nucléaires.

0. Introduction[1]

Context

In 2014, the European Union (EU) Council adopted directive 2014/87/EURATOM amending the 2009 Nuclear Safety Directive to incorporate lessons learned following the accident at the Fukushima Daiichi nuclear power plant in 2011. Recognising the importance of peer review in delivering continuous improvement to nuclear safety, the revised Nuclear Safety Directive introduces a European system of topical peer review which will commence in 2017 and will be repeated every six years thereafter. The purpose is to provide a mechanism for EU Member States to examine topics of strategic importance to nuclear safety, to exchange experience and to identify opportunities to strengthen nuclear safety. The member states, acting through the European Nuclear Safety Regulators Group (ENSREG), have decided that the topic for the first topical peer review is **Ageing Management**.

Scope

It was decided that topical peer review will cover the following types of nuclear installations:

- Nuclear power plants;
- Research reactors with a power equal to 1 MW_{th} or more;

that are still in operation on 31st December 2017 or under construction on 31st December 2016.

Ageing Management

Ageing and ageing management are defined as below in the terms of reference of the Topical Peer Review :

- **Ageing** is considered as a process by which the physical characteristics of a structure, system or component (SSC) change with time (ageing) or use (wear-out).
- **Ageing management** is understood as the engineering, operations and maintenance actions undertaken by a licensee to prevent or to control within acceptable limits ageing degradation of structures, systems and components (SSC) of its installation. With regard to safety it ensures the availability of required safety functions throughout the service life of the plant, with account taken of changes that occur with time and use and by considering all service conditions.

Topical Peer Review Process

National Assessment Report (NAR)

The first stage of the peer review process is the production of a national assessment report for each country participating in the topical peer review. Each member state has to make its NAR available by December 2017.

The objectives of the National Assessment Report (NAR) are to:

- describe the overall ageing management program including:
    - Programmatic aspects;
    - Implementation of overall ageing management program;

[1] This section is directly inspired from :

- "Terms of Reference for Topical Peer Review Process" on Ageing Management of Nuclear Power Plants for the Topical Peer Review 2017, by ENSREG (2016)
- "Technical specifications for the national assessment reports for the Topical Peer Review 2017 on Ageing Management", by WENRA RHWG (2016).

    - Experience of the application of ageing management;
- assess the outcomes to identify main strengths and weaknesses;
- identify actions to address any significant areas of improvement;
- produce a report in sufficient detail to allow a meaningful peer review.

One of the key objectives of ENSREG is to improve the overall transparency of issues relating to the safety of nuclear installations

Full specifications for achieving the NAR were provided by ENSREG.

Peer Review

The second stage of the peer review process is the peer-review itself during which National Assessment Reports will be examined by other countries in order to share operating experience, identify good practices, common issues and follow-up actions to address the challenges posed by ageing management of nuclear facilities. This second stage will extend from the publication of the National Assessment Report to mid-2018. A report on the implementation status is foreseen at the end of a follow-up phase, in December 2023.

Full details on the complete process for developing the Topical Peer Review are presented in the terms of reference and technical specifications by ENSREG.

A new topical peer review will take place in 2024 on another topic, still to be defined.

1. General information

1.A Nuclear installations identification[2]

1.A.1 Nuclear Power Plants (NPPs)

Seven units, spread over the Doel and Tihange sites, were put into service between 1974 and 1985 in Belgium: four of them at the Doel Nuclear Power Plant and three at the Tihange Nuclear Power Plant. The Figure 1A-1 here under shows the locations of the Nuclear Power Plants and the research reactors in operation in Belgium.

Figure 1A-1 : Nuclear sites in and near Belgium

The Doel and Tihange nuclear power plants are operated by Electrabel, a member of the ENGIE group. The tables *1A-1* and *1A-2* give the main characteristics of the 7 Belgian NPPs:

[2] This section is inspired from the 2016 Belgian report for the Convention on Nuclear Safety

Licensee	Units	Type	Thermal power (MWth)	Date of first criticality	Scheduled shutdown	Containment building characteristics	Designer
ENGIE Electrabel	Doel 1	PWR (2 loops)	1 312	1974	2025*	Double containment (steel and concrete)	Westinghouse
ENGIE Electrabel	Doel 2	PWR (2 loops)	1 312	1975	2025*	Double containment (steel and concrete)	Westinghouse
ENGIE Electrabel	Doel 3	PWR (3 loops)	3 064	1982	2022	Double containment with inner metallic liner	Framatome
ENGIE Electrabel	Doel 4	PWR (3 loops)	3 000	1985	2025	Double containment with inner metallic liner	Westinghouse

Table 1A-1 : Main characteristic of the units located at the Doel Site

Licensee	Units	Type	Thermal power (MWth)	Date of first criticality	Scheduled shutdown	Containment building characteristics	Designer
ENGIE Electrabel	Tihange 1	PWR (3 loops)	2 873	1975	2025*	Double containment with inner metallic liner	Framatome / Westinghouse
ENGIE Electrabel	Tihange 2	PWR (3 loops)	3 054	1982	2023	Double containment with inner metallic liner	Framatome
ENGIE Electrabel	Tihange 3	PWR (3 loops)	2 988	1985	2025	Double containment with inner metallic liner	Westinghouse

** under LTO*

Table 1A-2 : Main characteristic of the units located at the Tihange Site

Belgium's seven nuclear power units in operation are pressurised water reactor types built either by Westinghouse or by Framatome, each time in partnership with Belgian manufacturers for the major equipment of the primary and secondary systems. These units were put into service between 1974 and 1985.

In Belgium, the nuclear license has no time limit. Since the design and commissioning, a Periodic Safety Review (PSR) is conducted every 10 years to reconfirm the design basis of the plant and upgrade the safety in line with current standards. For the last PSR (2012-2015), an assessment of the 14 Safety Factors according to IAEA methodology SSG-25 [2J] has been performed. The last PSR includes therefore a special chapter on the assessment of the IAEA Safety Factor on Ageing.

A law relating to the phase-out of nuclear energy was voted by the Belgian parliament in January 2003. Nevertheless Electrabel studied and prepared the possibility of LTO for the three first units since 2009. By 2012, the Safety Authority and Electrabel finalised their evaluation and actions plans were agreed for a potential LTO for these units. In July 2012 a governmental decision was taken to allow Tihange 1 to operate until 2025. An action plan was launched for the improvement of Tihange 1 at that moment. On 28 June 2015, the phase out law of 2003 was finally modified to allow also the LTO of Doel 1 & 2 up to 2025. The action plan prepared for Doel 1 & 2 since 2012 but not applied due to the decision to definitively shut-down the two reactors was then finally launched.

Upon request of the Belgian Government, IAEA peer review missions on safe long-term operation (SALTO) were conducted to review the programs and activities of the Doel Nuclear Power Plant units 1 and 2 and Tihange Nuclear Power Plant unit 1, respectively in 2017 and 2015. The results and conclusions of these IAEA peer review missions are publically available on the FANC website.

1.A.2 Research reactors

Several research reactors were operational in Belgium (5 at the Nuclear Research Centre SCK•CEN and 1 at the University of Gent). At this moment 3 of these are still in operation, all of them at the SCK•CEN site in Mol.

- VENUS

VENUS is a zero power critical facility. The VENUS research reactor, which stands for Vulcan Experimental Nuclear Study, has been operational since 1964. This flexible facility has already been renovated and modernised several times. VENUS was initially used to study the optimal nuclear fuel configuration for various reactors. In 2008 SCK•CEN began the complete rebuilding of VENUS for research into accelerator driven systems. The project started in 2010 and since then the reactor has been known as VENUS-F.

Due to the zero power of VENUS this facility is not considered in the scope of the Belgian NAR.

- BR1

The BR1 is a natural uranium graphite reactor, comparable to the reactors ORNL X-10 (USA) and BEPO (Harwell, UK). The reactor went critical for the first time in 1956. The core is composed of a pile of graphite blocks. The reactor is air cooled. The fuel is metallic natural uranium with an aluminium cladding. Its design thermal power is 4 MW_{th}. However, since the start of BR2, this high power was no longer needed and since 1965 the BR1 is operated at a maximum thermal power of 1 MW using only the auxiliary ventilation system.

Because SCK•CEN limited its periodic safety review submitted in 2016 to a maximum thermal power of 1 MW, FANC subsequently decided in 2017 to formalize the power reduction of the BR1 and to definitively limit the maximum thermal power at 1 MW by a change in the license of SCK•CEN. In line with this development, it was proposed by the licensee SCK•CEN and the FANC to not consider the BR1 in the national report for the topical peer review on ageing.

- BR2

The BR2 is a heterogeneous thermal high flux test reactor, designed in 1957 for SCK•CEN by NDA [Nuclear Development Corporation of America - White Plains (NY - USA)]. Its first criticality dates from 1961 and operation of the reactor started in January 1963.

The reactor is cooled and moderated by pressurised light water (~12 bars) in a compact core of highly enriched uranium positioned in and reflected by a beryllium matrix. The maximum thermal flux approaches 10^{15} neutrons / (cm^2.s) and the ultimate cooling capacity, initially foreseen for 50 MW_{th}, has been increased in 1971 to 125 MW_{th} by replacement of the primary heat exchangers.

The reactor was originally designed for material and fuel testing. This still is an important activity. A number of irradiation devices are available. However during the last years isotope production (Mo-99, Ir-192 and others) has become an important activity. Besides this, two irradiation facilities for silicon doping are available.

The beryllium matrix swells under neutron irradiation due to the formation of gas (helium and tritium). This swelling causes cracking of the beryllium which is a brittle material. Furthermore, the build-up of the helium-3 isotope results in neutron poisoning. Due to these effects the lifetime of the beryllium matrix is limited. The possibility and the need for the matrix replacement was foreseen from the design and commissioning of the BR2. Three replacements were already performed. The first one took place in 1979 and the second one in 1996. The third replacement was done in 2015-2016.

The BR2 license included at the commissioning an end-of-life date after 30 years of operation. However in 1986 the end-of-life date was withdrawn from the license and replaced with the regulatory requirement to conduct a PSR process every 5 years. Later on the license was updated to require a PSR every 10 years similarly to nuclear power plants. In this framework during the lifetime of the reactor, continuous modernization projects have been executed. On the occasion of the last PSR (2016), a major refurbishment program was realized, including a third beryllium matrix replacement. Major works include the replacement of all underground piping, improvement of the electricity and instrumentation cabling by installing additional separate cable routes and the upgrade of hoisting devices by making them single failure proof. For a number of components and systems, design upgrades are made, to improve the safety and reliability.

Upon request of the Belgian Government, an IAEA expert peer review mission on safe long-term operation was conducted in November 2017 to review the programs and activities concerning ageing of the SCK•CEN research reactor BR2. The results and conclusions of this IAEA peer review missions will be published on the FANC website when available.

1.B Process to develop the National Assessment Report

In order to fulfill the requirements of the technical specifications of the topical peer review on ageing management, the Belgian Safety Authority involved several licensees in the process to develop the National Assessment Report.

In Belgium, licensees have the prime responsibility for nuclear safety. The regulation (SRNI-2011) requires them to set up and Ageing Management Program for **Ageing Management** of Systems, Structures and Components (SSCs) important to nuclear safety.

Thus the Belgian Safety Authority required them to perform a self-assessment of their ageing management program with regards to the specifications of the topical peer review process. In particular the Belgian Safety Authority requires the licensees to provide the basis content of the Belgian National Assessment Report requiring to provide for each topic an extended report on their activities, following the structure recommended in the NAR specifications.

The Belgian Safety Authority

Since 1 September 2001 the supervision of nuclear activities in Belgium is within the responsibility of the Federal Agency for Nuclear Control (FANC), which constitutes the Safety Authority. According to this law, the FANC may call upon the assistance of recognized bodies for health physics control, or on legal entities especially created to assist it in the execution of its missions. The FANC has made use of this provision and, in the case of the nuclear installations covered by this National Report (nuclear power plants and research reactors), created Bel V in September 2007 as a FANC subsidiary. Bel V is given a mandate to perform regulatory missions that can be legally delegated by the FANC. The FANC delegates different tasks to Bel V, a.o. on site routine inspections or review of the safety report by the licensees.

In the framework of the Topical Peer Review process, the FANC requested Bel V :

- To assess the specific reports provided by the licensee;
- To verify that the information provided by the licensee is comprehensive;
- To perform an independent assessment of the ageing management programs and their implementation on the specific SSCs based on the experience and the day-to-day analyses of the Safety Authority;
- To assess the appropriateness of the proposed action plan to address possible areas of improvement;
- To identify eventual gap/weakness in the Ageing Management Programs and address them by adequate new recommendations.

Bel V provided, for each topics, specific reports covering the regulator's assessment and conclusions as inputs for the NAR.

The FANC was responsible for the overall supervision of the National Assessment Report on the Topical Peer review on Ageing Management. The FANC activities consist firstly in questioning and verifying the completeness of the licensee's reports and to discuss with Bel V their assessments and reports.

Regular meetings between each licensee and the Safety Authority took place during the year 2017 in order to discuss the content of the reports but also assess the preliminary versions of the licensee's reports. Potential recommendations were particularly considered during the assessment phase by the Safety Authority. However the licensee of the nuclear power plants took the opportunity of its own self-assessment to already launch actions aiming at covering the identified gaps in its ageing management program. Therefore most small gaps identified during the TPR process are already resolved.

In this framework the NAR includes the final version of the licensee reports taking already into account the comments but also some recommendations from the Safety Authority.

Based on the self-assessment and the reports prepared by the licensees and on the assessments by Bel V, FANC redacted the National Assessment Report.

Finally the Belgian Safety Authority applied a graded approach to its licensees for the self-assessment meaning that the requirements for Nuclear Power Plants are more extensive than those for research reactors. Consequently the level of details on the ageing management is limited for the research reactors in the NAR. As a direct consequence, it was decided in order to maintain a certain readability to separate each chapter of the present report in two parts. Part A addresses the NPPs while par B addresses the BR2 research reactor.

2. Overall Ageing Management Program Requirements and Implementation

2.1 National Regulatory Framework

For the Belgian power and research reactors, the federal authorities define the regulatory framework regarding public health and safety:

- The Federal Agency for Nuclear Control (the FANC), under the supervision of the Minister of Home Affairs is in charge of enforcing the regulations for the protection of the population and the environment against the hazards of ionizing radiation and the regulations in nuclear safety. The FANC delegates some of its oversight missions to its subsidiary body, Bel V, for carrying out inspections in nuclear power plants and in other nuclear installations, and for the safety assessment of nuclear projects. Bel V is involved in all phases of a nuclear power plant lifecycle, from design and construction to operation and dismantling.
- The Federal Public Service (FPS) Employment, Labour & Social Dialogue is in charge of enforcing all applicable legislations regarding the systems' pressure boundary (integrity, control, repairs, etc.). The Authorized Inspection Agency (AIA) report to this FPS.

In addition, the Flemish and Walloon regions define the non-radiological environmental framework for plant operation for the Doel NPPs and SCK•CEN and for Tihange NPPs respectively. Finally, local authorities oversee local public services such as police, fire brigade and rescue services.

The oversight of Ageing Management of Nuclear Power plants and research reactors belongs to the competences of the FANC and Bel V. More details are given in the following paragraphs.

2.1.1 Licenses

The operation of each Belgian nuclear facility has been licensed by Royal Decree. These licenses were delivered based on the principle of fulfilling the requirements of the Safety Reports, and also contain legal provisions allowing operation:

- Compliance with Belgian regulation regarding boilers and pressure vessels
- Compliance with the Safety Analysis Reports of the nuclear units
- Periodic Safety Reviews
- Provisions for management of modifications that have an important impact on safety

It was decided initially to develop these Safety Reports, as a general principle, based on the US NRC rules (details here under).

The respect of conditions attached to the licenses is also verified by FANC and Bel V.

2.1.2 USNRC rules

It was decided in 1975 that the USNRC rules should be followed for the construction of the four last NPP units (Doel 3 and 4, Tihange 2 and 3).

Accordingly, the design and safety analysis of these units have been done following the US NRC rules and all the associated documentation (regulatory guides, standard review plans, ASME Code, IEEE standards, ANSI, ANS, etc.) in order to ensure a consistent approach. Compliance with the withheld

US NRC rules is documented in the Safety Analysis Report, deviations are identified and justified. For non-mandatory rules, the Safety Analysis Report documents how they have been implemented, in compliance with the safety objectives.

For safety-related pressure vessels, a specific derogation to the Belgian pressure vessel regulations ("Règlement général pour la protection du travail") was elaborated, in order to allow the use of the US rule based ASME Code sections III and XI. A transposition of the ASME Code has been written to cover organisational aspects like the definition of an inspector, of the Authorised Inspection Agency (AIA), etc …

2.1.3 Periodic Safety Review / License time-limit

2.1.3.1 Nuclear Power Plants

In Belgium the Nuclear License has no time limit. Since the design and commissioning, a Periodic Safety Review is conducted every 10 year to reconfirm the design basis of the plant and upgrade the safety in line with current standards. For the last PSR (2012-2015), an assessment of the 14 Safety Factors according to IAEA methodology has been performed. The last PSR includes therefore a specific chapter on the assessment of the IAEA Safety Factor on Ageing.

However, even if ageing was not assessed as a specific chapter of the first PSR (1985) and the next ones (1992-1995 and 2002-2005), identified ageing issues were addressed during these PSR. For example, for the first PSR of Tihange 1, differential settlements of the buildings, corrosion or RPV ageing were appropriately addressed.

2.1.3.2 BR2

The BR2 license included at the commissioning an end-of-life date after 30 years of operation. However in 1986 the end-of-life date was withdrawn from the license and replaced with the regulatory requirement to conduct a PSR process every 5 years. Later on the license was updated to require a PSR every 10 years similarly to nuclear power plants.

2.1.4 Royal Decrees of 20 July 2001 and of 30 November 2011

The current governing regulation regarding radiation protection is the Royal Decree of 20 July 2001 [1B] (amended) laying down the "General Regulations regarding the protection of the public, the workers and the environment against the hazards of ionising radiation". This Royal Decree transposes the European Directive on Basic Safety Standards 1996/29/Euratom (and 2013/59/Euratom in the next future) into the Belgian regulations.

The Western European Nuclear Regulators Association (WENRA) issued in 2008 the Reactor Safety Reference Levels [2G] to harmonize the safety requirements for European reactors. These reference levels have been included in Belgian regulation by the Royal Decree of 30 November 2011 [1A] on the Safety Requirements for Nuclear Installations (SRNI-2011).

Ageing Management of Systems, Structures and Components (SSCs) important to nuclear safety is addressed in Articles 10 of SRNI-2011, stipulating that the licensee of a nuclear installation shall set up an Ageing Management Program (transposition of the RL I1.1). Article 24 adds specific requirements for NPPs on Ageing Management.

These royal decrees are enforced by the FANC.

2.1.4.1 Specific Requirements in the Framework of Long Term Operation

In January 2003, the Belgian government issued the Nuclear Power Phase-out Law [1I], stipulating that all NPPs should be shut down after 40 years of operation.

In 2009, the FANC issued a "Strategic Note on Long Term Operation" [1H] including its expectations for Long Term Operation of Doel 1&2 and Tihange 1, in case the government would allow those plants to extend their period of operation [1I].

The FANC made use of article 13 of the GRR-2001 (the "General Regulations regarding the protection of the public, the workers and the environment against the hazards of ionising radiation") to propose a license amendment for Tihange 1 and Doel 1 & 2 in 2015 in order to enforce the Long Term Operation action plans. The license amendment also included particular requirements before the reactor could start its first LTO reactor cycle. The amendment to the license has been signed by the King on 27th of September 2015.

2.A Nuclear Power Plants

Although no formal Ageing Management Program existed at the time the Belgian units were commissioned (between 1975 and 1985), ENGIE Electrabel soon started ageing management activities on specific topics that appeared early in the life of the plant or that were known from the design phase.

Since then, the Ageing Management methodology developed progressively around individual projects dealing with particular degradation mechanisms, leading to the first Ageing Summaries in 1998. Based on its continuous improvement activities, ENGIE Electrabel decided to launch a formal and structured Ageing Management Program in 2004.

The Ageing Management Program ensures that safety-related SSCs maintain their high level of safety during the plant's entire lifetime, that plant availability is maximized, that ageing information is centralized and that management has a global view on the ageing management strategy and the associated cost on the medium and long term.

This program is inspired by the NS-G-2.12. and other international standards and guidance, is in line with the applicable rules and regulations and integrates ongoing ageing management initiatives in different domains. This chapter provides more details on the context and approach of ENGIE Electrabel Ageing Management Program for all of its Belgian Nuclear Power Plants.

2.A.2 International Standards and Guidance

2.A.2.1 Historical Situation

The initial design of nuclear power plants is based on the U.S. nuclear codes and regulation 10 CFR 50 relating to the entire nuclear island. The subsequent introduction of the ASME code placed a particular focus on nuclear circuits and made the safety requirements for each circuit more stringent. Since then, the IAEA developed a series of standards and guidance.

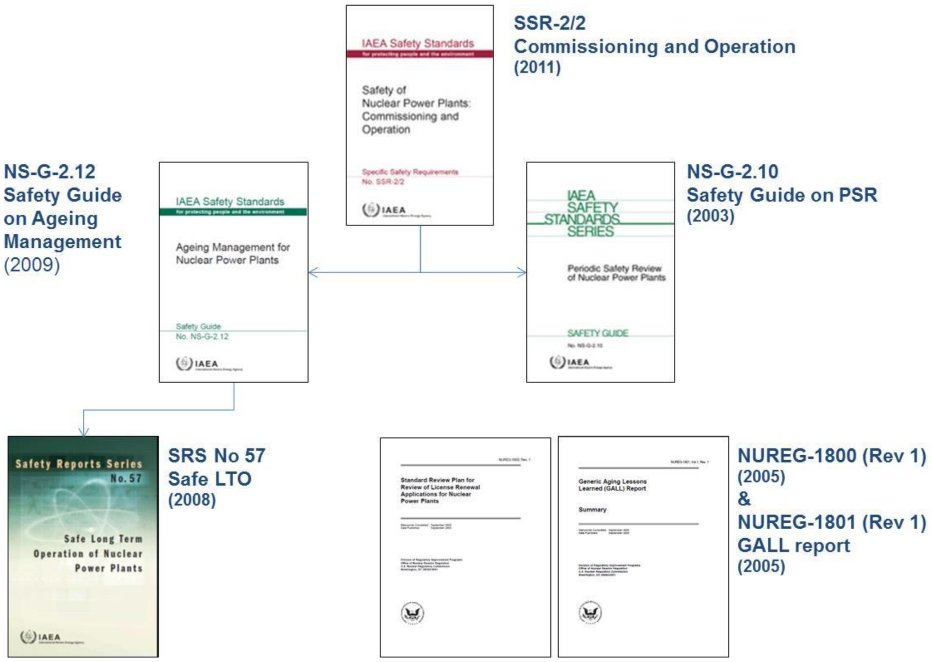


Figure 2A -1: IAEA & U.S. NRC standards and guidance related to Ageing Management and PSR considered for Belgian NPPs

In the framework of PSR, the following standards have been considered for the assessment of Ageing Management (Figure 2A-1):

Standards

- U.S. NRC 10 CFR 50
- IAEA Specific Safety Requirements report SSR-2/2 [2H]

In addition, the following guidance inspired the assessment of Ageing Management during PSRs:

Guidance

- IAEA Safety Guide NS-G-2.10 [2F]
- IAEA Safety Standard NS-G-2.12 [2C]

As requested by the FANC, the latest PSR of all Belgian NPPs has been performed according to IAEA Safety Guide NS-G-2.10 [2F], which has been superseded by Specific Safety Guide SSG-25 [2J] in 2013. In this methodology, Ageing Management is reviewed in the framework of Safety Factor 4:

"The objective of the review is to determine whether ageing in a nuclear power plant is being effectively managed so that required safety functions are maintained, and whether an effective ageing management program is in place for future plant operation."

The assessment of Safety Factor 4 for all Belgian NPPs has been performed considering IAEA Safety Guideline NS-G-2.12 on Ageing Management [2C]. This safety guideline gives practical information and guidance on how to implement requirement 14 on Ageing Management stipulated in the top-level IAEA Specific Safety Requirements report SSR-2/2 [2H]. Requirement 14 states that:

"The operation organization shall ensure that an effective ageing management program is implemented to ensure that required safety functions of systems, structures and components are fulfilled over the entire operating lifetime of the plant".

As explained in paragraph 0 of this document, the Belgian regulatory framework for Ageing Management (applicable to all Belgian NPPs) considers the WENRA Safety Reference Levels [2G] related to the Ageing Management Program. The Royal Decree was signed in 2011 and came immediately into force for articles specific to Ageing for Nuclear Power Plants (articles 10 and 24) while a transition period of 5 years was granted to nuclear research reactors.

The ageing aspect of this Royal Decree was assessed by the Safety Authority during the gap analysis set up prior to the promulgation of the Royal Decree. No gap was identified for articles specific to Ageing. The Ageing Management Program was assessed in details by the Safety Authority in the framework of the last PSRs (2012-2015 depending of the unit) and was the object of a specific inspection campaign by the FANC by the end of 2016.

LTO units

For LTO units Doel 1&2 and Tihange 1, the FANC requested to assess Ageing during Long Term Operation in line with the methodologies described in U.S. NRC's report NUREG-1800 [2L] and IAEA's safety report SRS-57 [2B].

In addition, in the framework of the LTO Study Program, physical ageing of all passive[3] SSCs in the scope of this NAR has been addressed according to the methodologies of 10 CFR 54 [2E]. A similar process was set up for addressing all active SSCs in the scope of this NAR.

2.A.2.2 Evolution of the International Standards and Guidance

Since the LTO project for Doel 1&2 and Tihange 1, and since the assessment of Safety Factor 4 "Ageing" in the framework of the latest PSR, there has been some evolution in the international standards and guidance, as illustrated in Figure 2A-2 for the IAEA standards.

As far as Ageing Management is concerned, an approved draft version of Safety Guide DS485 [2I] is now available. This new version will replace Safety Guide NS-G-2.12.

Safety Guide DS485 includes information from two additional supporting reports: the old SRS-57 [2B] (focusing on scope setting and organizational issues) and the more recent SRS-82, the International Generic Ageing Lessons Learned (IGALL) report [2D] that deals with AMPs and Time-Limited Ageing Analyses (TLAAs).

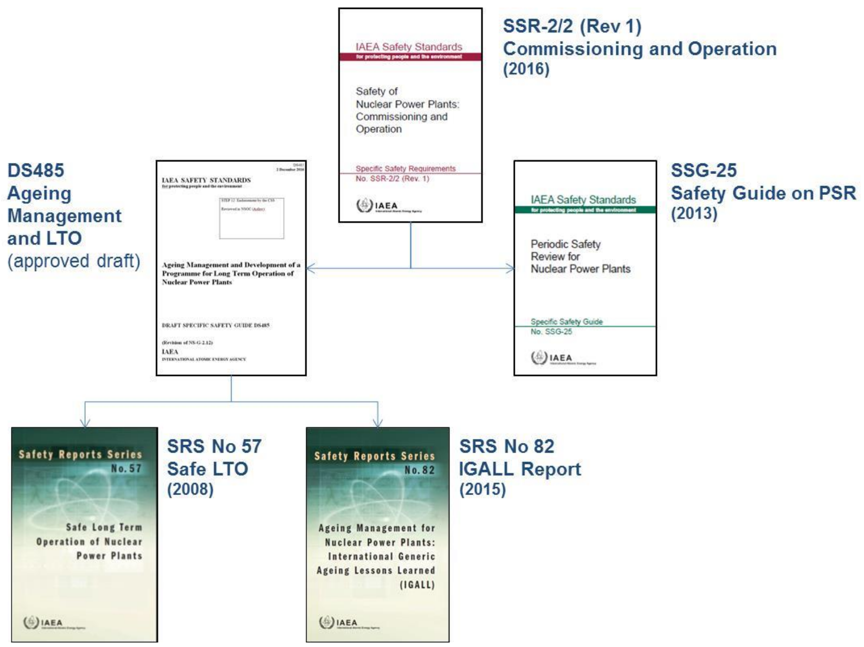


Figure 2A-2: IAEA standards and guidance related to Ageing Management and PSR as per 2017

[3] The definition of passive SSCs slightly differs between the Belgian regulatory framework (SRNI-2011) and the IAEA or American guides definitions.

The evolution of the international standards and guidance is followed and discussed by ENGIE Electrabel Ageing Management Committee to be progressively applied to all Belgian units. In particular, Safety Guide DS485 and the evolution of the IGALL report will be analysed. The current aim is to update the Ageing Management Program for units Doel 3 and 4, and Tihange 2 and 3 similar to the program installed for the LTO units, though based on these more recent standards and guidance.

2.A.3 Description of the Overall Ageing Management Program

2.A.3.1 Scope of the Ageing Management Program

The Ageing Management Program for all ENGIE Electrabel NPPs is based on six programs that are managed at site level (see Figure 2A-3):

- Maintenance Program
- In-Service Inspection Program (ISI)
- Equipment Qualification Program (EQ)
- Surveillance and Monitoring Program
- Water Chemistry Control Program
- Obsolescence Program

The first five programs are related to Physical Ageing, the latter is related to Non-Physical Ageing (outside the scope of the NAR).

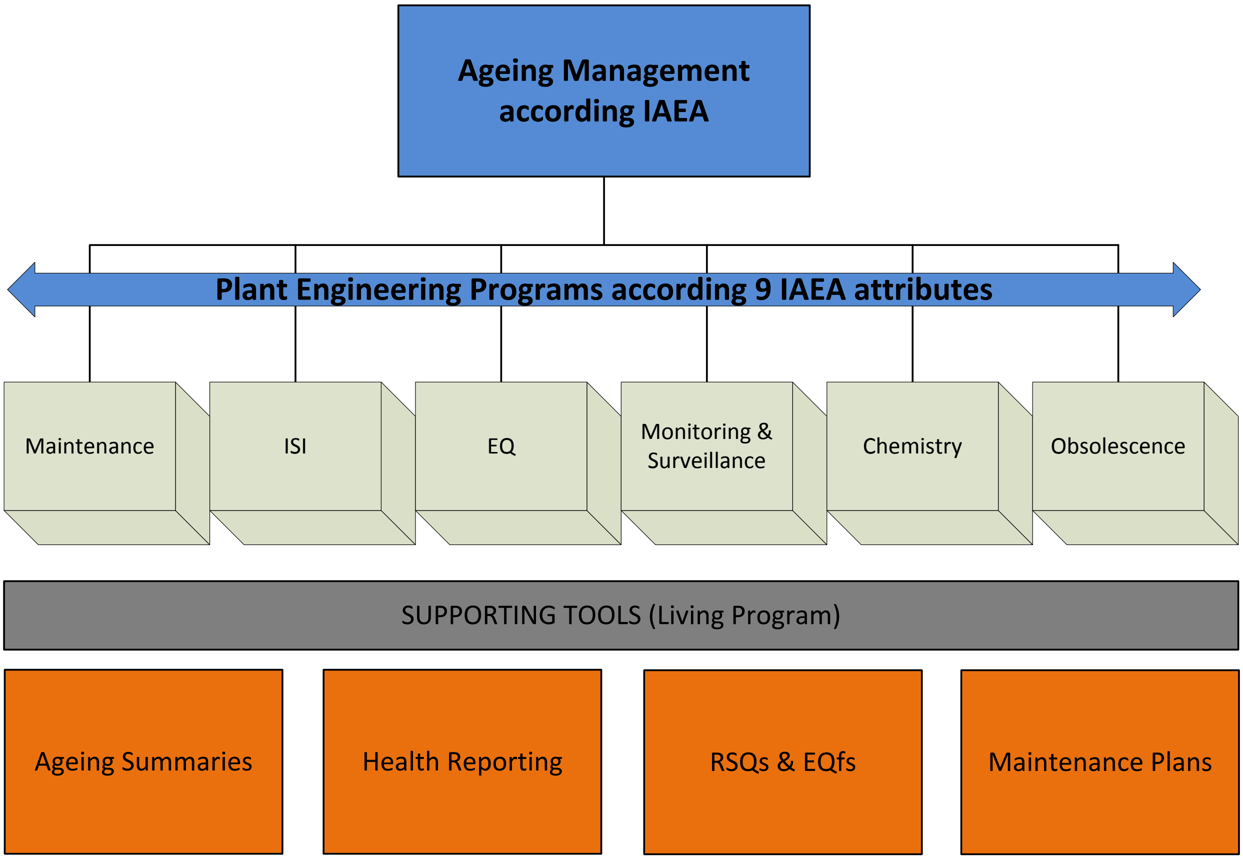


Figure 2A -3: ENGIE Electrabel Ageing Management Program for all units

2.A.3.1.1 Supporting tools for all units

The plant engineering programs are supported by a series of specific tools that keep the programs living:

- Ageing Summaries (AS)
- Health Reports
- RSQs and EQfs
- Maintenance plans

Each of these tools is explained in more detail in the following subsections.

Ageing summaries

The scope of the Ageing Management Program for all Belgian NPPs is largely based on expert judgement, driven by internal and external operating experience. The main supporting tools were the Ageing Summaries (AS).

Ageing Summaries address ageing degradation mechanisms affecting a particular component or group of components, including internal operating experience. AS cover technical and economic risks, mitigation actions, surveillance methods, acceptance criteria, and repair and replacement options. In each AS, a specific chapter is devoted to the actual situation in each of the Doel and Tihange units.

The first AS were issued in 1998. At that time, there was little guidance (from IAEA or other organizations) regarding SSC ageing management. Since then, the scope of the Ageing Management Program has grown significantly, as is illustrated by the increasing number of AS shown in Figure 2A-4.

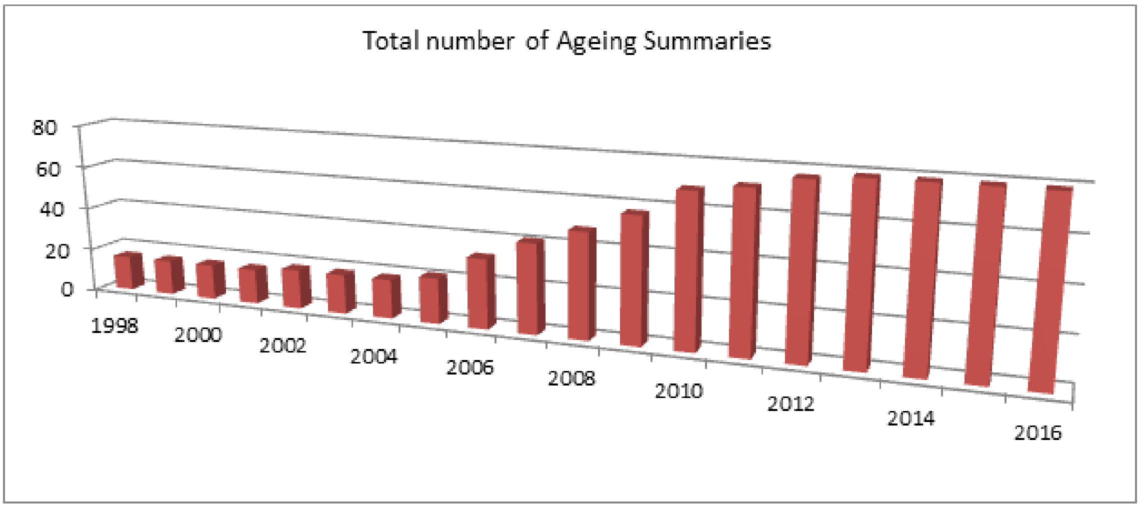


Figure 2A-4: Evolution of the number of AS covering safety-related SSCs in Belgian NPPs

Health reporting

Health Reports give an objective status of the actual condition of a system or component, based on a global analysis of collected and stored data, as well as trending of maintenance history and operational data. The reports are managed at plant level and are updated on a regular basis.

The Health Reports also define, where required, action plans to improve the health of systems or components or programs. These action plans are monitored on a periodic basis by the Plant Health Committee.

The status of a system or component as indicated in the Health Reports is a good basis for the evaluation of the effectiveness of the implementation of Ageing Management.

RSQs and EQfs

RSQs for the EI&C components

An Equipment Qualification (EQ) program is required for all equipment that must perform a designated safety function (1E) on demand and under postulated service conditions.

The ENGIE Electrabel EQ programs are based on the IEEE 323-74 standard (endorsed by RG 1.89). For each safety-related equipment, an RSQ (Rapport Synthétique de Qualification) describes the specific qualification tests (e.g. vibrations, temperature, pressure, jet impingement, radiation, humidity) that have been performed during the EQ program.

One of the objectives of the EQ program is to determine the equipment's qualified life, which is the amount of time the equipment can perform satisfactorily under normal operating conditions and during subsequent PIE (Postulated Initiating Events).

Once the equipment is qualified, it is important to preserve its qualified status. Before the end of its qualified life, the equipment must be replaced, its lifetime-limiting components must be renewed, or a new, longer qualified lifetime has to be established. This process is known as 'preserving EQ'.

To preserve the qualification of its equipment, ENGIE Electrabel has installed various onsite procedures, such as 'strategy documents' and 'letter of recommendations'. These documents describe:

- The maintenance actions that must be performed to ensure the equipment's initial qualification. Those maintenance requirements are described in the RSQ (maintenance activities e.g. replacement of coils, capacitors or gaskets, etc.).
- A check of the maintenance plans, including an analysis of the operating experience, any possible degradation or failures to verify whether the ageing effects are as predicted during the EQ program (as defined during the establishing EQ process).

The qualified life can be affected by any unforeseen change in the environmental parameters among other things. For this reason, feedback from Operations, Maintenance and System Health Reports (walk-downs) are crucial to detect any changes that could impact the qualified life. In some cases (e.g. for cables), dedicated walk-downs are performed to check the environment.

ENGIE Electrabel has installed temperature and humidity data loggers on various locations to verify whether the environmental conditions are still equivalent to the original design inputs. These measurements will be used to recalculate the qualified life, if necessary.

> ***Remark***: All qualified EI&C components of Doel 3 and 4 and Tihange 2 and 3 are covered by RSQs since the time of construction, and in line with the governing IEEE standards and Regulatory Guides. These RSQs cover all 1E functions in mild as well as harsh environmental conditions.

EQfs for the active mechanical components

Equipment Qualification files (EQfs) will be implemented for active mechanical components that need to remain operational after PIEs under harsh service conditions and a mission time longer than 3 minutes. When establishing these EQfs, the correct execution of the required maintenance activities is also verified. This tool is new and still under development.

Maintenance plans

The objective of ENGIE Electrabel maintenance program is to guarantee the availability and reliability of its NPPs' critical components (important to safety and power generation).

In order to achieve this, ENGIE Electrabel has developed robust Maintenance Plans (MPs) for its critical components, based on more than 30 years of experience. These MPs cover all preventive and remedial measures that are necessary to detect and mitigate the degradation of a functioning SSC or to restore the performance of design functions of a failed SSC to an acceptable level. The preventive maintenance activities have the appropriate frequency and extent to ensure that the levels of reliability and functionality of the plant's safety-related SSCs remain in accordance with the design assumptions and intent.

The MPs are developed and periodically reviewed based on:

- OEM recommendations
- Operational conditions
- Regulatory requirements:

Safety reports and technical specifications
RSQs or EQfs
ASME code (e.g. ISI)
Other regulations (AREI, RGPT, ARAB, etc.)
Insurers' regulations or requests

- Ageing input:

Ageing Summaries
Recently developed AMPs (Ageing Management Program) for passive components

- International and national operating experience (EPRI, NUREG, etc.)

Finally, the MPs are periodically evaluated for effectiveness in maintaining the intended function of each critical component. The effectiveness of the maintenance plan is monitored using condition-based assessment via the Health Reporting tool.

2.A.3.1.2 Specific supporting tools for LTO units

ENGIE Electrabel Ageing Management Program of LTO units Doel 1&2 and Tihange 1 is described in a Corporate Ageing Management policy document. Physical ageing of all active and passive SSCs in the scope of this NAR has been addressed in the framework of the LTO Study Program, in line with FANC's Strategic Note [1H], and according to the methodologies of 10 CFR 54 [2E], NUREG-1800 Rev.1 [2L], and NUREG-1801 Rev.1 [2A].

For the LTO units Doel 1&2 and Tihange 1, ENGIE Electrabel has applied additional tools (see Figure 2A-5):

- Scoping, screening and Ageing Management Reviews (AMRs)
- Ageing Management Programs (AMPs)
- Time-Limited Ageing Analyses (TLAAs)
- Reliability-Centred Maintenance (RCM)

These tools are explained in more detail in the following subsections.

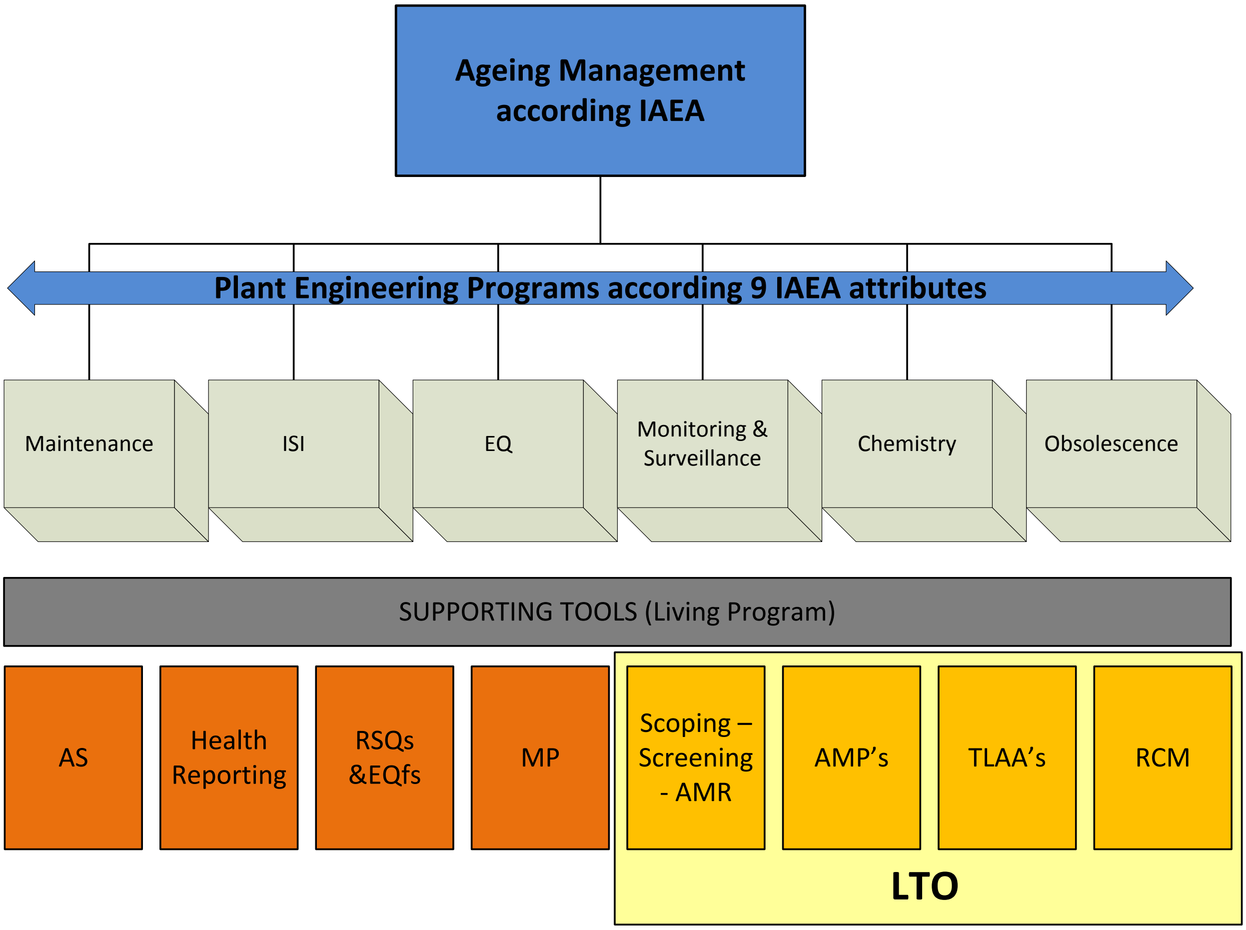


Figure 2A-5: ENGIE Electrabel Ageing Management Program for LTO units

Scoping, screening and Ageing Management Reviews (AMRs)

The LTO Ageing approach is based on the Integrated Plant Assessment (IPA) process. As is illustrated in Figure 2A-6, the first step in the LTO Ageing Management Program is scoping, identifying the passive and active SSCs that are in the scope of LTO according to the criteria set forth in 10 CFR 54.4 [2E].

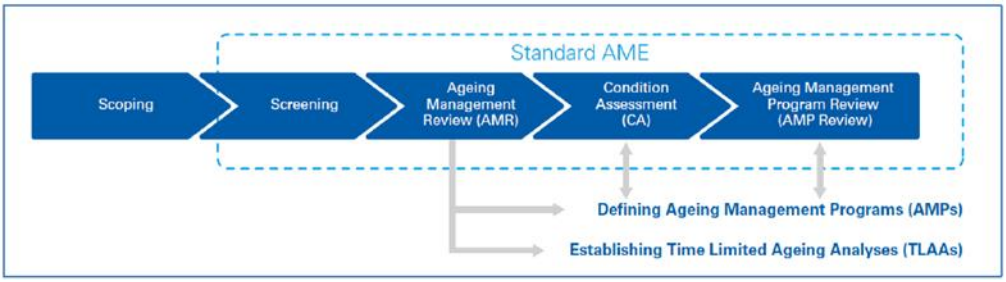


Figure 2A-6: Simplified IPA scheme

As such, the SSCs that are within the scope of LTO are:

- Criterion 1: Safety-related SSCs, which need to remain operational during and following design-basis events
- Criterion 2: Non-safety-related SSCs, whose failure could prevent safety-related components to accomplish their safety function
- Criterion 3: SSCs that accomplish specific functions such as fire protection, etc.

The definition of passive and active components is based on the U.S. NRC in 10 CFR 54.21. For passive components, the definition in U.S. NRC in 10 CFR 54.21(a)(1)(i) : "…perform an intended function, ..., without moving parts or without a change in configuration or properties…" is nevertheless slightly different from the Belgian regulatory definition: "… a component whose operation does not depend on external energy supply".

The **Scoping** phase resulted in a database containing all SSCs in scope of LTO Ageing for three domains: Mechanical, EI&C and Structural (including civil works). The scoping consisted of three steps:

- Identification of the SSCs for each of the three domains mentioned above
- An additional Specific Function Analysis investigating the impact of five specific functions (fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram, and station blackout) on the three domains mentioned above
- A cross-check of the performed scoping between the three domains

Next, a **Screening** phase was performed to identify the SSCs in scope that are subject to Ageing Management Review (AMR).

In view of the Scoping and Screening analysis in each of the three domains, the components were **grouped** as follows:

- <u>Mechanical</u>: Scoping was performed based on the systems of the plant, i.e. including any mechanical equipment such as valves, pumps etc. that is part of the system
- <u>Structures</u>: Scoping was performed based on the structure types listed in Table 2.1.5 of NUREG-1800 [2L], covering mechanical as well as civil works structures. This list is as follows:

No.	Type of structure	Structure grouping
1	Category 1 structures	Civil works or concrete structures
2	Primary containment structure	Civil works or concrete structures (Tihange 1) Primary metal containment and joints
3	Intake structures	Civil works or concrete structures
4	Intake canal	Civil works or concrete structures
5	Other non-cat. 1 structures within the scope of LTO	Civil works or concrete structures Other
6	Equipment supports and foundations	Civil works or concrete structures
7	Structural bellows	Primary metal containment and joints
8	Controlled leakage doors	Other
9	Penetration seals	Primary metal containment and joints
10	Compressible joints and seals	Primary metal containment and joints
11	Fuel pool and sump liners	Civil works or concrete structures
12	Concrete curbs	Civil works or concrete structures
13	Off-gas stack and flue	Civil works or concrete structures
14	Fire barriers	Fire protection (structural parts)
15	Pipe whip restraints and jet impingement shields	Pipe whip restrains and jet impingement shields
16	Electrical, Instrumentation and Control penetrations assemblies	Primary metal containment and joints
17	Instrumentation racks, frames, panels and enclosures	Panels, racks, frames, cabinets and other enclosures
18	Electrical panels, racks, cabinets and other enclosures	Panels, racks, frames, cabinets and other enclosures
19	Cable trays and supports	Cable trays and supports
20	Conduits	Civil works or concrete structures Primary metal containment and joints (for metallic conduits)
21	Tube rack	Cable trays and supports
22	Reactor vessel internals	See mechanical domain
23	ASME class 1 hangers and supports	Hangers and supports
24	Non-ASME class 1 hangers and supports	Hangers and supports
25	Snubbers	Snubbers
26	Load handling systems	Load handling systems

Table 2A-1: Structure types listed in Table 2.1.5 of NUREG-1800 [2L]

- EI&C: Scoping was performed based on the list of standard commodity groups and component types given in U.S. industry guideline NEI 95.10 [2M], but adapted to the Belgian context and practices:

Commodity group	Component types	Description
73	Alarm Unit	Alarm systems
74	Analysers	Moisture, hydrogen and toxic gases analysers
76	Batteries	Batteries
77	Cables and Connections, Bus, electrical portions of Electrical and EI&C Penetration Assemblies	Electrical cables and connection components
78	Chargers, Converters, Inverters	Rectifiers, inverters and stabilizers
79	Circuit Breakers	High voltage (150 kV), medium voltage (6 kV) and low voltage (230 V, 400 V and 690 V)
80	Communication equipment	The telephone system, the address system, the intercom system and the walkie-talkies used in the event of fire
81	Electric Heaters	Rod heaters for the pressurizer and heating elements in systems and rooms
82	Heat Tracing	Tracing systems whereby retaining fluid heat in the pipe is a safety function and where other, classified gear cannot achieve an equivalent effect
83	Electrical Controls and Panel Internal Component Assemblies	Control panels, relay cabinets and processing cabinets located in the relay rooms, auxiliary control rooms, main control room, emergency control room GNS and diesel generator rooms GNS.
84	Elements, RTDs, Sensors, Thermocouples, Transducers	Temperature sensors based on resistance measurement and thermocouples
85	Fuses	Fuses and fuse holders
86	Generators, Motors	Diesel and motor generators of rotating gear and actuation of valves
87	High voltage insulators	High voltage insulators
88	Surge arresters	Voltage limiter, surge arrester, lightning arrester
89	Indicators	Indicators (e.g. pressure and level indicators, and various measurement parameters)
92	Loop controllers	Controllers with feedback
95	Radiation Monitors	Instrumentation channel for radiation detection and measurement
96	Recorders	Recorders

Commodity group	Component types	Description
97	Regulators	Voltage regulators (for the start transformers)
98	Relays	Control relays, power relays, protective relays and single relays
100	Solenoid Operators	Pneumatic or solenoid actuated valves
101	Solid-State Devices	Electronic components and modules based on semiconductors, with or without programd logics or processors or (micro) computers
102	Switches	Pressure, pressure difference, level, temperature and flow switches to regulate pressure, flow, temperature and level, or to sound an alarm to generate actions, and the limit switches for indicating the position of the valves.
103	Switchgear	Electrical switch panels
104	Transformers	Power transformers and test transformers
105	Transmitters	Pressure and differential pressure transmitters used to measure pressure, flow, position, speed and level.
128	Pressure reducers	Flow/pressure reducers, positioners, converters, release valves and boosters for the pneumatic actuator of valves and registers.
129	Instrumentation tubing	Piping, couplings, manifolds, flexibles and connections belonging to the equipment contained in CG100, CG 102 Switches, and CG 128
131	Packages	Functional units

Table 2A-2: The list of standard commodity groups and component types given in U.S. industry guideline NEI 95.10 [2M]

For all passive components of the LTO units, identification of potential ageing mechanisms and their possible consequences was done through a systematic **Ageing Management Review** (AMR) as part of the Ageing Management Evaluation (Figure 2A-6). The AMR links component, component material and the environment to which the material is exposed, to the corresponding ageing effects that may apply to them. The bases for this AMR were the AMR tables included in GALL Rev.1 [2A]. Those tables consolidate OE from U.S. utilities that operate PWRs and BWRs regarding degradation mechanisms and ageing effects that may be encountered in different SSCs in the plant. They also identify which AMPs and TLAAs have to be developed to manage ageing effectively.

An **Ageing Management Program Review** was performed to surely capture any ageing mechanism specific to Doel 1&2 or Tihange 1 that would be missing in the initial AMR tables, from the results of the Condition Assessment.

Ageing Management Programs (AMPs)

In the framework of the LTO Study Program, AMPs have been developed:

- Generic AMPs, applicable to LTO units Doel 1&2 and Tihange 1, are managed at Corporate level
- Reactor AMPs, specific to the reactor and linking the recommendations of the generic Ageing Management Program to the specific reactor design, are managed at site level by System, Component and Program Engineers

The input for AMPs is provided by the GALL and experts from ENGIE Electrabel, Tractebel and ENGIE Laborelec.

Based on the AMR analysis, ENGIE Electrabel has developed additional Specific AMPs for ageing degradation mechanisms not covered in the GALL. This work has been considered in the IGALL AMR tables.

Time-Limited Ageing Analyses (TLAAs)

Original TLAAs are contained by reference in the Current Licensing Base.

On top of the AMPs, additional TLAAs have also been developed during the LTO Study Program. TLAAs are plant-specific safety analyses that consider time and ageing and involve SSCs within the scope of ageing management.

As described in 10.CFR.54, TLAAs are license calculations and analyses that meet the following six criteria:

- Involve passive SSCs in scope of the license renewal
- Consider the effects of ageing
- Involve time-limited assumptions defined by the current operating term
- Were determined to be relevant by the licensee in making a safety determination
- Involve conclusions or provide the basis for conclusions related to the capability of the SSC to perform its intended functions
- Are contained or incorporated by reference in the Current Licensing Base (CLB)

In the framework of the LTO program, ENGIE Electrabel reviewed and evaluated the TLAAs meeting those six criteria. TLAAs are managed at the plant level.

Reliability-Centred Maintenance (RCM)

In the framework of LTO Ageing, ageing management of active mechanical and active non-qualified EI&C SSCs is addressed through Reliability-Centred Maintenance (RCM). RCM results are integrated in the plants' Maintenance Programs.

The RCM approach starts with the identification of the leading failure modes for each type of equipment, followed by a criticality analysis and evaluation of the failure modes, to optimize preventive maintenance programs, surveillance and inspection programs. In this way, the identified failure modes are covered and critical functional failure can be prevented.

The RCM tool is implemented at Tihange 1 and Doel 1&2, focusing on optimal reliability of safety-related SSCs.

During the period of Long Term Operation, RCM will be periodically evaluated for effectiveness in maintaining the intended function of each SSC in scope. Its effectiveness will be monitored using condition-based monitoring via Health Reporting, integrating internal and external OE.

2.A.3.2 Ageing Assessment

2.A.3.2.1 Use of standards, guidance and manufacturing documents

The identified degradation mechanisms and their potential ageing effects are managed through AS for all units, and additionally by AMPs for the LTO units. In both cases, and whenever possible, acceptance criteria are developed in line with applicable US NRC documents or according to international standards, guidance, OE or internal R&D programs. Examples can be found in the specific topics on electrical cables, concealed pipework, RPV and concrete containment.

ENGIE is actively involved in the following working groups and international conferences related to SSC ageing management:

- International Cooperative Group - Environmentally Assisted Cracking
- Steam Generator Management Program from EPRI
- FROG Steam Generator Technical Committee
- PWROG Materials Subcommittee
- International Irradiation Assisted Stress Corrosion Cracking Advisory Committee
- Eurocorr (annual European Corrosion Congress organized by the European federation of corrosion)
- Fontevraud Symposium on Contribution of Materials Investigations and Operating Experience to Light Water Reactors Safety, Performance and Reliability, organized in France every 4 years
- International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, organized every 2 years
- International Light Water Reactor Materials Reliability Conference and Exhibition, organized every two years by EPRI
- Chemistry program from EPRI
- FROG Chemistry Working Group
- FROG Ageing and Corrosion Working Group
- IAEA IGALL Working Groups
- OECD Working Groups on Integrity and Ageing of Components and Structures

Information that is collected through the participation to conferences and working group meetings is then taken into account when AS or AMPs are updated by Tractebel or ENGIE Laborelec (at least every 5 years).

2.A.3.2.2 Use of R&D programs

Within ENGIE

Within ENGIE, R&D programs are described around 11 domains from which two are related to SSC ageing: Operational Excellence and Long Term Operation.

Operational Excellence

The research topics, focused on the improvement of plant availability and reliability, are defined in function of the daily experience of the plants and the international operating experience feedback. They are collected by interviews with the power plant managers or proposed by ENGIE Laborelec. The topics are prioritized and approved by the plants' top management. The research is performed by ENGIE Laborelec or through collaboration with research centres like CEA or SCK•CEN, or through European projects.

During the last decade, different projects regarding the improvement of component ageing management have been realized. Some examples:

- **Zinc injection in the primary circuits** has been realized at Doel 3 to reduce the primary dose rate (in 2011 for the first time). It also limits PWSCC initiation. Zinc injection has been studied during a research program in collaboration with CEA. Tests with materials used in the Belgian power plants revealed that zinc does not affect the crack growth rate. Zinc is still injected at Doel 3 for dose rate reasons coupled to PWSCC mitigation as a possible side effect.
- **Hydrogen (H_2) concentration in the primary circuits** might affect PWSCC initiation and propagation. A research program has been launched to study this effect, and ENGIE Laborelec took part in different working groups regarding this topic. Due to ambiguous laboratory results, no change in the actual practices has been recommended up to now.
- **Ethanolamine (ETA) injection in the secondary circuits** might reduce the iron transport and protect some components from Flow Accelerated Corrosion. The distribution coefficient between liquid and steam phases is better than for ammonia treatment resulting in a higher pH in wet steam phase mitigating FAC.

In addition, the limitation of iron transported in the secondary circuits might also limit fouling and denting in the steam generators.

- **Injection of dispersant (PolyAcrylic Acid) in the secondary circuits** has been tested in different phases. PAA has been used first at Doel 3 prior to shut-down and during the 2012 outage, with positive effect on the amount of sludge extracted before and during shutdown.

- **Diesel monitoring**: from fault detection to online diagnosis system. Development of an online diagnosis system for the diesel systems for predictive maintenance reasons as recommended by WANO experts.

Long Term Operation

The long-term ageing and material degradation mechanisms are part of the research domain for Long Term Operation. The R&D topics are related to the following degradation mechanisms: fatigue, corrosion, thermal effects and irradiation. As non-metallic materials, concrete and polymer/composite materials, are extensively used in power plants and have shown some ageing issues, additional research programs have been launched. For example:

- ENGIE participated in European project ADVANCE (Ageing Diagnostics and Prognostics of low-voltage I&C cables)
- An important research program in the field of irradiation embrittlement has been supported by ENGIE at the Belgian Nuclear Research Center SCK•CEN since the 1980s, as further described in the RPV chapter of this NAR

International programs

In addition to specific research topics, ENGIE takes part in different users groups meetings and periodic conferences to assure the technology watch and the integration of the findings from international research activities in the actual practices:

- Participation in the EPRI chemistry program gives access to the most recent research results and guidelines for the chemical conditioning of the primary and secondary circuits.
- Participation to the EPRI cable users group meeting.
- Participation to EPRI NDE conferences and the EPRI Steam Generator Management Program (SGMP) gives access to recent research results and operating experience from many plants in the world.

  In addition, here are some examples of practical implementations:

Development of sizing curves for Eddy Current inspections
Improvement of the steam generator inspections by the use of RevospECT, an automated analysis software

- Participation to EPRI's Checworks Users Group (CHUG) dealing with the follow-up of Flow Accelerated Corrosion.
- Participation to the European Network for Inspection and Qualification (ENIQ).
- Funding of R&D programs carried out by AREVA in the framework of the FROG Ageing and Corrosion working group gives access to research results related to various degradation phenomena such as (Primary Water) Stress Corrosion Cracking and (environmentally assisted) fatigue.
- Sponsor of SCK•CEN to take part in the INCEFA+ project addressing Environmentally Assisted Fatigue.

2.A.3.2.3 Use of internal and external operating experience

Existing plant processes have identified potential degradation mechanisms and their potential consequences since the start of plant operation. Some degradation mechanisms, for example irradiation embrittlement of the RPV, were already known at the design stage. Other ageing mechanisms appeared through time, for instance Stress Corrosion Cracking of Alloy 600, and were identified by the follow-up of internal and external OE. Since many years, ENGIE Electrabel has a process in place to manage Operating Experience (OE).

OE is collected through various sources: reports of internal events (meldingsfiches in Doel or Fiches d'Expérience in Tihange), reports of external events from external organizations (WANO, IAEA, NRC, etc.), recommendations from audits (ECNSD, QA, INSC, etc.), observation files (plant visits, etc.), seminar, peer review and workshops feedback, self-assessment reports, daily lessons learned, good practices, all internal orders, etc.

All ageing-related tools (maintenance plans, AMPs, AS, etc.) consider external and internal operating experience when they need to be updated (see section 2.A.3.3).

External operating experience process

ENGIE Electrabel has set up a Corporate Committee for Operating Experience (CCOE). This Committee meets every week and gathers OE Managers from Doel, Tihange and Corporate. The objectives of this Committee are as follows: screening external operating events and reports, adding data in a common KCD-CNT-Corporate database for OE, called OESAP, sharing internal events to the other site, and providing advice to the local committees that are in charge of external OE. These latter meet twice a month and gather the OE manager of the site and the SPOCs of the OE department on site. The purpose of those meetings is to discuss the external OE provided by CCOE and to decide whether further analysis is needed or not and if the information should be shared internally in the organization.

The ENGIE representatives that participate in a working group meeting or conference have to draft a report in which they indicate the potential links or consequences of the interesting collected information (OE) for the Belgian NPPs. The most interesting conclusions also have to be introduced in the OESAP-database with an indication of what group has to get the information and how this latter will be further processed.

When Tractebel, ENGIE Laborelec or another service provider updates or develops a new Ageing Summary, he must consult health reports and the OESAP system database to ensure that specific OE is taken into account and is analysed correctly.

Internal operating experience process

At Doel and Tihange, internal OE is recorded via 'Meldingsfiches (MF)' and by 'Fiches d'Expérience (FE)'. All MFs or FEs are reviewed during daily coordination meetings of each unit. MFs or FE's that are specific to a unit are reviewed by the person responsible for the operational coordination of this unit while the other MFs or FEs are reviewed by the Continuous Improvement Management representative. Operational coordinators determine whether MFs or FEs need to be further analysed or not and indicate which department oversees the action. Several multidisciplinary committees are then in charge of the review of MFs or FEs and of the defined analyses and actions.

Besides, internal operating experience is also collected and analysed through Health Reporting.

2.A.3.3 Monitoring, Testing, Sampling and Inspection Activities

ENGIE Electrabel Ageing Management Program relies on input from a large number of monitoring, testing, sampling and inspection activities performed on SSCs. Together, they make up the plant's Surveillance Program. This program is set up and kept up-to-date:

- In line with the plant's Technical Specifications
- In line with the applicable legislation (e.g. ASME XI In-Service Inspection program, legal inspections of pressure equipment, etc.)
- In line with instructions from manufacturers and vendors
- Following relevant internal or external OE
- According to industry codes, standards or good practices (e.g. follow-up of Boric Acid Corrosion, Flow Accelerated Corrosion, Water Chemistry program, etc.)

The objectives of the Surveillance Program are to check the proper functioning of safety-related systems, as defined in the design bases, as well as to detect any equipment degradation and malfunction in a timely manner by executing diagnostic testing and periodic inspections. The Surveillance Program is covered by many plant-specific procedures.

To identify unexpected degradation as soon as possible, many augmented inspection or monitoring programs have been launched after a first event was reported in OE. This has been the case for example with the inspection of Alloy 600 locations prone to Stress Corrosion Cracking, or the monitoring of all main transformers to detect degradation of the electrical insulation as early as possible.

For many safety-related systems, monitoring and inspection data are periodically evaluated with respect to allowable values and trending is performed to detect potential performance degradation in the framework of Health Reporting. Health Reports also cover aspects such as internal and external OE, feedback from the Ageing Management Program, Bel V inspection reports, safety and availability issues, etc. In this way, an evaluation can be made of the global actual status of the entire system and its SSCs.

Different third parties are used to monitor and test the installations, for example:

- Mandatory third parties:

Vinçotte verifies the conformity of electrical components;
Vinçotte follows the QA process for mechanical components;

- In some specific cases, third parties are contracted for their specific expertise:

Westinghouse/AREVA expertise is used for the vessel;
The original equipment manufacturer performs the maintenance and monitoring activities for diesel generators and large pumps.

2.A.3.4 Preventive and Remedial Actions

ENGIE Electrabel applies various maintenance policies:

- Preventive maintenance is performed periodically based on OE, RSQ requirements, etc.
- Predictive maintenance is typically used for domains such as vibrations, oil analyses and thermography. The maintenance policies are modified based on the results of this monitoring.

- Corrective maintenance is used whenever a component failure is detected during tests, maintenance, daily operation, etc.

The overall ENGIE Electrabel maintenance strategy is to avoid corrective maintenance. In practice, maintenance plans can be considered as the way to implement preventive actions.

Preventive and remedial actions are systematically addressed in AS and AMPs.

Examples of preventive and remedial actions are described in the chapters on concrete containment, RPV and electrical cables.

2.A.3.5 Roles and Responsibilities

ENGIE Electrabel Ageing Management Program is managed by the Corporate Ageing Manager, who defines the ageing policy and coordinates the fleet ageing program. He is assisted by the Ageing project managers from Tractebel and ENGIE Laborelec.

At site level, the local Ageing Management Program is managed by the site Ageing Manager, who is responsible for the application of the Ageing Management Program on site. Oversight of the Ageing-related topics is ensured by Site and Plant Health Committees.

At Corporate level, oversight is ensured by the Corporate Ageing Management Committee. It coordinates the objectives and priorities of the different programs in view of improving ageing management effectiveness. This Committee meets twice a year.

2.A.4 Review and Update of the Overall Ageing Management Program

This chapter explains in which way ENGIE Electrabel continuously reviews and updates its overall Ageing Management Program.

2.A.4.1 Living Ageing Management Program

The objective of ENGIE Electrabel Ageing Management Program is to ensure effective ageing management of the SSCs throughout their entire service life. Therefore, it relies on a systematic approach for coordinating all plant programs and activities relating to the understanding, control, monitoring and mitigation of ageing effects of the SSCs illustrated by the closed 'PLAN-DO-CHECK-ACT' cycle given in IAEA Safety Guide NS-G-2.12 [2C].

This 'PLAN-DO-CHECK-ACT' cycle is implemented in ENGIE Electrabel organization and processes for Ageing Management (Figure 2A-8). The applicable Corporate procedures explain in detail the role of the plant's Engineering, Maintenance and Operations departments, and of service providers such as ENGIE Laborelec and Tractebel in this continuous improvement process.

For all units, the scope of the Ageing Management Program is updated whenever relevant OE is encountered or the installation has been modified. In the modification process a checklist is used to check whether changes must be made to the monitoring processes. In case the modification requires a change of the maintenance plan, the program engineer evaluates the impact of this change on the ageing management program.

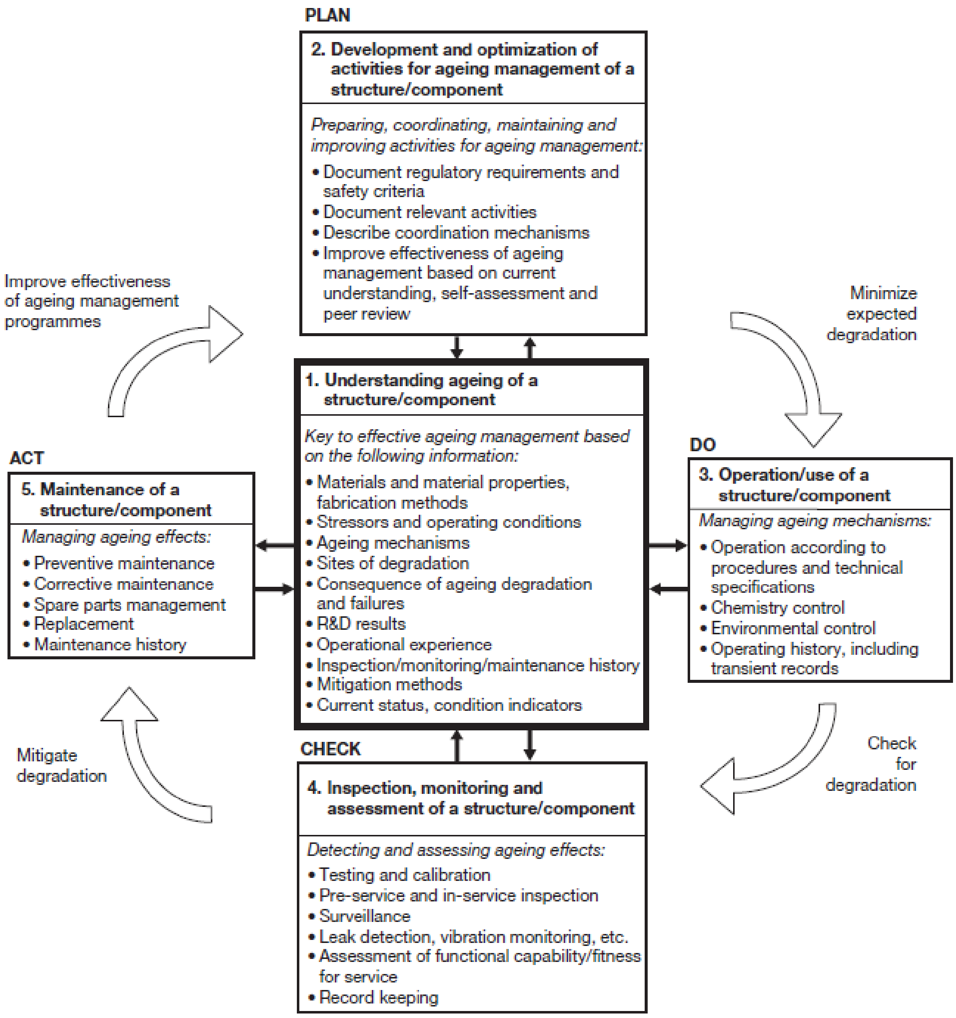


Figure 2A-7: PLAN-DO-CHECK-ACT approach for Ageing Management (taken from [2C])

For each of the supporting tools, explained in paragraph 2.3.1 of this document, a process is in place ensuring a living Ageing Management Program.

Tools for all units

- **Ageing Summaries** are periodically evaluated for effectiveness in maintaining the intended function of each SSC in scope. They will be reviewed at least every five years, or whenever relevant OE is encountered.
- **Health Reports**: The scope of system, component and program health reports evolves based on OE and new insights. All reports in scope are periodically updated and include condition monitoring of the SSCs.
- **RSQs and EQfs**: For the EI&C components, the EQ program is living; for the mechanical EQfs, the tool is launched. This is known as 'preserving EQ' (see paragraph 2.3.1.1 of this document).
- **Maintenance Plans**: The Preventive Maintenance Program is continuously updated and improved based on analyses of health reports, After Action Reviews after outages, internal and external OE, etc.

Tools for LTO units

- **Scoping, Screening, AMR**: The AMR database is updated whenever the Scoping & Screening database is updated following modifications of the installation and/or when relevant internal or external OE is encountered.
- **AMPs** are periodically evaluated for effectiveness in maintaining the intended function of each SSC in scope. They will be reviewed at least every five years, or whenever relevant OE is encountered.
- **TLAAs** are managed at plant level, according to specific plant procedure. ENGIE Electrabel Corporate monitors the publication of new or review existing TLAAs within IGALL, and transfer this information to Doel and Tihange.
- The **RCM** tool is continuously updated and improved based on analyses of health reports, After Action Reviews after outages, internal and external OE, etc.
- In addition, the **Health Reporting** tool continues the Condition Assessment that was performed in the context of the IPA process for the LTO units.

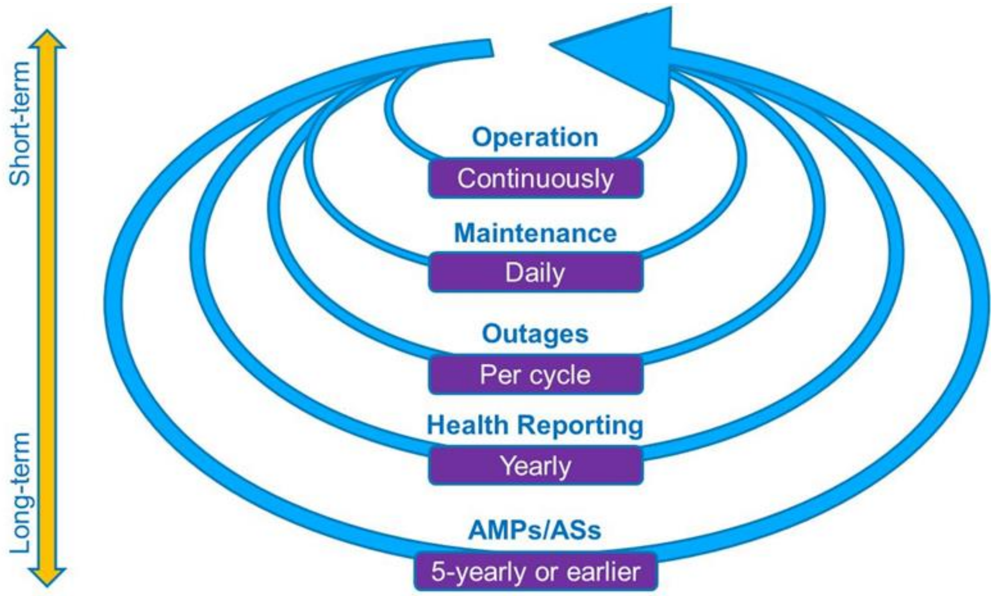


Figure 2A -8: Continuous improvement of Ageing Management in ENGIE Electrabel NPPs

Additional living short-term closed-loop processes contribute to the continuous improvement of the Ageing Management Program (Figure 2A-8):

- **Operation**: Anomalies, test results, and incidents monitored and analysed through logbooks, DCE meetings, post-job debriefings, G-factor analyses may lead to notifications and corrective actions such as adaptations to procedures and maintenance plans
- **Maintenance**: Feedback from preventive and corrective maintenance activities, post-job debriefings, re-work analyses may typically lead to corrective actions such as adaptations to maintenance plans, maintenance strategy documents, AMPs, or may initiate modification projects
- **Outages**: In a similar way, feedback on works performed during the cyclic outages may lead to similar adaptations as for daily maintenance

2.A.4.2 Review during Periodic Safety Reviews

The overall Ageing Management Program has been reviewed in the framework of the latest PSR (2012-2015), under Safety Factor 4 "Ageing" according to IAEA Safety Guide NS-G-2.10 [4G].

The assessments performed in the framework of the PSRs showed that the scope of the overall Ageing Management Program is complete. They cover all ageing mechanisms and can be managed in line with international standards (IAEA's Safety Guide NS-G-2.12).

2.A.4.3 Licensee Self-assessments and Audits

Several self-assessments and audits have been performed by ENGIE Electrabel in recent years:

- In 2011, Electrabel Plant Performance Management department performed an audit of the Ageing Management Program covering also the links with PSR, Health Reporting, performance and conditions of SSCs, and LTO.
- During the Ageing Coordination Meeting of December 2013, a self-assessment of the Ageing project was performed.
- In 2014, an internal audit on the Ageing Management Program at the Tihange NPP has been carried out by Electrabel.
- SALTO peer review missions were organized at Tihange 1 in January 2015 and at Doel 1&2 in 2017. The IAEA teams concluded that those plants have worked extensively in the field of LTO and ageing management. A follow-up mission was conducted for Tihange 1 in December 2016 and almost all issues were solved. For Doel 1&2 the follow-up mission is planned for 2019.
- In December 2016, the FANC conducted an inspection campaign on the management of maintenance and ageing at the Tihange and Doel NPPs. It concluded that the general policy is mature and conform to the regulatory expectations (Royal Decree of November 30th, 2011).
- Last PSR (see paragraph 2.A.4.2).
- ENGIE Electrabel has developed KPIs to monitor the progress of the AMP processes. The most important parameter to assess ageing management effectiveness is the status of the systems following the health reports. ENGIE Electrabel is developing new KPIs to continuously improve the monitoring.

2.A.4.4 Gap Analysis GALL versus IGALL

In view of a future update of its overall Ageing Management Program in line with current IAEA reference documents, ENGIE Electrabel performed a preliminary gap analysis between GALL Rev. 1 and IGALL [4I] in terms of AMPs, TLAAs and AMR tables (In the framework of the preconditions for LTO for the first units). The analysis revealed no significant gaps for most AMPs (mostly editorial changes). However, some AMPs, which were not part of GALL, need to be developed. AMPs are updated on a 5-yearly basis in line with the IGALL evolution.

2.A.4.5 Modifications in Current Licensing or Regulatory Framework

ENGIE Electrabel checks regulatory changes related to nuclear safety on a continuous basis through a process referred to as Regulation Watch [4U]. This process addresses changes in international requirements (e.g. IAEA), European and national regulations, and US regulations (US-Nuclear Regulatory Commission).

For the LTO-units Doel 1&2 and Tihange 1, ENGIE Electrabel used this process to identify potential design modifications caused by changes to relevant regulations or requirements that were further considered in LTO Design. This review was carried out in the framework of the last PSR.

2.A.5 Licensee's Experience of Application of the Overall AMP

2.A.5.1 Ageing Management since the Start of Operation

Although no formal Ageing Management Program existed when the Belgian units were commissioned (between 1975 and 1985), ageing management activities started very soon on specific topics that appeared early in the life of the plant or that were known from design. Typical examples of such topics are: embrittlement of the RPV, irradiation assisted stress corrosion cracking (IASCC) of baffle-former bolts, stress corrosion cracking of steam generator tubes, and fatigue due to thermal stratification in the pressurizer surge line.

Up to 2000, the Ageing Management methodology developed progressively around the backbone of existing individual projects dealing with particular degradation phenomena affecting mainly safety-related mechanical equipment. This was also the case in the domain of civil works, where in 1993 a procedure was issued for the follow-up of ageing of concrete structures belonging to seismic Category I buildings. This procedure was further developed to include also non-category I structures important for plant availability.

In 1998, the first synthesis reports on specific ageing phenomena were issued, called Ageing Summaries (see also Figure 2A-4).

Several factors led to the development and the implementation of a structured Ageing Management approach:

- Between 2000 and 2004 several new ageing issues appeared in foreign PWRs, such as PWSCC in Alloy 182/600 locations in primary components, RPV underclad defects, etc. which also had to be addressed for the Belgian NPP
- The PSR (2002-2005) of all Belgian units addressed several topics related to ageing and renovation of components
- In addition, there was a growing interest in Ageing Management of equipment that is critical for plant availability

2.A.5.2 A Structured Ageing Management Program since 2004

For all the reasons mentioned here above and inspired by the NS-G-2.12, in 2004, a formal and structured Ageing Management Program was launched, covering and integrating all ongoing ageing management initiatives in different domains.

The Ageing Management Program objectives were to ensure that safety-related SSCs maintain their high level of safety during the plant's entire lifetime, to maximize plant availability, to centralize ageing information and to give management a global view on the ageing management strategy and the associated cost on the medium and long term.

The Ageing Management Program was implemented through the following organization:

- Five pilot groups, with representatives from both the Doel and Tihange NPPs and Tractebel, addressing the ageing of primary mechanical equipment, secondary mechanical equipment, civil engineering, electrical equipment and I&C systems
- Coordination meetings ensuring coordination between the pilot groups
- A Steering Committee at the level of ENGIE Electrabel management, that decides on the evolution of the AMP, that provides the necessary resources, and that defines Ageing Management Program priorities and approves mitigation or repair strategies

The scope of the Ageing Management Program in the domain of each pilot group was mainly based on expert opinions guided by a review of different criteria such as importance for nuclear and classical safety, importance for availability, repair costs in case of failures and the consequence of a failure for the environment. Internal and external operating experience has also been an important driver of the AMP. Some examples:

- Stress corrosion cracking of Alloy 182/600 locations affecting primary equipment in NPPs worldwide triggered enhanced inspection programs, preventive measures and replacements in the Belgian NPPs
- The failure of a transformer at Tihange 2 due to early degradation of the electrical insulation triggered a close follow-up of the actual status of the entire fleet of main transformers, and initiated an Ageing Summary (AS) on transformers

As shown in Figure 2A-3, the number of AS gradually increased as of 2004. This evolution clearly illustrates the extension of the scope of the Ageing Management Program.

AS were the main deliverables of the Ageing Management Program. They address a specific ageing phenomenon affecting a particular component (or a family of components) and provide a summary of the technical issue and associated risks, surveillance techniques, remedial actions, the plant-specific situation of the component and recommendations for ageing management.

2.A.5.3 A Reinforced Corporate Follow-up since 2009

Since 2009, the Ageing Management Program is managed by an Ageing Manager at corporate ENGIE Electrabel level. However, management of both nuclear sites had the final responsibility regarding the implementation of the actions that resulted from the Ageing Management Program.

At that time the Ageing Management Program was organized as follows:

- Experts from both the Doel and Tihange NPPs, Tractebel and ENGIE Laborelec identified existing or potential ageing issues that needed to be addressed, evaluated them and proposed medium and long-term strategies to cope with them
- Ageing Management Program representatives from the Doel and Tihange NPPs, Tractebel and ENGIE Laborelec were responsible for the implementation and follow-up of the validated strategies within their own organization

  Representatives of the Ageing Management Program covered five domains: primary mechanical equipment, secondary mechanical equipment, civil engineering, electrical equipment and I&C systems. A sixth domain, "HVAC, fire and flooding protection", is currently added.

HVAC: the strategy is under development.

  - Fire: 2 AMPs exist and are updated periodically (every 5 years).
  - Flooding: the flooding protection measures that have been taken following last PSR and the stress test evaluation will be integrated/gradually in the Ageing Management Program.

- The Ageing Coordination Committee monitored the Ageing Management Program deliverables and defined priorities. For the main ageing issues, it validated the proposed strategies in view of presenting them to the Strategic Committee on Nuclear Safety Projects.

  Two main tools were introduced for the coordination of the Ageing Management Program:

The follow-up of the Ageing Management Program deliverables and priorities was done through the Ageing Master File (Excel file) managed by Tractebel and available to all participants
In a similar way, the Follow-Up Actions file was used for the follow-up of the actions that were proposed in the Ageing Summaries

- Decisions on strategic nuclear safety projects (including ageing) were taken during Nuclear Generation Team (NGT) meetings.

2.A.5.4 Enhanced Ownership and Integration by ENGIE Electrabel since 2017

With three older than 40-years units and four other units that have been in operation for 32 to 35 years, ENGIE Electrabel experiences various Ageing Management challenges. Among these challenges are the integration of LTO Ageing for Doel 1&2 and Tihange 1 into a living Ageing Management Program for those units, setting up a similar living Ageing Management Program for Doel 3 and 4 and Tihange 2 and 3, managing an increasing number of ageing-related projects of increased complexity, and closing the 'PLAN-DO-CHECK-ACT' cycle given in IAEA Safety Guide NS-G-2.12.

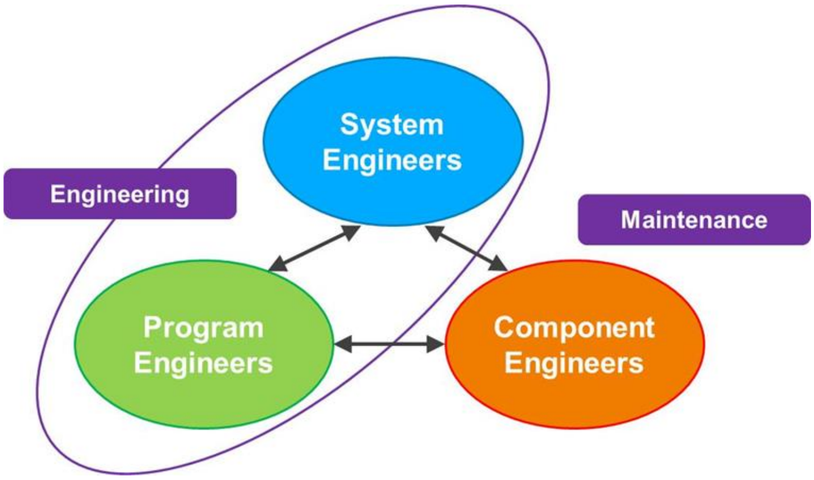


Figure 2A-9: Ageing Management relying on Program, System and Component Engineering

To address these challenges, ENGIE Electrabel will focus on the integration of the different ageing-related programs, on management of obsolescence and on compliance with the plants' design bases. To do so, ENGIE Electrabel is reorganizing its Engineering and Maintenance departments. As far as Ageing Management is concerned, the new organization relies on three major pillars and their mutual interactions: Program Engineering, System Engineering, and Component Engineering, as depicted in Figure 2A-9. The first two rely on staff from the Engineering department, the latter on staff from the Maintenance department.

- **Program Engineering** addresses plant programs such as ageing management, ISI, Flow Accelerated Corrosion, Boric Acid Corrosion, Repair & Replacement, etc., and groups the Program Owners that are responsible for the compliance of these programs with legal requirements, internal strategies, etc.
- **System Engineering** groups a team of dedicated System Engineers that fully concentrate on monitoring, reporting and correction of performance of the systems assigned to them (System Health Reports)
- **Component Engineering** brings together expertise on particular components (Component Health Reporting) and also addresses Obsolescence and supply of quality assurance components and spare parts

In the future, ENGIE Electrabel plans to move towards an Ageing Management Program for Doel 3 and 4 and Tihange 2 and 3 based on the program installed for the LTO units, and based on the most recent international guidelines. The scoping/screening will be performed using the methodology as defined in guideline IAEA NS.G. 2.12.

2.A.6 Regulatory Oversight Process

The aims of the regulatory oversight were synthetized in Section 1.B.

The overall ageing management program of ENGIE Electrabel has been the subject of a particular attention since several years. In particular:

- The overall ageing management program is discussed on a yearly base with ENGIE Electrabel since more than 10 years;
- Specific identified ageing-related concerns were considered in the framework of the PSR process since the first PSR in the 1980s.
- The overall ageing management was a specific safety study of the last PSRs in Tihange and in Doel (2012/2013 for Tihange 2 and Doel 3 and 2015/2016 for the other units). It can be noted that the topic 'Ageing' is one of the 14 Safety Factors selected in the IAEA Safety Guide NS-G-2.10 which is the guidance considered in the Belgian framework;
- Specific actions related to ageing assessment and performed by the licensee in the frame of the Long Term Operation (LTO) of Doel 1&2 et Tihange 1 were followed up by Bel V;
- FANC requests for SALTO missions at Doel 1&2 and Tihange 1 in order to confirm that the Ageing Management Program is implemented in line with IAEA standards.
- A specific inspection on the overall ageing management program has been carried out by the FANC in December 2016 in Tihange and in Doel. The followed-up inspection on this topic is planned in 2018 for both nuclear power plants.

The Safety Authority benefits from all this past experience, and therefore, no additional specific reviews have been performed for the assessment of the overall ageing management program in the frame of the TPR. Bel V has verified that the information provided by ENGIE Electrabel is in accordance with the latest information received by Bel V in the different interactions mentioned above.

In the framework of the structuration of the Global Inspection Plan by the Belgian Safety Authority that comes into force in 2018, new specific inspections on Ageing Management must be periodically conducted on both power and research reactors.

2.A.7 Regulator's Assessment of the Overall Ageing Management Program and Conclusions

2.A.7.1 International standards

The Safety Authority considers that ENGIE Electrabel effectively tries to follow the latest evolution of the international standards and guidance related to ageing management program and that the reference and standards used by ENGIE Electrabel are appropriate.

2.A.7.2 Description of the overall ageing management program

2.A.7.2.1 Scope of the overall AMP

The ageing management program for all ENGIE Electrabel NPPs is based on six programs that are managed at the site level[4]: maintenance program, ISI program, equipment qualification program (EQ), surveillance and monitoring programs, water chemistry control program and obsolescence program.

However, the supporting tools used for these plant engineering programs are a bit different and more developed for the LTO units (Tihange 1 and Doel 1&2) with respect to the other units (Tihange 2&3

[4] The ageing program of all the Tihange and Doel units is developed and managed by Engie Electrabel Corporate in close cooperation with the sites Ageing representatives and with specialists.

and Doel 3&4). In the case of Doel 3&4 and Tihange 2&3, the scope of the overall ageing management program is based on Operating Experience (OE) and on expert judgement guided by different criteria such as importance for nuclear safety and availability. For the LTO units, a specific Integrated Plant Assessment (IPA) approach was developed using additional tools such as:

- Scoping, screening and Ageing Management Reviews (AMRs)
- Ageing Management Programs (AMPs)
- Time-Limited Ageing Analyses (TLAAs)
- Reliability-Centred Maintenance (RCM)

Besides these tools, common tools are used for all units: Ageing Summaries (AS), Health reports, equipment qualification files for E&IC components and active mechanical components and maintenance plans.

The quality assurance aspects related to the scope of the AMP are not addressed in the section, but some information is nevertheless provided in the section 2.A.4.3. Moreover, this point was also covered by ENGIE Electrabel during the specific inspection on ageing and maintenance: specific Key Performance Indicators (KPIs) related to different aspects are currently developed at the ENGIE Electrabel Corporate level and they will be used to assess the quality and the effectiveness of the AMP process.

As already mentioned, the Safety Authority evaluated the scope of the overall ageing management program in the framework of the last PSRs and during the specific inspections on ageing and maintenance in Doel and Tihange (December 2016).

There are no other specific remarks on the scope of the AMP than the ones already addressed for the last PSR exercises (gap between LTO units and non-LTO units). It is a fact that some of the future challenges of ENGIE Electrabel will be to integrate into a living AMP the LTO Ageing of Tihange 1 and Doel 1&2. The current aim is to set up a similar living ageing management program for Doel 3&4 and Tihange 2&3. The improvements identified by the Safety Authority in the framework of the PSR have been gathered in an action plan.

This action plan is currently in progress and is thoroughly followed-up by the Safety Authority.

2.A.7.2.2 Ageing assessment

The identified degradation mechanisms and their potential ageing effects on all safety-related SSCs are managed through AS for all units, and additionally by AMPs for the LTO units. AS and AMPs address ageing mechanisms affecting a particular component or group of components. They cover technical and economic risks, mitigation actions, surveillance methods, acceptance criteria, repair and replacement options and operating experience (internal and external).

In both cases, and whenever possible, acceptance criteria are developed in line with applicable US-regulatory documents or according to international standards, guidance, OE or internal R&D programs (see section 2.A.3.2 for the description of these processes).

The licensee developed relevant tools to obtain an effective and adequate ageing management program. Moreover, internal and external OE is closely followed up, and the licensee is generally involved in dedicated R&D activities.

2.A.7.2.3 Monitoring, testing, sampling and inspection activities

ENGIE Electrabel ageing management program relies on the input from a large number of inspection and monitoring activities performed on SSCs. Moreover, these activities are generally addressed in the AS and/or AMPs as well.

The main activities are followed up by Bel V in the framework of specific or thematic inspections performed in Doel and Tihange NPPs. For example, regular meetings (typically twice a year) are held between ENGIE Electrabel and Bel V regarding specific mandatory inspections such as ASME XI inspections on Class 1 components.

The activities performed by "third party certification organisations" (AIA, EQB, etc.) are briefly described in the section 2.A.3.3. These activities are performed in concertation with Bel V. In this case, the roles of the third party certification organisations are described in a specific document which transposes to Belgium the regulatory aspects of ASME XI. For the other cases, these roles are described in the site procedures related to inspection and maintenance activities.

2.A.7.2.4 Preventive and remedial actions

Preventive and remedial actions are systematically addressed in the AS and the AMPs. For this TPR exercise, links are just made to the specific chapters on RPV, concrete containment and electrical cables. The link between maintenance plans and these actions is briefly described by ENGIE Electrabel in the section 2.A.3.4.

Also, it can be noted that ENGIE Electrabel will focus more on the integration of the different ageing-related programs, on management of obsolescence and on compliance with the plants design bases by reorganizing its Engineering and Maintenance departments in 2017 (see section2.A.5.4). By doing so, the link between Program Engineers, System Engineers and Component Engineers is clearer and will facilitate the exchange between departments.

2.A.7.3 Review and update of the overall AMP

In the framework of the last PSR of the 7 Belgian NPPs, the status of the ageing management program was assessed by ENGIE Electrabel (see section 2.A.4.2). The Safety Authority reviewed these assessment reports and transmitted their synthesis report afterwards to ENGIE Electrabel.

Globally, topics such as the overall ageing management program, the scope and the results of the LTO exercise (if applicable), the quality of the AS and the AMPs, the physical obsolescence and the results for the most important SSCs were assessed by ENGIE Electrabel and reviewed by Bel V. Out of these assessments, many strengths and areas of improvement were issued and the most significant ones were retained to be incorporated in action plans validated by both ENGIE Electrabel and Bel V.

Here are some of the main conclusions related to the overall ageing management program within ENGIE Electrabel:

- For Tihange 1 and Doel 1&2, the LTO Ageing project upgraded the global AMP by means of a systematic and comprehensive process (scoping, screening, AMR, AMPs, condition assessment, etc.). This is currently not the case for Tihange 2&3 and for Doel 3&4. However, ENGIE Electrabel takes currently action for these units to move towards an ageing management program based on the program installed in the LTO units and on the most recent guidelines.
- As mentioned in the section 2.A.4.1, ENGIE Electrabel relies on a systematic PLAN-DO-CHECK-ACT approach (given in IAEA Safety Guide NS-G-2.12 [2C]) for coordinating all

plant programs and activities related to the understanding, control, monitoring and mitigation of ageing effects of the SSCs. The Corporate procedures now explain in detail the different roles of the plant's Engineering, Maintenance and Operations departments as well as service providers like ENGIE Laborelec and Tractebel. For the non-LTO units, actions and recommendations issued from the AS and the system health reports (SHRs) were finally integrated into the PLAN-DO-CHECK-ACT approach to make the overall ageing management program even more living and effective.

- A certain number of AS still needs to be improved in order to obtain the required level of maturity. For the non-LTO units, AS have the same role as AMPs but the structure is just a little different. However, the attributes of these AS are relevant (identification and evaluation of potential ageing degradation, effective program for timely detection and mitigation of the ageing processes and/or degradation effects, acceptance criteria to determine the need, the type and timing of the corrective actions, etc.).

- In terms of organization of the ageing management program, it was noticed that this was not always adequate at the ENGIE Electrabel sites (Doel and Tihange) and Corporate to implement recommendations of the IAEA Safety Guide NS-G-2.12 [2C] (Doel and Tihange procedures were not in line with the Corporate AMP procedure, for example). Consequently, reorganization between the departments involved in the ageing management program is taking place in 2017 on-sites (see section 2.A.5.4 for more details).

- The SALTO missions of the IAEA performed in Tihange 1 (2015) and in Doel 1&2 (2017) also defined some recommendations. For each site, these will be incorporated in specific action plans: the program for LTO and ageing management related activities (such as the maintenance program) should be followed-up in a systematic manner to ensure that the required safety functions of SSCs are fulfilled over the plants entire LTO period. These actions and recommendations were taken into account by ENGIE Electrabel (see sections 2.A.3.5 and 2.A.5.4 for more details).

2.A.7.4 Licensee's experience of application of the overall AMP

ENGIE Electrabel describes in section 2.A.5 the evolution of its overall ageing management program and the reasons for which it is still evolving:

- The integration of LTO Ageing for Doel 1&2 and Tihange 1 into a living ageing management program is required;

- The ageing management programs of Doel 3&4 and Tihange 2&3 is currently being updated based on the program installed for the LTO units, and using the most recent international standards and guidance.

Taking these modifications into account, the overall ageing management programs will be adequate once the (minor) actions from the PSR exercises will be carried out.

As already mentioned, the adequacy of the application of the current overall ageing management program has been assessed during a specific inspection by the Safety Authority in December 2016. It was observed that several tools were recently developed or are still in development in the frame of the overall ageing management program. The Safety Authority recognized that these tools would provide a beneficial contribution to the ageing management in the future. However, as a consequence of the very recent development of these tools and of their relative poor maturity, it was felt difficult to judge their efficiency. Therefore, the Safety Authority planned to carry out a new specific inspection on this topic in the coming years.

2.A.7.5 Conclusions

The ageing management program by ENGIE Electrabel is in line with the international standards (particularly the IAEA Safety Guide NS-G-2.12 [2C]) and should ensure an adequate management of the ageing of the safety-related SSCs during the rest of lifetime of the NPPs.

The Safety Authority considers that recent and still ongoing reinforcement of this ageing management program in the framework of the LTO program for the three first units is a significant improvement. This new ageing management program for the LTO units (Tihange 1 and Doel 1&2) is considered as complete and based on a systematic and comprehensive approach. The current planned extension of this program to the ageing management program of Tihange 2&3 and Doel 3&4 based on the program installed for the LTO units, and using the most recent international standards and guidance, is positive but is a challenge for the forthcoming years.

Nevertheless some past and recent events still highlights ageing issues and an inadequate monitoring of the ageing of some components and of the remedial actions in the Belgian Nuclear Power Plants (i.e. a recent case of concrete degradation in a building housing auxiliary systems at Doel 3 in October 2017[5]). ENGIE Electrabel and the safety authorities have decided to investigate whether and how the ageing programs need to be adapted.

The Safety Authority acknowledges that the reinforced ENGIE Electrabel Corporate follow-up since 2009 and the enhanced ownership in Tihange and Doel since 2017 allowed a better integration of all the ageing management activities in the different domains of the mechanical, electrical, I&C and civil structures domains.

The SALTO missions conducted by IAEA in Doel and Tihange concluded that the Ageing Management by ENGIE-Electrabel is globally implemented in line with international standards.

It can be concluded that the overall ageing management program is in line with international standards but that its effective implementation on site is improvable by ENGIE Electrabel. The Safety Authority considers that the licensee has taken some necessary steps in this direction. Some tools have been recently developed and their efficiency will be judged on the long term.

Although this topic has been thoroughly followed-up by the Safety Authority in the recent years (PSRs, regular meetings with the licensee and specific inspections), the Safety Authority is aware of the increasing importance of the ageing management, consequent to their age, on the safety of the nuclear power plants. In consequence, the Safety Authority plans to continuously enhance the follow-up until the final shut-down and the dismantlement of the nuclear power plants.

Finally, the Safety Authority underlines the added value of the Topical Peer Review leading the licensee ENGIE Electrabel to conduct complete and constructive self-assessment of their ageing management program, and to proactively take measures identified as adequate by the Safety Authority in order to cover the few identified gaps regarding the international good practices.

[5] For more details, see the FANC website

2.A.8 Action plan

In the framework of this TPR, no additional action or improvement has been identified by the Safety Authority for the overall ageing management program. The Safety Authority considers that on this topic the ongoing action plans set up in the framework of the last PSRs (2012-2015) or of the LTO for the first units, in addition to the actions already performed in 2017 by the Licensee arising from its self-assessment in the frame of the TPR, are sufficient to achieve a complete ageing management program.

2.B BR2 Research reactor

2.B.2 International standards

For defining the overall Ageing Management Program, the following IAEA standards have been used by SCK•CEN :

- The use of a graded approach in the application of the safety requirements for research reactors (in version Safety Standard DS351, 20 November 2009);
- Ageing management for research reactors (SSG-10, 2010);
- Application of reliability centred maintenance to optimize operation and maintenance in nuclear power plants (IAEA-TECDOC-1590, May 2007);
- Guidance for optimizing nuclear power plants maintenance programs (IAEA-TECDOC-1383, December 2003);
- Implementation strategies and tools for condition based maintenance at nuclear power plants (IAEA-TECDOC-1551, May 2007);
- Maintenance, periodic testing and inspection of research reactors (Safety Guide NS-G-4.2, 2006).

2.B.3 Description of the overall ageing management program

2.B.3.1 Scope of the overall AMP

Responsibilities

The overall Ageing Management Program is conducted by a dedicated group of persons (Plant Asset Management – PAM) under the final responsibility of the BR2 reactor manager. This group was launched in 2012-2013 in the framework of the preparation for a LTO for the BR2 as well as the preparation for the last PSR and the transposition of the WENRA Reference levels in the Belgian regulation. The PAM group started the Ageing Management Program project by making the classification of all SSCs. This classification was done in multidisciplinary meetings with persons from operation, maintenance and safety groups. In a second phase a more detailed analysis is made to define the inspection and maintenance strategy for all SSCs with safety importance. Support is given by the groups responsible for the maintenance (mechanical, electrical, nuclear instrumentation and non-nuclear instrumentation). In a final step, maintenance and inspection procedures are developed by the groups responsible for the respective SSCs.

All information about the Ageing Management Program is kept in databases maintained by the PAM group.

Scope

The project includes all structures, systems and components which have an importance for the operation of the reactor. This includes not only on the SSCs important to safety, but also SSCs related to reliability such as the cooling towers. However, the classification is made such that a safety related item always gets priority. The fire detection and firefighting equipment and the security system is not included in the BR2 AMP, since these assets are under responsibility of the department of health physics and safety. Assets not directly related to the operation of the reactor, such as workshops and storage areas, are not included either.

Methods used for identifying SSCs with the scope of overall AMP

The method for identifying SSCs is known as the Asset Configuration Management (ACM). The process started with the choice of a general subject, such as the mechanical parts of the main primary loop. All SSCs were listed using the flow sheets of the installation. These were recently checked and updated. With this information, meetings were organized with personnel from maintenance and operation in order to classify all these assets. During these meeting three scores were given to the listed SSCs:

- Score on safety
  - Score 0 for a safety critical asset - the asset could be cause of an accident or a significant release of radioactivity;
  - Score 1 for an asset that is important to safety – failure means the reduction of a defense in depth barrier;
  - Score 2 for assets without safety importance.
- Score on reliability
  - Score 1 for an asset that would cause a long shut down of the reactor (more than 1 month);
  - Score 2 for an asset that would cause the loss of 1 operation cycle of the reactor (between 5 days and 1 month);
  - Score 3 is given to assets that are straightforward to replace and would cause only a short shut down (less than 5 days).
- Score for economy and replaceability
  - Score 1 is given to an asset that is not replaceable (or very difficult to replace);
  - Score 2 is given if the replacement cost is high;
  - Score 4 for an asset with a medium replacement cost;
  - Score 6 for an asset with a low replacement cost.

A remark must be made concerning the safety score. The score is not always determined by the failure to function. For example, a primary pump is not required for safe shutdown of the installation, however the pump is a part of the pressure retaining boundary.

Grouping methods of SSCs in the screening process

In order to obtain a grouping of all SSCs the score for safety (SS), for reliability (RS) and for replaceability (RS) are multiplied. In doing this, a figure (Total Impact TI) between 0 and 36 is obtained. This figure is used to classify all assets into 4 classes:

- Class A: $0 \leq TI \leq 4$
- Class B: $5 \leq TI \leq 18$

- Class C: 19 ≤ TI ≤ 24
- Class D: 25 ≤ TI ≤ 36

This classification has the following properties:

- Components which are important to safety will always be in class A or class B.
- Assets with a very high importance for safety will always be in class A, even if their replacement cost is low and if they can be replaced on the spot.
- Classes C and D are components which play a role for the reliability of the installation.

This classification is shown in Figure 2B-1

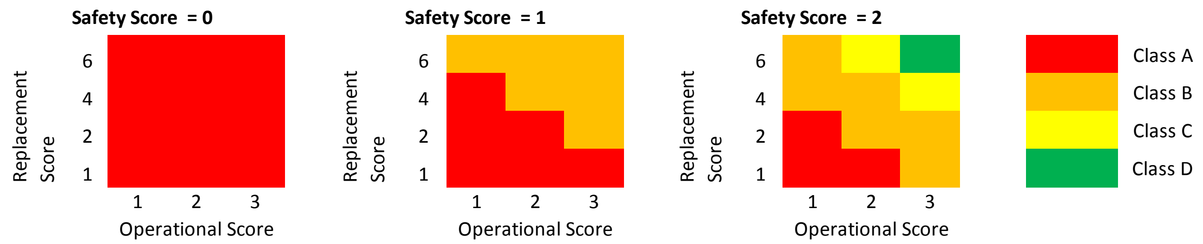


Figure 2B-1:The classification of components according to their score

The distribution of the different SSC according to these classes is given in Figure 2B-2.

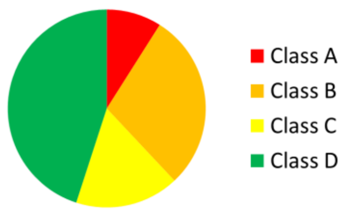


Figure 2B-2: The distribution of the classification of different components

Nearly half of the assets (47%) are in class D which means that they have no safety importance, have a limited impact on the reliability and are straightforward to replace. In this way major attention is given to SSCs important to safety (classes A (9%) and B (29)).

Some examples of the classification of components are:

- Class A: the valves in the primary loop that, in case of loss of coolant, isolate the part of the loop inside the reactor building from the part outside.
- Class B: instrumentation gauges that measure parameters important for safety (pressure, temperature, flow), but for which a redundancy exists.
- Class C: service building ventilators.
- Class D: cooling towers.

Ageing management required by the operating license and other legal requirements

The actual operating license of BR2, which dates from June 1986, has a number of requirements for ageing management of specific components. These conditions are:

- Periodic testing of the leak tightness of the reactor building;
- Periodic testing of the leak tightness of the penetrations through the reactor building;
- Monthly functional testing of the systems for reactor building isolation;
- The neutron dose of guide tubes for control rods is limited;
- Follow up of deformation and cracks of the beryllium matrix;
- Inspection of the vessel in case of replacement of the beryllium matrix.

General regulation on the safety of industrial installation has also a number of requirements on follow up of ageing. Amongst these are:

- Periodical inspection of pressure vessels;
- Periodical inspection of steam vessels (pressurizer and pre-heater);
- Inspection of electrical installations.

Introduction of the ageing management program

In the past, most of the test and maintenance actions were decided by the groups responsible for the components. It was mostly done using their expertise and judgement. A list of important inspections was available in the safety assessment report.

However, since 2011, Ageing Management of Systems, Structures and Components (SSCs) important to nuclear safety is addressed in Articles 10 of SRNI-2011 [1A], stipulating that the utility shall set up an Ageing Management Program (transposition of the RL I1.1).

In the framework of the strategic note describing the fundamentals expectations of the Belgian Safety Authority for a Long Term Operation for the Belgian research reactors published by the end of 2011, the FANC explicitly requested the Licensee to develop an overall Ageing Management Program.

Consequently, the formal Ageing Management Program (known as Plant Asset Management – PAM) for BR2 started in 2011 **and is still under development**. The implementation of the ageing management program (including the work order documents for class A and B components) has to be completed before June 2019.

Quality assurance

All documents related to the PAM project are stored in the general document management system. This system gives each document a unique identification. It also keeps track of all changes to a document and each day a back-up is made. The documentation will further be completed with older information about certain SSCs.

All classifications are verified by the department of health physics and safety, in accordance with their tasks defined in the regulatory framework. Special attention is paid to components important to safety.

2.B.3.2 Ageing assessment

The Installation Concept Management (ICM) phase of the PAM project defines the further actions for testing and maintenance of the class A and B components. For these components a risk based maintenance concept is used. The class C and D components are not subjected to formal ageing management. For Class C components only a review of the maintenance and replacement is made, so that it can be executed if necessary. These assets are subjected to "Run to Failure". Class D components undergo only on the spot maintenance, repair or replacement.

Maintenance concepts and risk profiles for Class A and B components.

All safety related components are analyzed using Reliability Centered Maintenance (RCM) approach and Failure Mode Effects and Criticality Analysis (FMECA). Time-dependent evaluation and ageing is taken into consideration. Following questions are to be answered during the evaluation process:

- What are the functions of the SSC? Some SSCs have a single function, others can have multiple functions. A typical example is a pump, which has to provide a specific flow rate and pressure, but which is also part of the pressure retaining boundary;
- What are the failure modes for each of the functions of the SSCs?
- What are the possible causes for each failure mode? Failure of the considered SSC due to a fault in another SSC is also taken into account;
- What are the indicators of each failure mode? There could be several different indicators for each failure, or some failures can remain hidden for a long period;
- In what way does each failure mechanism matter? Some failures will only lead to a disturbed operation or the stop of the installation, while others can increase the probability of an accidental condition. The probability of failure (depending on the mean time between failure) is taken into account;
- Is the technology used by the SSC obsolete or will it become obsolete in the near future? This question is of particular importance due to the age of the installation. A positive answer to the question will have an impact on the maintenance measures. It may even lead to a design modification of the installation, if maintenance or replacement is no longer possible;
- What can be done to prevent each failure mechanism? The answer to this question is the definition of a preventive maintenance program. Where available, codes and guidelines will be used. However in a number of cases, engineering judgement needs to be used;
- What are the corrective measures that can be taken in cases where preventive maintenance cannot be done or in the case of unexpected failures? This item defines the requirements for stocks of spare parts or the identification of potential suppliers for replacement parts;
- In the previous questions, it was assumed that no redundancy is available. It was assumed that the failure of the SSC leads to abnormal conditions. However, in most cases a certain redundancy present and the unavailability of certain SSC's for a limited period of time is acceptable. This makes repair possible. Maximum unavailability periods of important SSCs are given in the Safety Assessment Report.

Preventive maintenance tasks

A preventive maintenance (PM) task is defined for failure modes with a major impact. The definition in asset classes is further detailed with a parameter that takes the Mean Time Between Failures (MTBF) into account. A score for Probability of Failure (PF) is given to each failure mode of a safety related asset. The PF score is given according to Table 2B-1:

PF Score	Qualitative meaning	Relation to $MTBF$
1	Very probable	$MTBF \leq 3$
3	Probable	$3 < MTBF \leq 10$
10	Improbable	$10 < MTBF \leq 30$
30	Very improbable	$MTBF > 30$
1000	Impossible	$MTBF \approx \infty$

Table 2B-1: Definition of the Probability Failure (with MTBF expressed in "years")

The risk score can then be calculated as the product of all scored parameters:

$$Risk\ Score = probability \times consequences = PF \times (SS \cdot OS \cdot RS)$$

Where *PF* : Probability of Failure; *SS* : Safety Score; *OS* : Operational Score and *RS* : Replaceability Score.

Every different failure mechanism of an SSC may lead to different consequences, thus it is necessary to analyze them all independently. In case the function that fails due to a specific failure mechanism is redundant, i.e. some other asset of the installation takes over the function, the failure shall be scored as if there was no redundancy. However the fact that there is redundancy will be taken into consideration, so that the information will be registered and can be used to choose an appropriate maintenance strategy to prevent the failure of the function.

The risk score allows to define 4 categories of failures, according to the Table 2B-2:

Risk Category	Relation to Risk Score (RS)
Red	$RS = 0$
Orange	$0 < RS \leq 180$
Yellow	$180 < RS \leq 240$
Green	$RS > 240$

Table 2B-2: Risk Categories for failures of Class A and B SSC's

For those failure mechanisms of SSCs within the red category, all kinds of preventive maintenance (PM) tasks are proposed (especially considering Predictive Maintenance (PdM), that involves condition monitoring, trend analysis and failure prediction) based on or adapted from the guidelines given by the standard maintenance codes. When the standard codes are not applicable or adaptable to the specific case of BR2, PM tasks are proposed according to the engineering judgment and experience available in the PAM team and the rest of the BR2 and SCK•CEN personnel.

For those SSCs within the orange category, all kinds of PM tasks except PdM are foreseen in the same way. For failure mechanisms within the yellow and green categories no dedicated preventive maintenance program is foreseen.

The ICM documents

All SSCs belonging to class A or B are reviewed in detailed. For each SSC (or type of SSC) an individual assessment is made by the PAM group in cooperation with the maintenance groups. This document collects all available information (ageing mechanisms, consequences, acceptance criteria, construction standards) of the SSC. Based on this documents the maintenance groups develop working procedures.

Use of R&D programs

Dedicated R&D programs were only used for two components:

- The mechanical properties of irradiated aluminium;
- The behaviour of beryllium under neutron irradiation.

The program for aluminum was necessary as a support to assess the life time of the vessel. More details are given in the dedicated chapter. On the occasion of the replacement of the first beryllium core, knowledge of irradiated beryllium was limited. At this moment sufficient information is available for the life time determination.

For all other SSCs sufficient information is available and no dedicated R&D programs are necessary.

Return of experience

Internal experience is documented on two levels. First of all, each maintenance group keeps the detailed operation history of the SSC's under their responsibility. If a SSC is the cause of a disturbance in the operation, the information is documented in the general Non Conformity Database.

Due to the unique design of the reactor, experience from other installations can very rarely be used.

2.B.3.3 Monitoring, testing, sampling and inspection activities

Monitoring, testing, sampling and inspections are executed according to current standards. The following general standards are used:

Mechanical components

For mechanical components the follow up of the SSCs is based on the ASME code. Rules are followed as good as possible in practice. The used inspections are :

- Internal inspection;
- External leakage inspection;
- Internal leakage inspection;
- Indication testing where the component is monitored using its own instrumentation, such as the position indicator of a valve;
- Valve stroke and pump startup time testing;
- Structural integrity inspection;
- Exercise testing.

These sets of inspections are completed by the set of legally required inspections, prescribed by general legislation or by license requirements. Further inspections and tests come from recommendations by the suppliers.

Electrical and electronic assets and components

The maintenance approach for electrical assets adopted by BR2 in the frame of the Plant Asset Management project is adapted from the Recommended Practice for Electrical Equipment Maintenance

2010 Edition, which is developed by the American National Fire Protection Association (NFPA®), an International Codes and Standards Organization. The maintenance approach was discussed in meetings with the engineers responsible for the electrical maintenance of the BR2 installation. The maintenance tasks and frequencies are defined for each of the different categories of electrical assets or SSCs, together with a reference to the concerned paragraph of the recommendation in which the tasks are explained to some extent.

As with the mechanical components, this set of inspections is also completed with legal requirements and recommendations by the supplier.

2.B.3.4 Preventive and remedial actions

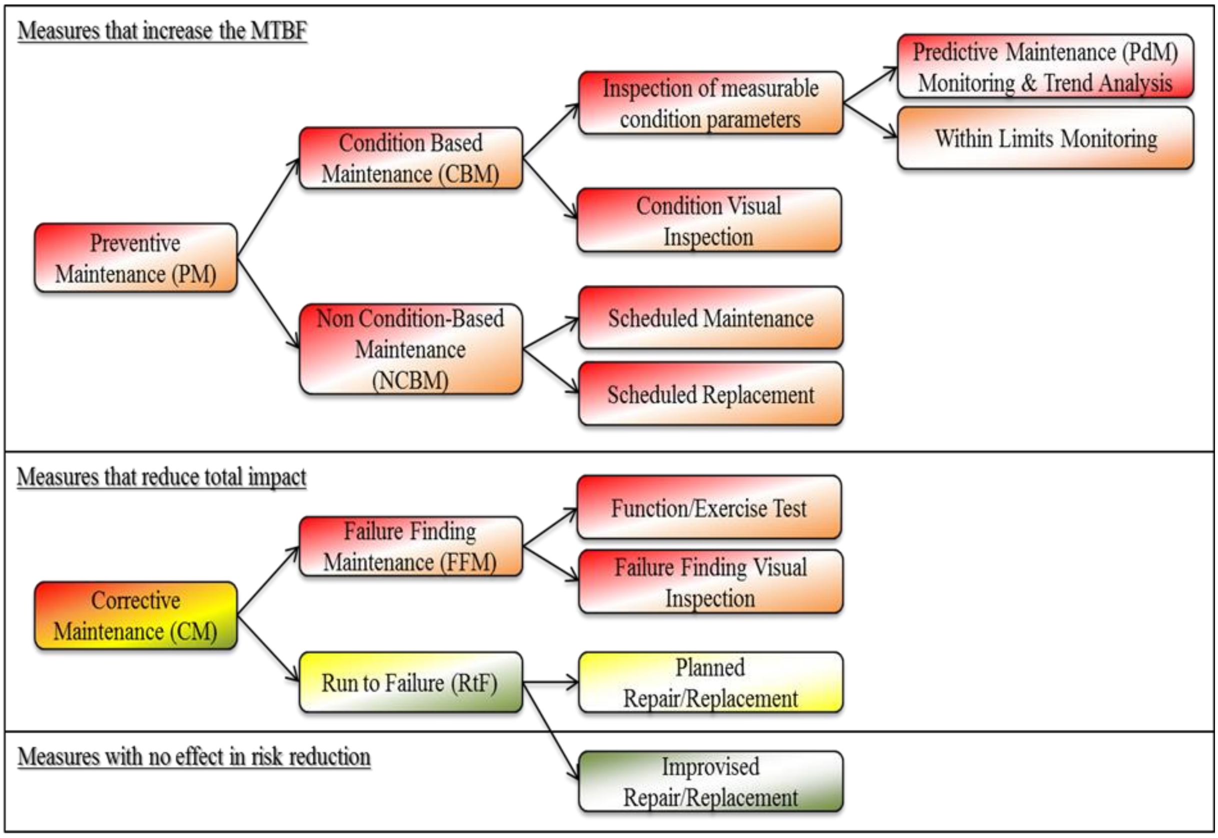


Figure 2B-3

Different types of maintenance concepts are used to prevent the failure of a component. The type of maintenance depends on the classification of the component. As it is shown in the Figure 2B-1, the maintenance measures can be divided in two big groups: **Preventive Maintenance (PM)** and **Corrective Maintenance (CM)**. The PM measures are those that help in increasing the Mean Time Between Failures (MTBF), i.e. reducing the probability of a failure happening by preventing it, while CM measures are those that help in reducing the consequences or the impact of the failure once it has happened. The colors of the boxes in *Figure 2B-1:The classification of components according to their score* reflect those of the risk categories color code in *Table 2B-2: Risk Categories for failures of Class A and B SSC's* in Section 0.

Preventive maintenance

Preventing a failure from happening can be achieved by two different maintenance strategies: **Condition-Based Maintenance (CBM)** and **Non Condition-Based Maintenance (NCBM)**.

Condition-based maintenance

CBM is maintenance when need arises. This kind of maintenance is performed when one or more indicators show that equipment is going to fail or that equipment performance is deteriorating. CBM is based on using real-time data to prioritize and optimize maintenance resources. Such a system will determine the equipment's health, and act only when maintenance is actually necessary.

To achieve that, every CBM task has three action phases: detection, analysis and correction.

In the detection phase, the assets that are in a deteriorating condition that can lead to a future failure shall be identified. For that, a scheduled program of inspections has to be organized in order to be able to detect or monitor the deterioration processes before the equipment fails. The inspections can be either visual, in which case clearances, settings, physical displacements, discontinuities, imperfections, loss of integrity, lost or missing parts, and signs of corrosion, wear or erosion can be investigated; or based on measurable equipment condition parameters, e.g. wall thickness or flaw size.

The objective of the analysis phase is to determine the exact condition of the equipment and defining the cause of the deterioration. Furthermore, it is in this phase when the decision to restore the equipment to its proper condition is taken. The decision can be taken either establishing a condition limit that will trigger the maintenance actions once it has been exceeded, or using the data acquired to analyze the trend of the condition to predict the future failure and take the actions before it happens, which is known as **Predictive Maintenance (PdM)**.

The correction phase is the one in which the real maintenance actions are taken to restore the equipment to its proper condition and eliminate the problem. The corrective action should be verified in order to make sure that the problem was actually fixed.

Non condition based maintenance

Non Condition-Based Maintenance (NCBM) or planned maintenance is a maintenance strategy in which maintenance events or actions such as greasing, painting, dust cleaning or part replacement are preprogramd or scheduled according to manufacturer recommendations, legislation or failure statistics. The period in between maintenance events is normally extrapolated from the estimation of running hours, switching limits, etc.

Corrective maintenance

Reducing the consequences of a failure can be achieved by two different maintenance strategies: **Failure Finding Maintenance (FFM)** or **Run to Failure (RtF)**.

Failure finding maintenance

FFM is a maintenance strategy that is used in order to determine whether a hidden failure has occurred or not in a specific asset. This is normally done by functional or exercise tests, e.g. testing if a pump starts up or not, or by visual inspections, e.g. if a piece of pipe or a junction leaks or not.

FFM does not prevent or avoid the failure from occurring, but it might help reducing its impact to the plant if the failure is found either before the function of the asset is completely gone (a small leakage has less consequences than a big leakage) or even before the function is actually needed (finding that a safety valve is not operational before an emergency situation happens).

Run to failure

RtF is a maintenance policy that allows an asset to run until it breaks down, at which point corrective maintenance may be performed. In this approach, the reactive maintenance actions (repair or replacement) can be either previously planned, in which case there is a reduction of the impact to the plant; or improvised, in which case the action has no effect in risk reduction at all.

2.B.4 Review and update of the overall AMP

The information contained in the PAM project is kept up to date, although care must be taken not to make unnecessary changes. Only if the scoring is not adequate or when new components are installed, changes will be made.

The BR2 reactor is subjected to 10 year periodical safety review since the last PSR (Every 5 years in the past since 1986). The content of this safety review is defined according the IAEA standards on periodical safety review of nuclear installations. Ageing management is on the list of items for review.

2.B.5 Licensee's experience of application of the overall AMP

The BR2 reactor is operating for more than 50 years. As a consequence, most of the SSCs have undergone some kind of ageing management, such as replacement, redesign, inspections, testing. A limited number of ageing requirements were prescribed in the license. Others are listed in the safety report. Some of the SSCs were followed using procedures, while others were tested and maintained based on the judgement and experience of the responsible persons. The main advantage of the AMP, which was started in 2011, is that all information is collected in a systematic manner. In this way it is guaranteed that all SSCs are treated and that all systems important to safety get the same necessary attention. The legal requirements are only concerned about SSCs important to safety. However, it is for an operator useful to take all SSCs into account, and to consider also aspects of reliability and cost. This will decrease the operational cost and will limit the unavailability of the reactor. In this case, the AMP must guarantee that safety has always priority above economical aspects.

2.B.6 Regulator's assessment on the overall ageing management program of BR2 and conclusions.

SCK•CEN was required to carry out a periodic safety evaluation and report on it before June 2016 (PSR 2016). The ageing management program of the BR2 was part of this PSR 2016. More specially the Safety Authority reviewed and approved in December 2016 the first stage of the ageing management program which concerned the overall methodology and the scoring of SSCs. Through the action plan resulting from the PSR 2016, SCK•CEN is required to complete the second and third stages (concerning respectively the development of inspection and maintenance strategy and the development of maintenance and inspection procedures) of the ageing management program before June 2019. A detailed and updated assessment by the regulator is provided in the following subsections.

During the IAEA Safety Review Mission on ageing management and continued operation of the BR2 carried out in November 2017 , several recommendations and observations relevant to the overall aging management program were provided[6]. SCK•CEN will be required to update its PSR 2016 action plan to address these issues.

2.B.6.1 International standards

During the 2016 PSR, the ageing management of BR2 has been assessed according to the latest IAEA standards and guidance. The recommendations of these standards have been incorporated as much as possible in the ageing management program of BR2, which is still under development and its complete implementation is foreseen for June 2019.

2.B.6.2 Description of the overall ageing management program

2.B.6.2.1 Scope of the overall AMP

The responsibilities in the overall ageing management process have clearly been identified by the SCK•CEN. The methods used for identifying SSCs within the scope were assessed during the last PSR by Bel V. These methods are globally adequate, with some exceptions. For instance, hot cells were not considered in the scope for ageing management. The Safety Authority recommends extending the scope of the ageing management program.

The ageing management program is still under development and its complete implementation is foreseen for June 2019. In that frame, the current ageing management programs for the identified ageing mechanisms will be improved in order to take into account all ageing effects.

Concerning the quality assurance, the collection and storage of data is appropriate. However indicators to assess the effectiveness of the process should be established in the frame of the development of the ageing management program.

2.B.6.2.2 Ageing assessment

The Installation Concept Management (ICM) phase of the Plant Asset Management started in the frame of the last PSR and is now finalized. In particular, several important ICM reports have been assessed. The third and last phase (Workorder Management & Skills) has started, and consists in implementing the measures identified in the existing maintenance procedures. The Safety Authority noted that this third phase is work-intensive and noted that the complete scope of SCK•CEN ageing management program includes both safety and non-safety related SSCs. The Safety Authority therefore required SCK•CEN to focus the third phase on safety related SSCs and to develop and implement a set of indicators that allow to follow-up the progress of the third phase.

2.B.6.2.3 Monitoring, testing, sampling and inspection activities

The ageing management program of BR2 foresees monitoring, testing, sampling and inspection activities for SSCs which are important to safety. These activities derive generally from recognized international codes, standards or practices.

2.B.6.2.4 Preventive and remedial actions

Preventive and remedial actions for the BR2 are defined according to a standard approach: the distinction is made between condition based maintenance, non-condition based maintenance, failure-

[6] The report of the IAEA Safety Review Mission on ageing management and continued operation of the BR2 carried out in November 2017 will be published on the FANC website when available: http://afcn.fgov.be/fr

finding maintenance and run-to-failure. The applicable action is derived from the scoring of the SSCs and is considered by the Safety Authority to be adequate.

2.B.6.3 Review and update of the overall AMP

SCK•CEN is required through article 10.2 of the SRNI-2011 to keep the ageing management program up-to-date. Furthermore, SCK•CEN is through its license required to carry-out a Periodic Safety Review every 10 years. A complete review of the ageing management program is part of this PSR. The action plan for the recently completed PSR 2016 included, as already indicated, the improvement and full-implementation of an ageing management program and this action is still on-going.

Procedures to review and update the ageing management program once established, and to measure its effectiveness, still have to be developed by the SCK•CEN.

2.B.6.4 Licensee's experience of application of the overall AMP

Although this was not considered at that time as part of an Ageing Management Program, the replacements of the Be matrix of BR2 that were foreseen from the design phase and performed adequately highlight that some ageing effects were considered from the very beginning at BR2.

2.B.6.5 Conclusions

The SCK•CEN is required to improve and implement the ageing management program of the BR2 to reach a level that is commensurable with the best international standards and practices for research reactors. The improvement of the ageing management program is still in progress and several recommendations have been formulated.

2.B.7 Action plan

The part of the action plan PSR 2016 related to the ageing management program, should be extended by SCK•CEN taking into account the following:

- Extend the scope of the ageing management program to include all SSCs relevant for safety that are present within the premises of the BR2. Notably the hotcells, experimental devices and spent fuel storage system should be included. In addition, spare parts for safety related SSCs that are in stock should also be included.
- Although the Safety Authority does not object to using the same methodology to develop an ageing management program for safety-related SSCs and non-safety related SSCs, there should be a clear separation between both. In particular the on-going development of the ageing management program should be focused on safety-related SSCs.
- Develop procedures to review and update the ageing management program once the current implementation phase is completed and to measure its effectiveness.

In addition to these recommendations, the recommendations and observations relevant to the overall ageing management program provided during the IAEA Safety Review Mission on ageing management will also be integrated in the PSR 2016 action plan.

3. Electrical cables

3.A Nuclear Power Plants

3.A.1 Description of Ageing Management Programs for electrical cables

This chapter focuses on the Ageing Management Programs (AMPs) for electrical cables as described in the Technical Specification for the National Assessment Report (NAR) [3B]. The electrical cables in scope of this report include:

- Electrical cables, including conductor, insulation, shield, jacket and sheath
- Termination arrangements (connections)

ENGIE Electrabel applies different scoping approaches for electrical cables and for termination arrangements (connections). The reason for these separate approaches is the fact that cables and termination arrangements (connections), for the same function, are not necessarily both located in an adverse environment.

3.A.1.1 Scope of Ageing Management for electrical cables

Since 2005, cable management practices at international nuclear facilities have improved. Previously, cables at these nuclear sites were considered passive systems, for which a system-level performance monitoring approach ensured the maintenance of safety functions. Operational experience has shown that cables are subject to ageing. For this reason, assessing the ageing-related degradation of most insulation requires more than the original system-level operating tests. This change in approach for managing cables is leading to the implementation of new regulations in certain countries [4K], and new standards (EPRI and IAEA) [5B] [5C] [2L] [4J]. In Europe, the WENRA Safety Reference Levels published in 2008 [3A] emphasize the need to correctly manage the ageing of all Systems, Structures and Components (SSC).

From 2009 on, ENGIE Electrabel published **Ageing Summaries** (AS) regarding cabling. AS were the main deliverables of the Ageing Management Program (see *Chapter 2.A.3.1.1 Supporting tools for all units* of this NAR).

Later, ENGIE Electrabel improved its existing cable management practices, leading to the development of an action plan in 2011 and the launch of a **Cable Management Program (CMP)** in 2013. The objective of the CMP is to identify and manage general ageing problems, anticipate failures and trace and transfer knowledge about this infrastructure (inventory, inspection results and maintenance history). ENGIE Electrabel has deployed the CMP in all seven Belgian nuclear units.

To support the CMP and facilitate its deployment, ENGIE Electrabel implemented a **software platform**. The tool is used to capitalize on information (e.g. REX, cable environments, etc.) and it encourages the standardisation of practices in ENGIE Electrabel NPPs. In addition, it enables better management of cable ageing by facilitating the exploitation of test results and inspections and by providing decision-making tools.

3.A.1.1.1 Methods and criteria used for selecting electrical cables within the scope of ageing management

All units

ENGIE Electrabel has decided to include all installed 6kV high-voltage (HV), 380V medium-voltage (MV) and I&C cables into the scope of the CMP. At the same time, ENGIE Electrabel has decided to limit the scope of the cables to be monitored (i.e. visually inspected and/or tested) to those located in an adverse environment, in accordance with international recommendations.

Figure 3A-1 illustrates the different stages in defining the scope of the cables.

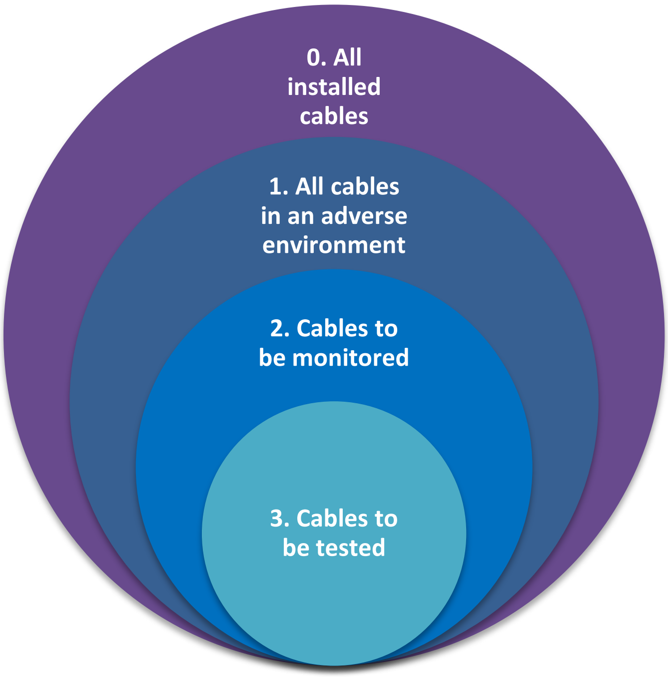


Figure 3A-1: Hierarchy of scopes in the Cable Management Program (CMP) – for all units

0. All installed cables

This population represents all cables installed in the NPP with or without a safety function.

1. Cables in an adverse environment

At least every four years, walk-downs are performed in all rooms with cables to identify any new adverse environments and to monitor the adverse environments that had already been identified. The environments identified as adverse are:

- Rooms in which the ambient or local average temperature exceeds 45°C for a long period
- Environments/rooms in which the absorbed dose rate exceeds 0.04Gy/h
- Environments with stagnant water (i.e. zones with periodic exposure to humidity or water, for a period exceeding a few days)
- Cables in contact with chemicals
- Cables exposed to considerable mechanical stresses

Each inspected room receives a score between 1 and 4, based on the guideline in the ENGIE Laborelec procedure. If the score is equal to 4 (strong adverse environment), a walk-down will be performed every year. In case of a score of 1 (mild environment), a walk-down will be performed every 4 years.

2. Cables to be monitored

Once the environment is defined as adverse, the condition of each cable is evaluated. When a cable is impacted by the adverse environment, ENGIE Laborelec gives a score to define whether the cables are degraded or not: scores range between 1 (as good as new) and 4 (to be replaced).

MV and I&C cables are visually inspected. The inspection periodicity varies in function of the score from once every year to once every 4 years.

The condition of representative HV cables is assessed by electrical testing as explained in the following section.

3. Cables to be tested

If both the environment and the condition of the cable indicate that the cable could age prematurely, a representative number of cables is tested periodically, following the methodology stated in the EPRI guidance (see Table 3A-1) [5B] [5C]. If cable measurement results surpass a certain threshold, the cable will be replaced, safety-related or not.

For certain cases, ENGIE Electrabel increases the sample size as stated in the EPRI guidance. For example, the insulation resistance is measured on all NIS cables (see *Chapter 3.A.1.3.3 NIS electrical cables*) instead of on a representative sample.

Size of sample k	0	1	2	3	…	k	…
Cable population	1	6	16	31	…	N(k)=1+2.5*k*(k+1)	…

Table 3A-1: Sampling rules deduced from EPRI guidance

LTO units

For LTO units Doel 1&2 and Tihange 1, compliancy with the scope defined in NUREG 1801 (GALL) [4H] was verified.

For the functions defined as criteria 1, 2 and 3 by the U.S. NRC (explained in *Chapter 2.A.3.1.2 Specific supporting tools for LTO units* of this NAR) but not qualified (10 CFR 50.49) [4B], compliance with the Ageing Management Programs (AMPs) E1, E2 and E3 and with the GALL were performed. The CMP and additional maintenance procedures cover the execution of the AMPs.

Figure 3A-2 shows that the CMP with additional maintenance procedure considers the AMP as defined in the GALL.

- **AMP E1:** Electrical cables and connections not subject to 10 CFR 50.49 – Environmental qualification requirements
- **AMP E2:** Electrical cables and connections not subject to 10 CFR 50.49 – Environmental qualification requirements used in instrumentation circuits
- **AMP E3:** Inaccessible 6kV cables not subject to 10 CFR 50.49 – Environmental qualification requirements

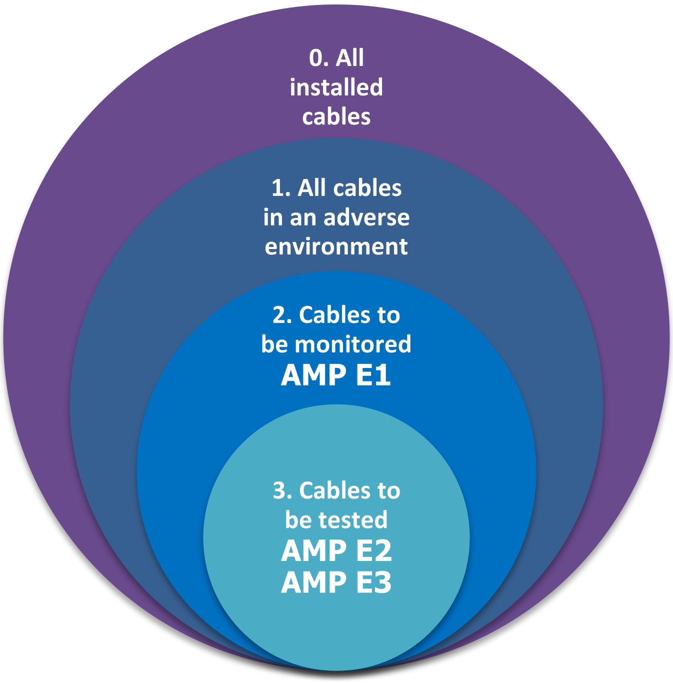


Figure 3A-2: Hierarchy of scopes in the CMP, including the AMPs as defined in the GALL – for LTO units

3.A.1.1.2 Methods and criteria used for selecting termination arrangements (connections) within the scope of ageing management

All units

The scope of the termination arrangements (connections) is defined by expert judgement and REX as discussed during multidisciplinary meetings based on a range of criteria:

- Rooms with high temperature
- Environment/rooms with high absorbed dose rate
- Environments with stagnant water
- Cables in contact with chemicals
- Cables exposed to significant mechanical stresses

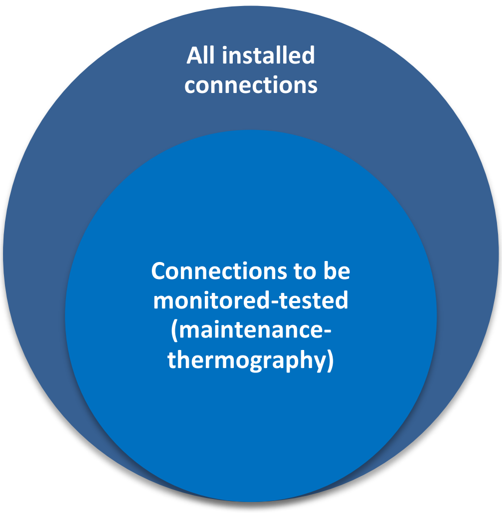


Figure 3A-3: Hierarchy of scopes for termination arrangements (connections) – for all units

Also, the results of the walk-downs are taken into account, as well as the thermographic measurements taken during the maintenance of the cells. The connections for safety-related equipment (criterion 1) are all qualified, independent of the environment they are located in.

LTO units

For the termination arrangements (connections) defined as criteria 2 and 3 by the U.S. NRC but not qualified (10 CFR 50.49), follow-up activities are described in the AMP E6 in accordance with the GALL. In addition, ENGIE Electrabel also considers criterion 1 in the scope of the AMP E6. The harshness of the environment was defined during walk-downs and based on expert judgement (see methodology as applied for all units).

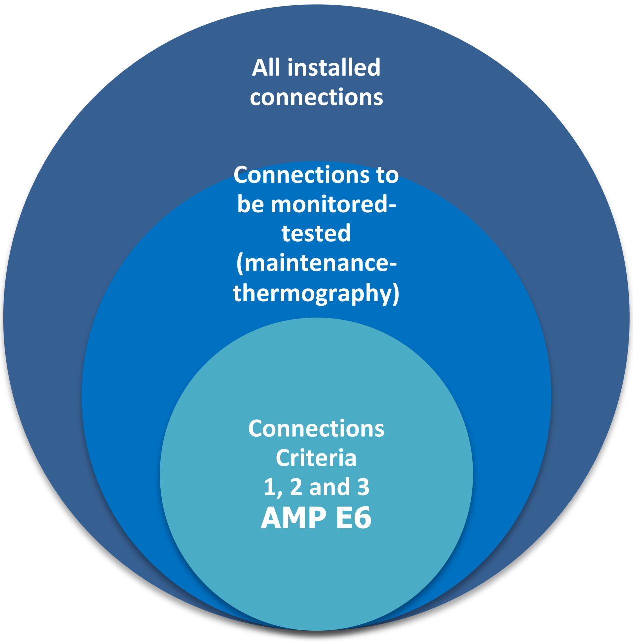


Figure 3A-4: Hierarchy of scopes for termination arrangements (connections), including AMP E6 – for LTO units

3.A.1.1.3 Processes and procedures for the identification of ageing mechanisms

Applied procedures for the identification of ageing mechanisms

Some ageing mechanisms (thermally and radiation-induced embrittlement and cracking) were already known at the design phase. Other mechanisms appeared over time and were identified through the follow-up of international (or national) operating experience, e.g. moisture intrusion.

In general, the ageing process of electrical cables and termination arrangements (connections) is evaluated based on four types of procedures.

1. **Qualification process**

The HV (6kV), MV (380V) and I&C cables in service in the Belgian NPPs have been qualified for the plants' lifetime. The purpose of the qualification process is to demonstrate the component's capability to perform a specific function, based on the simulation of accelerated ageing phenomena.

The preferred and most applied method of qualification is type testing, used to demonstrate that the cable is able to perform its class 1E function after being subjected to stresses caused by specific service conditions (mainly temperature and radiation) and Postulated Initiating Events (PIE), e.g. LOCA. The generation 1 HV, MV and I&C cables (see Table 3A-2) are being qualified in the framework of the LTO programs.

2. Cable deposits

The materials used in common types of cables have been studied extensively. Their degradation mechanisms (chemical reactions and their kinetics as a function of time, temperature, radiation dose rate, oxygen concentration, etc.) are well known [10] [11]. This knowledge ensures that the environmental conditions to which cables are exposed during accelerated ageing are representative for the conditions during natural ageing.

An efficient way to cope with this threat is to check the validity of the qualification by storing a number of selected cables in a cable deposit, an area in the reactor building with a high dose rate and high temperature. At predefined time intervals, samples are extracted and tested to assess their state and, consequently, deduce their ability to meet their specified performance requirements in operation.

In 2013, ENGIE Electrabel installed two cable deposits in the reactor buildings of Doel 3 and Tihange 2. Since 1993, there is also a cable deposit at Doel 2. The cable deposits contain generation 3 and 4 MV and I&C cables from (see Table 3-2).

3. Walk-downs

Based on the different walk-downs in the framework of the CMP and the different research projects, ENGIE Laborelec has developed a procedure with examples of the evolution of cable degradation and the related condition score: scores range between 1 (as good as new) and 4 (to be replaced). The cables are assessed according to the procedure and expert judgement to verify the integrity of the jacket. Inspections are always executed by two experts. If only visual inspection results are available, the evolution of cable degradation is characterized based on a comparison of the findings of previous walk-downs and the application of the ENGIE Laborelec procedure. If visual inspection is combined with cable testing, both results will be taken into account to determine the condition of the cable.

4. R&D programs

ENGIE participates in various international R&D programs such as the European ADVANCE project [7A], the IAEA-coordinated research project [2T], the thesis work with SCK•CEN and the set-up of the CAST project in the framework of Horizon 2020 [7B].

Different mechanical and chemical assessment techniques for MV (380V) cables were studied during these projects for various materials (e.g. XLPE, EPDM-EVA) used in cables. The following assessment techniques were considered (see also *Chapter 3.A.1.3 Monitoring, Testing, Sampling and Inspection Activities*):

- **Mechanical** assessment techniques: tensile testing (elongation at break) and indenter testing (compressive stiffness).

The elongation at break enables the assessment of the structural integrity of the cable's insulation.
Indentation is a non-destructive condition monitoring method based on measurement of the indenter modulus, which is associated to the compressive stiffness of polymeric cable materials. The instrument is clamped around the cable jacket and the probe only penetrates the surface of the test material a few hundred microns. Based on a model, the indenter modulus is linked to the elongation at break.

- **Chemical** assessment techniques: thermogravimetric analyses and oxidation induction time measurements by Differential Scanning Calorimetry (DSC) and infrared measurements.

During thermogravimetric analyses, the sample weight is monitored as the sample is heated at constant rate. These analyses enable the characterization of the cable's polymeric materials (glass-transition and melting temperature).
Oxidation induction time measurements enable the measurement of the remaining levels of antioxidants and the extent of the polymer's oxidation which is linked to the material's degradation.

Infrared measurements enable the observation of changes in the structure of the materials which are linked to the formation of new chemical bonds that have different light absorption characteristics in comparison to the bonds in the original unaged material. Such measurements can result in a better comprehension of the ageing phenomena.

Ageing mechanisms

The following ageing mechanisms are known for the **insulation** and **jacket** of electrical cables:

- Thermally induced embrittlement and cracking
- Radiation-induced embrittlement and cracking
- Water treeing and moisture intrusion
- Chemical attack
- Mechanical damage
- Radiolysis and photolysis (UV-sensitive materials only) of organics

The following ageing mechanisms are known for the **shield**:

- Corrosion and oxide formation of copper tapes or wires
- Thermally and radiation-induced embrittlement and cracking on semi-conducting polymers

For the **conductor**, no problematic ageing mechanisms have been identified up to this point. However, issues such as poorly made connections, deteriorated terminations or uninsulated connections contaminated with corrosive sulphur might affect conductor performance and are therefore monitored.

For the **termination arrangements (connections)**, the following ageing mechanisms and their potential effects are covered:

- Thermal and mechanical fatigue due to thermal cycles, Joule effect losses, mechanical cycles (loading and unloading), vibrations, electromagnetic forces caused by electrical transients, mechanical damage during tests, installations or maintenance activities
- Corrosion/oxidation due to humidity and condensation

3.A.1.1.4 Grouping criteria for ageing management purposes – Electrical cables

Identification of cable families

The cables are divided into two main groups: HV cables on the one hand, MV and I&C cables on the other hand. Within these groups, the cables are divided into the following cable families, based on similar ageing phenomena and the timeframe of manufacturing.

Cable group	Cable family	Description
HV	HV GEN1 PVC (1970-1977)	Cables with PVC insulation manufactured by Cablerie de Charleroi (CDC). Installed at Doel 1&2.
	HV GEN2 EPR (1977-1983)	Cables with EPR insulation manufactured by CDC.
	HV GEN2 XLPE (1977-1983)	Cables with XLPE insulation (types NU-EAXCW and NU-EXCW manufactured by CDC, types EXeCeWG manufactured by Alcatel Cable Benelux, previously CDC.
	HV GEN3 XLPE (1983-now)	Cables with XLPE insulation, type NU-EXCWG, manufactured by Eupen.
	HV GEN4 XLPE NU-EXCR(L)R (2007-now)	Cables with XLPE insulation, type NU-EXCR(L)R and NU-2XShX(L)HX, manufactured by Eupen.

Cable group	Cable family	Description
	HV GEN4 XLPE LC8205R (2014)	Cables with XLPE LC8205 insulation, type NU-EXCWG, manufactured by Eupen.
MV / I&C	MV, I&C GEN1 (1970-1977)	Cables with PVC insulation manufactured by CDC. Installed at Doel 1&2 and Tihange 1.
	MV, I&C GEN2 (1977-1983)	Cables with PRC insulation manufactured by CDC. In the mid-70s, CDC started to use PRC (cross-linked polyethylene or XLPE) instead of PVC in the manufacturing of cables for nuclear applications. At that time, the nuclear qualification of cables in accordance with IEEE 323 & 383 began.
	MV, I&C GEN3 (1983-2002)	Cables with EPDM rubbers manufactured by Eupen.
	MV, I&C GEN4: (2002-2009-now)	Cables with EPDM-EVA rubbers manufactured by Eupen. Due to an obsolescence of the jacket material, another jacket material with only small changes in the material was used from 2009 on.

Table 3A-2: Cable families

Sampling process for testing cables

The number of tested cables (sampling) is chosen in function of the family. The sampling methodology complies with the EPRI criteria (see Table 3A-1) [5B] [5C] defining a minimum number of cables to be tested in view of having a representative sample. Based on REX, a sample can be extended.

The final sampling process is defined to meet the following objectives:

- Comply with feasibility constraints

Consider international recommendations while complying with the EPRI criteria regarding the number of cables to test as a function of the cable population

3.A.1.1.5 Grouping criteria for ageing management purposes – Termination arrangements (connections)

Identification of termination arrangement (connections) families

The termination arrangement (connection) family is categorized based on type, technology and environment. For example, in Tihange 1 the connection families are:

Cl = Clipped	Sc = Scewed	Cr = Crimped	Co = Connected

	HV power supply				MV power supply				I&C			
High temp.	Cl	Sc	Cr	Co	Cl	Sc	Cr	Co	Cl	Sc	Cr	Co
Humidity	Cl	Sc	Cr	Co	Cl	Sc	Cr	Co	Cl	Sc	Cr	Co
Vibrations	Cl	Sc	Cr	Co	Cl	Sc	Cr	Co	Cl	Sc	Cr	Co
Irradiation	Cl	Sc	Cr	Co	Cl	Sc	Cr	Co	Cl	Sc	Cr	Co

Table 3A-3: Categorization of termination arrangements (connections)

Sampling process for testing termination arrangements (connections)

In the specific case of termination arrangements (connections) ENGIE Electrabel uses the most recent version of IGALL's AMP 206 (for LTO units): a sample is considered representative if the associated sample size represents 20% of the population considered, namely a connection family, without exceeding a maximum of 25 components.

3.A.1.2 Ageing Assessment of Electrical Cables

ENGIE Electrabel assesses the ageing of all types of electrical cables. For the purpose of this NAR, the text only addresses the following components:

- HV (6kV) cables
- MV (380V) cables, buried or in trenches
- Neutron Flux Instrumentation System (NIS) electrical cables
- Termination arrangements (connections)

The assessment of each of these components is discussed in the following subsections.

3.A.1.2.1 High-voltage (6kV) cables

The scope in the CMP concerns all HV (6kV) cables at the power station. However, the strategy defined in the context of the CMP determines that tests need to be performed on cables exposed to an adverse environment and in accordance with international regulations.

The scope of the cables to be tested (as explained in paragraph *3.A.1.1.1 Methods and criteria used for selecting electrical cables within the scope of ageing management* of this NAR) can then be extended by identifying the cables that are in an advanced state of ageing-induced degradation and for which the impact of failure is the largest.

The program is not limited to safety-related cables. However, if a representative sample must be defined for a family of cables exposed to the same environment, the selection of safety-related cables will be prioritized. In addition, the program covers 6kV cables in both adverse and non-adverse environments. This gives us a global view on the condition of a larger sample of the fleet's 6kV cables, to compare ageing of cables located in adverse and non-adverse environments and to anticipate any currently unknown ageing phenomena that might arise in the future. In the past, this made it possible to identify a change in the composition of the insulation material influencing the tan δ and insulation resistance of the cable.

International feedback (GL2007-01) shows that unfavourable operating conditions due to an adverse environment can lead to accelerated ageing of high-voltage cables and to a loss of function likely to impact the safety and availability of the unit. Consequently, ENGIE Electrabel developed its CMP.

The following ageing mechanisms have been identified:

- Water treeing and moisture intrusion

One of the most significant ageing mechanisms for HV cables is caused by a prolonged exposure to a wet environment. When the cable is energized, the insulation can undergo a type of degradation called water treeing. The presence of humidity and an intense electrical field can cause micro-cracks, especially in areas where there are imperfections in the insulation. This microscopic degradation spreads slowly and progressively. This phenomenon does not directly lead to total failure, but causes an increase in leakage current and a decrease in insulation capacity, which can lead to cable failure.

- Thermally and radiation-induced embrittlement and cracking

A prolonged exposure of the insulation to a high temperature can cause the insulation and jacket to become fragile and even crack. The degradation of the insulation will then gradually reduce its dielectric efficiency and cause an increase in leakage current. Exposure of a cable to a high radiation dose rate can also cause the jacket and insulation to become fragile, leading to a loss of dielectric functions.

HV cables can also be exposed to heating from the Joule effect, likely to reduce their service life. This heating is considered when designing the cable, and only specific cases of abnormally high resistance (due to corroded connections for example) or usage outside the design specifications can lead to degradation.

3.A.1.2.2 Medium-voltage (380V) cables, buried or in trenches

The program for MV cables buried or in trenches is not limited to safety-related cables. All cables exposed in this kind of environment are monitored.

International feedback (GL2007-01) shows that unfavourable operating conditions due to an adverse environment can lead to accelerated ageing of cables and to a loss of function likely to impact the security and availability of the unit. Hence, the need for an effective cable ageing management strategy that depends on the type of environment, buried or in trenches (see paragraph *3.A.1.3.2 Medium-voltage (380V) cables, buried or in trenches*).

While MV power supply and I&C cables' ageing mechanisms and effects due to significant moisture are limited (e.g. lower instrument and control voltage levels do not support water or electrical tree formation), operating experience has shown that insulation degradation may occur if I&C cables are exposed to continuous wetting or submergence (as described in AMP 203 IGALL[7]).

In addition, variances may exist in the ageing mechanisms and effects depending on electrical insulation material, manufacturing, and application. Therefore, periodic actions (see paragraph *3.A.1.3 Monitoring, Testing, Sampling and Inspection Activities* of this NAR) are necessary to minimize the potential for insulation age degradation due to significant moisture.

3.A.1.2.3 NIS electrical cables

The cables from the Neutron Flux Instrumentation System (NIS) of ENGIE Electrabel PWR plants each consist of two source chains, two intermediate chains and four power chains. Figure 3A-5 gives an example of a NIS configuration inside the reactor building. The NIS cables run from the reactor building through the annular space towards the electrical building.

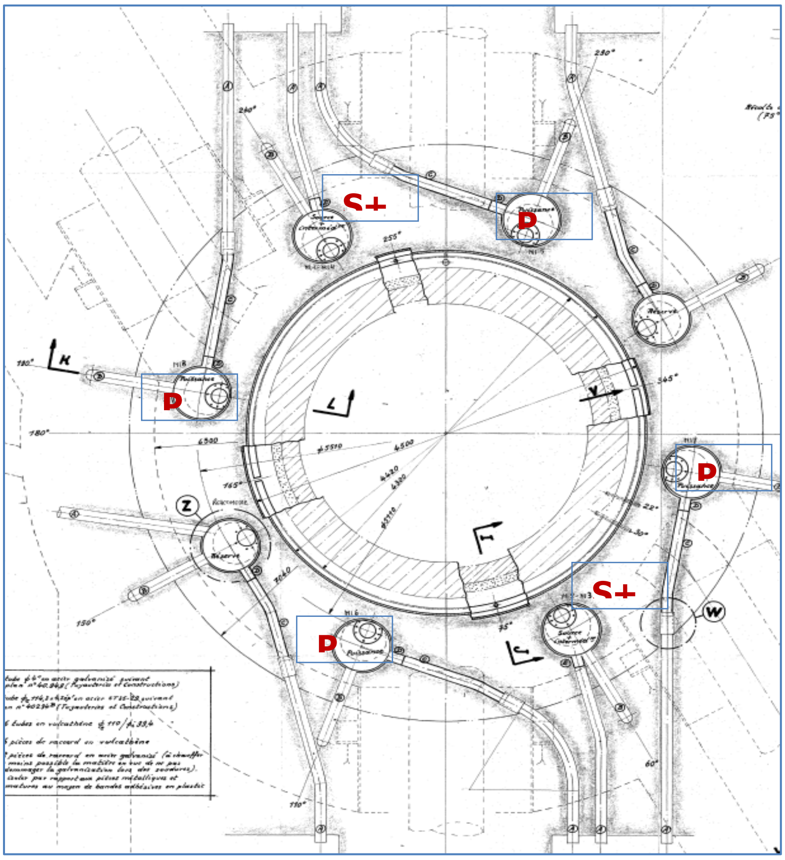
The ageing mechanisms considered for the NIS cables are:

- Thermally induced embrittlement and cracking
- Radiation-induced embrittlement and cracking

The cables located between the detector and the junction box are exposed to more severe ageing factors (temperatures and radiation) compared to cables between the junction box and the electrical penetration. The cables running from the junction box are protected from high temperatures by copper tubes shielding.

The ageing mechanisms were considered during the cables' qualification process. The cables are qualified for a period of 33 years (RSQ 512.04). As the source cables were installed around 2012 they are relatively new. No specific maintenance program must be performed due to the successful qualification.

[7] It should be noted that GALL's original AMP E3 did not consider inaccessible 380V cables.



S = Source / **I** = Intermediate / **P** = Power

Figure 3A-5: Illustration of the configuration of the NIS electronic cabinets around the reactor vessel

3.A.1.2.4 Termination arrangements (connections)

Ageing assessment is applied on all types of termination arrangements (connections) regardless of the environment and safety function. Potential issues with termination arrangements (connections) are well-known throughout the industry. Hence, maintenance programs for termination arrangements (connections) have existed since the start-up of the NPPs.

It should be noted that in the ENGIE Electrabel NPPs all connections that perform a safety function during a PIE are qualified following 10 CFR 50.49 [4B], regardless of their environment. Based on this approach, the connection lifetime is defined and if necessary additional maintenance activities are specified in the RSQ.

This Ageing Management Program covers the following ageing mechanisms and their potential effects:

- Thermal and mechanical fatigue due to thermal cycles, Joule effect losses, vibrations, electromagnetic forces caused by electrical transients, mechanical damage during tests, installations or maintenance activities
- Corrosion/oxidation due to humidity and condensation

The associated ageing effects are as follows:

- Increase in electrical resistance
- Release/loss of contact of the connection
- Overheating of the connection
- Degradation and loss of structural integrity of the connection

3.A.1.3 Monitoring, Testing, Sampling and Inspection Activities

3.A.1.3.1 High-voltage (6kV) cables

The condition of the 6kV cables and connections is assessed by visual inspections, thermography and cable tests.

Visual inspection

Visual inspection is one of the most used and most effective methods for the identification of visible degradation mechanisms induced by certain environments. However, it applies only to the exterior jacket of the cable, and therefore cannot be used to pinpoint degradation affecting only the insulation, such as water treeing. Nevertheless, the kinetics of ageing of the jacket exposed to high temperatures or strong dose rates exceeds that of the insulation. A visual inspection of the jacket can therefore make it possible to anticipate any harmful ageing of the insulation.

Cables exposed to adverse environments will be inspected visually during the walk-downs to identify any ageing phenomena linked to the cable's operational environment. Any observation linked to the cable's operational environment can also lead to the identification of a new adverse environment. The identification of 6kV cables in an adverse environment helps prioritize the testing sequence.

When a cable inspection is performed, the condition of the termination arrangements (connections) is assessed as well. In case of anomalies, a report is made. For example, a specific visual inspection campaign was carried out at the Tihange NPP due to white deposit found on the terminations. The affected termination arrangements (connections) have been replaced. This observation was not found at the Doel NPP.

Figure 3A-6: Example of a termination arrangement (connection) at the Tihange NPP

Frequency • As explained, a walk-down is performed at least once every four years in the rooms of the power station to identify any new adverse environment and monitor the zones already identified. HV (6kV) cables in adverse environments are identified and are set with a higher priority for condition monitoring.

Infrared (IR) thermography

IR thermography is used to detect local heating (hot spots) in the cable.

Frequency • The connections are inspected by infrared thermography on average every 18 months following the maintenance procedures of the Doel NPP and the Tihange NPP. The thermographic inspections are executed in the framework of predictive maintenance.

Cable testing

The purpose of the cable tests is to identify, evaluate and monitor non-visible faults. The monitoring tests applied to HV (6kV) cables are the following:

- Tests performed systematically

Reflectometry test in the time domain
Insulation resistance test and calculation of the polarisation index
Tan δ test (dielectric losses)

- Tests performed on a case-by-case basis and based on the results of the previously described tests

Partial discharge test
Dielectric test
Test of outer sheet (only for underground cables)

The **reflectometry test** (or TDR, Time Domain Reflectometry) is used to determine the cable length and is used to locate faults or wet zones in cables by detecting impedance fluctuations. However, this technique cannot be used to assess the degradation of the cable and must therefore be used as a supplement to other diagnostic techniques.

The test is based on the same principle as radar: a high-frequency pulse is applied to a cable, which is disconnected at both ends. The time taken by the pulse to return is measured and converted into distance. Knowing the propagation characteristics specific to insulation materials, an experienced operator can detect and locate the degradations that have appeared since the last inspection.

The **insulation resistance test** is a standard technique widely used to determine the condition of the insulation. The insulation resistance is measured using a Megger device on a cable that is disconnected at both ends. This test consists of measuring the resistance of insulation while applying a continuous DC voltage between the conductor and the earthing. This test is based on the following principle: when a continuous voltage is applied across the insulation, it is possible to measure a low current. The total current crossing the insulation is the sum of the capacitive charge current, leakage current and dielectric absorption current.

These three currents change over time. The capacitive charge current and the dielectric absorption current have a high value at the start of the test, when the insulation is exposed to a significant difference in potential. The insulation acts as a capacitor, after being charged, it discharges and the currents decrease until they cancel each other out. However, leakage current acts differently. Typically, it starts at zero and rises, then stabilizes at a value after a certain length of time. If the condition of the insulation is very degraded by humidity or by chemical contamination, the leakage current is higher than that of insulation in good condition. Consequently, the total current passing through the tested cable is high at the start, and then varies in different ways depending on the condition of the insulation over the next ten minutes. To take this behaviour into account, the resistance of the insulation is measured twice: after one minute and after ten minutes of testing. The ratio of the two measurements is calculated; this is called the polarisation index. After this test, a considerable discharge time of three times the charge time is needed due to the capacitive effect.

The **tan δ test (or electric dissipation factor)** is performed at a low frequency on HV cables to assess their degradation via quantifying dielectric leaks caused by characteristic degradations of these cables. This is one of the most effective and widely used tests for identifying degradations caused by exposure to a wet environment such as the appearance of water treeing or partial discharges. In the case of HV (6kV) cables exposed to high temperature and elevated dose rates, the performance of a tan δ test will follow the identification of degradation during visual inspections. In those areas, the performance of tan δ tests will then be systematic on a representative sample of cables.

This test is performed with a signal generator and a spectrum analyser. The instruments apply different voltages in the frequency band [0.1 Hz, 0.01 Hz] and they are connected between the cable conductor and the screen. The response of the resulting current is measured and then recorded for further analysis. The tan δ does not theoretically depend on the applied voltage: the tan δ of a cable in good condition must be constant. However, if a bend in the curve is seen, a partial discharge test must be used to jointly analyse the results.

The **partial discharge test** is performed in case of high tan δ values. This test consists of applying a high voltage to the cable, which results in partial discharges in the insulation in terms of discontinuities or degradations. It is then possible to locate these discharges and accordingly the associated faults. This test can be performed at the operating frequency (i.e. 50 Hz) or at very low frequency (below 1 Hz) or at undulated frequency (50Hz-800Hz).

Partial discharges are generally on the order of picocoulombs (pC). They can be measured using an oscilloscope connected to the cable. They can be located by measuring the gap between the pulses sent and pulses reflected from the place where the discharges are produced.

The **dielectric test** is only used if there is a doubt about the condition of the cable. It is a potentially destructive test that can be used to confirm the dielectric withstand of the insulation. This test causes the cable to breakdown if it is in a weak condition.

Frequency • The tests are performed periodically in function of the results of previous tests, the age of the cable and whether the cable passes through an adverse environment or not.

3.A.1.3.2 Medium-voltage (380V) cables, buried or in trenches

During the design phase of the Tihange NPP, all cables were laid in trenches. Each year, Tihange carries out a **visual inspection** of all trenches. In case water is present, it is extracted using pumps. If the water cannot be extracted, the necessary actions are taken to ensure a dry and sustainable situation to safeguard the insulation of the electrical cables. Based on the cables' condition and environment, additional insulation resistance measurements are performed.

At Doel, however, visual inspections are not possible as there are no accessible trenches with safety-related MV cables.

Hence, Doel performs **insulation resistance measurements** on buried MV (380V) cables every five years. Up till now, no significant cable degradation has been observed. The insulation resistance of the cable may not be lower than 500kΩ.

The cables' health is also monitored by the **health reports** (see *Chapter 2.A.3.1.1 Supporting tools for all units* of this NAR). If a system would be impacted by a failure of concealed cables, the system or component engineers will detect it and actions will be defined. Cables (buried or in trenches) are typically grouped for some function located outside the normal buildings. These circuits are followed by the system engineer.

More generally, ENGIE Electrabel also looks at the condition of the MV and I&C cables in non-adverse environments during room monitoring. If cables show signs of degradation, the cables will be inspected periodically as well, even when these cables are not located in an adverse environment.

3.A.1.3.3 NIS electrical cables

The purpose of the cable tests is to identify, evaluate and monitor non-visible faults that could appear regardless of the cables' qualification (they have been qualified for a period of 33 years). The sequence of qualification tests is presented in Table 3-4.

Initial tests	Visual inspection + insulation resistance
Thermal ageing	Ageing at elevated temperature
	Visual inspection + insulation resistance
Radiation ageing	Ageing by means of γ-radiation
	Visual inspection + insulation resistance
Final tests	Visual inspection
	Flexibility test
	Insulation resistance
	Voltage withstand test
	Insulation resistance
	Visual inspection

Table 3A-4: Overview of typical test program for qualification purposes

During **visual inspection**, degradations of the cable's exterior jacket are checked.

The **insulation resistance** test is a standard technique widely used to determine the condition of the insulation. The test is performed in accordance with IEC 60502-1 §17.1.1 and ICEA T 27 581 NEMA WC 53-2008 §2.3 [7C] [7D]. The test consists of measuring the resistance of insulation while applying a continuous voltage between the shielding and each conductor. The test is based on the following principle: when a DC voltage is applied across the insulation, it is possible to measure a low current. The total current crossing the insulation is the sum of the capacitive charge current, leakage current and dielectric absorption current. The results before thermal ageing are used as a reference throughout the qualification program and it is checked whether the insulation resistance is maintained.

The **flexibility test** is used to verify the lack of embrittlement in accordance with IEEE 383 [7E]. First the samples are straightened (if coiled) and then coiled with an inside diameter not exceeding 20 times the overall cable diameter.

The **voltage withstand test** is performed in accordance with IEEE 383. First, a coiled sample is immersed in tap water at room temperature for at least one hour. Next, while the coil is still immersed, a specific voltage (according to the manufacturer test) is applied for 5 minutes. During the test, the electrical continuity must be maintained.

The monitoring tests applied to the NIS cables are as follows.

- The **insulation resistance test** is a standard technique widely used to determine the condition of the insulation. These tests have been applied to check the condition of the cables. The results were satisfactory.
- **Continuity and capacity tests** (functionality check of chain) have also been carried out.
- During the normal operation of the units, the **thermal balance** is checked on a daily and monthly basis. In addition, the **quadrant imbalance** is compared weekly. All these comparisons are used to identify any change in the measurement chain. These checks are mandatory.
- To detect possible anomalies, a **calibration of the entire chain** is performed after each reloading during an outage.

As cables continue to age, ENGIE Electrabel will execute additional tests based on REX or IAEA recommendations.

Currently, the cables' insulation resistance is monitored yearly. The source cables have been replaced around 2012. The power flux cables and intermediate flux cables will be replaced in the coming years. For the LTO units, projects have been launched and will be realized by the end of 2019.

3.A.1.3.4 Termination arrangements (connections)

When a degradation of the accessible parts of cable termination arrangements (connections) is detected during a **visual inspection** (e.g. in the context of maintenance work), it is immediately followed up. Possible degradations are: a change in colour of material close to the termination arrangement (connection), a change in colour of the contact, oxidation, corrosion, cracking, visible surface deposits, etc. The cause of the degradation will be analysed and a solution will be provided.

Infrared (IR) thermography is used to detect local heating (hot spots) in a cable or connection.

If a questionable measurement is noted, the tightening torque of the bolts is adjusted.

The HV and MV connections are inspected by IR thermography on average every 18 months following the maintenance procedures of the Doel NPP and the Tihange NPP.

The permissible values for **tightening torques** can be found in the manufacturer documentation or in the maintenance procedures. They are performed during normal maintenance and the frequency depends on the kind of equipment the connectors are on (motor, motor operated valve, cabinet, etc.).

The values from **connection resistance measurements** (MV and I&C) are assessed based on the type of connection and on the maximum permissible current flowing through this connection. The measurements are performed on the different types of connections (clipped, screwed, crimped or connected).

No **loss of continuity** is acceptable and any significant measurement drift must be investigated.

For LTO units

For LTO units Doel 1&2 and Tihange 1, various projects have been realized or will be realized.

At Tihange 1, the conformity of the qualification of the connections and cabinets-boxes for the entire LTO period has been analysed. The inspection program is described in AMP E6 (GALL) and integrated in maintenance plans.

- Cabinets-boxes: approximately 450 cabinets-boxes inside containment (electronics boxes, splitter boxes, panel boards, etc.) and 250 boxes outside containment were inspected.
- Connections: the connections were inspected during walk-downs. The ageing degradation of the connections on the components was considered in the qualification of the specified component. The ageing degradation of the connections on the different racks was verified during periodic maintenance activities or during spot verifications.

Similar walk-downs on a representative number of boxes/panels are being performed at Doel 1&2. Till now, no significant issues have been revealed.

3.A.1.3.5 Establishment of acceptance criteria related to ageing mechanisms

For the electric tests, the criteria to adopt depend on the type of cable (insulation material, length, etc.). The criteria defined by ENGIE Laborelec specialists are based on the measurements over several years and on the international review of test criteria. Since the test criteria depend on the cable insulation material, the criteria cannot be copied from international studies. Nevertheless, the same approach can be adopted to define the criteria.

The test specifications and test criteria are compared with the specification of the cable manufacturer. Based on the results of all tests carried out on the cable, the ENGIE Laborelec engineer attributes a condition score of 1 (non-aged) to 4 (degraded).

Acceptance criteria for inspections are defined by the observation that the cables are not submerged or immersed in standing water. During visual inspections, the cable jacket is checked to make sure it is free form unacceptable surface anomalies such as embrittlement, discoloration, cracking, melting, swelling, or surface contamination due to the ageing mechanism and effects of significant moisture. If the above indications are present, additional testing (insulation resistance test) may be warranted to verify that the cable's electrical insulation is managed adequately.

3.A.1.4 Preventive and Remedial Actions for Cables and Termination Arrangements (Connections)

HV cables

The main preventive actions to be taken aim at limiting the harshness of the environment (water, chemical spray, UV light, etc.) by installing or repairing the thermal insulation and by installing systems to provide protection against these environments. Also important is the repair of the outer jacket when the outer jacket is damaged, e.g. due to the pulling of the cable.

When measurement results for 6kV cables surpass a certain threshold, the cables will be replaced. Consequently, the cables in scope and in a similar situation (age, material, environment) must be monitored to verify the expected state of ageing.

When the cable status is still acceptable, cable monitoring will be planned in the future. Based on the thermographic analyses, actions can be launched when anomalies are detected.

MV (380V) cables, buried or in trenches, and NIS electrical cables

As described in *Chapter 3.A.1.2.3 NIS cables*, the necessary actions are taken to ensure a dry and sustainable situation to safeguard the isolation of the 380V electrical cables in trenches. No other preventive and remedial actions are possible for these groups of cables.

Termination arrangements (connections)

Any observed anomaly that might be detected during walk-downs must be the subject of a notification and an intervention, and must be resolved. The likelihood of encountering the same problem on other connections is assessed and, where appropriate, the quality of the potentially affected connections (same environment or same type, depending on the cause of the anomaly) is verified.

The replacement or repair of the affected termination arrangement (connection), the extension of the sample to be inspected or an increase in inspection frequency can be envisaged.

The visible and accessible sections of both power and I&C cables are visually inspected.

Electrical measurements of cables can reveal degradation of termination arrangements (connections) and give rise to the appropriate remedial actions.

3.A.2 Licensee's Experience of the Application of AMPs for Electrical Cables

Every year, ENGIE Electrabel and/or ENGIE Laborelec take part in the EPRI Users group meeting to exchange information with other users on the general approach of cable ageing and on 6kV measurements. Several exchanges were done with EDF to discuss cable ageing aspects.

ENGIE Laborelec has performed several investigations on cables which help in analysing the cable measurement results. Studies were done regarding the influence of UV light on the outer jacket made of polyethylene.

In 2013, ENGIE Laborelec investigated the exploded terminations of the cooling water pumps (non-1E equipment) of Doel 1. These investigations revealed the presence of partial discharges at the terminations, therefore several terminations were replaced preventively. Due to the exploded termination, a TEV measurement campaign was launched for about 20 terminations at pumps and for 80 cabinets at Doel 1&2 to identify possible similar degradation phenomena on other cables.

Due to the 6kV cable management in the framework of the CMP, ENGIE revealed that the terminations of 6kV cables in a wet environment are affected by accelerated ageing. The CW pumps (non-1E equipment) from Doel 3 and 4 and terminations are situated outdoor and therefore the cable terminations are situated in an adverse environment. Testing and lab testing revealed the degraded condition of the cable terminations. A project is running to replace several cable terminations.

In Tihange, based on the test results of some non-safety related cables, several terminations of the external power supply transformer cables will be replaced. The terminations are located outdoor in a wet environment and are exposed to UV light.

The CMP and AMPs managing the different ageing mechanisms that could potentially affect cables and termination arrangements (connections) can be considered as adequate. This is shown by the timely detection of indications on different HV (6kV) cables by electrical testing and of degradations observed visually on MV (380V) and I&C cables.

In the future, ENGIE Electrabel has planned to move towards an Ageing Management Program for Doel 3 and 4 and Tihange 2 and 3 in line with the program installed for the LTO units and based on the most recent international guidelines.

ENGIE will continue to maintain international contacts (EPRI user group, IAEA workgroups, exchanges with other international users, etc.) and to participate in international cable ageing studies. The purpose is to continuously improve the applied approach and to maintain compliancy with international guidelines.

For MV and I&C cables, ENGIE Electrabel and ENGIE Laborelec will investigate whether its currently applied testing methodology can be extended with more quantitative condition assessment techniques.

In laboratory conditions and on a case-by-case basis, ENGIE Laborelec will continue to investigate the replaced cables to improve the knowledge about the various degradation phenomena and the cause of degradations being observed.

3.A.3 Regulator’s assessment and conclusions on ageing management of Electrical Cables

3.A.3.1 Regulatory oversight process

The ageing management of the cables has been followed by Bel V since several years. In particular:

- The ageing management of electrical cables and connections is addressed in Safety Factor 3 (equipment qualification) and in Safety Factor 4 (ageing) during the last PSRs (2012-2015);
- For the preparation to LTO for Doel 1&2 and Tihange 1 units (2012-2015), Bel V reviewed the Integrated Plant Assessment (IPA) related to electrical cables and connections and is following the realization of the associated action plans proposed by the licensee in order to upgrade the ageing management of the electrical cables and connections;
- Electrical cables and connections management enter in the scope of the yearly meetings with ENGIE Electrabel about ageing.

The Electrabel self-assessment report for this TPR was received and discussed. Specific attention was given to the procedures for identifying the rooms with adverse environment.

3.A.3.2 Regulator's assessment

3.A.3.2.1 Scope of ageing management for electrical cables

All installed 6 kV (high-voltage, HV), 380 V (medium-voltage, MV) and I&C cables, safety related or not, are within the scope of the ageing management. The scope of the cables to be monitored (visually inspected and/or tested) is at least the cables located in an adverse environment. This environment is assessed at least every four years during walk-downs. The strategy of assigning higher importance for the monitoring of cables located in adverse environment is in line with international recommendations (e.g. [2N]).

The termination arrangements (connections) to be monitored are defined by expert judgment based on return of experience, on the characteristics of the environment assessed during walk-downs and as a result of the previous measurements done during maintenance operations.

In the framework of LTO programs, the licensee has performed an upgrade of the cable management program to be in compliance with the AMP E1, E2, E3 and E6 defined in NUREG 1801 (cables and connections not subject to Environmental qualification requirements of 10 CFR 50.49). In the framework of the LTO program, a verification of the conformity of the qualification of cables and connections has also been performed.

The approach of the licensee for defining the scope of ageing management of cables follows the good practices.

3.A.3.2.2 Ageing assessment of electrical cables

The cables ageing management process used by the licensee is derived from the environmental qualification process, the follow-up of cable deposits, the follow-up of international and national operating experience and participation in R&D programs.

The qualification process includes an accelerated ageing for demonstrating the capacity of a component to fulfil its safety function in the conditions that will occur at that time. From this qualification process, requirements are defined for the use and maintenance of the cables and connections to ensure that they remain qualified (e.g. qualified life, temperature conditions or other specific conditions). The qualification of the cables of generation 1 (PVC cables, see table 3-2) is being done in the framework of the LTO programs. The other generations of safety related cables as well as the different types of termination arrangements are all qualified.

The presence of cable deposits in harsh environment (near the pressuriser) is a good practice in order to complement the results of the cable qualification and to compensate for the fact that "hot spots" may be difficult to localise in the reactor building (due to e.g. dose for the personnel or due to cables located in inaccessible locations). Nevertheless no cable deposits are presents for the cables of generations 1 and 2.

The ageing mechanisms identified by the licensee are in line with international guidelines. The licensee has identified the relevant ageing mechanisms of cables and connections.

3.A.3.2.3 Monitoring, testing, sampling and inspection activities for electrical cables

The general principle is that cables located in an adverse environment are visually inspected and, depending on their conditions, a sample of those cables is tested. Some specific approaches for the monitoring of the different examples are presented in 3.A.1.1.

The general methodology for monitoring the cables is satisfactory

Concerning the nuclear instrumentation (NIS) cables, the functional checks mentioned by the licensee (periodic functional checks, calibration of the chain, …) do not really allow assessing the condition of the cable and cannot be used to detect ageing degradations before the cables fail. Nevertheless these cables are relatively new (2012) or will be replaced in the coming years. The Safety Authority judges that acceptable. Let us note that some of the power and intermediate flux cables will be in fact replaced because they currently lack of a qualification file.

Only one method of condition monitoring is used (i.e. insulation resistance measurement) for the NIS cables and the MV cables buried or in trenches at the Doel NPP. Therefore, as mentioned by the licensee, the Belgian Safety Authority recommends to investigate for MV and I&C cables whether the testing methodology can be extended with more quantitative condition assessment techniques.

3.A.3.2.4 Preventive and remedial actions for electrical cables

The preventive and remedial actions presented by the licensee are considered satisfactory.

3.A.3.3 Conclusions

The licensee has identified the relevant ageing mechanisms of cables and has developed the means to identify currently unknown ageing mechanisms. The general method for following the cables' environment and for the selection of cables to be monitored follows the international guidelines and is satisfactory.

For the LTO units, a systematic and comprehensive process of upgrade of the AMP related to the cables was performed. ENGIE Electrabel plans for Tihange 2&3 and for Doel 3&4 to move towards an ageing management program based on the program installed in the LTO units and on the most recent guidelines.

The cable management program is adapted to the specificities of those different types of cables and to their importance for safety.

The ageing management of the cables is considered satisfactory, with the evolution foreseen by the licensee.

3.A.4 Action plan

The current improvement foreseen by the licensee is considered to be sufficient for the Belgian Safety Authority. No further improvement has been identified by the Safety Authority for the ageing management of the cables and connections.

As noted by the licensee, it could however be valuable to investigate whether the currently applied testing methodology can be extended for the MV and I&C cables, in addition to their current ageing management program.

3.B BR2 Research reactor

3.B.1 Description of ageing management program for electrical cables (BR2)

3.B.1.1 Scope of ageing management for electrical cables (BR2)

The safety of BR2 depends only in a limited way on electrical cables. After a scram of the reactor, the cooling can be guaranteed by natural convection if the automatic block valves in the main primary circuit are placed in the correct position. Electrical signals are needed for a very short time to generate the necessary commands. Electricity is further needed for monitoring, lighting, ventilation and other support systems. In case of severe loss of coolant, which could endanger the natural convection, additional electrical power is needed for pump capacity.

The electrical cables of BR2 consists of the following groups:

- The 3 kV feed of the main pumps: the operation of these pumps is not necessary after scram of the reactor. The cables have no safety function.
- The 10 kV normal feed cables: these cables feed the normal and the vital electrical boards. The vital boards can also be fed by diesel generator with a no break take over. The safety function of these cables is limited.
- The normal internal 380 V AC distribution systems: these cables have no safety function.
- The vital internal 380 VAC distribution systems: these cables have a safety function for support after scram.
- The 110 V DC system: the system is fed by batteries and can also serve a limited number of AC users.

3.B.1.2 Ageing assessment for electrical cables (BR2)

The main electrical feed cables are in use since the start of the reactor. These cables will be replaced in the framework of the renewal of the diesel generators. The new building for the generators is chosen to be closer to the transformers, thus minimizing the length of the feed cables.
The internal 380V AC electrical system uses still the original cables. There is no separation between signal cables and power cables. However, no problems occurred until now. Since 2016 all systems needed for the safety of the reactor are fed by a newly installed redundant cable system. These cables follow a different route than the normal cables trays and the cables are fire retardant. The new cable trays can resist the reference earthquake of the installation.

The 110V DC system is fed by battery. It is a distributed system, with a main battery group and several local battery units. The grid has a floating voltage level. In this way, it is tolerant against one isolation fault. This 110V DC grid has a continuous isolation measurement of the isolation, such that this single fault does not remain hidden.

The cables of the neutron measurements chains pass through the reactor pool and are well protected against radiation by the water. The water temperature of the pool is low (maximum 35 °C). The instrumentation is checked before every operation cycle (typically 5 per year). The cables can easily be replaced in case a fault is detected.

3.B.1.3 Monitoring, testing, sampling and inspection activities for electrical cables (BR2)

Legal requirements (non-nuclear)

The electrical power cables are checked according to the (non-nuclear) legal requirements. High-tension cables are checked every year, low tension every 5 year. The measurements are mostly the isolation and the earthing system.

Additional monitoring

Every year a thermography of the connections is made, alternating with the reactor in operation and during shutdown when different systems are in operation.
Since 2014 a measurement program with trend follow up of the important power cables is started. Isolation values and continuity of the cables is measured.

The 110 VDC system is continuously monitored for isolation faults. This, together with the tolerance against a single isolation faults, makes the system very reliable.

3.B.1.4 Preventive and remedial actions for electrical cables (BR2)

Two actions as a consequence of ageing of electrical cables are taken:

- The essential cables of the 380 VAC system and the essential instrumentation cables have an alternative back up cable, following a route different from the normal cable route.
- The important high tension feed cables, which are more than 55 years in operation without any problems, will be replaced at the occasion of the construction of the new emergency generators (changing in routing of the cables).

3.B.2 Licensee's experience of the application of AMPs for electrical cables (BR2)

No problems due to cable ageing were observed in the past. None of the electrical cables is located in places with severe operating conditions (pressure, temperature, radiation). The actual follow up program, based on usual industrial standards is sufficient for early detection of safety problems due to ageing. A few measures (replacement or doubling of cables) have been taken by precaution.

3.B.3 Regulator's assessment and conclusion on ageing management of electrical cables (BR2)

3.B.3.1 Regulatory oversight process

The ageing management of cables was assessed during the last PSR in Safety Factor 3 (equipment qualification) and in Safety Factor 4 (ageing).

Also, in the framework of the TPR, a thematic inspection at the SCK•CEN was conducted on the Ageing Management Program, covering also the electric cables.

3.B.3.2 Regulator's assessment

3.B.3.2.1 Scope of ageing management for electrical cables (BR2)

No formal ageing management program exists at SCK•CEN for electrical cables. However, different actions are undertaken to manage their ageing. The extent of these actions varies for the different groups of electrical cables. Those actions are evaluated in the following sections.

3.B.3.2.2 Ageing assessment of electrical cables (BR2)

According to the licensee, the following elements justify partly the absence of a formal ageing assessment for the electrical cables:

- There are by design no safety related cables located in an adverse environment of the reactor. In particular, the neutron flux cables are shielded by a pool and their environment temperature does not exceed 35°C;
- The High Voltage (HV) feed cables will be replaced in the framework of the renewal of the Diesel generators. An ageing follow-up program for those cables will be defined after their replacement;
- The Medium Voltage (MV) safety related cables have been replaced in 2016;
- By design, the 110 V DC network is tolerant to one isolation fault. Further, this network has a continuous measurement of isolation in order to detect the first isolation fault.

However the cables did not follow a specific environmental qualification program, i.e. a program that demonstrates (e.g. based on type tests including an accelerated ageing) the capacity of the cables to fulfil their safety function in the environmental conditions (such a dose rate, temperature or humidity) that will occur at that time. The qualification to accident conditions mentioned by the licensee is in fact only related to their fire resistance. In consequence, the Safety Authority recommends that the licensee demonstrate that the safety related cables can fulfil their safety function in all relevant accidental conditions.

The Safety Authority considers that all safety related electrical cables (including I&C cables) should be included in the ageing management program including cables that have recently been replaced.

In addition the Safety Authority recommends performing walk-downs in order to confirm the absence of adverse environment (e.g. wetting or abnormal local cable's heating mechanisms) and to update then the ageing management program accordingly.

3.B.3.2.3 Monitoring, testing, sampling and inspection activities for electrical cables (BR2)

The Safety Authority considers that the functional checks of the neutron flux instrumentation cables do not allow assessing the condition of the cables before they fail. The development of a formal ageing assessment for those instrumentation cables, followed by an eventual upgrade of the ageing management program (and in particular the monitoring program) is judged therefore to be especially relevant by the Safety Authority.

3.B.3.2.4 Preventive and remedial actions for electrical cables (BR2)

In general, replacement is foreseen for cables that fail. The Safety Authority considers this strategy feasible and acceptable taking into account the design features limiting the impacts of cable failures (backup cable for essential 380 V and instrumentation cables).

3.B.3.2.5 Licensee's experience of the application of AMPs for electrical cables (BR2)

The licensee reports that no particular problems related to cable ageing were identified in the past and considers the actual current program as adequate. Nevertheless, replacement of cables or doubling of cables has been taken as precautionary measures.

3.B.3.3 Conclusions

A formal ageing assessment program for the electrical cables has not yet been established by SCK•CEN. Nevertheless, a monitoring program exists for some types of cables and cables were recently renewed or their renewal is planned in the near future. Furthermore SCK•CEN claims that by design no safety related cables are located in an adverse environment of the reactor. To support this claim and the development of the ageing assessment program for the electrical cables, the Safety Authority requires SCK•CEN to perform walk-downs in order to confirm the absence of adverse environment (e.g. wetting or abnormal local cable's heating mechanisms), in particular for the safety related cables that were not recently replaced or that are not planned to be replaced in the future (for instance the 110V cables and instrumentation cables).

It is also concluded that the qualification of the cables in accidental conditions is limited to fire resistance. The nuclear Safety Authority asks the licensee to demonstrate that the safety related cables can fulfil their safety function in all relevant accidental conditions.

3.B.4 Action plan

The part of the action plan PSR 2016 related to the ageing management program, in particular electrical cables, should be extended by SCK•CEN taking into account the following:

- Establish a formal ageing assessment program for all safety related electrical cables. Priority should be given to neutron flux instrumentation cables and the 110V cables. The ageing management program should then be upgraded in accordance.
- Perform walk-downs in order to confirm the absence of adverse environment (e.g. wetting or abnormal local cable's heating mechanisms) and in support of the development of the ageing assessment program for the electrical cables. Particular attention is needed for the safety related cables that were not recently replaced or that are not planned to be replaced in the future (for instance the 110V cables and instrumentation cables).
- Demonstrate that the safety related cables can fulfil their safety function in all relevant accidental conditions.

4. Concealed Pipework

4.A Nuclear Power Plants

4.A.1 Description of Ageing Management Programs for Concealed Pipework

This chapter focuses on the Ageing Management Programs for concealed pipework as described in the Technical Specifications of the National Assessment Report [3B]. The concealed pipework in scope of this report includes:

- Pipework buried in soil or concrete, cast in concrete and placed in covered trenches
- The connections of these pipes, including welds, flanges and bolts

4.A.1.1 Scope of Ageing Management for Concealed Pipework

ENGIE Electrabel progressively manages the ageing of concealed pipework since more than ten years. Non-destructive testing, inspection programs and modification of the concealed pipework have been introduced in order to deal with the ageing phenomena.

From 2009 on, ENGIE Electrabel has published Ageing Summaries regarding concealed pipework. AS were the main deliverables of the Ageing Management Program (see Section 2.A.3.1.1 Supporting tools for all units of this NAR).

Later, ENGIE Electrabel implemented AMPs in accordance with the NUREG 1801 rev1 [4H] in the framework of the Long Term Operation of Tihange 1 and Doel 1&2.

Through the years, the definition and scope evolved. In 2016, ENGIE Electrabel launched a global Buried Piping Program and started structuring all past initiatives in order to centralize the management of such pipes for its nuclear power plants.

In this Buried Piping Program, ENGIE Electrabel groups safety-related pipework located in areas with restricted access for inspection. This includes pipework buried in soil, cast in concrete, installed in covered trenches, manholes and wells where human access or movement is limited.

4.A.1.1.1 Methods and criteria used for selecting concealed pipework within the scope of ageing management

In the 2009 developed AS, identification of the safety-related pipework was largely based on expert judgement, driven by internal and external operating experience.

In the framework of the LTO of Tihange 1 and Doel 1&2, the safety-related pipework has been identified systematically using the criteria specified in the scope definition of the U.S. License Renewal rule [4A].

In analogy with the passive mechanical scoping methodology applied for the LTO units, the safety scope of units Tihange 2 and 3 and Doel 3 and 4 is based on:

- Class 1 seismic circuits (quality group A, B and C)
- Effluent treatment circuits
- Diesel auxiliary circuits
- Fire extinguishing circuits

The technical details of the above scope are being elaborated per pipework section, based on:

- Interviews with experts
- Analysis of technical documents, technical specifications, safety analysis reports, drawings, piping and instrumentation diagrams (P&IDs), isometrics and existing databases
- Walkdowns
- Synergy between Doel and Tihange

ENGIE Electrabel is summarizing the collected data in a table in order to progressively build a complete Master Database of the piping in scope. This Master Database will group all necessary information such as dimensions, environment, installed materials and service conditions.

The Master Database will also include links to the recorded isometrics, which will make it possible to locate every connection (welds, flanges, bolts, etc.) in scope.

As required by the Technical Specifications of this NAR, this chapter only addresses the following types of pipework:

- Fuel transfer pipes
- Fire water pipes
- Cooling water to safety-related SSC pipework

Site	Circuit	Environment	Material	Internal fluid	Coating
Doel	Fire water	Soil / Air indoor uncontrolled	Carbon steel	Raw water	Bitumen (and cathodic protection)
					PE (and cathodic protection)
	Cooling	Soil / Air indoor uncontrolled	Carbon steel	Raw water	Bitumen
	Diesel	Air indoor uncontrolled	Stainless steel	Fuel	Isolation and tracing
		Soil	Carbon steel		Double wall
			Stainless steel		
Tihange	Fire water	Soil	Ductile cast iron	Raw water	External bitumen-based zinc layer Internal centrifuged cement mortar
			Glass-reinforced epoxy		/
	Cooling	Air indoor uncontrolled	Carbon steel	Raw water	/
					Rustproof paint
	Diesel	Air indoor uncontrolled	Stainless steel	Fuel	/

Table 4A-1

4.A.1.1.2 Processes and procedures for the identification of ageing mechanisms related to concealed pipework

Ageing Summaries describe, for the systems in scope of the AS, the problem and its significance, which allows to identify the main ageing mechanisms in place and their contributing causes.

For the LTO units, the systematic screening and Ageing Management Review process allowed to identify the ageing mechanisms and ageing effects in place, potentially affecting the integrity of each component of the concealed pipework, as required by the GALL [2], and this in function of each material/environment combination. The results of this process are one or several AMPs, which are being implemented for the in-scope pipework, ensuring that the required functions are maintained, reliable and available for the lifetime period.

As part of the Buried Piping Program, ENGIE Electrabel has decided to extend the exercise to the other units. This exercise is based on the same methodology as applied for the LTO units: the GALL report [4H]. However, in order to complete the existing ageing mechanisms, other external sources such as EPRI [5A] and IAEA [2P] have been used, together with other site-specific ageing mechanisms.

4.A.1.1.3 Grouping criteria for ageing management purposes

In the framework of the Buried Piping Program, the pipework and their connections are grouped based on the following characteristics:

- Component part (pipe, elbow, reducer, connection, etc.)
- Construction material (carbon steel, stainless steel, glass-reinforced epoxy, ductile cast iron, high-density polyethylene)
- Internal (raw water, fuel oil, treated water) and external environment (air indoor uncontrolled, soil)

The categorization of the characteristics is based on the GALL report [4H], this methodology has also been used in the framework of the LTO projects.

Based on the:

- Type of each mechanism (electrochemical, thermal, mechanical)
- Localized or generalized nature of the failure associated to the ageing mechanism
- Inspection and control techniques applied to qualify the impact of the mechanisms

… the ageing mechanisms have been grouped into the following macro ageing mechanisms (more details in Appendix A of this chapter):

- General corrosion
- Localized corrosion
- Internal and external mechanical damage
- Structural deformation
- Mechanical loss of material
- Fouling

4.A.1.2 Ageing Assessment for Concealed Pipework

4.A.1.2.1 Ageing mechanisms

In the framework of LTO and the Buried Piping Program ageing mechanisms are being identified and evaluated for all units. The main ageing mechanisms affecting the NAR examples are:

- Internal and external general corrosion

- Internal localized corrosion (pitting, crevice and microbiologically-influenced corrosion)
- Fouling

The organization of the report was slightly modified as compared to the proposed table of contents of the Technical Specifications of the NAR. It was found that it would enhance the readability of the report to treat the three following aspects for each of the above-mentioned types of pipework:

- Identification and assessment of ageing mechanisms
- Preventive monitoring, testing, sampling and inspection activities
- Corrective actions

For each NAR example, external ageing is managed by the Buried Piping and Tank Surveillance AMPs. For internal ageing, specific AMPs are identified in each subsection hereunder, depending on the internal environment.

In addition, the differential settlement of the buildings' foundations has been monitored, allowing the assessment of the ground movements and the potential structural deformation.

The situation for the three types of pipework is discussed in the following subsections.

4.A.1.2.2 Fuel transfer pipes

Identification and assessment of ageing mechanisms

In addition to the Buried Piping and Tank Surveillance AMPs, internal ageing of fuel transfer pipes is managed by the Fuel Oil Chemistry Program. This program aims to mitigate the loss of material on the internal surface due to general, pitting, crevice, microbiologically-influenced corrosion and fouling. It also reduces the risk of exposure of the fuel system's internal surface to fuel contaminated with water and microbiological organisms.

Regular monitoring and fuel oil control is performed to check whether fuel oil quality meets the acceptance criteria. These acceptance criteria are in accordance with the plant's technical specifications and the guidelines of the American Society for Testing and Materials.

Preventive monitoring, testing, sampling and inspection activities

Today, external corrosion is monitored through **external inspections** while internal corrosion and fouling are monitored through **fuel analysis** and **periodic testing.**

External inspections

According to the Buried Piping and Tank Surveillance AMPs, the fuel transfer pipes and their connections are visually inspected during excavation works or opening of the trenches. During these inspections the pipework is checked for possible leakages, and the mechanical and structural integrity of the pipework and its welding joints and supports are verified.

Fuel analysis

The fuel quality is assessed based on:

- Fuel supply monitoring
- Diesel fuel sampling

The supplied fuel must meet well-defined physicochemical characteristics. Every parameter is evaluated based on strict specifications limits and a certificate of conformity must be provided upon delivery.

In accordance with the criteria specified in the technical specifications, fuel samples are collected periodically to monitor the concentration of corrosion and fouling reagents. The following parameters are verified:

- Viscosity
- Sediment concentration
- Water concentration

This approach is in conformity with the technical specifications.

Every external inspection and fuel analysis is evaluated and deviations are treated. Up to now, no major deviation has been observed.

Periodic testing

In addition, the proper functioning of the diesel engines is frequently tested according to the technical specifications.

The fuel circuit is monitored through the Health Reporting process and main parameters of the circuit are monitored by the Operations department.

Corrective actions

In function of the results of the predefined preventive inspections, ENGIE Electrabel implemented corrective actions based on good engineering practices and industrial recommendations. For example, inspections on carbon steel single wall fuel pipework showed corrosion in moist environments. Hence, ENGIE Electrabel decided to adapt the installations:

- Replacement of the concealed fuel pipework with stainless steel piping
- Treatment of the trenches to make them leak-tight regarding fuels
- Modification of the pipework trajectory to make it as visible as possible
- Channelling potential leaks to a collection tank
- Double wall piping with leak detection

4.A.1.2.3 Fire water pipes

Identification and assessment of ageing mechanisms

In addition to the Buried Piping and Tank Surveillance AMPs internal ageing of fire water pipes is managed by the Fire Water System AMPs. These AMPs focus on managing the loss of material due to corrosion, microbiologically-influenced corrosion (MIC) or biofouling of carbon steel and cast iron components in fire protection systems exposed to water. To ensure no significant general and localized corrosion or biofouling has occurred, periodic flushing, system performance testing, and inspections are conducted.

Preventive monitoring, testing, sampling and inspection activities

Today, external and internal corrosion are monitored through inspections, while internal fouling is monitored through periodic flushing and testing.

Inspections

The condition of the buried fire water pipes is monitored based on:

- Visual inspection of the external pipe surface during excavation works. Prior to entering a period of extended operation, staff verifies that there has been at least one inspection or a focused inspection the past ten years.
- Visual inspection of the inner wall during equipment maintenance. To limit unavailability of the pipework, these inspections are always combined with revisions and maintenance of the section valves.
- Monthly local electrical measurements to check the effectiveness of the cathodic protections of the carbon steel cast iron pipework against any signs of corrosion.
- Monitoring and conditioning of the fire water system's water quality.
- Control room monitoring of the condition and pressure of the charging pump of the buried circuit.

Periodic flushing and testing

In addition, the fire water system is subjected to the following testing programs:

- Periodically, the fire water stations are tested and flushed at full charge.
- Every five years, the fire water circuit is tested at full capacity with pressure drop and flow measurements during normal operating conditions. A flow is forced and the pressure drop is measured. This gives an indication of the flow obstruction and the inner wall condition.

In addition, the fire water system is monitored through the Health Reporting process and main parameters of the circuit are monitored by the Operations department.

Corrective actions

Tihange

Through the years, ENGIE Electrabel launched various preventive and remedial actions, especially at Tihange, where the fire water circuit was frequently used for other purposes than in Doel, and hence was more prone to corrosion. To resolve this issue, ENGIE Electrabel installed:

- An industrial water circuit to avoid the use of water for other purposes than firefighting (i.e. road cleaning, fliter cleaning, etc.) and, as such, limit fouling and corrosion of the fire water pipework.
- A new system to condition (the water from the river Meuse is either acid or basic) and distribute the water in the fire water circuit. A ground water pump, which fills the fire water circuit for the three Tihange units with clean water, limits fouling and corrosion of the fire water circuit.

The buried fire water circuit of Tihange 1 consists of ductile cast iron pipes, coated externally with a bitumen-based zinc layer and cladded internally with centrifuged cement mortar. In harsh soils, a 0.2 mm polyethylene protection provides additional protection against external corrosion. The fire water circuit for Tihange 1 was replaced in 1988. At the end of 2006, a laboratory inspection of the joints together with internal and external surface control of the pipework confirmed the pipework's good general condition with no particular degradation.

The fire water circuit of Tihange 2 and 3 was a buried carbon steel pipework embedded in a protective asphalt layer. Due to the deposit of corrosion products, ENGIE Electrabel replaced the Tihange 2 and 3 fire water circuit and piping towards the Meuse with glass-reinforced epoxy (2012-2016). Glass-reinforced epoxy is corrosion-resistant and has a long life expectancy. In addition, the physical properties of these filament wound pipes, in combination with the used resin and glass quality, makes the glass-reinforced epoxy resistant against the most severe chemical and mechanical forces.

Doel

The buried circuit of Doel consists of carbon steel pipework, coated externally and cathodicaly protected. Based on careful maintenance of the cathodic protection system, the circuit has maintained its good general condition throughout the years. However, parts of the circuit of Doel 1&2 have been replaced after excessive use.

At Doel, the fire water system is either filled up with town water or pre-treated lake water which, compared to Tihange, decreased internal corrosion significantly.

4.A.1.2.4 Cooling water to safety-related SSC pipework

In Tihange, two distinct parts of the cooling water to safety-related SSC pipework are in scope:

- The pipework behind the safety groundwater pumps of Tihange 1
- The cooling of the safety diesel engines of Tihange 1

For Doel, the system in scope is the raw water circuit, which is used for cooling down the intermediate circuit in case the circulation water circuit is not available.

Identification and assessment of ageing mechanisms

In addition to the Buried Piping and Tank Surveillance AMPs internal ageing of the cooling water circuits is managed by the Open Cycle Cooling Water AMPs. These AMPs address the ageing effects of fatigue, material loss and fouling due to micro- and macro-organisms and various corrosion mechanisms generally found in Open Cycle Cooling Water systems. These ageing effects are managed by a combination of preventive, condition, and performance monitoring activities..

Preventive monitoring, testing, sampling and inspection activities

Today, external and internal corrosion are monitored through inspections, while internal fouling is monitored through periodic testing and chemical analysis.

Inspections

- Visual inspections of the external pipe surface are conducted during excavation works or periodic inspections.
- Anomalies due to internal corrosion and fouling are monitored through visual inspections, mainly during maintenance works on active components.
- The structural integrity of the pipes is inspected using UT wall thickness measurements.

Periodic testing and chemical analysis

- The raw water is not only filtered, its chemistry is also closely monitored. In addition, the raw water circuits can be subjected to biocide treatment (sodium hypochlorite, antifouling, etc.) to prevent the proliferation of micro-organisms such as algues, bacteria and mussel seeds.
- Circuit performance and fouling are tested periodically.

In addition, the raw water circuits are monitored through the Health Reporting process and main parameters of the circuit are monitored by the Operations department.

Corrective actions

Tihange

The MIC program allows to precisely identify where local corrosion points are located. In function of their number and depth, corrective actions are performed. These actions can go from thickness monitoring, to local repair and even pipework section replacement.

Doel

The Doel raw water system is currently ongoing a global ageing project. Thorough endoscopic inspection allowed to characterize the condition of the pipework. Corrective actions are being put in place and spread over time in function of their importance:

- If coating imperfections should occur, ENGIE Electrabel will launch the appropriate corrective maintenance works, for instance local repairs or a complete reapplication after sandblasting
- In case of advanced ageing, pipework sections replacements or global repairs

4.A.2 Licensee's Experience of the Application of AMPs for Concealed Pipework

ENGIE Electrabel issued the Buried Piping Program in 2016 and relevant AMPs to manage the different ageing mechanisms that could potentially affect concealed pipework and their connections. These programs are based on various internal and external inspections, thickness measurements, chemical analyses and treatments, periodic testing and proactive repairs and replacements. At both the Doel and Tihange NPP sites, dedicated engineers have been appointed to finalize and execute the Buried Piping Program.

ENGIE Electrabel will continue to monitor the preventive testing, sampling and inspection activities in order to anticipate, prevent and treat degradations of the concealed pipework in its NPPs. In addition, international operating experience will be gathered throughout updates of the AMPs and workgroup participation. For instance, ENGIE Electrabel will closely monitor the model-supported development of non-destructive testing methods for defect detection in composite parts such as glass-reinforced epoxy (ENGIE Laborelec is a member of the involved Advisory Board).

In the future, ENGIE Electrabel plans to let the Ageing Management Program for Doel 3 and 4 and Tihange 2 and 3 evolve in line with the program installed for the LTO units and based on the most recent international guidelines.

4.A.3 Regulator's assessment and conclusions on ageing management programs for concealed pipework

4.A.3.1 Regulatory oversight process

Up to now, Bel V has never performed a specific follow-up of the ageing management activities regarding concealed pipework. Nevertheless, several issues related with concealed pipework were analyzed and followed adequately on the basis of the day-by-day activities of the Safety Authority. As part of its analysis, Bel V has verified conformance of the ageing management program of ENGIE Electrabel to the international standards [2D, 4H, 5A].

4.A.3.2 Regulator's assessment

4.A.3.2.1 Scope of ageing management for concealed pipework

In the ageing management programs implemented at ENGIE Electrabel, also pipework outside the scope of this TPR is managed, including non-safety related pipework, and pipework which is underground but not strictly concealed. Since more than 10 years, ENGIE Electrabel has progressively installed ageing management programs which manage the ageing of concealed piping.

Originally, the methods and criteria used for selecting concealed pipework within the scope of the ageing management were based on expert judgment, driven by internal and external operating experience. The ageing management programs managing this selected pipework were called 'Ageing Summaries'. In the framework of the LTO of Tihange 1 and Doel 1&2, the safety-related concealed pipework has been identified more systematically using the criteria specified in the scope definition of the U.S. License Renewal rule 10CFR54.4 [3]. The programs managing ageing for these LTO-units were called 'Ageing Management Programs' or 'AMPs'. The scoping methodology from the LTO-units is currently being extended to the other Belgian units in the framework of the Buried Piping Program, covering all Belgian Units. The finalization of this Buried Piping Program is foreseen at the end of 2017.

The pipework is further categorized according to its material and environment. The current grouping envelopes adequately these piping segments. Also the ageing mechanisms related to concealed pipework have been grouped according to several criteria. The identification of these ageing mechanisms is consistent with the methodologies described in [2D, 4H, 5A].

4.A.3.2.2 Ageing assessment for concealed pipework

ENGIE Electrabel identified the main ageing mechanisms for the concealed pipes.

For the first units in LTO, the Buried Piping and Tank Surveillance AMPs were started in 2016. In addition, specific AMPs exist for internal ageing. For the other units, no such specific ageing management programs for these pipes formally exist. Only fragments of these ageing management programs are included in different 'Ageing Summaries'. However ENGIE Electrabel plans to implement the same approach on these units as applied for the LTO units in the upcoming years.

The existing AMPs listed above, and the relevant parts of the Ageing Summaries are in line with internationally accepted standards and guidance [2-5].They are updated according to the latest R&D results and results obtained by international workgroups in which ENGIE Electrabel participates (e.g. [4]). Also relevant internal and external operating experience is taken into account. This is in line with attribute 8 of the IAEA IGALL.

4.A.3.2.3 Monitoring, testing, sampling and inspection activities for concealed pipework

ENGIE Electrabel lists all monitoring, testing, sampling and inspection activities currently implemented in its units to manage ageing for concealed pipework. These activities concern external as well as internal ageing management of the considered pipework.

In the framework of its Buried Piping Program, ENGIE Electrabel will perform a gap analysis between the ageing management programs for the different units and extend the monitoring, testing, sampling and inspection activities where necessary, in order to reach consistent ageing management programs for all units. Such gap analysis and extension of the monitoring, testing, sampling and inspection activities is effectively needed, because the ageing management programs for the different units currently do not have the same level of adequacy.

The results from all the monitoring, testing, sampling and inspection activities are used as an input for the Health Reporting process which monitors the long term behavior of a system. Health Reporting processes exist for each of considered pipe types. This type of trending analysis helps to timely detect anomalous situations.

4.A.3.2.4 Preventive and remedial actions for concealed pipework

In order to avoid ageing of concealed pipework, the licensee specifies two main preventive measures, i.e. material choice (e.g. protecting external and internal claddings, glass fibre reinforced plastic) and water conditioning. Preventive replacements have been performed in Doel as well as in Tihange throughout the years in order to improve the resistance of the existing concealed pipework against ageing.

Despite the preventive measures, ENGIE Electrabel also lists a number of corrective replacements which were needed due to ageing degradation which was discovered by the monitoring, testing, sampling and inspection activities.

The preventive and remedial actions are conform to the recommendations of the IAEA IGALL, attribute 2 and 7 [2D].

4.A.3.2.5 Licensee's experience of the application of AMPs for concealed pipework

ENGIE Electrabel gives some examples of preventive replacements which were performed after non-critical ageing degradation was detected by the monitoring, testing, sampling and inspection activities. This demonstrates that some ageing phenomena proceed as predicted by the ageing management program of the licensee, and that the frequency and acceptance criteria of the monitoring, testing, sampling and inspection activities are effective in their purpose to timely detect ageing degradation.

However, there are still some cases where ageing degradation is not timely detected and corrective actions are necessary. In order to enhance the ageing management program for concealed piping, the licensee will continue to update the AMPs based on external and internal operating experience and plans to let the Ageing Management Program for Doel 3, Doel 4, Tihange 2 and Tihange 3 evolve in line with the program installed for the LTO units and the most recent international guidelines. The Buried Piping Program which is currently ongoing is consistent with this strategy.

The proposed actions are sufficient to further enhance the ageing management program for concealed piping.

4.A.3.3 Conclusions

The existing AMPs of the LTO units, and the relevant parts of the Ageing Summaries of the non-LTO units are in line with the internationally accepted standards and guidance. It has been demonstrated that in several cases, ageing is effectively managed. For some cases however, improvement is possible.

In order to enhance the ageing management program for concealed piping, the licensee will continue to update the ageing management programs based on external and internal operating experience and plans to let the ageing management program for Doel 3, Doel 4, Tihange 2 and Tihange 3 evolve in line with the program installed for the LTO units and the most recent international guidelines. ENGIE Electrabel drew the conclusions of its own self-assessment performed in the framework of the TPR and already launched adequate actions in order to fill the gaps identified. A Buried Piping Program is then ongoing which is consistent with this strategy and which will manage the ageing of the buried piping in a centralized and consistent manner for all units.

4.A.4 Action Plan

The Buried Piping Program and the already existing plans to extend the AMPs for the LTO units towards the non-LTO units are appropriate actions in order to enhance the current ageing management program for concealed piping at the Belgian NPPs.

4.B BR2 Research reactor

4.B.1 Description of ageing management program for concealed pipework

4.B.1.1 Scope of ageing management for concealed pipework

Following concealed pipework is present at BR2:

With safety significance due to potential ground contamination:

- Piping for transporting pool water from the reactor building to the storage basin under the ventilation building (piping placed in a concrete cellar).
- Piping for radioactive contaminated waste water and industrial waste water (double walled buried piping with leak control).

With importance for operation (all this piping is buried):

- Piping for city water.
- Feed water line for firefighting in the reactor building.
- Feed line for cooling water from the lagoon.
- Fuel transport piping for the diesel generators and the heating.

Support system without safety function (out of the scope for this report):

- Natural gas feed piping for heating.
- Secondary cooling piping (both main cooling and experiment cooling).
- Industrial waste water.

The piping for transfer of pool water between the buildings and the radioactive waste water piping contain contaminated water and are a potential cause of ground contamination. The water is used to refill the reactor pool, also in case of a loss of coolant accident. However, alternative water sources are available. In the past, this piping was buried in soil.

Piping with safety significance for operation are the fuel line and the cooling water for the diesel generators. This piping is buried in soil. If the flow through the piping is insufficient, due to breakage or severe leakage, the generators could become unavailable because the day tanks in the generator building will not be refilled.

Diesel generators can be cooled using city water or surface water. In case of unavailability of one of the lines due to severe leakage, the other system serves as a back-up. A storage tank that contains sufficient water for at least 24 hrs working of the generators is also available on site. The piping of both circuits is standard buried pipework. Due to the redundancy, this is considered sufficient by SCK•CEN.

The fuel line for the diesel generators fills the day tanks in the building. Failure of these lines will not lead to a stop of the generators on a short time.

Although this piping serves safety systems, they are not critical, due to the redundancy and the diversity.

4.B.1.2 Ageing assessment for concealed pipework

Various ageing phenomena of buried piping were observed during the last years. No specific standards were used for assessing the ageing, since all these piping is under replacement. This replacement was decided due to a number of leaks that occurred in the piping of the secondary circuit. Although the secondary circuit has no safety function, these leaks have an important impact on the reliability and it was decided to replace the whole circuit. Leakage occurred in places where the piping passes under a road. This road is salted in winter time and this is probably the reason of an increased corrosion rate at this location.

The piping for transfer of pool water between the reactor building and the storage pool under the ventilation building was replaced based on return of experience from another research reactor where contamination of ground water by tritium was caused by leakage of this type of piping.

4.B.1.3 Monitoring, testing, sampling and inspection activities for concealed pipework

Problems with concealed pipe work will never endanger the safe shutdown of the reactor and the reactor can remain in a safe state for a longer period. The risks due to failure of concealed piping are ground contamination by radioactive isotopes or by diesel fuel. The new piping system has a leak detection system:

- The transfer piping for pool water is placed in a concrete cellar. The cellar has a water detection. Presence of water could be caused by leaking piping. In this case, samples of the water will be taken and analysed by gamma spectroscopy to identify the origin of the water. The stainless steel piping is visible and can be inspected. However, at this moment no inspection program is foreseen due to the low loading of the piping (demineralized water at low temperature and low pressure). Due to a design change, this piping is no longer to be considered as concealed piping
- The radioactive waste water piping is double walled with leak detection and alarm in case of presence of water in the double wall.
- Fuel lines are double walled with leak detection, as required by regulation on environmental protection (norm EN 12285-1, class A component).

4.B.1.4 Preventive and remedial actions for concealed pipework

All important concealed pipework has been replaced as indicated in 4.B.1.2. The only buried piping that is not replaced is the industrial waste water and the feed lines for drinking water.

The old piping dated from the construction of the reactor and could not be inspected unless excavated. Return of experience from 1995 and later shows that the lifetime of the piping is at least 30 years, but degradation becomes potentially significant after 50 years, especially if environmental conditions are not favourable.

The piping for transfer of pool water was replaced in 2016. The new piping is constructed according to the standard ASME B31.1 and is made in stainless steel.

New pipework for transfer of radioactive contaminated fluids is double walled with leak control or is placed in a cellar. The parts located on the BR2 domain were replaced in 2016. Parts outside the BR2 site towards the waste treatment installations of Belgoprocess were replaced a few years earlier.

The pool water feed line, which contains no contaminated water, has been replaced in 2017.

4.B.2 Licensee's experience of the application of AMPs for concealed pipework

No AMP currently exists for the concealed pipeworks. Single wall underground piping cannot be inspected. Although probably the major part of the piping was still in good condition, this could not be proven and replacement was necessary.

4.B.3 Regulator's assessment and conclusion on ageing management of BR2 concealed pipework.

4.B.3.1 Regulatory oversight process

Up to now, Bel V has never performed a specific follow-up of the ageing management activities regarding concealed pipework. Nevertheless, several issues related with concealed pipework were analysed and followed as part of the regular activities.

4.B.3.2 Regulator's assessment

4.B.3.2.1 Scope of ageing management for concealed pipework

The scope for the ageing management for concealed pipework is considered to be complete. All relevant pipes are considered by the SCK•CEN.

4.B.3.2.2 Ageing assessment for concealed pipework

The apparent need of SCK•CEN to replace concealed pipework because ageing-related degradation, indicates that the ageing phenomena are not well understood or that the current ageing assessment was not sufficiently effective. It is therefore concluded by the Safety Authority that the ageing assessment should be revised for safety related concealed piping also taking into account the various ageing phenomena observed for the old concealed piping.

4.B.3.2.3 Monitoring, testing, sampling and inspection activities for concealed pipework

The only monitoring, testing, sampling and inspection activity reported on by SCK•CEN is the leak detection system. No other monitoring, testing, sampling and inspection activities for concealed piping are foreseen. The justification provided by SCK•CEN for this limited strategy is that problems with concealed pipework will never endanger the safe shutdown of the reactor and that the reactor can remain in a safe state for a longer period. In addition, SCK•CEN points out that due to the redundancy and the diversity of the systems, the concealed pipework is not critical, although some of these pipings are safety related.

Leak detection does not result in timely detection of ageing degradation and the Safety Authority consider that for some safety related concealed piping, additional monitoring, testing, sampling and inspection activities as the ones suggested in attribute 5.23 of IAEA SSR-10 [2] should be performed. More specifically:

- Piping for transport of pool water from the reactor building to the storage basin under the ventilation building is to be used during a LOCA accident, and is thus safety-related. In addition, this piping is also safety-related due to the potential for ground contamination. Given this, other monitoring, testing, sampling and inspection activities (e.g. periodic visual inspection) should be defined for periodic evaluation of the ageing status of this segment. In addition it is noted that this piping segment is easily accessible.

- Piping for radioactive contaminated waste water and industrial water is buried in soil and has been replaced by new piping with double walls and leak detection. This piping is safety related due to the potential for ground contamination. The double walled piping in combination with leak detection is considered by the Safety Authority to be a sufficient strategy to avoid the only risk which is ground contamination.
- Fuel transport piping for the Diesel generators and the heating is buried in soil and has been replaced by a double walled piping with leak detection. It is safety related because it fills the day tanks of the Diesel generators. Failure of these lines will not lead to a stop of the generators on the short time, but may lead to problems on the longer term when the day tanks are empty. There is no redundant system foreseen which fills the day tanks when the fuel transport piping is unavailable. Since the fuel transport piping is required to keep the reactor in a safe state in the short and long term in case of a loss of offsite power, and since no redundant system is foreseen to provide the fuel for the Diesels, the Safety Authority considers that other monitoring, testing, sampling and inspection activities besides leak detection should be defined for this concealed piping segment.
- Piping for city water and feed line for cooling water from the lagoon consist of two sets of standard piping buried in soil which and none of this piping contains leak detection. Both are safety related since they are used for cooling of the Diesel engines. Although there is a high level of redundancy and diversity for the cooling of the Diesel engines, the Safety Authority suggests giving consideration to monitoring, testing, sampling and inspection of these lines.
- The concealed part of the feed water line for firefighting in the reactor building consists of standard buried piping without leak detection. It is safety related since it is needed to mitigate the effects of a fire. Since the water line for firefighting is required to get the reactor in a safe state on the short term in case of a fire, and since no redundant system is foreseen deliver the water for firefighting, the Safety Authority considers that appropriate monitoring, testing, sampling or inspection activities should be defined as part of the ageing management program.

4.B.3.2.4 Preventive and remedial actions for concealed pipework

The preventive and remedial actions which are carried out by SCK•CEN in order to avoid or correct ageing of concealed pipework consist mainly of replacement of the considered concealed pipework. Up to now, all pipework in the scope of the TPR has been replaced since its construction, except for the piping for city water. It is mentioned by SCK•CEN that return of experience has shown that the lifetime of the piping is at least 30 years, but degradation becomes potentially significant after 50 years, especially if environmental conditions are not favourable.

The Safety Authority suggests performing an ageing assessment of the concealed city water piping, and depending on its results, that a replacement of the concealed city water piping be considered since this part has not been replaced since the construction of the reactor more than 55 years ago.

4.B.3.2.5 Licensee’s experience of the application of AMPs for concealed pipework

With regard to the licensee’s experience of the application of AMPs for concealed pipework, it is indicated by SCK•CEN that safety-related equipment is replaced correctively or preventively when the good condition of a concealed pipework could not be proven.

The Safety Authority notes that feedback from the assessment on replaced piping has not been explicitly used as information to demonstrate the adequacy of the current ageing management program or to adapt it where necessary.

4.B.3.3 Conclusions

Most safety related concealed piping was recently replaced and the most relevant sections of piping are now double walled and have leak detection. To a large extent SCK•CEN´s strategy to rely on leak detection and replacement is adequate according to the Safety Authority. However, leak detection does not necessarily result in timely detection of ageing degradation and the Safety Authority considers that additional monitoring, testing, sampling or inspection activities should be implemented for the piping for transport of pool water from the reactor building to the storage basin under the ventilation building, the fuel transport piping for the Diesel generators and the concealed part of the feed water line for firefighting in the reactor building. For the piping for city water and feed line for cooling water from the lagoon such additional monitoring is also suggested. In addition and based on the projections provided by SCK•CEN, the need for a replacement of the piping for city water should be investigated.

Finally, the Safety Authority suggests that SCK•CEN updates or validates its ageing management program based on data obtained from the piping that was removed.

4.B.4 Action plan

The part of the action plan PSR 2016 related to the ageing management program, in particular concealed piping, should be extended by SCK•CEN taking into account the following:

- Additional monitoring, testing, sampling or inspection activities should be implemented for:
  - the piping for transport of pool water from the reactor building to the storage basin under the ventilation building,
  - the fuel transport piping for the Diesel generators and
  - the concealed part of the feed water line for firefighting in the reactor building.
- The need for a replacement of the piping for city water should, based on the projections provided by SCK•CEN, be investigated.

Furthermore it is suggested to:

- Consider additional monitoring, testing, sampling or inspection activities for city water and feed line for cooling water from the lagoon;
- Update or validate the ageing management program for concealed piping based on data obtained from the piping that was removed.

5. Reactor Pressure Vessels

5.A Nuclear Power Plants

5.A.1 Description of Ageing Management Programs for RPVs

5.A.1.1 Scope of Ageing Management for RPVs

Description of the RPVs of the Belgian nuclear power plants

The RPV is typically constituted of the following components (Figure 5A-1, corresponding to Doel 3 and Tihange 2 RPVs).

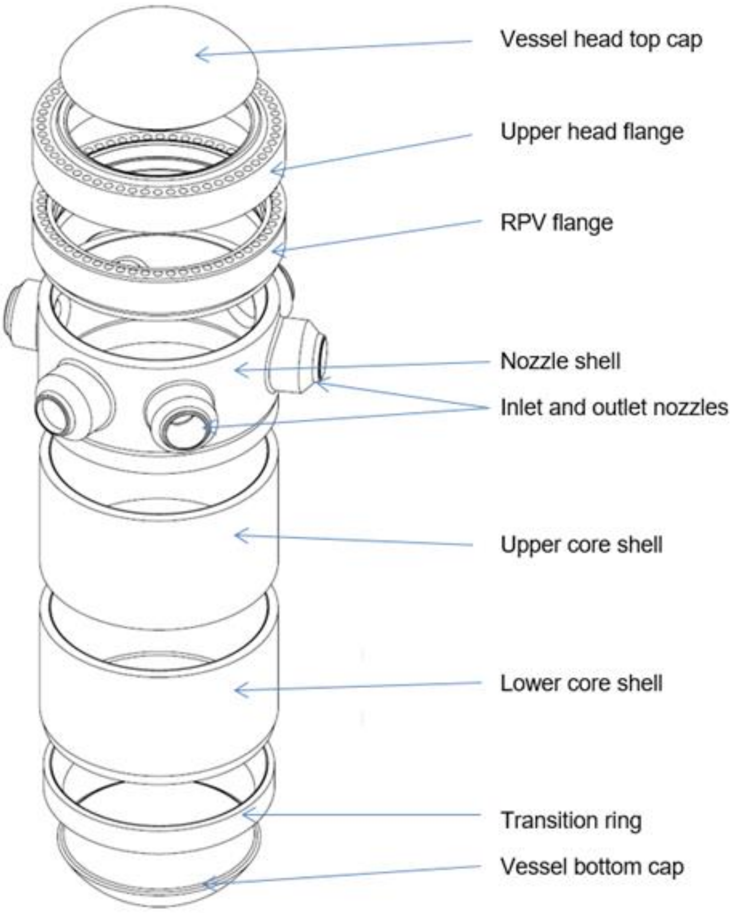


Figure 5.A-1: Main components of the reactor pressure vessel (example of Tihange 2, Doel 3)

Although the size of the different parts may vary between the Belgian units, the general design is the same. The RPV is constituted of several forged rings assembled by circumferential welds, with a hemispherical bottom and a flanged and gasketed removable upper head (Figure 5A-1). The body of the RPV is made of low-alloy carbon steel, and inside surfaces in contact with the coolant are clad with austenitic stainless steel to minimize corrosion.

However, there is a difference in design for the RPVs of Doel 4 and Tihange 3: in these cases, the nozzle shell and flange are combined into one piece, and the nozzles are of the set-on type. Also, the replacement RPV heads that were installed in 2015 have been forged in one piece.

A more detailed description of the RPVs of each Belgian nuclear power unit are given in appendix B.

Methods and criteria used for selecting components within the scope of ageing management

During the design of the Belgian nuclear power plants, the RPV had already been identified as an essential component of the primary circuit, for which a specific ageing management program was necessary. Hence, all parts of the RPV that constitute the pressure boundary are in scope.

Processes and procedures for the identification of ageing mechanisms for the different RPV materials and components

The ageing mechanisms identified for the different materials and components of the RPVs are described below. Some of them were already known at the design stage, like the irradiation embrittlement issue, which is monitored since the start-up of the units through the surveillance program, or low cycle fatigue, considered at the design stage as required by the ASME Code for class 1 components.

- The surveillance program for irradiation embrittlement is one of the key aspects of the RPV ageing management and is required by the U.S. rules applied in Belgium according to 10CFR50 app.H [4F]. It has been implemented in all units since start-up.
- The In-service inspection program as required by ASME Code Section XI is applied in the Belgian units since start-up, with enhanced programs for some specific issues.

Other ageing mechanisms appeared in the course of time and were identified by the follow-up of international (or national) operating experience. Some of the main events are listed below:

- Underclad defects in nozzles (detected during fabrication of the French RPVs at the end of the 1980s).
- Underclad defects in the RPV shell (detected in French unit Tricastin 1, 1999).
- Stress corrosion cracking in Ni-based alloy RPV upper head penetrations (Bugey, 1991; several French units, 1990s; U.S. units and worldwide, after 2000).
- Stress corrosion cracking in Ni-based alloy Bottom Mounted Instrumentation nozzles (South Texas 1, 2003; Gravelines 1, 2011; Palo Verde 3, 2013).
- Primary water stress corrosion cracking of dissimilar metal welds (nozzle to safe-end welds) (V.C. Summer, 2000; Ringhals 3-4, 2000; Ohi 3, 2008; several U.S. units).
- Boric Acid Corrosion of low-alloy steel in case of primary coolant leak (Davis Besse, 2002).

Some potential mechanisms were also evaluated, although no actual degradation was reported in operating experience, like the thermal ageing of low-alloy steel.

For each identified degradation mechanism, an Ageing Summary was established, describing:

- The specific component and degradation mechanism considered
- The current knowledge and operating experience on this degradation mechanism
- Potential significance of this degradation for the Belgian units
- The contributing causes
- The inspection techniques/schedule
- The acceptance criteria
- The possible mitigation and repair actions
- The situation of each individual Belgian unit regarding this degradation mechanism (operating experience, inspections performed and main findings, mitigation and/or repair actions implemented, recommended further actions)
- Administrative information like the reference of past projects, where more detailed information can be found, as well as persons in charge, participation to international working groups or projects

The Ageing Summaries procedure was initiated in the 1990s and new summaries were developed progressively when new degradation mechanisms were identified based on international operating experience and on R&D results. The summaries are revised regularly, with a frequency depending on the subject but not exceeding five years, in order to include the latest operating experience or R&D results. A revision is made independently of the planned schedule if a significant event occurs.

In the framework of the studies for the Long Term Operation of Doel 1&2 and Tihange 1 units, the more systematic approach codified in the U.S. License Renewal rule was followed. The whole RPV was obviously in the scope since it constitutes an essential part of the pressure boundary. The Generic Ageing Lessons Learned (GALL) Report [4H] was used to identify the degradation mechanisms potentially affecting the different parts of the RPVs, based on the Ageing Management Review tables. This confirmed that the Ageing Summaries covered all the identified degradation mechanisms for the RPV components.

Ageing Management Programs (AMPs) for the management of the ageing mechanisms during the LTO period were established along the guidelines of the GALL.

- M1: ASME XI In-service inspection, Subsections IWB, IWC and IWD.
- M3: Reactor head closure studs
- M10: Boric Acid Corrosion
- M11: Ni-alloy nozzles and penetrations
- M31: Reactor vessel surveillance
- Specific AMP M101: RPV nickel alloy Bottom Mounted Instrumentation (BMI) nozzles and welds

Time-Limited Ageing Analyses (TLAAs) were also established for the degradation phenomena involving a time hypothesis like irradiation embrittlement or fatigue.

Later, the Tihange 1 AMPs were revised to adopt the format that was recommended by the IAEA [2C][2D]. This was not done for Doel 1&2, although all generic IAEA AMP attributes are addressed.

When the International Generic Ageing Lessons Learned Report (IGALL) was issued by IAEA in 2015 [2D], a new review was performed to check that no additional degradation mechanisms had to be considered for the RPVs. It is planned to revise the AMPs to make them conform with the IGALL requirements, but for the AMPs relevant for the RPVs no new AMP is necessary and the needed modifications are relatively minor or purely editorial.

Provisions for identifying unexpected degradation

For the RPV, the in-service inspection program according to ASME XI and the augmented programs implemented by ENGIE Electrabel to address specific issues such as the PWSCC of Ni-based alloys or the underclad defects ensure the identification of degradation mechanisms inducing material cracking. A typical example is the discovery of the hydrogen flaking issue in the Doel 3 and Tihange 2 core shells during an inspection to detect any potential underclad defects. In this respect, it must be mentioned that hydrogen flakes are not an ageing mechanism but fabrication phenomena.

The inspection technique is qualified by the third-party certification organization EQB (Electrabel Qualification Body). In specific cases, Defect Tolerance Analyses have been performed in order to define the in-service inspection performance requirements.

The national and international operating experience is closely monitored by the regular participation to international groups like the PWROG (Pressurized Water Reactor Owners Group), the FROG (Framatome Owners Group), and by the participation to conferences. Sources of information such as the NRC Information Notices, International Incident Reporting System (IRS), WANO Significant Event Reports (SER), Westinghouse Nuclear Safety Advisor Letter (NSAL), Infograms and Technical Bulletins, etc. are also followed-up.

Many augmented inspections programs in the Belgian units were launched shortly after a first event was reported in the international operating experience, like for example the inspection of RPV upper head penetrations or of dissimilar metal welds.

For aspects related to the RPV irradiation embrittlement, in addition to the different media mentioned above, the regular participation to the International Group on Radiation Damage Mechanisms provides information on the latest scientific developments in the field of irradiation damage.

5.A.1.2 Addressing RPV Ageing Mechanisms

The different ageing mechanisms considered for the RPVs are:

- RPV irradiation embrittlement
- Low cycle fatique
- Underclad defects in RPV shells
- Underclad defects in nozzles
- PWSCC of BMI nozzles
- PWSCC of RPV upper head penetrations
- PWSCC of dissimilar metal welds
- Thermal ageing of low-alloy steel

As required in the technical specifications, the following components will be covered:

- The steel vessel including base metal, cladding and welds
- The vessel head and the lower dome including penetrations
- Inlet and outlet nozzles

The organization of the report was slightly modified as compared to the proposed table of contents of the technical specification. It was found that it would enhance the readability of the report to treat the three following aspects for each of the above-mentioned degradation mechanisms:

- Ageing assessment
- Monitoring, testing, sampling and inspection activities
- Preventive and remedial actions

The situation of all Belgian RPVs is addressed in the paragraphs dealing with each degradation mechanism considered.

5.A.1.2.1 RPV irradiation embrittlement

Ageing assessment of RPV irradiation embrittlement

Low-alloy carbon steel embrittlement occurs in the beltline region (the region that is exposed to a neutron fluence larger than 10E17 neutrons / cm² (E > 1 MeV)) of the RPV due to neutron bombardment. It increases the material's yield and ultimate strengths, and decreases material ductility and fracture toughness. Prevention of RPV failure depends primarily on maintaining RPV material fracture toughness at levels that avoid brittle fracture (with margins) during plant operation.

RPV irradiation embrittlement is one of the most significant RPV degradation mechanisms. It has been identified and followed since the early days of the nuclear industry and there is a vast amount of information on this topic.

In all Belgian units, RPV irradiation embrittlement is monitored through a surveillance program as required by 10CFR50 App.H [4F] and specified in ASTM E185 [6C].

For the LTO units (Doel 1&2 and Tihange 1) an AMP was established initially along the guidelines of the GALL [4H] and later on adapted to the format recommended by IAEA. TLAAs were established along the guidelines of the GALL [4H] to justify the acceptability of RPV irradiation embrittlement for 50 years of operation.

Surveillance programs

Test specimens, representative for the beltline region (shells, weld, heat-affected zone), are encapsulated and placed inside the RPV in holders fixed to the thermal shield to duplicate as much as possible the irradiation history of the vessel's inner surface (Figure 5A-2). The surveillance capsule is exposed to a neutron flux that is two to three times higher than the flux at the RPV surface.

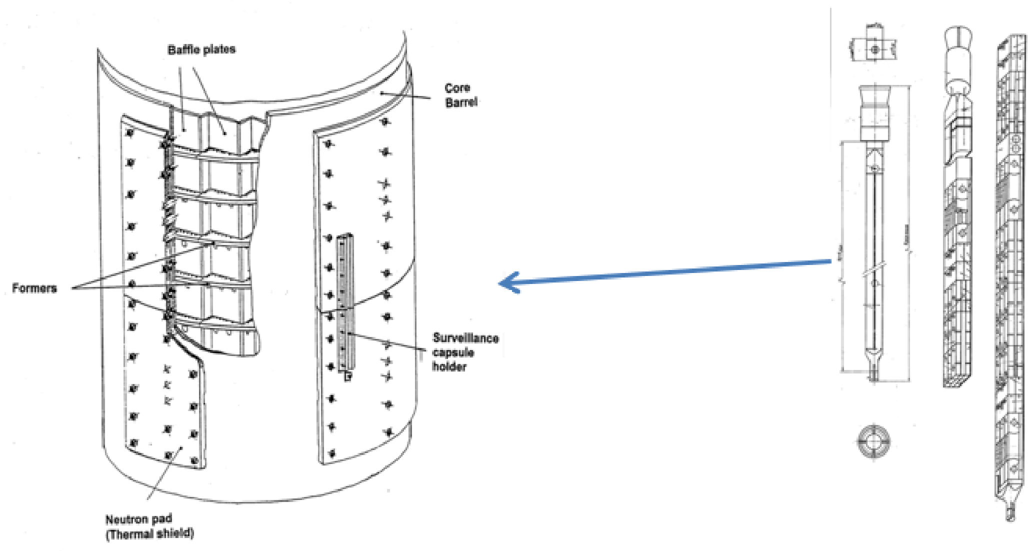


Figure 5A-2: Surveillance capsule holder on thermal shield and surveillance capsule

The surveillance capsules contain mainly Charpy-V and tensile specimens and a limited number of fracture toughness specimens.

Fluence is monitored by various types of dosimeters contained in the capsules. Irradiation temperature is monitored through low melting point eutectic alloys (with melting points of 304 and 310°C).

Surveillance capsules are withdrawn and tested according to a pre-defined schedule, in order to gather information on RPV irradiation embrittlement and verify that the material is evolving as expected.

Key design, manufacturing and operations documents used to prepare the AMP

The appropriate follow-up of irradiation embrittlement requires precise documentation of the manufacturing of the RPV core shells and welds and of the corresponding surveillance coupons. Particularly for welds, the surveillance blocks should be made with the same lot of welding products as the RPV welds, and the surveillance blocks should undergo the same heat treatments as the RPV.

Key design and manufacturing documents are:

- Detailed drawings, identifying the content of each surveillance capsule as well as the position of each specimen in the capsule
- Detailed drawings of the sampling position of each specimen in the surveillance coupons
- Materials test reports of the core shells and their welds, giving the unirradiated material properties as well as the chemical composition
- RPV manufacturing history, including the different heat treatments

Internal and external operating experience

The problem of irradiation embrittlement was recognised at the design stage. All RPVs worldwide are affected to various extents, and the Belgian RPVs are no exception.

There is a vast amount of experimental data from international RPV surveillance results, which made it possible to develop trend curves for some classes of material, predicting irradiation embrittlement (in terms of RT_{NDT} shift) in function of fluence, chemical composition (mainly Cu, Ni and P, but other elements like Mn are considered in some trend curves) as well as irradiation conditions in terms of flux or irradiation temperature [4O][6D][6E].

There is also a vast amount of internal operating experience in Belgium since a total of 35 surveillance capsules out of the 36 initially present in the seven Belgian reactors have been withdrawn and tested. Specific to the Belgian surveillance programs is that the surveillance capsules withdrawal schedule was adapted to provide RPV irradiation embrittlement information for a fluence corresponding to 60 years of operation with the last surveillance capsule. Surveillance results at high fluence are therefore available for most materials.

The situation of the Belgian units is summarized in Ageing Summaries.

R&D programs

RPV irradiation embrittlement has been the object of intensive research worldwide since the phenomenon was first identified in the 1960s. This led to an improved understanding of the mechanisms as well as to the development and improvement of microstructural techniques. The use of microstructural techniques that provide structural and/or compositional information at, or near, the atomic scale allows to characterize the microstructure evolution under irradiation. Parametrisation of the irradiation-induced changes as a function of material and compositional variables can then be used to underpin the mechanistic understanding of the embrittlement process. Historically, this mechanistic understanding has evolved in parallel with the development in microstructural techniques.

Large irradiation programs on RPV steels and model alloys or model steels in test reactors also helped to unravel irradiation mechanisms.

In parallel, research also focused on the evolutions of the mechanical properties under irradiation and on the development of enhanced characterization techniques.

ENGIE has been supporting an important research program in the field of irradiation embrittlement at the Belgian nuclear research centre SCK•CEN since the 1980s. This has resulted in the development of the enhanced surveillance approach [7H] which will be described in the section *Monitoring, testing, sampling and inspection activities for RPV irradiation embrittlement*.

Establishment of acceptance criteria

For historical reasons related to the post-world war II state-of-the-art scientific knowledge and pragmatic engineering solutions, the Charpy-V-V impact test is the reference test used to evaluate the degradation of RPV materials. In particular, the key property that is used for RPV integrity assessment is the ductile-to-brittle transition temperature (DBTT). The procedures that were developed within the ASME Code are essentially empirical and rely on a number of correlations. One of these correlations is the indexation of the lower bound fracture toughness curve through a single variable, the reference temperature of nil ductility transition (RT_{NDT}). By knowing only this single material property, the RT_{NDT}, the static and dynamic lower bound fracture toughness curves necessary for structural integrity assessments can be determined.

The initial RT_{NDT} is determined as specified in ASME NB-2331 [6B] by combining Pellini drop weight tests in accordance with ASTM E-208 [6A] to determine the temperature of nil ductility transition and of Charpy-V impact tests for the DBTT. In irradiated condition, the RT_{NDT} is defined as the initial value plus the shift under irradiation (ΔRT_{NDT}) measured by the shift of the Charpy-V curves at the 41J level as illustrated in Figure 5A-3 (only the static initiation K_{Ic} curve is shown for clarity, the same procedure applies to the crack arrest K_{Ia} curve).

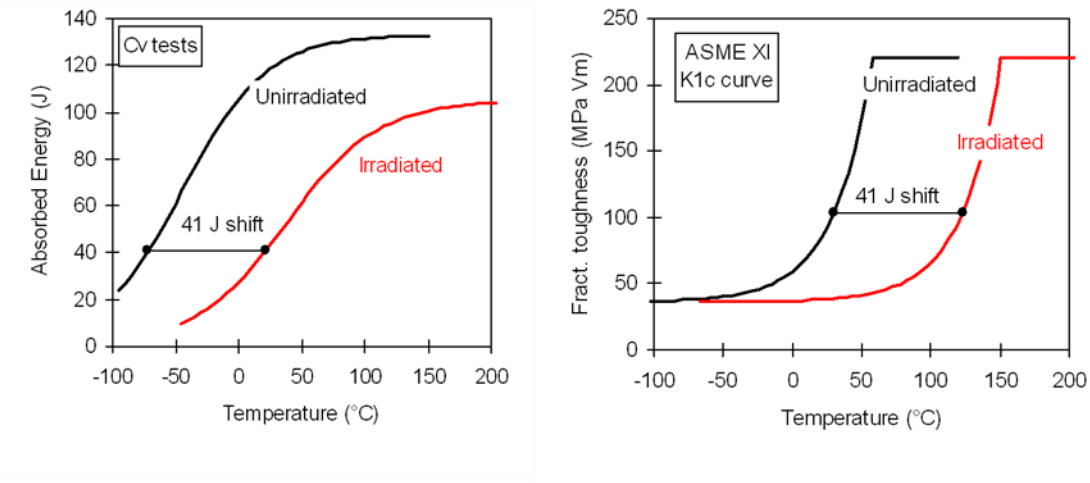


Figure 5A-3: Conventional evaluation of the fracture toughness in irradiated condition

The structural integrity of the RPV is governed by the fracture toughness of the irradiated material. Since the reference fracture toughness curves of the ASME Code are indexed to a reference temperature T-RT_{NDT}, the shift under irradiation of the fracture toughness curve is equal to the shift of the RT_{NDT}. In addition, the maximum acceptable value of RT_{NDT} must be defined in order to guarantee a sufficiently low risk of vessel failure in case of a Pressurized Thermal Shock (PTS) event.

The PTS transient is characterized by an initial rapid cooldown, initiated for example by a Safety Injection (IS) following a Small Break Loss Of Coolant Accident (SBLOCA). This cooldown induces high thermal stresses in the vessel wall close to the inside surface, and the cooling of the RPV wall reduces the fracture toughness of the material. If a significant pressure is maintained during the transient or re-established later, the combined pressure and thermal stresses, together with the reduced toughness, can in presence of a defect lead to a crack extension and potentially to a vessel failure.

A limiting value of RT_{NDT} was established in the early 1980s in the U.S. based on a generic study valid for different PWR types, which led to the so-called PTS screening criterion of 10 CFR50.61 [12]. The RT_{NDT} limit values for the acceptability of the risk of vessel failure in case of PTS are 132°C for base materials and axial welds and 149°C for circumferential welds.

It was recognized since the beginning of the 2000s in the U.S. that these criteria are very conservative and the progress in fracture mechanics and in the knowledge of defects distribution in RPVs made it possible to develop an alternative criterion 10 CFR50.61a [13] which is less conservative, but its application is subjected to a number of conditions and its use is optional. This less conservative alternative criterion is not used for the Belgian units.

The RT_{NDT} for a given fluence is determined by prediction curves (trend curves), after confirming that the trend curve conservatively envelops the experimental results (in terms of ΔRT_{NDT}) from the surveillance programs.

Monitoring, testing, sampling and inspection activities for RPV irradiation embrittlement

Monitoring and trending of irradiation embrittlement occurs through the regular withdrawal and testing of surveillance capsules as required by 10CFR50 App.H [4F]. The surveillance program is designed and implemented in accordance with ASTM E-185 [6C].

Surveillance capsules are withdrawn according to a predefined schedule and the contained specimens are tested. The irradiated Charpy-V energy results are fitted by a hyperbolic tangent function and compared to the similar curve established in non-irradiated condition (Figure 5A-3 left).

In irradiated condition, the RT_{NDT} is defined as the initial value plus the shift of the Charpy-V energy curve measured at the conventional level of 41J. The shift of the fracture toughness curves is considered to be equal to the RT_{NDT} shift (Figure 5A-3 right).

The irradiated RT_{NDT} obtained from the surveillance program is compared to the regulatory trend curves to confirm that these trend curves are conservative.

Different trend curves are used in Belgium:

- The Regulatory Guide 1.99 revision 2 trend curve [4O] is used only for Doel 1&2 where the limiting material is the circumferential weld with a relatively high Cu content. This formula is not appropriate for the low Cu materials of the other Belgian units and tends to underestimate embrittlement at high fluence for these materials.
- For the Tihange 1, Tihange 3 and Doel 4 units the French FIS formula is used [6D] and conservatively envelops the surveillance results. This formula was selected because the Belgian RPVs are very similar to those of the French 900 MW series.
- For Tihange 2 and Doel 3 the FIS formula was used until 2015. However, a specific formula was developed in the framework of the Safety Cases to justify the restart of both units following the 2012 discovery of hydrogen flaking in the RPV core shells [8B] to [8H]. This formula is based on the more recent RSE-M trend curve [6E] with an additional term conservatively covering the potential effect of hydrogen flakes [6R].

Surveillance results are available for fluences higher than those currently considered for the units' end of operation (50 years for Doel 1&2 and Tihange 1, 40 years for the other units) and the conservativeness of the used trend curves is demonstrated well beyond these limits.

The regulatory acceptance criteria are:

- The 10CFR50.61 PTS criteria [4C] specify limit values of RT_{NDT} for the acceptability of the risk of vessel failure in case of a PTS event. The limiting RT_{NDT} values are 132°C for forgings and 149°C for circumferential welds.
- There are also requirements on the upper shelf energy, which should be at least equal to 102 J before irradiation. After irradiation, it should at any time exceed 68 J [4E].
- The pressure-temperature (p-T) limits for heat-up and cooldown and the protection against low temperature over-pressurization (LTOP) are calculated according to the rules of ASME XI appendix G. The analysis uses the fracture toughness curves indexed to the RT_{NDT} predicted by the trend curve (upper bound) for the fluence corresponding to the time limit considered for the analysis. The fluence is calculated at the tip of the postulated reference crack, taking into account the attenuation through the vessel wall.

R&D program: enhanced surveillance

The enhanced surveillance approach applied in Belgium [7H] is the result of a vast R&D program that has been carried out at the Belgian nuclear research centre SCK•CEN since the end of the 1980s.

Key aspects of this development were:

- The development of a reconstitution technique to obtain new Charpy-V specimens by welding extension pieces to the broken half-Charpy-V specimen. This makes it possible to obtain additional results from the limited inventory of surveillance specimens available in a surveillance capsule, or to measure different properties, in particular fracture toughness.

- The development of the instrumented Charpy-V test, during which the load applied to the specimen is recorded vs. the time during the fracture process. This makes it possible to identify a number of critical parameters on the load trace, which enable a much finer interpretation of the test results. This also provides a quality assurance tool to identify outlier results.
- The development of test techniques on miniature specimens (mainly tensile but also mini-CT fracture toughness specimens and mini-Charpy-V specimens), with the same objective of optimizing the use of the limited quantity of material.
- The tensile properties are measured in an extended temperature range, to provide the necessary information for the validity criterion of the Master Curve (see below) and for modelling purposes. The yield stress increase is also monitored, although there are no regulatory criteria, and compared to the RT_{NDT} increase to check the consistency. Based on the Belgian data, the expected ratio between RT_{NDT} shift and yield stress increase is around 0.5 for forgings and between 0.5 and 1 for welds. This is consistent with information from expert literature.
- The development of a phenomenological hardening model predicting the material yield stress increase in function of fluence. This was supported by a number of RPV steel irradiation campaigns in the BR2 reactor (the so-called Radamo program).
- The development of fracture toughness measurements on small Charpy-V-size specimens tested in three-point slow bending, as well as extensive validation of the Master Curve technique [6F] for the interpretation of these tests. The increase in T_0 temperature of the Master Curve is also compared to the RT_{NDT} increase. Based on the Belgian data, the expected ratio between T_0 shift and RT_{NDT} shift is around 1.15 for forgings, consistent with what is reported in literature, while more scattered value are found for welds. A ratio close to 1 is expected based on international experience.

Preventive and remedial actions for RPV irradiation embrittlement

Low leakage core loading patterns are used in all units in order to limit the fluence on the vessel wall as much as possible considering the other constraints on the loading pattern.

One preventive action to limit the severity of the thermal shock events is the pre-heating of the SI water:

- It is implemented at Doel 1&2 because the embrittlement of the RPV weld between the two core shells is significant based on the conservative regulatory approach (but still meets the regulatory criteria), although more advanced techniques (direct fracture toughness measurements evaluated by the Master Curve) show that there are large margins.
- It is also implemented at Doel 3 (since 2015) in order to increase the margins in the structural integrity evaluations performed in the framework of the Safety Case to justify the restart of the unit in spite of the presence of hydrogen flakes in the core shells.

The rate of the heat-up and cooldown of the primary circuit is adapted in function of RPV irradiation embrittlement according to the rules of ASME XI app. G.

5.A.1.2.2 Low cycle fatigue

Ageing assessment of low cycle fatigue for the RPV

Fatigue crack initiation is considered in the design of ASME Class 1 components (typically components of the reactor coolant pressure boundary) using the cumulative usage factor (CUF). CUF is a conventional parameter for evaluating fatigue damage and is used to assess the likelihood of initiating a fatigue crack. Fatigue crack initiation is assumed to occur when the CUF reaches the value of 1.

The CUF is very low in all parts of the RPV. For the Tihange 1 RPV for example, the CUF is below 0.1 in all locations, except in the RPV head (junction between flange and head) where it is 0.41 for 40 years and for the RPV studs (0.85 for 40 years, but it remains below 1 for 50 years considering a realistic number of heat-up and cooldown transients). Similar values are obtained for Doel 1&2.

Low cycle fatigue is negligible in the RPV core shells.

Because the evaluation of the CUF considers the number of thermal and pressure cyclic transients imposed on the component, CUF evaluations are typically TLAAs. Such TLAAs have been executed for the LTO units (Doel 1&2 and Tihange 1) demonstrating that the fatigue remains acceptable for 50 years of operation. For the other units, the analysis performed at the design stage remains valid as long as the number of occurrences of the design transients is not exceeded.

In the last ten years, the potential effect of the environment on the fatigue design curves (defining the allowable number of cycles in function of stress intensity) became an important topic. These design curves are based on tests performed in air, but some experimental programs have shown that the fatigue life was reduced in the light water reactor environment. In the U.S. approach applied in Belgium, an environmentally-adjusted cumulative usage factor (CUF_{en}) is used to assess the impact of the reactor coolant environment. The CUF_{en} accounts for the effects of material type, strain rate and chemistry. This is presently required only for the LTO units.

The RPVs are made of low-alloy steel and their inner surfaces are clad with austenitic stainless steel (mean thickness = 5 mm). When the LWR environmental effects are considered, only the cladding needs to be evaluated since the low-alloy components are not exposed to the environment. It was demonstrated that the cumulative usage factor CUF_{en} for the cladding in the Doel 1&2 outlet nozzle was only 0.014. In conclusion, there is practically no impact of environmental effects on the RPVs' fatigue life.

Key design, manufacturing and operations documents used to prepare the AMP

- Stress reports and drawings
- Materials characteristics
- Environment characteristics
- List and characteristics of design transients
- Transient bookkeeping since unit start-up (including the hot tests phase)

Internal and external operating experience

A failure in service of a large ASME Class 1 component due to low cycle fatigue has never been reported.

R&D programs

The Belgian nuclear research centre SCK•CEN takes part in an ongoing international R&D program investigating the environmental effects regarding fatigue (INCEFA).

Establishment of acceptance criteria

The acceptance criteria are those of the ASME Code section II, subdivision NB which specifies design rules for preventing fatigue failure. The CUF must remain below 1 in all locations of the component.

Monitoring, testing, sampling and inspection activities for low cycle fatigue

In each unit, a bookkeeping of the operational transients is performed to ensure that the number of occurrences considered at the design stage is not exceeded.

The parameter values used in the analysis to account for the reactor water environment (based on the CUF_{en} approach), such as the dissolved oxygen content, are monitored.

Preventive and remedial actions for low cycle fatigue

There is no known remedial action to restore fatigue damage, except replacement of the component. The design conditions considered fatigue for 40 years of operation. However, there are large margins for the RPV, and the RPV would not be a limiting component from the low cycle fatigue point of view in case of extension of the operating period to 50 or 60 years. The RPV is not subjected to unexpected design transients such as thermal stratification.

Regarding preventive measures, a rigorous bookkeeping of the design transient must be performed.

5.A.1.2.3 Underclad defects in RPV shells

Ageing assessment of underclad defects in RPV shells

Underclad cracks in RPVs are known since the 1970s. This degradation mechanism was initially detected at the Rotterdam Dockyard Manufacturing company during magnetic particle inspections of a RPV in January 1971. Subsequent evaluations by Westinghouse concluded that these underclad cracks would not have an impact on the integrity of reactor vessels for a full 40 years of operations. The analysis was later extended to 60 years [7I].

No underclad defects were reported during fabrication of the Belgian RPV shells.

The first type of underclad cracking identified in the 1970s is referred to as 'reheat underclad cracking'. It has been reported only in forgings and plate material of ASME SA-508, Grade 2 Class 1 (formerly known as SA-508 Class 2) composition made to coarse-grain practice. High-deposition-rate welding processes, such as the submerged-arc wide-strip process and the submerged-arc 6-wire process, result in coarse-grains in the weld metal. Reheat underclad cracking occurs when the adjacent layer of high-heat-input cladding is added (or during post-weld heat treatment), which reheats the heat-affected zone (HAZ) in the carbon steel, causing a liquid phase along the grain boundaries and grain boundary separation upon cooling [4L]. Reheat underclad cracking was not observed in ASME SA-508 Grade 2 Class 1 materials clad by low-heat-input processes nor in clad ASME SA-533 Grade B Class 1 plate material, which is produced to fine-grain practice, regardless of the welding process used. The maximum depth of underclad cracks due to reheat cracking is 7.5 mm according to [7I].

Another type of underclad cracking is cold cracking, which results from the combination of:

- A quench microstructure sensitive to cold cracking in the HAZ
- Stresses generated during the cooling
- Hydrogen that diffuses in the ferritic steel from the deposit

Even though the crack sizes considered in reference [7I] are slightly smaller, some considerations in this document are applicable to the cold cracking type of underclad cracks.

The type of underclad cracking affecting potentially the Belgian units, following the discovery of such cracking in the French RPV of Tricastin 1, is cold cracking.

Key design, manufacturing and operations documents used to prepare the AMP

- Detailed drawings of the RPV and materials certificates
- Detailed cladding deposition procedure, including pre- and post-heating conditions
- Recordings of the applied welding parameters
- Manufacturing history of the RPV including the different heat treatments

Internal and external operating experience

In February 1999, the entire core beltline region (including base materials) of the Tricastin 1 RPV (900 MWe NPP) was inspected through ultrasonic testing (UT) using a new device capable of accurately inspecting the first 25 mm of the material wall thickness. These inspections revealed the presence of about ten longitudinal underclad defects. The maximum depth of the defects was 10 mm [2S]. They were attributed to cold cracking in the RPV low-alloy steel material. After extensive inspections on the other French 900 MW units (34 units in total), only three other shells were found to have some (limited) underclad defects.

The same indications were re-inspected four cycles later, in 2003 [6W], revealing no significant dimension change, taking account the process accuracy of +/- 1.3 mm in depth and +/- 5mm in length. No evolution of underclad defects was reported by EDF since the initial discovery in 1999.

The problem of underclad cracking in the core beltline and in the RPV nozzles is treated in two Ageing Summaries.

The situation in the Belgian units is the following:

The significance of this problem for the Belgian NPPs has been analysed in terms of metallurgical and fabrication data. The study concluded that considering the cladding processes and the chemical composition of the base materials, one could not exclude the existence of underclad cracks in the core belt region of the Belgian RPVs.

Therefore, a fracture mechanics analysis has been performed for the different Belgian units to determine the maximum allowable defects that would not induce brittle fracture under any service or accident transient condition. The fracture mechanics analyses showed that the largest possible underclad defects (taken conservatively as a 12.5mm deep semi-elliptical defect) remained acceptable with the required margins for all design and accident conditions.

After discussions with the Safety Authority, it was agreed to include a UT inspection of the RPV beltline in the program of the next Periodic Safety Review (PSR), starting with Doel 3 in 2012.

- The Doel 3 inspection was performed in 2012 and no underclad defects were found. However, the inspection did reveal large numbers of indications deeper in the shells. Since the Tihange 2 core shells had been manufactured at the same time and by the same supplier as the Doel 3 RPV shells, a similar inspection was performed at Tihange 2 in September 2012, with similar results. Therefore, both plants remained in cold shutdown with their core unloaded. A series of examinations, tests and inspections were performed leading to the conclusion that the indications were hydrogen flakes [41], that they were quasi-laminar in orientation and that they did not affect the structural integrity of the RPVs, regardless of the operating mode, transient or accident condition. The issue of hydrogen flakes is treated in paragraph 5.1.3.
- Tihange 1 and Tihange 3 RPV core shells were inspected in 2013 and no underclad defects were found.
- Doel 4, Doel 1&2 RPV core shells were inspected in 2015 and no underclad defects were found.
- No hydrogen flakes were found at Doel 1&2, Tihange 1, Tihange 3 and Doel 4 RPVs.

R&D programs

Large-scale experiments were performed by various organizations or international programs in order to investigate the behaviour of underclad cracks.

One of these experiments is reported in [6Y]. A 10.7 x 42.1 mm semi-elliptical defect was introduced in a large-scale specimen (7010 x 300 x 70 mm) of a nuclear grade German RPV steel similar to A508 Cl. 2, which was afterwards clad by 2 layers of strip cladding, each with a thickness of 4 mm, and thermally treated in accordance with the relevant nuclear standards. The specimen was tested under combined mechanical (3-point bending) and thermal loading until complete failure. The clad face was exposed to liquid nitrogen while the opposite face was electrically heated to approximately room temperature. The cladding remained intact until final failure of the specimen which occurred without preliminary crack extension.

A second test was performed with a similar specimen, this time with a through-clad crack of initially 13.8 x 57.3 mm in the base metal. In this test, crack extension and arrest occurred, increasing the area of the initial crack times four approximately (27.6 x 113.7 mm at arrest). The cladding remained intact during this crack extension. The test was continued to final fracture and again the cladding remained intact until final failure. This shows clearly the high resistance and the beneficial effect of the cladding (restraining the crack opening). Numerical simulations were in good agreement with the experimental results.

A research program was also carried out at the Belgian nuclear research centre SCK•CEN with the financial support of ENGIE, with the following objectives:

- Characterization of the cladding properties to provide validated data for the fracture mechanics analyses, using a cladded block coming from the RPV core shell of the decommissioned Lemoniz plant (Spanish plant which was never operated). The cladding was characterized in non-irradiated and irradiated condition.
- Development and validation of a new specimen type for the validation by dedicated experiments of the PTS analysis techniques (thick circular specimens with a semi-elliptical fatigue crack under biaxial bending loading simulating real PTS conditions [6I]), to investigate:

Transfer of material properties from specimens to structure (constraint effects).
Effect of the cladding on the crack propagation.
Additional validation of the Warm Pre-Stressing (WPS) concept. According to this concept, crack initiation does not occur when the stress intensity factor is decreasing after reaching a maximum at high temperature.

The program was completed successfully. The beneficial effect of WPS has been clearly identified, the stress intensity factor at fracture being always larger for WPS transients than for isothermal loading.

Testing the effect of cladding on underclad cracks showed that cladding plays a significant role and contributes to an additional safety margin. The cladding reduces the driving force by restraining the crack opening. In addition, cladding increases the potential for crack arrest.

Establishment of acceptance criteria

For existing defects (indications found during ISI), the criteria to be met are those of ASME Section XI, IWB-3600, which impose criteria in level A/B and level C/D conditions, either on the stress intensity factor or on the crack size.

Underclad defects play a key role in the risk of vessel failure in case of PTS.10 CFR 50.61 [4C] defines maximum RT_{NDT} values for which the risk of vessel failure in case of PTS is acceptable (corresponding to a vessel failure probability of 5 10-6per reactor-year):

- RT_{NDT} < 132°C (shells and longitudinal welds)
- RT_{NDT} < 149°C (circumferential welds)

These generic criteria were established based on a probabilistic study.

Monitoring, testing, sampling and inspection activities for underclad defects in RPV shells

UT inspection of the base materials is not required by the ASME code that is adopted in Belgium, the mandatory inspections being limited to the welds (with a few cm of base materials on each side). Therefore, there were no inspection results for the base materials in the Belgian units when the issue was identified at Tricastin 1.

UT inspections to detect this type of defects are performed using specific transducers providing a high sensitivity in the first 25 mm of the RPV shell. The zone to be inspected corresponds to the core height increased by 200 mm on each side. The technique is qualified to detect defects of a minimal size of 5x25 mm.

The volumetric inspection technique is qualified by the third-party certification organization: EQB.

In agreement with the safety authorities, a UT inspection of the core shells of the Belgian units: was planned during the PSRs.

Between 2012 and 2015 the entire beltline region of all Belgian RPVs was inspected and no underclad defect was found. Since underclad defects are fabrication defects and do not arise in service, no further inspection is planned, as agreed with the Belgian Safety Authority. Hence, no monitoring or trending for RPV shell underclad defects is required. However, specific inspection techniques are applied for the Tihange 2 and Doel 3 RPVs affected by hydrogen flakes.

Preventive and remedial actions for underclad defects in RPV shells

Underclad defects are fabrication defects and prevention of their occurrence can only be achieved by taking the right precautions during the fabrication.

Since no underclad defects were found in the dedicated UT inspections performed in all Belgian units, no specific preventive actions are necessary.

Mitigation solutions: low leakage fuel loading patterns are already in use in Belgian units. IS water preheating is already implemented at Doel 1&2 (35°C) and Doel 3 (45°C) to reduce the severity of the thermal shocks and increase the safety margins for the PTS issue.

5.A.1.2.4 Underclad defects in nozzles

Ageing assessment of underclad defects in nozzles

Underclad defects in nozzles can occur due to cold cracking in ASME SA-508 Class 3 forgings (in this case the RPV nozzles), after deposition of the second and third cladding layers, when insufficient (or none at all) pre-heating and post-heating was applied during the cladding procedure.

Such cold cracking is attributed to residual stresses near the yield strength in the weld metal/base metal interface after cladding deposition, combined with a crack-sensitive microstructure in the HAZ and high levels of diffusible hydrogen in the austenitic stainless steel or Inconel weld metals. The hydrogen diffused into the HAZ and caused cold (hydrogen-induced) cracking as the HAZ cooled.

Key design, manufacturing and operations documents used to prepare the AMP

- Detailed drawings of the RPV nozzles and materials certificates
- Detailed cladding deposition procedure, including pre- and post-heating conditions
- Recordings of the applied welding parameters
- RPV manufacturing history, including the different heat treatments
-

Internal and external operating experience

During a manufacturing inspection in 1979, a new form of underclad cracking (cold cracking) which affects the base metal immediately beneath the clad of the vessel nozzles was observed in French RPVs. After this discovery, a large number of metallurgical investigations were carried out by the manufacturer (Framatome) to evaluate the impact on the integrity of the reactor vessels.

Systematic UT inspection of the nozzles of all French RPVs under construction or installed at that time revealed that all the nozzles were affected by cold cracking in various degrees. Particularly the bores of all nozzles were affected.

For the vessels under construction, Framatome made no repairs. A fracture mechanics study based on conservative assumptions demonstrated that the defects would not pierce the cladding and that their extension in the base metal is extremely low for the duration of design life (40 years) and for the normal operating conditions. The conclusions of this safety analysis were accepted by the French safety authority.

In the EDF fleet, around 200 underclad cracks are known, from 6 to 10 mm depth, affecting 23 nozzles (mainly inlet nozzles) in 15 units. All these defects are periodically inspected. No evolution has ever been detected [6V].

The problem of underclad defects in nozzles and the similar problem of underclad cracks in the RPV core beltline region are treated in Ageing Summaries.

The situation in the Belgian units is the following:

- Inspections in Doel 1&2 and Tihange 1 did not reveal any underclad defect. In these units, the cladding process was different from the one used in the affected French units. In Doel 1&2, the strip cladding was deposited longitudinally in a single layer in the nozzles area with pre- and post-heating. In Tihange 1, the cladding was deposited in two layers by the wire twin arc process, and a stress relief heat treatment (550°C 1hour) was applied between the deposition of the two layers. In the French units, the cladding was applied by strip cladding, deposited circumferentially, in two or three layers without pre-heating for the deposition of the second and third layers.
- The upper part of the Doel 3 and Tihange 2 RPVs (including the nozzles) was manufactured by Framatome. The second cladding layer of the vessel nozzle was deposited automatically with preheating to avoid cold cracking. The situation is therefore more favourable than in the French units. Nevertheless, the manually deposited cladding was done without preheating. Inspection campaigns detected indications on the bores of inlet (G) and outlet (H) nozzles of the two vessels. However, the population of detected indications differs from the overall population of French vessels affected by the cracking phenomenon: the number of indications of high amplitude is smaller and there is a concentration of indications in the overlapping area of manual / automatic cladding in the H1 and H3 nozzles of Doel 3, and the H1 and H3 nozzles of Tihange 2. The amplitudes of the detected echoes prove that the height of the defects is very limited (4 mm maximum). Hence, the study conducted for the French vessels is certainly conservative for defects present in the nozzles of Tihange 2 and Doel 3.
- The vessels of Tihange 3 and Doel 4 were under construction when the problem of underclad defects appeared in France. The welding procedures used were similar to those adopted by Framatome after 1979 to correct the appearance of cracking during the cladding and therefore the existence of cold cracking under the cladding was not expected. Ultrasonic inspections using the same procedures as those used in the context of underclad cracking for Tihange 2 and Doel 3 were used during the manufacturing of the Tihange 3 and Doel 4 RPVs. These inspections revealed the existence of discontinuities at the cladding / base material interface in the area of overlapping between manual and automatic cladding. An inspection carried out before and after the heat treatment showed, in some nozzles, the emergence of new signals after heat treatment. The characteristic signals after the preliminary heat treatment do not match with those of cold underclad defects. Investigations on defects removed in fabrication (boat sample) showed that these cracks were formed during the annealing relaxation (so-called hot underclad cracking). These indications were acceptable by ASME XI standards and studies performed in the U.S. for similar cracks have concluded that they had no detrimental effects on the structural integrity [4L].

R&D programs

After the discovery of underclad defects, an experimental program on large specimens was launched in France. The objectives of this test program were to investigate a number of points that were not yet well known at the time of the initial safety analysis for underclad defects. Tests performed on specimens with underclad cracks revealed that the measured number of cycles for cladding perforation was 10 to 15 times larger than the number of cycles calculated with the assumptions of the initial safety analysis. The defects had not even started to grow after the number of cycles corresponding to the cladding perforation study in the initial study [7K] to [7M].

In addition to the studies and tests dedicated to underclad defects, some tests were also carried out to validate the calculation methods in fracture mechanics applied to the reactor vessels. They again showed the high degree of conservativeness of the initial Framatome study.

Establishment of acceptance criteria

The criterion used in the initial Framatome study was the absence of cladding piercing after 40 years of operation.

Structural integrity analyses to justify the acceptability of underclad defects are performed according to the rules of the ASME XI appendix A.

Monitoring, testing, sampling and inspection activities for underclad defects in nozzles

Inspections of the nozzle's inside radius were performed in the framework of the ASME XI inspection program in all Belgian units and did not reveal any indication.

Preventive and remedial actions of underclad defects in nozzles

These underclad defects are fabrication defects which can be prevented by an adequate pre- and post-heating treatment for all cladding layers. No preventive action is possible for existing vessels.

5.A.1.2.5 PWSCC of BMI nozzles

Ageing assessment of PWSCC of BMI nozzles

BMI penetrations allow the passage of the thimble tubes through the pressure boundary. The thimble tubes enable the passage of neutron flux detectors to access the fuel assemblies.

PWSCC of alloy 600 RPV BMI penetrations and of their welds in alloy 182 might occur. This is a generic problem that potentially affects all units built with these materials.

The problem is similar to that of the RPV upper head penetrations cracking but with the difference that the temperature at the level of the BMI penetrations is the cold leg temperature. The situation is more comparable to the "cold" RPV upper head penetrations, which are also exposed to the cold leg temperature.

This is one of the factors that could probably explain the very limited number of cracking incidents experienced on these penetrations worldwide.

Internal and external operating experience

In April 2003, an external Bare Metal Visual (BMV) inspection in South Texas unit 1 (STP1) has evidenced small boric acid deposits (a few mg) on 2 BMI penetrations [6N]. The deposits were identified as originating from the primary water (presence of boron and lithium) and the leak was considered to be between three to five years old. After core unloading, UT and eddy current inspections evidenced cracks and leak paths. Destructive analyses were performed to confirm the origin of cracking. It was concluded that PWSCC cracks initiated at a site where a lack of fusion at the weld / base metal interface was present.

Following this event, the NRC issued bulletin 2003-02 [5R] recommending BMV inspections of all concerned U.S. units. Visual examinations have been required by the NRC since 2008 following the ASME Code Case N-772. [6J].

U.S. plants performed the examinations required by the NRC and no other event occurred until 2013, when dry boric acid was found on a BMI penetration during the inspection of the Palo Verde unit 3.

Another event concerns the Takahama unit 1 in Japan, where an axial indication of 32 mm long and less than 1 mm deep was detected in 2003 by performing an eddy current inspection of the internal surface area of the BMI penetrations. The area concerned was ground during the next outage (2004) and subsequently waterjet peening was applied. The PWSCC origin has not been confirmed.

Most French units had their BMI penetrations inspected since 1992 and no degradation was found until 2011, when a crack was discovered in a BMI penetration during the third PSR of Gravelines 1. The crack was nearly through-wall starting at the inner wall and was attributed to PWSCC growth of two axial manufacturing defects. The solution implemented by EDF was a mechanical plug combined with a leakage monitoring program able to detect a leak of about 1 liter per hour. At the end of 2015, the BMIs of 91% of the 900 MWe French NPPs and of 50% of the 1300 MWe French NPPs had been inspected using UT, and no indications were detected.

The Belgian units are potentially concerned by PWSCC of the BMI penetrations and the situation in the Belgian units is summarized in an Ageing Summary.

For the LTO units Doel 1&2 and Tihange 1 a specific AMP was established initially along the guidelines of the GALL [4H] and later on adapted to the format recommended by IAEA.

The AMP is based on a combination of:

- Availability of analytical tools to evaluate the acceptability of detected indications
- Augmented in-service inspections using qualified techniques and with an inspection schedule ensuring detection of cracks and leaks

Indications have been reported after UT inspections performed at Doel 1 in 2015 and at Tihange 1 in 2016. Based on fracture mechanics analyses to estimate the severity and potential growth of those indications, it was concluded that Doel 1 could operate normally for the next ten years (i.e. until the final shutdown of the unit) but an action is required for Tihange 1 where the acceptability of the indications was justified for at least three years. The AMP will be adapted accordingly.

Key design, manufacturing and operating documents used to prepare the AMP

- Detailed drawings and manufacturing information of the RPV bottom head and BMI penetrations. One factor that influences the residual stress level and hence the sensitivity to PWSCC is the fact that the penetrations were stress-relieved with the RPV or not.
- Possible non-conformities documented in fabrication (lack of verticality and internal diameter deformation), which required straightening and re-machining. This could have introduced residual stresses after the stress-relief heat treatment.

R&D programs

Destructive examination of the cracked BMI penetrations of South Texas were performed and confirmed that the cracking was due to PWSCC.

Several laboratories worldwide conducted extensive testing to characterize the crack growth rate in Ni-based alloys. These data were used by EPRI in the framework of the Materials Reliability Program (MRP) to develop disposition curves for the base material alloy 600 [7F] and for the corresponding welds in alloy 82, 182 and 132 [7G].

The vendors have also performed R&D to develop repair or mitigation techniques.

Establishment of acceptance criteria

No specific acceptance criteria are presently defined in the U.S. for BMI penetrations. To overcome this lack of rules, the existing criteria for reactor vessel head penetrations were adapted, for instance the evaluation procedure and acceptance criteria defined in the article IWB-3600 of the 2004 edition of the ASME code, Section XI.

Monitoring, testing, sampling and inspection activities for PWSCC of BMI nozzles

For the LTO units Doel 1&2 and Tihange 1, the following inspection program is applied:

- ASME VT-3 inspection of the inside surface every ten years
- BMV inspections every two cycles
- Qualified volumetric inspection technique to detect cracks in the penetration and the weld to base material interface every ten years

This inspection was performed in 2015 at Doel 1 in accordance with the AMP and detected some indications which were justified as acceptable for at least the next ten years.
A similar inspection had already been performed at Doel 2 in 2008 and no indications were detected.
A volumetric inspection of all BMI penetrations has been performed at Tihange 1 in 2016 in accordance with the AMP. Axial and circumferential indications were found in three penetrations. Their acceptability was justified for continued operation and a reinspection program is established based on conservative crack growth rates.

- VT inspection of the J-groove welds and outer diameter of the nozzles every ten years (coupled with the volumetric inspection)

For the other units, BMV inspections are performed every three years.

The volumetric inspection technique is qualified by the third-party certification organization: EQB.

Each BMI penetrations inspection is planned based on the component's susceptibility to PWSCC and of previous examination results. The resulting program is submitted for approval by the Belgian Safety Authority.

Preventive and remedial actions for PWSCC of BMI nozzles

Preventive measures to mitigate PWSCC are in accordance with PWR water chemistry guidelines for primary coolant systems, as established in EPRI Report 1014986 [5D]. The program description and the evaluation and technical basis of monitoring and maintaining reactor coolant water chemistry are presented in an AMP related to water chemistry.

The enhanced ISI program can be considered as a preventive action, not in terms of influencing the cracking mechanism, but by the timely detection of cracking well before a leak occurs.

5.A.1.2.6 PWSCC of RPV upper head penetrations

Ageing assessment of PWSCC of RPV upper head penetrations

The PWSCC of alloy 600 RPV upper head penetrations and of their welds in alloy 182 is a generic problem potentially affecting all units with these materials. The cracking can start sooner or later during the plant's lifetime depending on plant-specific conditions (stress, temperature) or material resistance (heat-to-heat variability), nevertheless all RPV upper head penetrations of this type are susceptible to experience cracking.

Internal and external operating experience

The PWSCC of RPV upper head penetrations in Ni-based alloy 600 was initially discovered at Bugey (FR) in 1991. This launched a systematic inspection program revealing that the same issue was affecting many other RPV heads. The issue was initially controlled in France through extensive analytical and experimental programs devised to predict the initiation and propagation of these types of cracks and hence to determine inspection, repair or replacement strategies. It eventually led to the systematic replacement of all affected RPV heads in France. The replacement heads have penetrations made of a different alloy (690) that has long been used with excellent results in steam generator tubes. No cracking has been reported with alloy 690 components in service up to now, although laboratory investigations showed that cracking could be induced under specific conditions.

The French operating experience motivated inspections in many other countries where the same problem was identified. This was also the case in Belgium where indications were found after UT inspection of the Tihange 1 head in 1992.

In the U.S., this issue was long considered as a "French illness" until in 2002 a large wastage cavity was discovered at a penetration nozzle on the Davis Besse RPV head. In the following years, a number of cracks and leaks were discovered on U.S. vessel heads. In the U.S. cases, the issue also affects weld material and cracks are of all types (internal, external, circumferential, axial). Repairs were applied as a short-term measure but the general strategy was to replace the affected heads by new heads with penetrations made of alloy 690.

The problem also affected some Belgian plants:

- Tihange 1: RPV head replaced in 1999.
- Doel 1: RPV head repaired in 2005 and planned to be repaired again in 2018 as new cracks and an evolution of existing cracks have been detected in some penetrations since 2012. In the meantime, the inspection program (using the most advanced measuring techniques) demonstrated that the repairs (using a technique that has been applied successfully on multiple RPV heads) ensure the safety of the RPV. The process has been applied on U.S RPV heads and is accepted by the U.S.NRC
- Doel 2: scratch-type shallow indications.

In the Belgian units, the cracks have been detected in the base material only.

Even if no cracking has been detected in the other Belgian units, international experience shows that there is a possibility that these heads could also experience cracking in the coming years.

Tihange 3 and Doel 4 heads have been replaced preventively in 2015 as they were deemed highly susceptible according to the U.S. ranking based on age and temperature. No indication was ever reported in those heads.

Doel 1&2 are also in the high susceptibility category but to a lesser extent.

Doel 3 and Tihange 2 are in the low susceptibility category according to the U.S. ranking, although it must be reminded that cracking can also occur in cold heads, where the temperature under head is the cold leg temperature.

The situation of the Belgian units is summarized in an Ageing Summary.

For LTO units Doel 1&2 and Tihange 1 an AMP was established initially along the guidelines of the GALL [4H] and later on adapted to the format recommended by IAEA.

The AMP is based on a combination of:

- Augmented ISI according to the applicable regulation, using qualified techniques and with an inspection schedule ensuring timely detection of cracking
- Follow-up of international operating experience feedback regarding alloys 600 and 690, and alloy 182

Key design, manufacturing and operations documents used to prepare the AMP

- Detailed drawings and manufacturing information of the RPV head. One factor that could potentially influence the residual stress level and hence the sensitivity to PWSCC is the volume of the weld, although this influence cannot be quantified.
- The temperature under head is the most important parameter influencing PWSCC sensitivity. This is very clear in laboratory crack growth rate tests, although operating experience does not show a clear temperature influence and some cold heads have also experienced cracking. Cold and warm heads refer to the fact that the temperature under head can be the cold leg

temperature or the hot leg temperature or a value in between. In the Belgian units, the temperature under head is 287°C in Tihange 2 and Doel 3, 307°C in Doel 1&2, and 318°C in Tihange 1, Tihange 3 and Doel 4.

R&D programs

In the U.S. and other countries destructive examination of cracked penetrations performed confirmed that the cracking was due to PWSCC.

Several laboratories worldwide conducted extensive testing to characterize the crack growth rate. These data were used by EPRI in the framework of the MRP to develop disposition curves for the base material alloy 600 [7F] and for the corresponding welds in alloy 82, 182 and 132 [7G].

The vendors have also performed R&D to develop repair or mitigation techniques.

Establishment of acceptance criteria

When the augmented inspections program reveals indications in the reactor vessel head penetration nozzles, an evaluation procedure is defined in article IWB-3660 of the 2007 edition of the ASME Code, Section XI [6L] and in Code Section XI Appendix O [6M]. These procedures and acceptance criteria are identical to those defined in ASME Code Case N-694-1 [6G] or previously in [6V] in 2003.

The maximum flaw growth is calculated for the considered evaluation period (generally the next fuel cycle) using conservative crack growth laws (fatigue and stress corrosion cracking), and the dimensions at the end of the evaluation period are compared with the maximum allowable flaw dimensions according to the acceptance criteria. All applicable loadings must be considered (including residual stress).

The ASME Code requirements only concern the base material (alloy 600) at present. No rules are given for cracks in the J-groove welds; on the contrary, application of the procedure of IWB-3600 is explicitly prohibited for the J-groove weld. Indications in the J-groove weld must be justified on a case-by-case basis.

The crack propagation laws to be used for the analysis are those defined in [7F] and [7G].

Monitoring, testing, sampling and inspection activities for PWSCC of RPV upper head penetrations

The inspection program monitors for cracking/PWSCC and loss of material/wastage in the upper vessel head penetration nozzles to ensure their structural integrity prior to a loss of their intended safety function.

The inspection program also monitors for evidence of reactor coolant leakage as a result of through-wall cracks that may exist in the upper vessel head penetration nozzles or their associated partial penetration J-groove welds. Evidence of reactor coolant leakage may manifest itself in the form of boric acid residues on the upper RPV head or adjacent components or in the form of corrosion products that result from rusting of the low-alloy steel materials used to fabricate the RPVs.

In Belgian NPPs, detection is accomplished through implementation of a combination of:

- BMV examination of the upper head surface to detect any suspect boric acid deposit or any loss of material that may be induced as a result of boric acid wastage
- Qualified volumetric examination techniques (UT), to detect cracking of the penetration itself
- Televisual inspection for the J-groove weld; the weld/penetration interface is also controlled by UT from the inside

The volumetric inspection technique is qualified by the third-party certification organization: EQB.

Inspection schedules and frequencies are implemented in accordance with the plant's susceptibility category, based mainly on the temperature under the RPV head, as well as on the presence of previously detected indications.

Preventive and remedial actions for PWSCC of RPV upper head penetrations

Preventive measures to mitigate PWSCC are in accordance with PWR water chemistry guidelines for primary coolant systems, as established in EPRI Report 1014986 [5D]. The program description and the evaluation and technical basis of monitoring and maintaining reactor coolant water chemistry are presented in an AMP related to water chemistry.

The enhanced ISI program can be considered as a preventive action, not in terms of influencing the cracking mechanism, but by the timely detection of cracking well before a leak occurs.

The most significant preventive action applied in the Belgian units was the preventive replacement of the Tihange 3 and Doel 4 RPV heads in 2015. Although no degradation was detected, these heads were classified as being highly susceptible according to the NRC order EA-03-009 [4P] based on temperature (318°C) and operating time (more than 26 Equivalent Full Power Years (EFPY) at the time of replacement, corresponding to more than 31 Equivalent Degradation Years (EDY)). RPV heads are classified as being highly susceptible above 12 EDY [4P].

These two heads were the world's only RPV heads manufactured by Cockerill, so there is no comparison available to determine whether a particular fabrication history explained the absence of degradation. The new heads have penetrations in Inconel 690.

5.A.1.2.7 PWSCC of dissimilar metal welds

Ageing assessment of PWSCC of dissimilar metal welds

PWSCC is a corrosion mechanism that forms cracks in susceptible materials in an aggressive environment (primary water in this case) and under tensile stresses. Ni-based alloy 182, used in some units to join the low-alloy steel RPV nozzle and the stainless steel safe-end, is known to be sensitive to PWSCC.

A PWSCC crack can take a long time to initiate, depending on the component's specific conditions. Once initiated, PWSCC cracks start to grow. Only limited field data on crack propagation exists, and there is a large variability in crack growth rates measured in laboratory tests. A conservative crack propagation law for alloy 182 is established in [7G].

The contributing factors are mainly:

- The stress level: The threshold value for PWSCC initiation is considered to be around 350 MPa. The RPV nozzle welds were stress-relieved during fabrication, which should reduce the PWSCC risk, but local effects not necessarily documented in the fabrication files (grinding for example), cannot be excluded. Repairs carried out on the weld after Post-Weld Heat Treatment tend to increase its susceptibility to PWSCC and most occurrences in international operating experience have been correlated with repairs made during fabrication, but not all.
- Temperature is a major contributor.
- Environment: primary water is necessary for PWSCC to occur.

Key design, manufacturing and operations documents used to prepare the AMP

- Detailed drawings and manufacturing information of the RPV nozzle to safe-end weld. One factor that favourably influences the residual stress level and hence PWSCC sensitivity is the fact that the welds were stress-relieved with the RPV in the Belgian units.
- Repairs in the dissimilar metal welds documented during fabrication. Repairs can introduce residual stresses after the stress-relief heat treatment.

Internal and external operating experience

The main events relative to PWSCC cracking in RPV nozzle welds are described hereafter:

- At the end of 2000, a primary water leak was discovered near one of the RPV outlet nozzles of the V.C. Summer NPP [4S]. The weld had a through-wall axial flaw with a small circumferential component and other small part-through-wall axial flaws. Destructive examinations of the piping and the weld material revealed that PWSCC caused the flaws. The low-alloy (ferritic) steel and the stainless steel at the ends of the weld arrested the axial crack growth. The component was repaired by cutting the weld and welding a spool piece.
- Nearly in the same period, significant part-through-wall axial cracks were detected at the same locations at units 3 and 4 of the Ringhals NPP. The examination of removed boat samples containing the cracks confirmed that they were caused by PWSCC. The cavities were left in place for two cycles, after which a weld inlay was applied at Ringhals 4 in 2002 and Ringhals 3 in 2003 [7N][4V].
- In Ohi 3, a small crack was detected using Eddy Current Testing (ECT) in the RPV outlet nozzle weld in March 2008. The ECT was performed as confirmatory examination prior to water jet peening. The length of the crack was 10 mm and the depth was estimated to be less than 5 mm based on UT. The crack was removed by progressive grinding, and it appeared that it was much deeper than expected. (20.3 mm). The maximum length of the crack was 13.5 mm at the depth of 5.5 mm. After grinding, water jet peening was applied as corrective action. Weld inlay repair was applied at the next outage. The origin of the cracking is suspected to be high residual stresses from machining of the weld inside surface during fabrication [6X][4V].
- In Salem 1, an indication was found in one of the hot leg safe-end welds (Inconel 182, weld thickness = 66.5 mm) in a pre-MSIP (Mechanical Stress Improvement Process) inspection in 2008 [4V]. The indication was 62 mm in length and 16.1 mm in depth (= 24 % of the thickness). MSIP was applied to all six RPV nozzles, including the one with the indication.
- Many of the concerned welds have been repaired during fabrication, but not all (although it is difficult to be certain, since some repairs may not have been properly documented). It is suspected that high residual surface stresses resulting from machining during fabrication may be sufficient to initiate PWSCC. The concerned units are generally over 20 years old.
- It must be noted that the reported indications are not necessarily due to PWSCC, since some were repaired without detailed characterization of the crack.

The situation in the Belgian units is the following:

- Doel 1&2 and Tihange 1 have stainless steel welds and are not concerned by this issue. Doel 3, Doel 4, Tihange 2 and Tihange 3 have dissimilar metal welds in alloy 182 between the RPV and the main coolant piping.
- Following the V.C. Summer operating experience, the fabrication records of the Belgian units were reviewed and an enhanced inspection program was implemented.

Indications have been found in several welds in Belgian units:

- Doel 3

One circumferential indication was detected in an outlet nozzle (23 x 2 mm) during the 2006 inspection (by UT and ECT). As no evolution was seen during the 2012 inspection (by UT and ECT), it is very unlikely that the indication is due to PWSCC.

- Doel 4

No indication of PWSCC.

- Tihange 2

One 10 mm long axial indication was detected in the outlet nozzle H2 weld during the 2003 inspection (by UT and ECT). The depth was not measurable but was considered to be 1 mm. The weld was re-inspected (by UT and ECT) in 2005, 2006, 2008 and 2014 without detecting any evolution. It is very unlikely that the indication is due to PWSCC.
One 5 mm long axial indication was detected in the inlet nozzle G2 weld during UT inspection in 2003. The depth was not measurable but was considered to be 1 mm. The weld was re-inspected (by UT and ECT) in 2014 without evolution. It is very unlikely that the indication is due to PWSCC.

- Tihange 3

Indications were detected in the 2006 inspection (by UT and ECT):
Several indications were found in outlet nozzle H2, some of which are circumferentially oriented. The maximum size of the circumferential indications (a x l) is 3.7 x 15 mm. The larger axial indication size (a x l) is 2 x 25 mm.
Two smaller indications are also present in outlet nozzle H3, one of which is axially oriented, a x l = 2 x 13 mm.
Circumferential indications were also detected in inlet nozzles G2 and G3, with a maximum depth of 2.3 mm.
The outlet nozzles were re-inspected in 2007, 2009, 2012 and 2015 and the inlet nozzles in 2009 and 2015, (in all cases by UT and ECT) without evolution. It is very unlikely that the indication is due to PWSCC.

R&D programs

There is an ongoing research program at SCK•CEN focused on the alloy 182 weld from the Spanish cancelled and decommissioned Lemoniz plant (same design of the dissimilar metal weld as in Tihange 2 and Doel 3). Several crack growth rate tests in various orientations confirm the conservativeness of the MRP-115 disposition curves [7G]. Crack initiation tests were performed on 4-point bend specimens sampled from the Lemoniz weld, including some specimens with Electric Discharge Machined (EDM) surfaces from the qualification blocks of a defect removal process. No specimen was broken (in 4-point bend tests) during the 36 months of exposure (12 months at 300°C, 12 months at 325°C and 12 months at 340°C) at stress levels varying from 310 to 626 MPa (yield stress is of the order of 500 MPa). Some cracks are visible on the EDM specimen surfaces but are most probably hot cracks from the fabrication and did not propagate.

Several research programs were performed in the framework of the FROG (FRamatome Owners Group) to evaluate aspects like the influence of surface finish and of water chemistry and in the PWROG (large program on the effect of Zn addition on PWSCC).

Destructive examination of the V.C. Summer cracked penetrations and of boat samples taken in the Ringhals 3 and 4 units were performed and confirmed that the cracking was due to PWSCC.

Several laboratories worldwide conducted extensive testing to characterize the crack growth rate. These data were used by EPRI in the framework of the MRP to develop disposition curves for the welds in alloy 82, 182 and 132 [7G].

The vendors have also performed R&D to develop repair or mitigation techniques.

Establishment of acceptance criteria

When indications potentially caused by PWSCC, or susceptible of giving rise to PWSCC, are detected, a crack growth analysis is performed based on the measured dimensions, and if necessary a specific follow-up program is implemented to confirm the absence of any evolution of the indication.

For the analysis, the crack growth law from MRP-115 [7G] was used.

For propagating cracks, the maximum allowable crack depth is 75% of the thickness.

Monitoring, testing, sampling and inspection activities for PWSCC of dissimilar metal welds

Following the V.C. Summer operating experience, an enhanced inspection program was implemented:

- Visual inspections are normally not performed under the ASME XI requirements but were introduced as additional measure in the augmented ISI plan.
- Liquid penetrant tests are required once every ten years under the ASME XI requirements and are increased in the augmented ISI plan.
- Ultrasonic tests: for the RPV safe-ends, the UT inspection is performed with the MIS-B (Machine d'Inspection en Service Belge) from the inside surface. The minimum detectable crack height initiated on the inside surface is considered to be 5 mm. Each such weld is examined once every ten years according to ASME XI. The augmented program introduced in 2002 requires one, two or three examinations of those welds every ten years, depending on the weld type. The precise inspection schedule is regularly updated to take into account the inspection findings and after discussion with the Safety Authority.
- Eddy current tests of the inside surface of the RPV safe-ends welds have been introduced by the augmented program to enhance the detection capability for surface flaws.

The volumetric inspection technique is qualified by the third-party certification organization: EQB.

Preventive and remedial actions for PWSCC of dissimilar metal welds

Preventive measures to mitigate PWSCC are in accordance with PWR water chemistry guidelines for primary coolant systems, as established in EPRI Report 1014986 [5D]. The program description and the evaluation and technical basis of monitoring and maintaining reactor coolant water chemistry are presented in an AMP related to water chemistry.

Short-term repair techniques were developed by ENGIE Electrabel soon after the discovery of the first indications in the Belgian units:

- Removal of small surface indications (max. 4 mm) by grinding
- Larger defect removal by EDM for RPV outlet nozzle (and following NDE of the cavity), maximum depth 35 mm

These techniques have not yet been applied since no propagation of the existing indications was observed in successive inspections. Both techniques could introduce detrimental residual stresses and as long as the indications are not evolving, it is preferable to leave them as is.

A number of possible PWSCC mitigation techniques were evaluated and the developments in this field are closely followed but there is presently no final decision on the qualification and application of a preventive or mitigation technique.

5.A.1.2.8 Thermal ageing of low-alloy steel

Ageing assessment of thermal ageing of low-alloy steel

RPV operating temperatures range from 283°C to 329°C. Because the operating temperatures are less than 0.3 times the absolute melting temperature, creep or other self-diffusion controlled embrittlement mechanisms are not of concern.

Thermal ageing embrittlement (also called temper embrittlement) of ferritic low-alloy steels is due to intergranular segregation of phosphorus and other impurities, which reduce the grain boundary cohesion.

The name temper embrittlement refers to this phenomenon occurring during the cooldown from the tempering temperature, during which the material can remain in the temperature range of 600°C to 400°C for a long time (several tens of hours). The phenomenon is reversible and the initial properties can be restored by a thermal treatment at 600°C followed by a rapid cooling, hence the name reversible temper embrittlement. The same phenomenon could occur at primary circuit temperatures during long periods of operation and is often called thermal ageing in this context.

Temper embrittlement is manifested as an increase in DBTT, due to the change from predominantly cleavage fracture to predominantly intergranular fracture along impurity segregation paths. Macroscopically, this can appear as an increase in DBTT measured in Charpy-V impact test.

The DBTT is found to be directly related to the grain boundary concentration of the impurities, usually to phosphorus in commercial low-alloy steels, but also to the metallurgical structure (weld, HAZ or base metal) and temperature.

A key feature of the mechanical property changes induced by phosphorus segregation is that they do not include increases in yield stress or hardness.

It must be underlined that the presence of intergranular fracture does not necessarily induce a catastrophic degradation of the material properties, in particular of fracture toughness.

The contributing factors are mainly:

- The operating temperature.
- The prior austenite grain size and impurity content. The sensitivity increases with increasing grain size and the presence of specific impurities, in particular phosphorus.
- The phosphorus segregation on grain boundary can also be strongly influenced by the heat treatment applied during fabrication, in particular the cooldown from the tempering temperature.

Key design, manufacturing and operations documents used to prepare the AMP

- Manufacturing information of the RPV components (material composition, heat treatments)
- RPV operating temperature

Internal and external operating experience

Thermal ageing data in expert literature has been reviewed by the SCK•CEN. In addition, the issue of thermal ageing of low-alloy steel is summarized in an Ageing Summary, which also compiles the phosphorus contents of the materials in the Belgian units (which are generally low).

During an NPP's forty years of operation, the RPV components are exposed to operation temperature for more than 300,000 hours. In literature, such data are not available, especially for temperatures higher than those of the RPV beltline (285°C), since the main concern is a potential thermal ageing contribution in the RPV irradiation embrittlement. There is little information to be found about long-term thermal ageing at higher temperatures, such as those of the pressurizers for example.

The long-term ageing data at operation temperature of around 285°C, for which the reported maximum exposure times are in the order of 200,000 hours [7O], indicate only a marginal effect on the mechanical properties. None of the available data indicate some concern about thermal ageing within the time range for which experimental results are available.

Regarding the RPV beltline region, the surveillance capsule specimens are exposed to the same temperature as the vessel wall and any thermal ageing contribution is included in the post-irradiation testing results.

No failure related to thermal ageing of large primary components in low-alloy steel was ever experienced in a PWR. The phenomenon was not considered in the design stage.

R&D programs

An experimental program was carried out by the FROG on the retired Ringhals 4 pressurizer which was replaced in 2011 after 27 years of operation at 345°C service temperature. No significant effect was seen on the base material (A533B class 1 plate) nor on the HAZ (no measurable change in Charpy-V impact transition temperature and upper shelf energy). The fracture mode remains essentially in cleavage. The phosphorus content is reported as 0.008 to 0.009% in the fabrication reports, but on the blocks lower values were measured (0.004 to 0.005%).

On the other hand the welds showed an unexpected behaviour with a significant DBTT shift but complementary investigations identified that this was likely due to a different heat treatment on the weld block used in the acceptance tests and on the pressurizer. This high Ni-based weld (1.5% vs ≈0.8% in Belgian units) is not representative of the welds in the Belgian RPVs.

Establishment of acceptance criteria

There are no acceptance criteria for thermal ageing, as it is not mentioned in the ASME Code.

Monitoring, testing, sampling and inspection activities for thermal ageing of low-alloy steel

Not applicable.

Preventive and remedial actions for thermal ageing of low-alloy steel

Not applicable.

5.A.1.3 Hydrogen Flakes in RPV Shells

The RPV shells of the Belgian NPPs Doel 3 and Tihange 2 have been the subject of extensive safety studies following the detection of hydrogen flakes in both RPVs' shells. Although the safety cases have revealed that hydrogen flakes have been present inside the RPV shells since fabrication, and inspection campaigns revealed no indication of flake growth, ENGIE Electrabel decided to report this phenomenon in the framework of this NAR.

Ageing assessment of hydrogen flakes in RPV shells

In 2012, a specific inspection for underclad defects was performed in Doel 3. The inspection revealed no underclad defects, but it did uncover large numbers of indications deeper in the shells. As the Tihange 2 core shells had been manufactured at the same time and by the same supplier as the Doel 3 RPV shells, a similar inspection was performed at Tihange 2 in September 2012, with similar results. Therefore, both plants remained in cold shutdown with their core unloaded. A series of examinations, tests and inspections were performed leading to the conclusion that the indications were hydrogen flakes [6Q], that they were quasi-laminar in orientation and that they did not affect the structural integrity of the RPVs, regardless of the operating mode, transient or accident condition. All this was documented in the Safety Case Reports issued in December 2012 [8B][8C] and the Safety Case Addenda issued in April 2013 [8D][8E]. Based on these reports and on the supporting documents, the FANC authorized the restart of both units and they were put back on-line in June 2013 [8A]. This authorization was conditioned: ENGIE Electrabel needed to implement a mid-term action plan addressing aspects such as the qualification of the applied UT procedure, and mechanical testing of irradiated specimens containing hydrogen flakes [8A]. Material from a rejected AREVA steam generator shell (identified as VB395) containing the same type of hydrogen flakes as the Doel 3 and Tihange 2 core shells was irradiated in the BR2 test reactor at SCK•CEN. The preliminary results of the post-irradiation tests showed an unexpectedly high shift of the fracture toughness curve (T_0 of the master Curve) and of the RT_{NDT} that did not confirm the hypotheses considered in the initial Safety Case Reports. Therefore, ENGIE Electrabel decided to immediately shut down both plants. To fully address this concern, the material test program was extended, including several RPV materials and covering additional irradiation campaigns. This led to a modification of the irradiation embrittlement trend curves considered in the structural integrity analysis [64]. The final Safety Case Reports

[8F][8G], confirming the fitness-for-continued operation of both RPVs, were submitted to the FANC in October 2015. FANC allowed the restart of both units on 17 November, 2015. All Safety Case Reports as well as the reports from international expert groups are available on the FANC website [8H].

The Tihange 1 and Tihange 3 RPV core shells were inspected in 2013 and no hydrogen flakes were found.

The Doel 4, Doel 1&2 RPV core shells were inspected in 2015 and no hydrogen flakes were found.

Key design, manufacturing and operations documents used to prepare the AMP

- Detailed drawings of the RPV and materials certificates
- Manufacturing specifications for the RPV shells
- Detailed fabrication records of the RPV shells, from the casting of the ingot to the forging of the RPV shell, including all heat treatments
- In fabrication and pre-service inspection records

Internal and external operating experience

Hydrogen flakes are well known in the forging industry but their detection in the Tihange 2 and Doel 3 RPVs was a first-of-a-kind for a nuclear component in service.

Following the discovery of hydrogen flakes in the Tihange 2 and Doel 3 RPVs, this issue was investigated by many other utilities at the request of their Safety Authorities. In some cases, the investigation was limited to a detailed review of the fabrication conditions, in order to justify that the occurrence of hydrogen flakes could be excluded, and a limited number of plants performed partial or complete inspections, but no other case of hydrogen flaking was reported.

It must be underlined that the U.S.NRC did not request specific inspections for hydrogen flakes in U.S. plants.

R&D programs

An extensive experimental program was launched at the Belgian Nuclear Research Center SCK•CEN in support of this issue and for the development of a specific conservative trend curve for the flaked material. The program consisted of four accelerated irradiations in the BR2 reactor, on different materials:

- A material affected by hydrogen flakes (VB395, a discarded steam generator shell manufactured by Areva), which exhibited an unexpectedly high sensitivity to irradiation embrittlement
- Another material affected by flakes (KS02, a German material from a RPV flange), which behaved as expected without influence of the flaking on the irradiation embrittlement sensitivity
- Reference materials not affected by flakes (Doel 3 and Tihange 2 materials from the surveillance programs, Doel 3 archive material from the nozzle shell and from the core shell)

In total, more than 1500 mechanical tests were performed.

Establishment of acceptance criteria

Due to the large number of hydrogen flakes detected in the core shells of the Tihange 2 and Doel 3 RPVs, it was not possible to apply the standard procedure of the ASME Code for the justification of flaws found during in-service inspections. Specific rules were developed for the grouping of closely spaced flaws and for their analysis by the rules of fracture mechanics.

First, a screening criterion was developed to identify all the flakes that were clearly innocuous for the structural integrity. This step resulted in the elimination of 99.75% of the flakes detected at Doel 3 and of 99.69% of the flakes detected at Tihange 2.

The remaining flakes were analysed in detail using advanced finite element and fracture mechanics techniques to justify that they respected the ASME XI criteria [6S][6T].

Monitoring, testing, sampling and inspection activities for hydrogen flakes in RPV shells

In addition to the irradiated material testing, another mid-term requirement of the FANC was to perform a formal qualification of the UT procedure used in the inspections. The qualification process led to an adaptation of the inspection procedure with increased sensitivity levels. The updated cartography of the flakes was established and the structural integrity assessments of both RPVs were revised accordingly [6S][66T].

The restart authorization of November 2015 required additional in-service inspections in the affected units: after the first cycle after restart, and afterwards at least every three years. The in-service inspections need to be performed, using the same techniques, to confirm the absence of evolution of the hydrogen flakes population.

No evolution of the flakes distribution was detected after the first fuel cycle.

Preventive and remedial actions for hydrogen flakes in RPV shells

Hydrogen flakes are fabrication defects and prevention of their occurrence can only be achieved by taking the right precautions during fabrication.

Mitigation solutions: low leakage fuel loading patterns can be used to reduce the RPV fluence and are already in use in the Belgian units. Safety Injection water preheating is already implemented at Doel 3 to reduce the severity of the thermal shocks for the flakes located close to the inside surface (more exactly close to the cladding-base material interface) and to increase the safety margins for Pressurized Thermal Shock. It was not implemented at Tihange 2 as the hydrogen flakes are located deeper in the vessel wall and sufficient margins were demonstrated without this preheating.

No specific actions are necessary in the other units since they were all inspected and no hydrogen flakes were detected.

5.A.2 Licensee's Experience of the Application of AMPs for RPVs

The AMPs implemented to manage the different ageing mechanisms that could potentially affect the RPVs can be considered as adequate.

This is shown by the timely detection of indications in RPV upper head penetrations, BMI penetrations, dissimilar metal welds, and of the hydrogen flakes in the Tihange 2 and Doel 3 core shells (although these are fabrication defects and not related to an ageing degradation). The acceptability of theses indications was justified and the augmented inspection program was adapted accordingly when and where necessary.

Regarding RPV irradiation embrittlement, the surveillance programs for the Belgian NPPs provide information for fluences well above the one corresponding to 50 years of operation (for Doel 1&2 and Tihange 1) or 40 years (for the other units). In addition, the experimental results are in line with the predictions of the regulatory trend curves.

5.A.3 Regulator's assessment and conclusions on ageing management of RPVs

5.A.3.1 Regulatory oversight process

The RPV being one of the most critical SSCs of a nuclear power plant, it has been subject to a particular attention by the Safety Authority for a long time, in particular by Bel V and by Vinçotte, which is the Belgian mandated organisation regarding inspection of steam components. The respective missions of Bel V and Vinçotte have been defined in the frame of the transposition to Belgium of the regulatory aspects of section XI, Division 1 of the ASME code.

In particular:

- Bel V and Vinçotte perform a close follow up of the ageing management of RPVs by the licensee through regular meetings (generally on a yearly basis), during which the scope and planning of the ASME XI and other supplementary inspections are discussed with the licensee;
- In the last years, all the RPVs of Belgian nuclear power plants have been inspected, providing up to date results. Those inspections covered the welds as required by the ASME B&PV code, but the Safety Authority asked to inspect also the base material. The first aim of this request was to confirm the absence of underclad cracking, as an answer to the return of experience of Tricastin. But during these inspections on Doel 3 and Tihange 2, the presence of hydrogen flakes has been highlighted, and the Safety Authority asked to enlarge the scope of the inspection on the base material, to confirm the absence of such indications on the other RPVs.
- Bel V was strongly involved in the assessment of several safety issues related to the RPV, amongst others:
    - the underclad defect issue;
    - the hydrogen flakes issue in Doel 3 and Tihange 2;
    - the RPV head replacements;
    - the surveillance program;
    - PWSCC issues;
    - TLAA analyses for LTO.

The Safety Authority benefits from all this past experience, and therefore, no additional specific reviews have been performed for the assessment of the ageing management program of RPV in the framework of the TPR.

5.A.3.2 Regulator's assessment

5.A.3.2.1 Scope of ageing management for RPVs

All parts of the RPV which constitute the pressure boundary are in the scope of the ageing management. Also, all known ageing mechanisms able to affect the RPV are documented and analysed in dedicated ageing summaries or AMPs and monitored through surveillance or inspection activities. These ageing mechanisms correspond to the ones identified in well-known international guidance (such as IAEA-TECDOC-1556), as well as the ones identified in recent return of experience. The scope and the procedures for the identification of ageing mechanisms developed by the licensee are adequate for the ageing management of the RPV.

5.A.3.2.2 Ageing assessment of RPVs

Understanding RPV ageing is essential for an effective RPV ageing management. This understanding consists in knowledge of RPV materials and material properties, stressors and operating conditions, likely degradation sites and ageing mechanisms, condition indicators and data needed for assessment. The understanding of RPV ageing is derived from the RPV baseline data, the operating and maintenance histories, and return of experience.

The licensee has a comprehensive understanding of an RPV, its ageing degradation, and the effects of the degradation on the ability of the RPV to perform its design and safety functions. This understanding is supported by the licensee's knowledge of the design basis, fabrication, operation and maintenance history, inspection results, generic operating experience and research results. This knowledge has been updated in the frame of the analysis of the hydrogen flaking issue. All known ageing mechanisms requiring management are assessed by the licensee and discussed with the Safety Authority on a regular basis, in different frames. For each of them, acceptance criteria have been developed on the basis of international practices and guidance and approved by the Safety Authority. Internal and external operating experience is closely followed up by the licensee, and he is generally involved in dedicated R&D activities. All relevant information is documented in dedicated ageing summaries or AMPs.

5.A.3.2.3 Monitoring, testing, sampling and inspection activities for RPVs

The RPV inspection and monitoring activities are designed to detect and characterize degradation before the safety margins are compromised.

The in-service inspection program for the Belgian RPVs is carried out according the ASME XI code, transposed to the Belgian situation. Also, a surveillance program to monitor the irradiation embrittlement has been established as required by 10 CFR 50 App. H and as specified in ASTM E185. Moreover, an augmented program has been established to cover all known specific issues such as underclad defects which would not be sufficiently addressed by the ASME XI code. These inspections are generally based on international practices and guidance and followed by a mandated organisation competent for vessel inspection. These inspection and surveillance activities are adequate to identify in a timely manner potential ageing degradation of the RPV.

5.A.3.2.4 Preventive and remedial actions for RPVs

For all known ageing mechanisms, the licensee takes the necessary mitigation measures to lower as much as possible the rate of ageing degradation. Also, when necessary, remedial actions have been taken early enough, for example in replacing the reactor pressure vessel heads.

5.A.3.2.5 Licensee's experience of the application of AMPs for RPVs

The experience shows that the AMPs and Ageing Summaries implemented to manage the different known RPV ageing mechanisms can be considered as adequate. Although the hydrogen flakes in the Tihange 2 and Doel 3 core shells have been detected lately, this does not put into question the adequacy of the existing AMPs, since hydrogen flaking is not considered as an ageing degradation. However an augmented inspection program has been asked by the Safety Authority to confirm that these indications do not grow with time. The augmented inspection program requires follow-up UT-inspections, at the end of the cycle after their first restart, and thereafter at least every three years.

5.A.3.3 Conclusions

All known ageing degradation mechanisms affecting potentially the RPV are covered by this ageing management program of ENGIE Electrabel, and are closely followed by the Safety Authority since their significance have been highlighted. The attributes of the RPV ageing management are very similar to international practices and guidance, and this is adequate to manage the different ageing mechanisms that could potentially affect the RPV.

5.A.4 Action plan

No need for further improvement has been identified for the ageing management of the RPV.

5.B BR2 Research reactor

5.B.1 Description of ageing management program for BR2 RPV

5.B.1.1 Scope of ageing management for BR2 RPV

Description and material characteristics

The pressure vessel of the BR2 reactor is made of sheets of an aluminum alloy, laminated and assembled together with circumferential and vertical welds. It has a length of about 7.5 m and its diameter goes from about 1 m in the center line to about 2.5 m in its extremities. The total weight of the vessel is, without its covers (top and bottom), about 9 tons. The vessel is mainly composed of 6 elements, from which two are forged and the other four are made of hot rolled plates. These rolled parts are:

- A diablo shaped shell in the center, the thickness of which is 21.4 mm in its cylindrical part in the middle with an inner diameter of 1107 mm and a height of 1524 mm (fig 2, item A).
- Two truncated conic shells (bottom and top) that have a thickness of 63.5 mm and 48 mm, respectively (fig 2, item B).
- A cylindrical shell with a thickness of 63.5 mm for the bottom part of the pressure vessel (fig 2, item C).

These shells are all made out of the aluminum alloy AISI 5052-gr.O (ASTM-GR20A). Aside from the lower cylindrical shell, they all have two vertical weld cords.

The forged elements are two annular seamless flanges (fig 2, item D), with an inner diameter of 2134 mm and a minimum thickness of 48 mm in an forged aluminum alloy of the type B54S (English standard, corresponding to alloy AISI 5083). The two annular flanges (top and bottom) are welded to the main body of the vessel.

The top and bottom covers of the pressure vessel are made in stainless steel type AISI 304 ELC. The fixing bolts are in stainless steel type AISI 416.

Cooling water enters the vessel through 2 inlet nozzles, located near the top (fig. 1, item 1). It flows downwards through the core and leaves the vessel though the 2 outlet nozzles near the bottom (fig. 1, item 2). Construction details of the nozzles is found in Figure 5A-3.

The shroud (the envelope around the center section of the pressure vessel for external cooling of the wall) has a thickness of 6.35 mm in the middle section and of 9.5 mm at the truncated conical sections. The shroud is also made in the alloy AISI 5052-gr.O and it is fixed to the pressure vessel by means of welding. The available volume in the shroud is in the order of 1 m^3. This shroud has no structural safety function. Pool water enters the shroud from the bottom, passes along the vessel wall and flows freely in the pool.

In the design of the reactor vessel, due consideration has been given to the loads resulting from:

- Pressure of the primary coolant circuit
- Temperature gradients between primary coolant and pool water
- Nuclear heat deposited in the vessel wall of the core region

- Weight of the reactor internals (matrix, channels, experiments)
- Hydrodynamic effects due to the primary coolant flow
- Hydrostatic pressure of the pool water

The main design parameters are:

- Internal pressure: 1.45 MPa (210 psi)
- External pressure: 0.15 MPa (22 psi)
- Temperature: 93 ºC (200 °F)

The design specification was established by the two consulting engineers of SCK•CEN, namely:

- NDA (Nuclear Development Corporation of America, White Plains, NY, USA)
- BEN (Bureau d'Etudes Nucléaires, Brussels, Belgium)

In the absence (period 1957-1960) of any codes governing the construction of aluminum reactor vessels, it was decided to build the BR2 reactor vessel in accordance with the most stringent conditions of the ASME Boiler and Pressure Vessel Code (Unfired Pressure Vessels). Association VINÇOTTE was committed as Inspector. It was their duty to adapt the ASME rules to Belgian laws and to more restrictive prescriptions in accordance with their own experience.

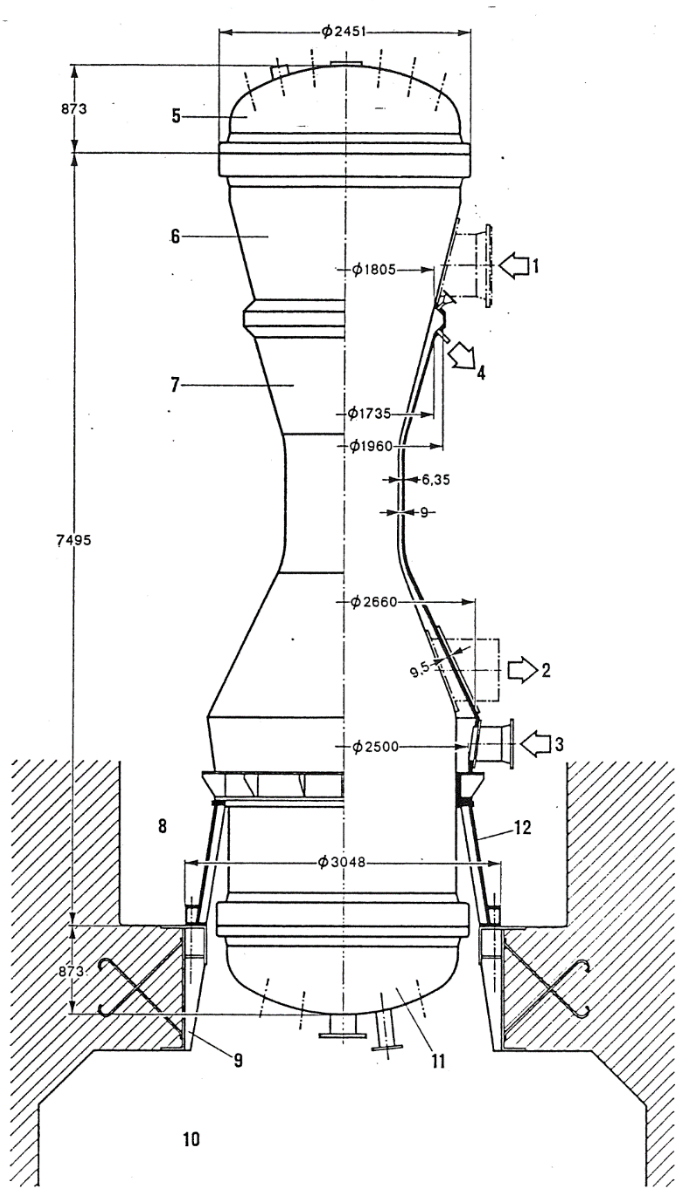


Figure 5B-1 : a half section of the vessel and the shroud, both in the reactor pool

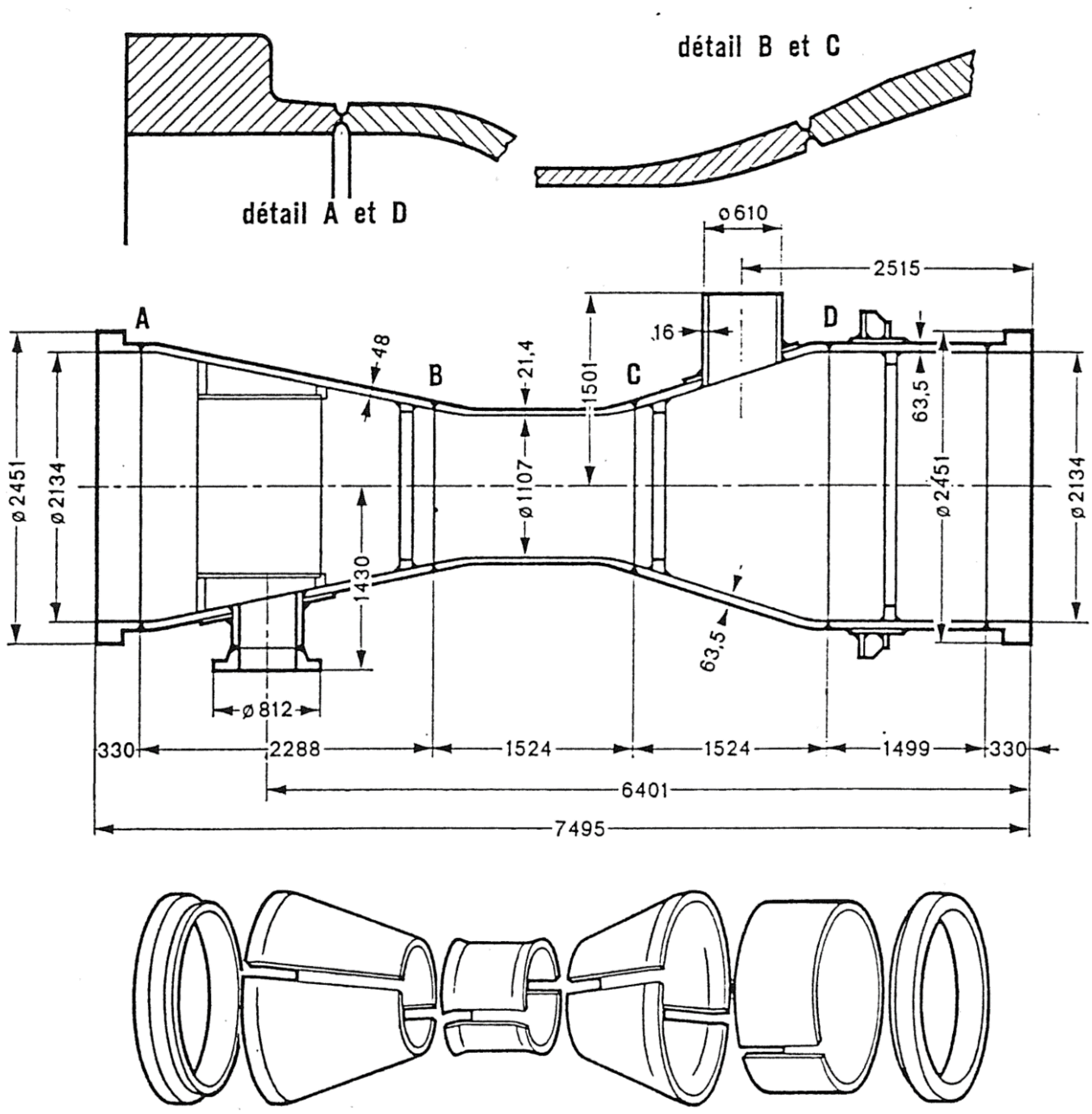


Figure 5B-2: main dimensions of the pressure vessel and an exploded view of its constructive elements

Criteria for selection of components within the scope of ageing management

a) Diablo shaped shell

Under normal operation conditions, the most solicited part of the vessel is the diablo shaped shell around the mid-plane of the core, where:

- The wall thickness is minimal: 21.4 mm
- The neutron fluences are the highest, resulting in the changes of its mechanical properties and metallurgical structure.
- The thermal stresses are highest due to the nuclear heating.

b) The remaining parts of the vessel wall

The remaining parts of the vessel wall (the two truncated conic shells, the cylindrical shell and the two forged flanges are not subjected to significant neutron irradiation.

c) Nozzles of the primary piping

The primary piping induces stresses on the vessel, mainly because of the variation of the momentum of the conveyed fluid and the thermal expansions. These stresses result in bending and torque moments applied to the inlet and outlet nozzles of the primary piping. The nozzles are not subjected to significant neutron irradiation.

d) The steel covers

The steel covers are not subjected to significant irradiation.

Identification of the ageing mechanisms for the different materials and components

The ageing mechanisms for the vessel are:

- Hardening of the aluminum due to neutron irradiation
- Low cycle fatigue due to start and stop of the reactor.
- Corrosion

Hardening of the aluminum occurs only in the central zone of the vessel. Following effects occur:

- Thermal neutron flux:
    - Production of silicon due to the reaction

$$^{27}Al\,(n,\gamma)\;^{28}Al \underset{\beta^-}{\rightarrow} {}^{28}Si$$

with an approximate efficiency of 0.2 wt% Si for a thermal neutron fluence of $1{\cdot}10^{22}$ n_{th}/cm^2.

- Fast neutron flux:
    - Damage in the crystal structure due to production of displacements per atom (dpa). According to the conversion studies, ~15 dpa are achieved for a fast neutron fluence of $1{\cdot}10^{22}$ $n_{(E>0.1\ MeV)}/cm^2$, with an effective threshold of 25 eV.
    - Production of gaseous atoms of ^1H and ^4He due to the reactions

$$^{27}Al\,(n,p)\;^{27}Mg \underset{\beta^-}{\rightarrow} {}^{27}Al$$

$$^{27}Al\,(n,\alpha)\;^{24}Na \underset{\beta^-}{\rightarrow} {}^{24}Mg$$

with an approximate efficiency of 16.34 appm of ^1H and 2.78 appm of ^4He for a fast neutron fluence of $1{\cdot}10^{22}$ $n_{(E>0.1\ MeV)}/cm^2$.

The studies performed by transmission electronic microscopy (TEM) gave the following results:

- The first dislocations appear after a fast neutron fluence of $1.5{\cdot}10^{21}$ $n_{(E>0.1\ MeV)}/cm^2$.

- Subsequently, a general structure of dislocations is created, as well as in-situ intergranular precipitation of fine particles of Mg_2Si. These particles are the product of the interaction between the Si produced due to transmutation and the Mg dissolved in the matrix.

- When the Mg is completely consumed, the Si continues to precipitate as a free element, given the fact that Si is not soluble in aluminum.

- There is a little amount of cavities produced and a little amount of preferential precipitation at the grain boundaries.

- The single phase alloy thus becomes a hardened alloy due to multiphase precipitation.

For fatigue analysis, two different types of failure mechanisms are taken into account. For the central part of the vessel, which is influenced by neutron irradiation, the material is considered as perfectly brittle. For the other parts of the vessel, plastic deformation is taken into account.

Corrosion is an issue and therefore the acidity has to be strictly monitored and controlled. The Al 5052-O alloy has an excellent corrosion resistance in pure water. The temperature of the primary water is low (less than 50 °C) and the water chemistry is strictly controlled for protection of aluminum. The acidity of the primary water is kept between 5.5 and 6.0 and the conductivity as low as possible (generally below 0.5 µS/cm). The use of materials that could release halogens (chlorine, fluorine, …) is not allowed in the primary of the pool water. Materials that could cause electrochemical corrosion (such as copper) are not allowed in direct contact with the primary or the pool water. The stainless steel covers are sufficiently far away from the core, such that no irradiation assisted corrosion effects will occur.

5.B.1.2 Ageing assessment of RPVs

Definition of the AMP for the reactor vessel

Inspection of the reactor vessel is, according to the license, required when the beryllium matrix is unloaded for replacement. Depending on the utilization of the reactor, this occurs every 15 to 20 year. Documentation about the initial inspection of the vessel is not available. The first documented inspected was done in 1979, the second in 1996 and the third in 2016. Each inspection was based on information and techniques of the previous one, completed with additional information and the use of up to date methodologies. The actual ageing management program of the BR2 reactor consists of four elements:

- The fracture mechanical evaluation of faults in the vessel wall. This calculation is used to determine the maximum allowable material fault.
    - Calculation of the maximum allowable fault was made using the R6 methodologies and the results were compared with the ASME XI method, with finite element calculations and with analytical methods. These calculations were made for the second inspection in 1996.
- The fatigue analyses.
    - Fatigue calculations are made according to the ASME XIII, div. 2 code.
- The material surveillance program, which is necessary due to the fact that the evolution of the mechanical properties of irradiated Al 5052-O were not known.
    - An extended research program was necessary in order to get information about the evolution of the mechanical properties of irradiated aluminum. A semi empirical model was composed and calibrated using test results with irradiated aluminum. Test material

came from samples of the shroud, used guiding tubes for control rods, samples from old vessel material of the HFR (Petten, Netherlands) and data from Oak Ridge National Laboratory – USA. The model is still being validated for higher fluences using test result of samples that are irradiated in the core.

- The non-destructive inspection of the vessel wall.
    - The inspection was prepared and executed by a company qualified for inspections according the ASME XI code. The tools used in previous inspections were reviewed and upgraded with new detectors. Inspection procedures were prepared by the inspection company. The whole inspection was witnessed and certified by the authorized inspection agency.

Fracture mechanics calculation

The stress intensity factor of an infinite long axial crack in the wall was calculated using the R6 method. The calculation shows that for a crack with a depth of 5 mm in the central part of the wall (which is the thinnest) can be accepted if the material toughness remains higher than 12 MPa.√m. The value gives a safety factor 2 against failure.

Fatigue analysis

For the fatigue analyses following load cycles are taken into account:

- Normal operation – emergency shutdown with depressurization to 5 MPa.
- Normal operation – scheduled shutdown (depressurized)
- Emergency shutdown – rest (depressurized)

The calculation showed that fatigue load in the central part is not a limiting factor for the lifetime of the vessel if no significant faults are present.

For the ductile part the highest load is found for the inlet piping at the top of the vessel. The most severe condition is the fast depressurization during an emergency shutdown. However the number of this load cases is very low. In case of an emergency stop of the reactor, normally the primary pressure is kept by reducing the secondary cooling flow.

5.B.1.3 Monitoring, testing, sampling and inspection activities for RPVs

The material follow up program

A major study program was devoted to this topic in the period 1992-95. The objective of the study was to develop trend lines concerning the embrittlement and the toughness associated to the reactor pressure vessel.

The final result of this study program proposes the acceptance criteria and provides de toughness evolution curve of the base material and welds of the reactor vessel. The executive summary is summarized hereunder:

"The static initiation fracture toughness of the BR2 aluminium vessel weld and base metals has been estimated by testing specimens cut from the aluminium shroud surrounding the vessel; this encompasses uniaxial tensile, notched tensile, three point slow bend, precracked Charpy and Charpy-V notch impact tests; to help extrapolate these data, similar measurements have been performed on BR2 control rod guide tubes and on remnants from the first HFR vessel. Chemical and microstructural characterization of the materials has also been done to provide the needed grounds for the data evaluation.

Two successive modelling steps have been adopted for this evaluation:

1) *Dislocation theory guided by transmission electron microscopy is used to define matrix strengthening in function of silicon production by thermal neutron and of atomic displacements induced by fast neutrons.*

2) *Two complementary micromechanical models are used to relate the toughness KI to the flow properties determined in the preceding step.*

It is concluded that the BR2 vessel toughness will remain adequate for the planned life extension beyond 1995, provided the scheduled in-service inspection reveals no unacceptable defect in the beltline welds. Due to the toughness extrapolation uncertainties, it is recommended to implement a vessel surveillance program using samples from the base and weld metals of the vessel shroud."

Samples of vessel material, taken from the shroud, were further irradiated. The model for prediction of the mechanical properties of the irradiated Al 5052-O is now validated for a neutron dose that will be reached far beyond 2026. The fracture toughness of the weld material is significantly lower than that of the base material. This is in any case a conservative result. The welds on the shroud were made in the field, while the vessel itself was welded in the shop. The quality of the vessel wall welds will be better than those of the shroud.

Non-destructive inspection of the vessel

The vessel wall can only be inspected from inside when the beryllium matrix is removed. This was done in 1979, 1996 and 2015. The outer wall is normally not accessible due to the presence of the shroud. The objective of the inspection is to detect indications larger than 3 mm. This gives sufficient margin compared to the limit of 5 mm. The irradiated part, with the 2 circular welds included, is completely inspected. For the unirradiated part only the welds are inspected. A limited part of these welds is not inspectable. The techniques used for the inspection are:

- Visual (with camera), ultrasonic and eddy current for the irradiated part of the vessel wall.
- Visual and ultrasonic for the welds of the unirradiated part.
- Visual and replica for the inlet nozzles
- Visual, ultrasonic and penetrant for the stainless steel head and bottom.

The inspection is done according to the methodology of the ASME V and AXME XI code, with the remark that the ASME XI code deals with power plants and is in principle not applicable to aluminium. However, the methodology can still be used.

The largest indication that was found has a depth of 5.7 mm and is located in a vertical weld of the bottom part. Although this value is larger than the general criterion of 5 mm, it is straightforward to show that it poses no risk for failure of the vessel due to the fact that on this part the vessel wall is much thicker than on the irradiated part (63.5 mm instead of 21.4 mm, see Figure 5B-2).

All indications that were found are compatible with the observations of the previous inspections. This shows that the indications date from the construction of the vessel and are not a consequence of ageing of the vessel. As a conclusion of the follow up program, the vessel is qualified for operation till at least 2026.

5.B.1.4 Preventive and remedial actions for RPVs

The condition of the vessel does not require special actions during the next years. In case of very long term operation objective (beyond 2030) the material irradiation program must be continued in order to have the model for prediction of the mechanical properties to be qualified. Next inspection will be done in case of replacement of the actual matrix.

The only issue that could occur in the next years is an abnormal (more than 1 per year) number of emergency stops with loss of pressure. The stops are the most severe load for low cycle fatigue and the fatigue analysis should be reviewed. However, up to now, these stops are very rare.

5.B.2 Licensee's experience of the application of AMPs for RPVs

The construction material of the vessel wall (Al 5052-O) has an excellent corrosion resistance and a good resistance against neutron irradiation. However, the knowledge about irradiated Al 5052-O was limited and a qualification program had to be defined. During the initial construction of the vessel, which dates from the late fifties of the previous century, no samples for material follow up were foreseen. This issue was solved by taking samples from the shroud, which is made from the same material. The irradiation history of the shroud is comparable to that of the vessel wall and by loading a number of samples in the core a lead factor on the irradiation dose could be obtained.

A remark about the design of the vessel, is that it is not easy to inspect:

- The vessel wall is only accessible from the inside, making the inspection only possible when the beryllium matrix is unloaded.
- A limited number of welds is not accessible, although these welds are not in the irradiated area.
- The nozzles are difficult to inspect.

5.B.3 Regulator's assessment and conclusion on ageing management of BR2 RPV.

5.B.3.1 Regulatory oversight process

It is noted that recently, the Beryllium matrix has been replaced. During this replacement, the RPV has been thoroughly inspected by the SCK•CEN. The inspection procedure and the results have been assessed by the Safety Authority.

In the framework of the TPR, Bel V performed a thematic inspection at the SCK•CEN, during which the Ageing Management Program related to the RPV was reviewed.

5.B.3.2 Regulator's assessment

5.B.3.2.1 Scope of ageing management for BR2 RPV

All parts of the RPV are considered in the ageing management. Also, the ageing mechanisms considered for the vessel are the ones which are expected since the design, because up to now no operating experience has highlighted new ageing mechanisms that should be taken into account. The scope and the procedures for the identification of ageing mechanisms developed by the licensee are adequate for the ageing management of the RPV.

5.B.3.2.2 Ageing assessment of BR2 RPV

The ageing mechanisms requiring management have been identified and their significance has been established. Acceptance criteria for the most significant ageing mechanisms have been derived by the SCK•CEN. Where needed, R&D programs have been developed, in particular for the surveillance program.

5.B.3.2.3 Monitoring, testing, sampling and inspection activities for BR2 RPV

In-service inspections of the vessel were carried out in 1979, 1996 and 2015. The main objective of these inspections was to detect the presence of defects in the walls of the reactor vessel, appearing either due to corrosion or low cycle fatigue, and to assess the possible evolution of flaws observed during previous inspections. During the last in-service inspection in 2015, the irradiated part of the vessel was fully inspected (base material and welds), while for the unirradiated part, only the welds were inspected. The techniques that were used are ultrasonic, eddy current and visual inspection. The results of this inspection, combined with the positive conclusion of the surveillance program, justified that the vessel could be used for the next expected lifetime of the beryllium matrix. No progressive deterioration has been detected up to now. This has been followed and approved by the Safety Authority.

The Safety Authority notes that the number of stops that provide the most severe load for low cycle fatigue is not explicitly tracked (although it is clear that such stops are very rare and that the upper bound has not been exceeded).

5.B.3.2.4 Preventive and remedial actions for BR2 RPV

Apart from maintaining the operation conditions as required to lower as much as possible the rate of ageing degradation, no specific preventive or remedial actions are necessary for the RPV of BR2.

5.B.3.2.5 Licensee's experience of the application of AMPs for BR2 RPV

Operational observations since the beginning of the reactor operation (1963) show the good condition or behaviour of the reactor vessel. This may be illustrated by the following facts:

- in the few cases where the primary circuit had suffered abnormal contamination due to defective fuel elements, the presence of small leaks in the vessel would have been revealed by the monitoring of the pool water. All observations made up to now have confirmed the tightness of the vessel.
- close control by analysis of the primary coolant water quality maintains confidence in the corrosion behaviour of the primary circuit as a whole;

5.B.3.3 Conclusions

The scope and the procedures for the identification of ageing mechanisms developed by the licensee are adequate for the ageing management of the RPV. The ageing mechanisms requiring management have been identified and that their significance has been established. Monitoring, testing, sampling and inspection activities are adequate. Operational observations since the beginning of the reactor operation show the good condition or behaviour of the reactor vessel.

The Safety Authority noted that the number of stops that provide the most severe load for low cycle fatigue is not explicitly tracked (although it is clear that this is very rare and that the upper bound has not been exceeded).

5.B.4 Action plan

The ageing management of the RPV of BR2 should be extended by explicitly tracking the number of stops that provide the most severe load for low cycle fatigue and compare this number with the established limit.

6. Calandria/pressure tubes

Belgian NPPs and research reactors are not CANDU-type. This chapter is therefore not in the scope of the Belgian NAR.

7. Concrete containment structures

7.A Nuclear Power Plants

7.A.1 Description of Ageing Management Programs for Concrete Structures

This chapter focuses on ENGIE Electrabel ageing management activities for the secondary and primary concrete containment structures of its Belgian nuclear power plants. Throughout the text, the Tihange 1 concrete containment structure is used as the representative example for the WENRA TPR [3A] [3B] analysis (unless stated otherwise), as it is the most exhaustive example relevant to the purpose of the review.

7.A.1.1 Scope of Ageing Management

Description of the containment's main functions

The concrete structures within the scope of this chapter are:

- **Primary containment**: Concrete containment structures, with or without a liner, designed to withstand the pressure associated with a significant leakage of coolant from the reactor cooling system (LOCA)
- **Secondary containment**: The concrete structure that surrounds:

A concrete containment structure as described in the first bullet
A (self-standing) steel containment designed to withstand the pressure associated with a significant leakage from the reactor cooling system

The seven Belgian nuclear power plants are located in Doel (four units) and Tihange (three units):

- The Doel 1&2 twin units are equipped with a self-standing steel primary containment, designed to withstand LOCA effects, and a reinforced concrete secondary containment.
- The other units are equipped with a pre-stressed[8] concrete primary containment – with internal steel liner – designed to withstand LOCA effects, and a reinforced concrete secondary containment.

The main functions of the containment are :

- Primary containment: Providing leak tightness for liquid, gas and fission products during operational and accidental (LOCA) situations
- Secondary containment: Providing a leak-tight annular space that captures possible leakage from the primary containment and protects the primary containment from extreme environmental and abnormal load cases such as tornadoes, design-basis accidents (explosions with projectiles, etc.) and reference airplane crash
- Both primary and secondary containments: Contributing to the overall biological shielding during operational as well as accidental situations

[8] For compatibility with TPR wording, the term "pre-stress" is used for post-tensioned primary containments.

7.A.1.1.1 Methods and criteria for selecting components within the scope of ageing management

Primary and secondary containments are safety-related structures that have been identified during the design stage as essential structures for which specific ageing management programs had to be implemented. Reference is made to the definitions, design requirements, testing and in-service surveillance requirements from the Standard Review Plan paragraphs 3.8.1 and 3.8.4 [30] for primary and secondary containments.

7.A.1.1.2 Processes and procedures for the identification of ageing mechanisms for the different materials and components of concrete structures

Primary containment pre-stress losses

It is a well-known phenomenon that pre-stress forces tend to decrease with time because of tendon steel relaxation and concrete long-term deformations such as drying shrinkage and creep effects. A correct evaluation of these pre-stress losses is therefore a key factor of a successful design.

Since the first application of pre-stressing systems (before World War II), many decades have been devoted to mastering the long-term pre-stress losses to safely include them in structural designs. The related theories, developed under the auspices of the Comité Euro-international du Béton (CEB, est. 1953) and the Fédération Internationale de la Précontrainte (FIP, est. 1952), now merged into the Fédération Internationale du Béton (FIB, est. 1998), are supported by extensive experimentation and may be considered as well established for more than 50 years [7Q]. The last 40 years were mainly devoted to further developing and progressively integrating the various existing models into unified standards, e.g. Eurocode 2 [7R].

From the design stage on, the follow-up of pre-stress losses has been organized as follows:

- When designing the primary containment, pre-stress losses have been fully accounted for, according to the applicable standards. Allowable lower bounds for pre-stress forces have been defined for raft tendons (if applicable), vertical and horizontal wall tendons, and dome tendons. These pre-stress lower bounds have been translated into theoretical expected behaviour for allowable upper bounds of long-term deformations of concrete. This theoretical expected behaviour was originally calculated at the design stage, and is expected to be reached in the long term. These lower bounds are not a limit as such, but a guide to follow any deformation. The evolution of the concerned parameter is recorded and compared on the long term with the behaviour that was expected at the design stage. The evolution trend is interpreted during the continuous follow-up.
- The Initial Structural Integrity Test (ISIT) has allowed to state the actual initial response of the structure to accidental conditions [1K].
- Long-term deformations of the concrete of pre-stressed structural elements have since then been monitored. The measured concrete deformations are assessed periodically.
- The pre-stress follow-up is completed by periodical lift-off tests on un-bonded tendons and destructive tests on witness beams where applicable.
- Concrete deformations are measured before, during and after each Integral Leak Rate Test (ILRT or Type A Test [1K]), allowing to check the instrumentation performance by comparison of the results with the containment's theoretical behaviour under pressure.

Primary containment – state of preservation

The primary containment's overall state of preservation is assessed through planned periodical visual inspections according to ASME XI Code – Division 1 [6M to 6O].

The primary containment's outer face inspection program mainly envisions the assessment of concrete cracking evolution as well as the observation of the possible onset of corrosion phenomena on the pre-stress system and reinforcement bars.

Internal steel liner

Components, examination type and frequency are defined by the inspection program, based on ASME XI code, edition 1992 requirements (Tables IWE-2500-1 EA to EP) [6N]. The examination schedule currently follows Program B requirements, as defined by IWA-2432 [6M] and the procedures specific to the units (the Tihange 1 reference document is cited as an example). If deemed necessary, surface areas may be subjected to augmented visual inspection if both sides are accessible or ultrasonically inspected if only one side is reachable.

The following components are subjected to examination:

- The 6-mm thick liner steel sheets as well as the venting system metal sheets
- The liner anchors: angles or studs embedded in the concrete, if accessible
- The polar crane supports
- The anchors of components attached on the liner
- The penetration sleeves
- All welds of these components on the liner

Examination categories, based on the ASME XI classification (Table IWE-2500-1), accurately defining the zones to be inspected are described in reference procedure (the Tihange 1 reference document is cited as an example).

Table 7A-1 gives an overall view on the follow-up programs for the Belgian NPPs' primary containments.

Units	Doel 3	Doel 4	Tihange 1	Tihange 2	Tihange 3
Extensometers:					
Operational	6 months (1)	6 months (2)	6 months (3)	6 months (4)	
ILRT (Type A test)	10 years (5)				
Lift-off tendons	5 years		-	5 years	
Witness beams	-	-	10 years	10 years	-
Concrete Outer Face Visual Inspection	5 years + Type A test				
Liner inspection	10 years (6)+ Type A test				

Notes

1. Doel 3: Extensometers are Huggenberger cells, DEMEC bases and vibrating wire extensometers, installed in 1997 on the external face of the containment, as a retrofit of the existing systems.
2. Doel 4: Extensometers are Huggenberger cells, DEMEC bases and vibrating wire extensometers, installed in 2011 on the external face of the containment, as a retrofit of the existing systems.
3. Tihange 1: Vibrating wire extensometers were embedded into the concrete for ISIT monitoring. In order to comply as much as possible with American regulation, it was decided later on to voluntarily reactivate this system for follow-up purposes. New vibrating wire extensometers have been installed on the external face of the containment in 2015 as a retrofit of the existing system.
4. Tihange 2 and 3: Extensometers are of the vibrating wire type, embedded into the concrete.
5. ILRT is not a structural integrity test. Nevertheless, it allows to assess the proper functioning of the instrumentation by comparing the measured results with the containment's theoretical deformation.
6. Within the ten-year interval, all accessible areas of the liner are progressively inspected during outages. A general visual examination is performed before each ILRT (Type A test).

Table 7A-1: The inspection periodicity of the primary containments of the Belgian NPPs

Secondary containment – state of preservation

The secondary containment is particularly exposed to outer environmental conditions (thermal actions, freeze-thaw cycles, concrete carbonation, chemical attack, etc.). Hence, its overall state of preservation is assessed through planned periodical (every 5 years) visual inspections, as specified in the LCO (Limiting Conditions for Operation) of the technical specification, complemented with material tests when necessary. Concrete surface conditions are examined for cracking, spalling, rebar corrosion, chemical attack effects, abnormal deformations, lack of air and water tightness, etc. Inspection procedures consider structures according to their relevance for nuclear safety as well as the accessibility for visual inspection.

The reactor building's overall settlements and relative displacements are monitored through three-yearly topographic surveys.

Findings are documented through relevant reports and Ageing Summaries (AS) according to their acceptability level, to organize a graded treatment: observation, augmented follow-up, corrective measures.

Primary and secondary containments – inspection procedures

The original inspection procedures have been updated in accordance with the continuous evolution of the regulatory framework, the results of the follow-up, and operational needs. Such procedures are referred to in AS.

For LTO units Doel 1&2 and Tihange 1, these procedures have been transposed into GALL [4H] AMPs. This transposition is expected to be updated according to the more recent IGALL terminology [2D] [2E].

In the future, a similar process is expected to take place for the other units.

Protective coatings

Protective coatings are used for different purposes, and the acceptance criteria of their ageing mechanisms are adapted to their specific purpose.

Inside the primary containment, for steel and concrete elements of "Service Level I" as defined by RG 1.54 [4M], the coatings have the following functions:

- Prevent or minimize loss of material due to corrosion
- Facilitate decontamination of concrete surfaces

- Avoid degradation that would lead to clogging of emergency core cooling systems suction strainers

The relevant functions apply to the steel liner of the primary containments.

Outside the primary containment, for concrete elements of the "Service Level II" as defined by the RG 1.54 [4M], the coatings have the following function:

- Facilitate decontamination of concrete surfaces

This function applies to the external face of the primary containment and to the internal face of the concrete secondary containment.

7.A.1.1.3 Representative sample of the containment designs

In the remainder of this document, the ageing management process – ageing assessment, monitoring, testing, sampling and inspection activities as well as preventive and remedial actions – is illustrated with reference to Tihange 1 (unless stated otherwise). The required appropriate justification is briefly reported hereunder.

Considering,

• the comparison between primary and secondary containments of the units within the scope of the review ;

• the purpose of the review, i.e. the ageing management of the concrete containment structures ;

• The units have been in operation for 42 years for CNT 1, compared to 35 years for KCD 3 and CNT 2 and 32 years for KCD 4 and CNT 3 ;

• That differences in the foundation systems for KCD and CNT units do not play a prominent role on the primary containment main purpose as a pressure barrier;

• That – while being part of the pressure barrier – the raft is exhibiting a fundamentally different structural behaviour than the cylindrical wall and spherical dome, acting as a bent plate rather than a membrane and further, far less sensitive to pressurization effects than these membrane elements;

• That differences in the raft geometry and design are therefore not to be regarded as justifying differentiated approaches unless from the raft pre-stress ageing management;

• The presence of a tendon gallery at KCD 3, KCD 4 and CNT 1 units, allowing a direct visual inspection of the vertical pre-stress anchor devices;

• The availability of witness beams for CNT 1 and CNT 2 units, allowing a direct visual evaluation of possible tendon corrosion phenomena, even for a primary containment protected by a secondary containment;

• The absence of lift-off tendons and raft sensors at CNT 1 unit;

• The retrofitted sensors at KCD 3, KCD 4 and CNT 1 units;

• The similarity of design between aeroplane crash proof secondary containments;

it is proposed to consider CNT 1 unit as the suitable NAR example for the WENRA TPR [1] [2] analysis of the Belgian nuclear power plants secondary and primary concrete containment structures, being the most exhaustive example relevant to the purpose of the review.

Nevertheless, in case considering the ageing of the lift-off systems would be required, further reference could be made to one of the other units – either KCD 3 & 4 or CNT 2 & 3. This position is *justified as the lift-off systems are quite similar and can be assessed irrespective the unit they are part of.*

7.A.1.2 Ageing Assessment of concrete structures

7.A.1.2.1 General

The transposition of the original inspection and ageing assessment procedures into GALL AMPs is organized as follows:

- Generic AMPs, valid for Doel 1&2 and Tihange 1, prepared according to the GALL program philosophy:

  AMP-S1: ASME XI – Subsection IWE – Steel containment program

  SAMP-S2: Concrete containment program also known as AMP PSP S2

  AMP-S4: 10CFR50 / J – Pressure retaining components leakage

  AMP-S8: Protective coating monitoring and maintenance program

- Specific AMPs, applicable to Tihange 1:

  CNT 1 – AMP-S1-CW: ASME XI – IWE– Steel containment program

  CNT 1 – SAMP-S2: Concrete containment program

  CNT 1 – AMP-S4: 10 CFR 50 / J - Pressure retaining components leakage

Generic and specific concrete containment programs deviate somewhat from the original GALL documents as the pre-stressed primary containments of Doel and Tihange are designed with grouted tendons instead of un-bonded tendons. Only lift-off test tendons of Doel 3 and 4, and Tihange 2 and 3 are un-bonded. As the recently issued IGALL document AMP 302 [2E] deals explicitly with pre-stressed concrete containments with grouted tendons, it is expected that the generic and specific concrete containment programs will be adapted accordingly in the future.

7.A.1.2.2 Ageing mechanisms

The main ageing mechanisms that might affect the containments are briefly listed hereunder. A more detailed description as well as their significance can be found in IAEA reference documents [2M] [2Q].

Concrete

- Chemical attack: Calcium leaching, sulphate attack, alkali-aggregate reactions (ASR), carbonation, delayed ettringite formation, chloride attack etc.
- Physical attack: Salts crystallization, freeze-thaw attack, abrasion, erosion, elevated temperature, settlement, etc.

Steel reinforcement

- Rebar and embedded items corrosion, stress corrosion cracking

Pre-stressing systems

- Pre-stressing tendons and anchor devices corrosion
- Loss of pre-stressing force due to concrete creep and shrinkage and pre-stressing steel relaxation
- Monitoring systems ageing: Corrosion, electrical connections oxidation, etc.

  This ageing mechanism neither impacts the primary containment's structural integrity nor its functionality as a pressure barrier in case of LOCA.

Internal steel liner

- Steel corrosion: Pitting, crevice corrosion, boric acid corrosion and induced loss of material, stress corrosion cracking
- Steel parts cyclic loading and induced cracking, mechanical wear of locks, hinges and closure mechanisms
- Elastomers deterioration of moisture barriers

Interactions of the liner with the concrete containment structure such as anchors to the concrete and barrel-to-basement junction

- Corrosion

Seals and protective coatings

- Flaking, blistering, chalking, cracking, peeling, discoloration

7.A.1.2.3 Acceptance criteria

The acceptance criteria for the ageing mechanisms affecting the concrete structures can be functional criteria (tightness or structural stability for example) as well as technical criteria (magnitude of defects for example)

It is important to state that the final decision about the acceptability of observed defects is the responsibility of the engineer in charge.

The acceptance criteria are listed in the relevant AMPs.

Environmental parameters – settlements

Limit values for this site-specific parameter are given in the SAR of Tihange 1 [1K], with due consideration of overall and differential settlements since the construction time.

The overall settlement observed during any period elapsed between two consecutive survey campaigns must remain lower or equal to the overall settlement observed during the previous period, with a 5 mm tolerance.

The residual differential settlements measured from 2017 may not exceed 10 mm, without prejudice of any previously observed displacement on the equipment, with a 5 mm tolerance.

Reinforced concrete

- Loss of structural integrity

  Return on experience indicates that the structural integrity may be endangered as soon as the steel reinforcement sectional loss due to corrosion reaches 5%. This value is an indication, which must be refined by additional studies if a sectional loss is acknowledged.

- Concrete surface general appearance

  ACI 349.3R, Chapter 5 [7P] standard prescriptions apply. They are transposed in the generic AMP: SAMP S2.

Pre-stressed concrete

The tendon anchor systems are deemed acceptable if there are no signs of steel cracking in the tendon ends or in the anchor plates, if there are no signs of active corrosion, if damaged strands would have been previously accepted through a specific analysis and if concrete crack widths are lower than 0.3 mm in the anchor plate vicinity.

- Pre-stress losses

Pre-stress losses due to steel relaxation and concrete creep and shrinkage are monitored by measuring the containment's long-term deformations. The theoretical expected behaviour has been established on these long-term deformations when designing the structure. Table 7A-2 summarizes the theoretical expected behaviour for Tihange 1.

Description	Raft [10^{-6}]	Wall [10^{-6}]		Dome [10^{-6}]
		Vertical	Horizontal	
Initial Pre-stress **After Concrete Elastic Shortening**	+63	+203	+357	+338
Concrete Shrinkage (2)	+100	+100	+100	+100
Concrete Creep (2)	+63	+175	+250	+246
Steel Relaxation (3)	-10	-10	-10	-10
Total (4)	+216	+468	+697	+674

Notes

1. A positive sign stands for a contraction deformation while a negative sign stands for an elongation. The values expressed are dimensionless.
2. These limits are set as the free – not restrained – values of these long-term concrete deformations.
3. The concrete deformation related to pre-stressing steel relaxation is an elongation.
4. Readouts from the extensometers – if present – must be compared to the sum of steel relaxation and concrete creep and shrinkage effects.

Table 7A-1: Pre-stressed concrete deformation - theoretical expected deformation (Tihange 1) (1)

Internal steel liner

In-service non-destructive examination results are compared to previously recorded inspection results. Accepting components for continued service is based on ASME XI – IWE-3120 and, more specifically, to Table IWE-3410-1 [6N].

The coated or painted areas are examined for evidence of flaking, blistering, peeling, discoloration or any other signs of distress.

Suspect areas are accepted by engineering evaluation or corrected through repairs or replacements. If necessary, additional examinations are performed.

Containment leak tightness

The SAR of Tihange 1 [1K] defines allowable leak rates for the double containment.

- Primary containment

  The primary containment internal steel liner provides the necessary leak tightness while the pre-stressed concrete structure provides the required resistance to the pressure and thermal conditions of LOCA. The acceptance criteria for the primary containment leak tightness tests (types A or ILRT, B and C) are fixed by the technical specifications of the unit [1K].

- Secondary containment

  The secondary containment leak tightness criteria are defined by the SAR [1K] in terms of a maximal allowable leak rate, with 6 mbar underpressure in the annular space.

7.A.1.2.4 Key standards, guidance, design, manufacturing and operations documents

Key standards and guidance as well as design, manufacturing and operations documents used for:

- Preparing the original inspection procedures
- The Ageing Summaries
- The AMPs for LTO units
- The acceptance criteria

are referred to hereunder:

- Key standards and guidance documents
- Design, manufacturing and operations documents as well as the references thereto

As stated in the specific AMPs, the operating experience has been accounted for, according to the SAR, § 13.4 "Self-Assessment and Audits" [1K] and the referenced procedures as well.

There is currently one R&D program linked to the ageing of the concrete structures. The aim of this program, performed at the SCK•CEN, is to study the extent of accelerated carbonation in concrete due to microcracks. The program was started in 2015 and extends to 2020. This program aims to assess the service life of a concrete structure by investigating and modelling how fracture properties (width, roughness) influence the carbonation rate and how, in turn, carbonation influences the fracture and transport properties (permeability, diffusion). In 2016, the program focused on evaluating the feasibility of producing cracked cement paste and performing experiments on the cracked cement paste samples.

In addition, the 2012 repair campaign for the Tihange 2 reactor building led to the development of a new type of mortar. The repair campaign had revealed issues with the concrete that was used during construction: heterogeneous, high porosity, low structural characteristics, etc. These issues hampered the adhesion of the repair mortar (reviewed by the University of Liège).

The parties involved in the repair campaign (Electrabel, Tractebel, ULg, AM Cop & Portier – Rinaldi, BASF) asked BASF to develop a new repair system, adapted to the specific situation of Tihange 2. After having tested various possible repair systems, a new type of repair mortar (MASTEREMACO S 5400) proved to be successful and was applied. It will also be used for future repairs.

As described in paragraphs 2.A.3.2.1 and 2.A.3.2.2, ENGIE Electrabel monitors various international R&D programs. Different experts within the ENGIE Group are assigned to share their experience with other operators during ageing meetings organized by the IAEA. The results of those meetings are the AMPs that are progressively implemented by ENGIE Electrabel. For example, the IAEA meeting regarding the Barsebäck incident led to the development of AMP S8 and the launch of maintenance programs to monitor any possible degradation of the protective coating inside containment [4T].

7.A.1.3 Monitoring, Testing, Sampling and Inspection Activities

The objective of the actions described in the following paragraphs is to detect the presence of ageing effects associated to the mechanisms described in paragraph 7.A.1.2.2.

The monitoring activities described below are supplemented by testing and sampling whenever deemed necessary by the engineer in charge.

7.A.1.3.1 Frequency of inspections

The frequency of inspections is established according to ASME XI – Division 1 - IWL [6O], the ACI 349.3R-02 [7P] and RG 1.90 [4N].

The normal frequency of inspections for the secondary containment and the settlement survey is every 3 years.

When deemed necessary by the engineer in charge, the frequency of inspections is adapted to the possible evolution of the ageing mechanism.

In the framework of the Tihange 1 LTO program, the settlement survey is (temporarily) conducted every three months for a period of three years that started in May 2017.

7.A.1.3.2 Primary containment inspection program – key features

Table 7A-1 provides an overall presentation of the inspection program for the Belgian NPPs' five pre-stressed primary containments, describing the related activities and required frequencies.

The acceptance criteria are mentioned in paragraph 7.A.1.2.3.

The inspection procedure is fully referenced. The main steps are briefly described hereunder:

Vibrating wire extensometers

- Vibrating wire extensometers' frequencies and temperatures are measured every six months to produce a periodical record of the concrete deformations on the long term
- The monitoring system's proper functioning is checked during type A tests (ILRT)

Witness beams

This ten-yearly destructive test aims at assessing whether a pre-stressed beam, subject to environmental conditions similar to those prevailing in the annular space, would show signs of any rebar or pre-stressing system (anchors and tendons) corrosion phenomenon. The main steps are:

- Visual examination for possible cracks and externally visible signs of corrosion
- Pre-stress anchor device examination, after concrete cover removal
- Rebar examination
- Tendon sections examination

It is important to note that this destructive test does not aim at assessing any order of magnitude of the residual pre-stress in the witness beam as this would never be representative of the actual pre-stress losses of the primary containment tendons.

Concrete outer face visual inspection

The five-yearly concrete outer face inspection program concerns:

- The areas designated for cracks examination, referred to during the initial structural integrity test (ISIT)
- The wall-raft junction (i.e. "barrel-to-base mat")
- The material hatch and the main penetrations
- The small penetrations that are grouped
- The main steam and water penetrations
- The dome pre-stressing anchors zone
- The polar crane level, and its local external pre-stress
- The buttresses and the pre-stressing anchors cover
- The pre-stressing gallery and the vertical tendons' pre-stressing anchors

Internal steel liner inspection

For Tihange 1, the liner inspection program is organized around nine zones, each of them limited by a 40° azimuthal angle. One or two zones are inspected during every outage, in such a way that the liner is fully inspected on a ten-year basis.

In each of these zones, the following items are examined:

- 100% of the surface of all plates in VT-3 according to Table IWE-2500-1, category EA
- 25% in VT-1 of all penetration welds of the steam and water systems (CVP and EAN), of supports of the polar crane and of the welds of the flanges and nozzles according to Table IWE-2500-1, category EB
- 100% in VT-1 of all containment surfaces requiring an augmented program (corroded zones and zone near the base plate) according to Table IWE-2500-1, category EC
- 100% in VT-3 of all seals, gaskets and moisture barriers according to Table IWE 2500-1, category ED
- 100% in VT-1 of all pressure retaining bolting (as well as the bolt torque if needed) according to Table IWE-2500-1, category EG

All examined components are described in a database, also keeping a summarized record of the results of each inspection.

7.A.1.3.3 Secondary containment inspection program – key features

Visual inspections

Inspections have been defined in the framework of a general ISI program, and inspection procedures are formalised in specific documents.

The objective of these inspections is to detect the presence of defects related to ageing mechanisms, and to follow their evolution in time. A specific software has been developed for those inspections, allowing to compare the inspected surfaces from one inspection to the other, and to detect potential through-wall cracks by comparing external and internal crack patterns.

Specific inspections

To complete the regular in-service visual inspections, more detailed visual inspections are organized for specific items, depending on the outcome of the regular inspections.

These inspections, requiring more sophisticated means (high-definition cameras, drones, etc.), are realised as “one-time inspections”.

It has been the case for the external face of the secondary containment of Tihange 1 in 2016, in relation with the LTO activities.

Sampling and testing

When deemed necessary by the engineer in charge, further inspection means will be activated. These means include the physical follow-up of cracks (crack monitoring areas, crack width meters, etc.), and the extraction of concrete core samples for testing related to concrete resistance, carbonation depth measurement, chemical degradation of concrete and other properties.

Topographical survey

The purpose of the topographical survey is twofold: it aims to evaluate the reactor building’s absolute settlement as well as its relative displacements with respect to the neighbouring buildings. This survey is related to the environment criterion as described in paragraph 7.A.1.2.3 of this document, and is performed according to the referenced procedure.

7.A.1.3.4 Primary containment surveillance – main results

Pre-stress losses

The pre-stress losses have been accounted for when designing the primary containment, in accordance with the regulations in force at that time (1970).

The response of the newly built structure has been monitored with embedded vibrating wire extensometers during the initial pressure testing in August 1974. The initial behaviour has been found in good agreement with the design assumptions.

The further evolution of the pre-stress has been monitored since then[9], regarding the initial calibration campaign performed early 1973 before pre-stressing. The extensometers' readouts therefore effectively integrate the wall and dome deformations from the very beginning of the structure's lifetime, including pre-stress initial effects, long-term concrete shrinkage and creep as well as pre-stressing steel relaxation phenomena.

This periodic monitoring, with an augmented six-monthly frequency since 2006, is showing concrete deformations that are not deviating from the forecast long-term behaviour. Currently, nearly 45 years after tensioning, the pre-stress losses remain in a steady state.

An extrapolation, performed in 2012 according to the most up-to-date standard models for concrete long-term deformations evaluation, is showing that pre-stress forces would remain within the theoretical expected behaviour for a 60 years' lifetime. Since then, the most recent monitoring results have fully confirmed this extrapolation.

Pre-stressing system and reinforced concrete – state of preservation

The periodic destructive tests on witness beams are showing the very good condition of the grouted tendons, anchor blocks and reinforcement, without any sign of corrosion, nearly 45 years after casting. The visual inspection also confirms the overall good condition of the primary containment's external face.

Internal steel liner

The main surveillance results are briefly presented below.

Almost all imperfections discovered during the last ten-year inspection interval (2006 – 2016) concern components – piping and cable tray supports, insulation, etc. – pressed against the liner with the ensuing potential risk of degradation. Repair campaigns have been carried out systematically for all these defects.

In 2008, surveillance revealed that some supports were damaged after having been hit by the polar crane. Some of these supports were pressed against the liner and some ventilation ducts were damaged. All these supports and ducts have been replaced.

Existing blistering zones were scrutinized during the 2014 inspection campaign. These zones, dating back to the plant's construction and duly justified at that time, are due to local out-of-plumb and

[9] As specified in the TLAA [74]: "Tihange 1 NPP primary containment has been designed and erected before 1974, complying with an own specification. Subsequent American regulation documents (…) were made applicable in Belgium after completion of the civil structure. At that time, the in-service inspection program was only based on the above referenced witness beams destructive tests. Nevertheless, vibrating wire extensometers were embedded into the wall and the dome to assess the as-built structural behaviour during the first pressure test. (…). It was not intended to use further these extensometers as a permanent monitoring system. From 1994 on, it has been voluntarily decided to organize the in-service inspection as much as possible in accordance with the American regulation (…): the Regulatory Guide 1.90 (Rev. 1, August 1977): 'Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons' and the ASME Code, Section XI, Subsection IWL: 'Requirements for Class CC Concrete Components of Light-Water Cooled Plants'. The existing vibrating wire extensometers were reactivated wherever possible. Systematic visual inspections and extensometers readouts are performed since then."

misalignments defects, that occurred during the concrete casting, without any structural consequences. No evolution has been highlighted. As part of the LTO project, a list summarizing the areas exhibiting these local blistering phenomena has been prepared.

Before the 2016 type A leak test (ILRT), a general visual examination of all accessible surface areas of the liner was carried out. No inacceptable defect has been highlighted.

7.A.1.3.5 Secondary containment surveillance – main results

Reinforced concrete – state of preservation

The external face of the secondary containment does not show any evolution of the imperfections already detected. The concrete surface shows some cracks and local spalling. In view of the License Renewal Application for Tihange 1, a rehabilitation program has been set up to prevent degradation of reinforced concrete for the extended lifetime period.

Reactor building settlements and relative displacements

The settlements and the relative displacements between the buildings that are part of the nuclear island show a very slow evolution, in accordance with the type of structures. Particularly, the differential settlements remain compatible with the plant’s operational conditions.

7.A.1.4 Preventive and Remedial Actions

7.A.1.4.1 Preventive actions

Preventive actions aim at avoiding onset or development of any ageing mechanism. These actions have already been used for some units, and are mentioned in the procedures.

Protection against environmental parameters

ENGIE Electrabel has applied protective coatings to avoid water leaks or infiltration. This has been the case for several units of the Doel NPP, for example.

Reinforced concrete

The application of a watertight coating or a water repellent treatment can avoid water ingress into the concrete, preventing ageing mechanisms such as reinforcement corrosion, ASR and leaching. These coatings or treatments are regularly inspected to evaluate their effectiveness on the long term.

The injection of products in concrete cracks is also considered as ageing prevention, as it limits the evolution of corrosion phenomena.

The treatment is adapted to the types of cracks:

- Active cracks, their width evolving with external thermal or mechanical actions
- Passive cracks, exhibiting no or negligible evolution with time

The injection material is adapted to the types of cracks, but also adapted to the cement properties of the existing concrete. Chemical and mechanical properties of the injection product are compatible with the concrete structure to avoid negative side effects.

Pre-stressed concrete

In addition to the preventive actions for reinforced concrete listed above, pre-stressing tendons anchor devices in the pre-stressing gallery have been protected by a grouting.

Internal steel liner

The AMP related to the steel containment is a surveillance program that does not specify typical preventive actions.

7.A.1.4.2 Remedial actions

Reinforced and pre-stressed concrete

Remedial actions are needed when the acceptance criteria defined in paragraph 7.1.2.3 of this document are exceeded, and if the components' required safety functions would be impacted in the short term due to the absence of such a remedial – or corrective – action.

The preventive actions listed above can also be considered as remedial actions for the acceptance criteria of the second level as described in ACI 349.3R [7P].

Visible defects on reinforced concrete that could induce reinforcement corrosion – even not visible from the outside – require adapted concrete refurbishment actions.

A general procedure describing the methods ensuring the quality of concrete repair works has been issued for this purpose:

- When concrete renovation involves reinforcement exposition, rebars will be inspected and, if deemed necessary by the engineer in charge, they will be repaired and/or replaced.
- The surface of the concrete zone affected by rebar corrosion is visually inspected before the application of the new repair material, checking the adequacy of the concrete and steel surface preparation. The new concrete material must be chemically, mechanically and physically compatible with the existing concrete.
- In case the component's structural integrity would be questioned, multipurpose corrective actions must be taken. The component's structural strengthening will be considered, taking into account the origin and the magnitude of the threat. It is for example not possible to replace or repair pre-stressing tendons in a sleeve injected with cement grout. Other solutions, e.g. installing new external pre-stressing tendons or strands, would have to be examined and specifically studied to demonstrate their feasibility and effectiveness. The preventive actions listed above aim to anticipate such situations, preparing in due time the corrective actions, before the component's structural integrity would be impacted.

Internal steel liner

The required remedial – or corrective – actions are designed according to ASME XI – Subsection IWE [6N], supplemented by Subsection IWA for pressure retaining components repair activities [6M], or justified after an engineering evaluation.

7.A.2 Licensee's Experience of the Application of AMPs

Recent AMPs have been created by updating existing inspection and repair-replacement procedures drawn up from the very beginning of plant operation as recorded in the AS.

This chapter highlights the main preventive and remedial actions performed at Tihange 1 according to the requirements of these existing procedures. Recent AMPs are expected to be applied in the same way in the future.

Vibrating wire extensometers expertise – 1994

To evaluate the proper functioning of the existing concrete deformation sensors, ENGIE Electrabel contacted TELEMAC, manufacturer of the extensometers installed at Tihange 1, 2 and 3, to perform an assessment. This resulted in improved insight into the reliability of the existing systems.

Internal steel liner blistering follow-up – 2014

As stated in paragraph 7.1.3.4 of this document, existing blistering zones were scrutinized during the 2014 inspection campaign of the internal steel liner at Tihange 1. The observations made during successive outages are compared through photogrammetric techniques. The results showed no evolution.

Pre-stressed concrete deformation follow-up system refurbishment – 2015

In accordance with the well-established Belgian ageing management practice ([2M], Appendix I and [2Q], Appendix III), the existing concrete deformation follow-up systems have been supplemented by additional vibrating wire extensometers, placed on the primary containment's external face.

The experience gained with the new instrumentation at Doel 3 (1997) was first used at Doel 4 (2011) and then, more recently, at Tihange 1. The new systems at Doel 3 and 4 are fully operational while the system installed at Tihange 1 is currently being calibrated. In any case, the existing systems are still to be used until the end of the plants' operational lifetime.

Secondary containment – concrete renovation

In perspective of the License Renewal Application of Tihange 1, the results of the regular inspections of the secondary containment have been examined in view of a longer lifetime for this structure. Inspection and maintenance programs have been extended, and preventive/corrective actions have been undertaken. A specific project is launched to resolve the issues that have been detected during the SALTO mission.

7.A.3 Regulator's assessment and conclusions on ageing management of containment concrete structure for NPP

7.A.3.1 Regulatory oversight process

The ageing management of the concrete containment structures s followed by the Safety Authority. In particular, the two following subjects were specifically addressed during the last Periodic Safety Reviews (2012-2015):

- The loss of prestressing;
- The renovation of buildings and structures.

In addition for the LTO of Doel 1&2 and Tihange 1 units (2015-2025), the Integrated Plant Assessment (IPA) related to concrete containment structures was reviewed. The action plans decided by the licensee in order to upgrade the ageing management is closely followed up by the Safety Authority.

In the framework of the TPR, some clarifications were asked and discussed during several technical meetings, amongst others concerning the design limits and the safety margins for the loss of prestress.

7.A.3.2 Regulator's assessment

The Tihange 1 concrete containment structure is appropriate as the representative example for the WENRA TPR.

The approach of the licensee for defining the scope of ageing management of concrete containment structures follows the good practices. The licensee adequately follows the international standards.

The results from all the monitoring, testing, sampling and inspection activities give a good overview of the long term behaviour of the concrete containment structures. Although some more or less recent events highlight that in the recent past the monitoring and the preventive and remedial actions for the ageing management of concrete structures in Doel and Tihange NPPs have to be improved, today the preventive and remedial actions performed by the licensee for the ageing management of the containment concrete structures of NPPs are satisfactory. However the ageing management program of the other concrete structures still need for improvement and follow-up (as highlighted by the recent event reported in section 2.A.7.5.

Licensee's experience of the application of AMP for concrete containment structures is the creation of one AMP that concerns the concrete containment. It updates existing inspection and repair-replacement procedures drawn up from the very beginning of plant operation.

This list of the preventive and remedial actions which were performed after non-critical ageing degradation results from the monitoring, testing, sampling and inspection activities. This demonstrates that some ageing phenomena proceed as predicted by the ageing management program of the licensee, and that the frequency and acceptance criteria of the monitoring, testing, sampling and inspection activities are effective in their purpose to timely detect ageing degradation. In some case, preventive and corrective actions have been undertaken.

7.A.3.3 Conclusions

The licensee has identified the relevant ageing mechanisms of concrete containment structures. The technical documentation and the operating procedures to cope with them and to monitor the ageing of the concrete containment structures are adequate.

Nevertheless, during the last decade, several events highlighted that the monitoring of the ageing of the concrete structures was a weakness of the ENGIE Electrabel ageing management program. The need for reparation of the containment concrete structure of Tihange 2 underlined the failure of the preventive and remedial actions for this type of component, as existing at that time, and the need for improvement of the ageing management of concrete structures. Although some measures were taken for improving the ageing management of concrete structures, some recent events at Doel NPP still show that this topic need further attention. Attention should be put on fully implementing the ageing management on safety important concrete structures of NPPs.

Based on the above elements, the ageing management of the concrete containment structures will receive even more attention and careful follow-up by the Safety Authority in the years to come.

7.A.4 Action Plan

Although some recent events highlight the necessity to extend and deepen the ageing management program of concrete structures, the Safety Authority acknowledges that the ongoing actions should answer these concerns. In consequence, no other actions than ensuring a complete implementation on the field of the ageing management program of concrete structure are identified by the Safety Authority.

7.B BR2 Research reactor

7.B.1 Description of ageing management program for BR2 concrete containment structures

7.B.1.1 Scope of ageing management for concrete containment structures

The BR2 containment consists of a steel shell supported on the inner side by a concrete structure. The steel shell and the concrete wall are separated by a layer of isolation material (Styrowanner®). The concrete cylindrical wall has a diameter of 30.5 m and a height of 33.0 m. The concrete wall supports the containment building floor slabs on reinforced concrete corbels which project from the inner circumference of the wall. The polar crane is also supported by a corbel at the top of the concrete wall. There are a number of penetrations through the wall. The most significant one is the opening for the vehicle air lock which is approximately 3.5 m wide and 4.0 m high.

The steel shell is placed on a reinforced concrete foundation plate with a thickness of 1.25 m. A reinforced concrete plate with a thickness of 4.11 m is placed on the bottom of the steel shell to support the concrete walls and the reactor pool.

The reactor pool is placed on concrete columns. Between the columns, a concrete wall with steel doors is placed in such a way that the area between the columns is a water tight area (the sub pile room).

The safety functions of the concrete structures are the stability of the containment wall and the stability of the columns for the pool wall. The water tightness of the sub pile room is also a safety function.

7.B.1.2 Ageing assessment for concrete structures

The environmental conditions of the concrete structure are very mild. All concrete is located within the steel shell and is thus protected from adverse external conditions (rain, freezing, wind, …). No severe degradation mechanisms are to be expected.

7.B.1.3 Monitoring, testing, sampling and inspection activities for concrete structures

During the refurbishment of 1996, an inspection of the condition of the concrete wall was done. The concrete wall was found to be in a good condition.

The leak tightness of the containment building is being tested twice a year. Severe degradation in the concrete wall would damage the steel shell and lead to a leak. Up to now, no such leaks have been detected.

The leak tightness of the sub pile room is also being tested twice a year. A number of leaks in the walls were found and repaired. No structural faults were detected. Leakage has been found in the walls, especially near penetrations of the walls (doors, tubing penetrations, …). These defects have been repaired.

A leakage in the pool will easily be detected since the pool wall is nearly completely visible and a leak will be seen early, before significant degradation occurs.

7.B.1.4 Preventive and remedial actions for concrete structures

The only remedial action that has been defined is a coating of the inner side of the sub pile room walls to improve the leak tightness, although this is not directly related to ageing management.

7.B.2 Licensee's experience of the application of AMPs for concrete structures

Due to the mild operation conditions, no degradation of the concrete structures is observed, even after more than 55 years of operation. As a consequence no dedicated AMP is foreseen.

7.B.3 Regulator's assessment and conclusions on ageing management of BR2 containment concrete structure

7.B.3.1 Regulatory oversight process

The ageing management of concrete structures has been assessed during a thematic inspection in October 2017. The conclusions of this inspection have been taken into account in the framework of the TPR.

7.B.3.2 Regulator's assessment

7.B.3.2.1 Scope of ageing management for concrete containment structures

The SCK•CEN considers that the concrete structures in the BR2 reactor and the steel shell are within the scope of this TPR. The principles drawing of the BR2 reactor and machine building can be found in Figure 7B-1.

However, no details were provided concerning the characteristics of the concrete, the reinforcements or the steel shell. The lack of details does not allow having a good description of the containment structures.

7.B.3.2.2 Ageing assessment of containment structures

On the basis of mild environmental conditions within the steel shell that protects the containment structure from adverse external conditions, SCK•CEN does not expect problems due to water intrusion with further damage due to corrosion of freezing. Consequently, SCK•CEN does not identify any ageing mechanism of concrete containment structures.

The Safety Authority acknowledges the benign environmental conditions but is of the opinion that the concrete containment structures should be included in the ageing management program.

7.B.3.2.3 Monitoring, testing, sampling and inspection activities for concrete containment structures

An inspection of the condition of the concrete wall was performed in 1993 for the refurbishment of 1996. No other inspection or assessment of the structures was performed since 1993.

SCK•CEN has neither a dedicated monitoring nor an established testing program. No sampling of the concrete structures at the time of construction is available.

SCK•CEN states that no inspection is performed, except occasional visual inspections.

The Safety Authority is of the opinion that SCK•CEN does not satisfactory perform inspection activities related to the ageing degradation mechanism of the concrete containment structures. It is strongly recommended to foresee such inspection activities in the frame of the ageing management program that is currently under development.

7.B.3.2.4 Preventive and remedial actions for concrete containment structures

A repair work is ongoing in 2017 concerning the coating of the inner side of the sub pile room walls to improve the leak tightness.

No other preventive or remedial actions were performed up to now.

The Safety Authority notes that preventive and remedial actions for concrete containment structures have not been included in the ageing management program and no significant actions have been undertaken in this area.

7.B.3.2.5 Licensee's experience of the application of AMP for concrete structures

On the basis of mild operation conditions, the SCK•CEN states that no concrete degradation is expected and therefore did not include this in the AMP. Consequentially SCK•CEN has no experience in the application of AMP in this particular instance.

7.B.3.3 Conclusions

The Safety Authority acknowledges the benign environmental conditions but is of the opinion that the concrete containment structures should be included in the ageing management program. Based on the ageing assessment, SCK•CEN should determine the need for monitoring, testing, sampling and inspection activities as well as for preventive and remedial actions.

7.B.4 Action Plan

The part of the action plan PSR 2016 related to the ageing management program should be extended by SCK•CEN to cover concrete containment structures. In particular the need for monitoring, testing, sampling and inspection activities as well as for preventive and remedial actions should be determined on the basis of a dedicated ageing assessment.

8. Pre-stressed concrete pressure vessels

Belgian NPPs and research reactors do not have pre-stressed concrete pressure vessels. This chapter is therefore not in the scope of the Belgian NAR.

9. Overall assessment and general conclusions

In Belgium some elements of Ageing Management exist since the construction and the commissioning of both power and research reactors, as for example maintenance and in-service inspection program. For nuclear power plants, it is obvious that the design of the most important components, like the ones acting as one of the confinement barriers (RPVs, concrete containment structures…) were conceived so as to allow a monitoring of the evolution of some critical physicochemical properties. It is indeed obvious for example that the NPP RPV irradiation embrittlement is monitored through a surveillance program, since the phenomenon is identified in the 1960s. In the same way, some choices were made from the design phase of the BR2 research reactor, like foreseeing the replacement of the Be matrix.

Over time some additional attention was given to the management of ageing mainly with establishing a qualification process for the lifetime of the systems, structures and components important to safety. Also, the degradation mechanisms identified over time and deemed relevant for the Belgian reactors are monitored by the licensee as a common practice.

However the need for structuring all activities relevant to Ageing Management in an Overall Ageing Management Program in order to ensure a complete program covering all SSCs is a relatively recent development in Belgium for both nuclear power and research reactors. Indeed at an international level, awareness of the importance of Ageing Management appears and develops essentially from the mid-2000s, in parallel with the first long term operation projects for nuclear power plants; not much later and due to long and unexpected outages that impacted the supply of radioisotopes, this development was extended to research reactors. Also in Belgium, a structured ageing management program was introduced in 2004 for NPPs, based on the NS-G-2.12. With the possible LTO for the first NPP units, the need for structuring the Ageing Management Program was exacerbated since 2009.

Since that time, a continuously growing attention is focused on Ageing Management by both the licensees and the Safety Authority.

Nuclear Power Plants

In line with the publication of international standards and the preparation for a LTO of its first units, the licensee ENGIE Electrabel developed a more structured and complete Ageing Management Program.

The ageing management program by ENGIE Electrabel is now in line with the international standards and should ensure an adequate management of the ageing of the safety-related SSCs during the rest of the lifetime of the NPPs.

The Safety Authority considers that recent and still ongoing reinforcement of this ageing management program in the framework of the LTO program for the three first units is a significant achievement. This new ageing management program for the LTO units (Tihange 1 and Doel 1&2) is considered as complete and based on a systematic and comprehensive approach. The planned extension of this program to the ageing management program of Tihange 2&3 and Doel 3&4 based on the program installed for the LTO units, and using the most recent international standards and guidance, is a positive initiative which will be a challenge for the forthcoming years.

Nevertheless some past and recent events still highlight ageing issues and an inadequate monitoring of the ageing of some components and an insufficiency of remedial actions in the Belgian Nuclear

Power Plants. Ongoing investigations must show if these recent findings are symptomatic of a structural or particular deficiency, in order to adapt in consequence the ageing management programs.

In the framework of this TPR, no additional action or improvement has been identified by the Safety Authority for the overall ageing management program. The Safety Authority considers that on this topic the ongoing action plans set up in the framework of the last PSRs (2012-2015) or of the LTO for the first units, in addition to the actions already performed in 2017 by the Licensee arising from its self-assessment in the frame of the TPR, are sufficient to achieve a complete ageing management program.

It can be concluded that the overall ageing management program is in line with international standards but that its effective implementation on site can be further improved by ENGIE Electrabel. The Safety Authority considers that the licensee has taken some necessary steps in this direction. Some tools have been recently developed and their effectiveness will be judged on the long term.

Research Reactor BR2

SCK•CEN is required to improve and implement the ageing management program of the BR2 to reach a level that is commensurable with the best international standards and practices for research reactors. The development of a complete ageing management program of the BR2 was therefore part of the Periodic Safety Review that was completed 2016. More specially the Safety Authority reviewed and approved in December 2016 the first stage of the ageing management program which concerned the overall methodology and the scoring of SSCs. Through the action plan resulting from the PSR 2016, SCK•CEN is required to complete the second and third stages (concerning respectively the development of an inspection and maintenance strategy and the development of maintenance and inspection procedures) of the ageing management program before June 2019.

Although the development of the ageing management program is still in progress, some gaps have been identified throughout the TPR assessment. The Safety Authority therefore formulates several recommendations in order to fill these gaps.

Regulatory Oversight

The Belgian regulatory oversight on Ageing Management increased in the last decade with the focus put in the framework of the PSR on the safety factor Ageing and on the LTO for the first units but also for the BR2. The regulatory requirements on ageing management have been clearly defined in the Belgian regulations for nuclear facilities. The strategic notes on LTO by the FANC and the PSR process to be performed accordingly to IAEA standards specifically focuses on the Ageing Management Program, while the Royal Decree of 30 November 2011 on the Safety Requirements for Nuclear Installations specifically requests an up-to-date and complete Ageing Management Program.

In order to verify the correct implementation of the international standards and the Belgian regulation on Ageing Management, the Safety Authority requested for IAEA review missions on Ageing Management at the nuclear power plants and at the BR2 reactors (2015 & 2017) and carried out an inspection campaign on this topic (2016) for all nuclear facilities.

The Safety Authority also decided in 2017 to restructure its overall inspection plan and to integrate in it the follow-up of Ageing Management via periodic inspections on this topic, as an improvement measure.

Although ageing management has been thoroughly followed-up by the Safety Authority in the recent years (review of LTO and PSR projects, regular meetings with the licensee and specific inspections), the Safety Authority is aware of the increasing importance of an adequate ageing management, to ensure further safe operation of the nuclear power plants and research reactors in Belgium. In consequence, the Safety Authority plans to continuously enhance the follow-up on this topic until the final shut-down and decommissioning of the nuclear facilities.

Finally, the Safety Authority acknowledges the added value of this Topical Peer Review allowing the licensee to conduct complete and constructive self-assessments of their ageing management program, and to proactively take measures identified as adequate by the Safety Authority in order to cover the few identified gaps regarding the international good practices.

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Annex 1 - Content

Annex 2 - Abbreviations

Abbr.	Definition
ACI	American Concrete Institute
ACM	Asset Configuration Management
AMP	Ageing Management Program
AMR	Ageing Management Review
ANA	Authorized Nuclear Agency
ARAB	Arbeidsreglement voor Arbeidsbescherming (General rules for safety at work)
AREI	Arbeidsreglement Elektrische Installaties (Work rules for electrical installations)
AS	Ageing Summaries
ASME	American Society of Mechanical Engineers
ASR	Alkali-Silica Reaction
ASTM	American Society for Testing and Materials
BMI	Bottom Mounted Instrumentation
BMV	Bare Metal Visual
BWR	Boiling Water Reactor
CAST	Cable Ageing Study
CCOE	Corporate Committee for Operating Experience
CDC	Cablerie de Charleroi
CEA	Commissariat à l'Energie Atomique et aux Energies Alternatives
CEB	Comité euro-international du béton
CFR	Code of Federal Regulations
CHUG	EPRI's Checworks Users Group
CLB	Current Licensing Base
CM	Corrective Maintenance
CM	Condition Monitoring (Electrical measurements)
CMP	Cable Management Program
CNT	Centrale Nucléaire de Tihange (Tihange Nuclear Power Plant)
CQ2	Contrôle Qualité 2 (Control quality 2)
CT	Compact Tensions
CUF	Cumulative Usage Factor
CUF_{en}	ENvironmentally-adjusted Cumulative Usage Factor
CVP	Circuit de Vapeur Principal (Primary steam circuit)
CW	Cooling Water

DEMEC	Déformètre mécanique (Micrometer dial gauge)
DSC	Differential Scanning Calorimeter
EAN	Eau Alimentaire Normale (Normal feed water)
ECT	Eddy Current Testing
EDY	Equivalent Degradation Years
EDM	Electric Discharge Machined
EI&C	Electrical Instrumentation and Components
EFPY	Equivalent Full Power Years
EM	Environmental Monitoring
ENIQ	European Network for Inspection and Qualification
EPDM-EVA	Ethylene Propylene Diene Rubber – Ethylene Vinyl Acetate
EPR	Ethylene Propylene Rubber
EPRI	Electric Power Research Institute
EQ	Equipment Qualification
EQB	Electrabel Qualification Body
EQf	Equipment Qualification file
FANC	Federal Agency for Nuclear Control
FAC	Flow Accelerated Corrosion
FE	Fire Extinguishing
FE	Fiches d'Expérience (document to record internal operating experience at Tihange NPP)
FIB	Fédération internationale du béton (International Federation for Structural Concrete)
FIP	Fédération internationale de la précontrainte
FIS	Formule d'Irradiation Supérieure
FMECA	Failure Mode Effects and Criticality Analysis
FPS	Federal Public Service
FROG	Framatome Owners Group
GALL	Generic Ageing Lessons Learned
HAZ	Heat-Affected Zone
HV	High voltage
I&C	Instrumentation and Controls
IAEA	International Atomic Energy Agency
ICM	Installation Concept Management
IEEE	Institute of Electrical and Electronic Engineers
IGALL	International Generic Ageing Lessons Learned
ILRT	Integral Leak Rate Test (Type A Test)
INCEFA	INcreasing Safety in NPPs by Covering gaps in Environmental Fatigue

	Assessment
INPO	Institute of Nuclear Power Operations
IPA	Integrated Plant Assessment
IR	Infrared
IRS	Incident Reporting System
ISI	In-Service Inspection
ISIT	Initial Structural Integrity Test
KCD	Kerncentrale Doel (Doel Nuclear Power Plant)
LCO	Limiting Conditions for Operation
LOCA	Loss of Coolant Accident
LTO	Long Term Operation
LTOP	Low Temperature Over-Pressurization
LWR	Light-Water Reactor
MF	Meldingsfiche (document to record internal operating experience at Doel NPP)
MIC	Microbiologically-influenced corrosion
MIS-B	Machine d'Inspection en Service Belge (Belgian in-service inspection device)
MRP	Materials Reliability Program
MSIP	Mechanical Stress Improvement Process
MP	Maintenance Plan
MV	Medium voltage
NAR	National Assessment Report (as a WENRA TPR 2017 requirement)
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NGT	Nuclear Generation Team
NIS	Neutron Flux Instrumentation System
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NS	Nuclear Safety
NSAL	Nuclear Safety Advisory Letter
NSSS	Nuclear Steam Supply System
NUREG	Nuclear Regulatory Group
OE	Operating Experience
OECD	Organization for Economic Co-operation and Development
OEM	Original Equipment Manufacturer
pC	Picocoulombs
PAM	Plant Asset Management

PdM	Predictive Maintenance
PIE	Postulated Initiating Events
PF	Probability of Failure
PM	Preventive maintenance
PSR	Periodic Safety Review
PSP	Plant-Specific Program
PTS	Pressurized Thermal Shock
PVC	Polyvinylchloride
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
PWSCC	Primary Water Stress Corrosion Cracking
RCM	Reliability-Centred Maintenance
REX	Return on Experience
RG	U.S. NRC Regulatory Guide
RGPT	Règlement Général pour la Protection du Travail (General rules for safety at work)
RL	WENRA Reference Level
RM	Room Monitoring
RPV	Reactor Pressure Vessel
RSQ	Rapport Synthétique de Qualification
RSE-M	Règles de Surveillance en Exploitation des Matériels Mécaniques
RT_{NDT}	Reference Temperature for Nil Ductility Transition
RW	Raw water
SALTO	Safety Assessment Long-Term Operation
SAR	Safety Analysis Report of an NPP
SBLOCA	Small Break Loss Of Coolant Accident
SCC	Stress corrosion cracking
SCK•CEN	StudieCentrum voor Kernenergie – Centre d'Etudes de l'énergie Nucléaire
SER	Significant Event Reports
SI	Safety Injection
SGMP	Steam Generator Management Program
SHR	System Health Report
SPR	U.S. NRC Standard Review Plan (NUREG-0800)
SSC	Systems, Structures and Components
TDR	Time Domain Reflectometry
TE	Tractebel Engineering
TEV	Transient Earth Voltage

TLAA	Time-Limited Ageing Analyses
UT	Ultrasonic Testing
UV	Ultraviolet
VT	Visual Testing
VT-1	VT-1 examinations are conducted to detect discontinuities and imperfections on the surfaces of components, including such conditions as cracks, wear, corrosion, or erosion. (ASME XI – IWA-2211)
VT-3	VT-3 examinations are conducted to determine the general mechanical and structural condition of components and their supports by verifying parameters such as clearances, settings, and physical displacements, and to detect discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. VT-3 includes examinations for conditions that could affect operability or functional adequacy of snubbers and constant load and spring type supports. (ASME XI – IWA-2213)
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators Association
WPS	Warm Pre-Stressing
XLPE	Cross-linked Polyethylene

A Appendix 4A – Ageing mechanisms / concealed pipes

General corrosion

General corrosion, also known as uniform corrosion, proceeds at approximately the same rate over a metal surface. Loss of material due to general corrosion is an ageing effect requiring management for low-alloy steel, carbon steel and cast iron in virtually any wetted environment, including outdoor environments [4]. Most components are built with a corrosion allowance that represents expected metal loss over the component life. Cathodic protection or the use of coatings or paints can also be used to address uniform corrosion.

Local corrosion

Pitting corrosion

Pitting corrosion is localized corrosion of a metal surface, confined to a point or small area, which takes the form of cavities called pits [4]. It often affects passive alloys and is influenced by both material and surrounding environment. Pitting can occur at surface deposits (under-deposit corrosion), due to electrical imbalances, or by other initiating mechanisms.

Crevice corrosion

Crevice corrosion is localized corrosion of a metal surface at, or immediately adjacent to, an area that is shielded from full exposure to the environment because of the close proximity of the metal to the surface of another dissimilar material. Crevice corrosion occurs in a wetted or buried environment when a crevice or area of stagnant or low flow exists that allows a corrosive environment to develop in a component. It occurs most frequently in joints and connections, or points of contact between metallic materials and non-metallic materials, such as on gasket surfaces, in lap joints and under bolt heads. Carbon steel, cast iron, low-alloy steels, stainless steel, copper, aluminium and nickel-based alloys are all susceptible to crevice corrosion. Steel can be subject to crevice corrosion in some cases after cladding/lining degradation [4].

Galvanic corrosion

Galvanic corrosion is accelerated corrosion of a metal because of an electrical contact with a more noble metal or non-metallic conductor in a corrosive electrolyte. It is also called bimetallic corrosion, contact corrosion, dissimilar metal corrosion or two-metal corrosion. Galvanic corrosion is an applicable degradation mechanism for steel materials coupled to more noble metals in heat exchangers; galvanic corrosion of copper is of concern when coupled with the nobler stainless steel [4].

Flow accelerated corrosion

Flow accelerated corrosion (FAC) is a corrosion mechanism which results in wall thinning in susceptible materials. An example is carbon steel piping exposed to moving, high temperature, low oxygen water such as PWR primary and secondary water. FAC is the result of dissolution of the surface film of the steel, which is transported away from the site of dissolution by the movement of water [4]. ENGIE Electrabel launched the FAC program to identify the affected circuits and manage the ageing mechanisms.

Microbiologically-influenced corrosion

Microbiologically-influenced corrosion (MIC) is any of the various forms of corrosion influenced by the presence and activities of such micro-organisms as bacteria, fungi and algae, and/or the products produced in their metabolism. Degradation of material that is accelerated due to conditions under a biofilm or microfouling tubercle, for example, anaerobic bacteria that can set up an electrochemical galvanic reaction or inactivate a passive protective film, or acid producing bacteria that might form corrosive metabolites [4].

Intergranular corrosion

Intergranular corrosion is a form of Stress Corrosion Cracking (SCC) in which the cracking occurs along the grain boundaries. It is most common on austenitic stainless steel and nickel-based alloys that have undergone sensitization (formation of chromium carbide precipitate at grain boundaries.) or cold working (when it may initiate as transgranular SCC) [4].

Selective leaching

Selective leaching is also known as de-alloying (e.g. dezincification or graphitic corrosion) and involves selective corrosion of one or more components of an alloy [4].

Stress corrosion cracking

SCC is cracking that requires the presence of a susceptible metal, a corrosive environment and a sufficiently high tensile stress (applied and/or residual). SCC is highly chemically specific in that certain alloys are likely to undergo SCC only when exposed to a small number of chemical environments under certain temperature ranges. SCC includes intergranular SCC, transgranular SCC, primary water SCC and low temperature crack propagation as degradation mechanisms. High-strength bolting materials with yield strength in excess of 1034 MPa exposed to corrosive lubricant such as molybdenum and humidity or water are also susceptible to SCC [4].

Mechanical damage

Creep

Creep, for a metallic material, refers to a time-dependent continuous deformation process under constant stress. It is an elevated temperature process and is not a concern for low-alloy steel below 370°C, for austenitic alloys below 540°C, or for nickel-based alloys below 982°C [4].

Fatigue

Fatigue is a phenomenon leading to fracture under repeated or fluctuating stresses having a maximum value less than the tensile strength of the material. Fatigue fractures are progressive, and grow under the action of the fluctuating stress. Fatigue due to vibratory and cyclic thermal loads is defined as the structural degradation that can occur from repeated stress/strain cycles caused by fluctuating loads (e.g. from vibratory loads) and temperatures, giving rise to thermal loads. After repeated cyclic loading of sufficient magnitude, microstructural damage may accumulate, leading to macroscopic crack initiation at the most vulnerable regions. Subsequent mechanical or thermal cyclic loading may lead to growth of the initiated crack. Vibration may result in high cyclic fatigue for components, as well as in cutting, wear and abrasion if left unabated. Vibration is generally induced by external equipment operation. It may also result from flow resonance or movement of pumps or valves in fluid systems. Crack initiation and growth resistance is governed by factors including stress range, mean stress, loading frequency, surface condition and the presence of deleterious chemical species [4].

Hydrolisis

Cleavage of chemical bonds by the addition of water. This ageing phenomenon affects polymeric material or composite material and is very dependant of the temperature level.

Structural deformation

Differential movement

Settlement of structures may occur due to changes in site conditions (e.g. water table, soil settlement and heaving). The amount of settlement depends on the foundation material [4].

Section distortion

The degradation mechanism of distortion associated with component supports can be caused by time-dependent strain or by gradual elastic and plastic deformation of metal that is under constant stress at a value lower than its normal yield strength [4].

Uprising/crushing

Uprising/crushing is the global deformation of the pipe due to a high level of soil loading or groundwater.

Mechanical loss of material

Cavitation

Cavitation is the formation and instantaneous collapse of tiny voids or cavities within a liquid subjected to rapid and intense pressure changes resulting in pitting. Cavitation caused by severe turbulent flow can potentially lead to cavitation damage [4].

Liquid droplet impingement

Liquid droplet impingement occurs when water droplets strike metal surfaces at high velocity. The phenomenon occurs in pumps, valves, orifices and at elbows and tees in pipelines [6].

Erosion

Erosion is the progressive loss of material from a solid surface due to mechanical interaction between the surface and a moving fluid, a multicomponent fluid or solid particles carried by the fluid. Attributed to cavitation, flashing, droplet impingement or solid particle impingement [4].

Fouling

Fouling

Fouling is an accumulation of deposits on the surface of a component or structure. This term includes accumulation and growth of aquatic organisms on a submerged metal surface or the accumulation of deposits (usually inorganic) on heat exchanger tubing and surfaces. Biofouling, a subset of fouling, can be caused by either macro-organisms (e.g. barnacles, various types of clams and mussels, and others found in fresh and salt water) or micro-organisms (e.g. algae). Fouling can also be categorized as particulate fouling from sediment, silt, dust and corrosion products, or marine biofouling, or macrofouling (e.g. peeled coatings, debris). Fouling in a raw water system can occur on the surfaces of piping, valves and heat exchangers. Fouling can result in a reduction of heat transfer or loss of material [4].

Microbiological fouling

The uncontrolled growth of micro-organisms, especially filamentous and slime-forming bacteria, results in the formation of a complex organic substance referred to as ‘slime’ or ‘biomass’ [8].

Macrofouling

Macrofouling is associated either with an accumulation of mollusks and other invertebrate organism or with an accumulation of inorganic solids such as sediments [8].

Tuberculation

Tuberculation is the development of small mounds of corrosion products on the inside of pipes. These mounds are reddish brown and of various sizes [6].

B Appendix 5A – RPVs of the Belgian NPPs

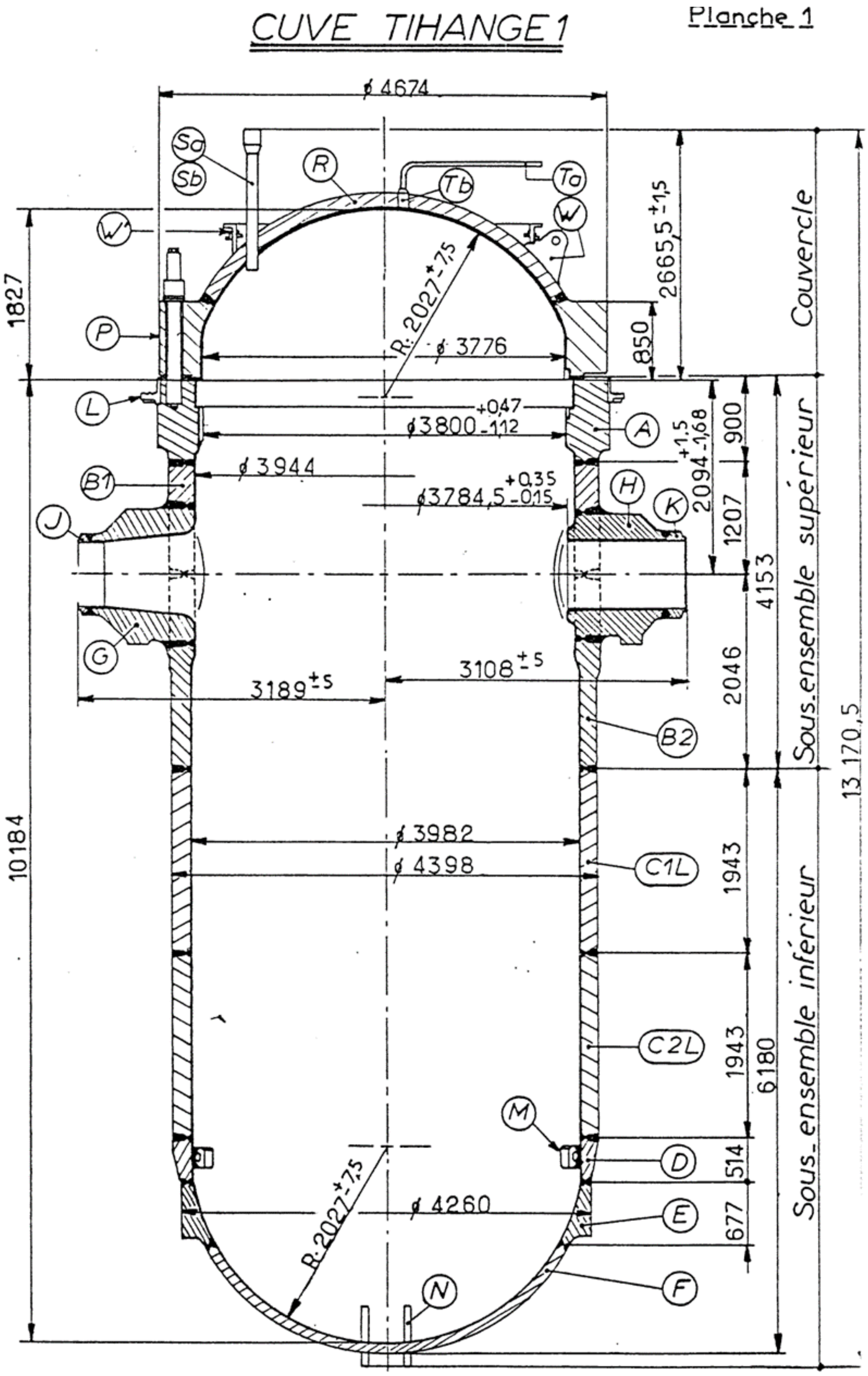

Figure App A5-1: Tihange 1 reactor pressure vessel

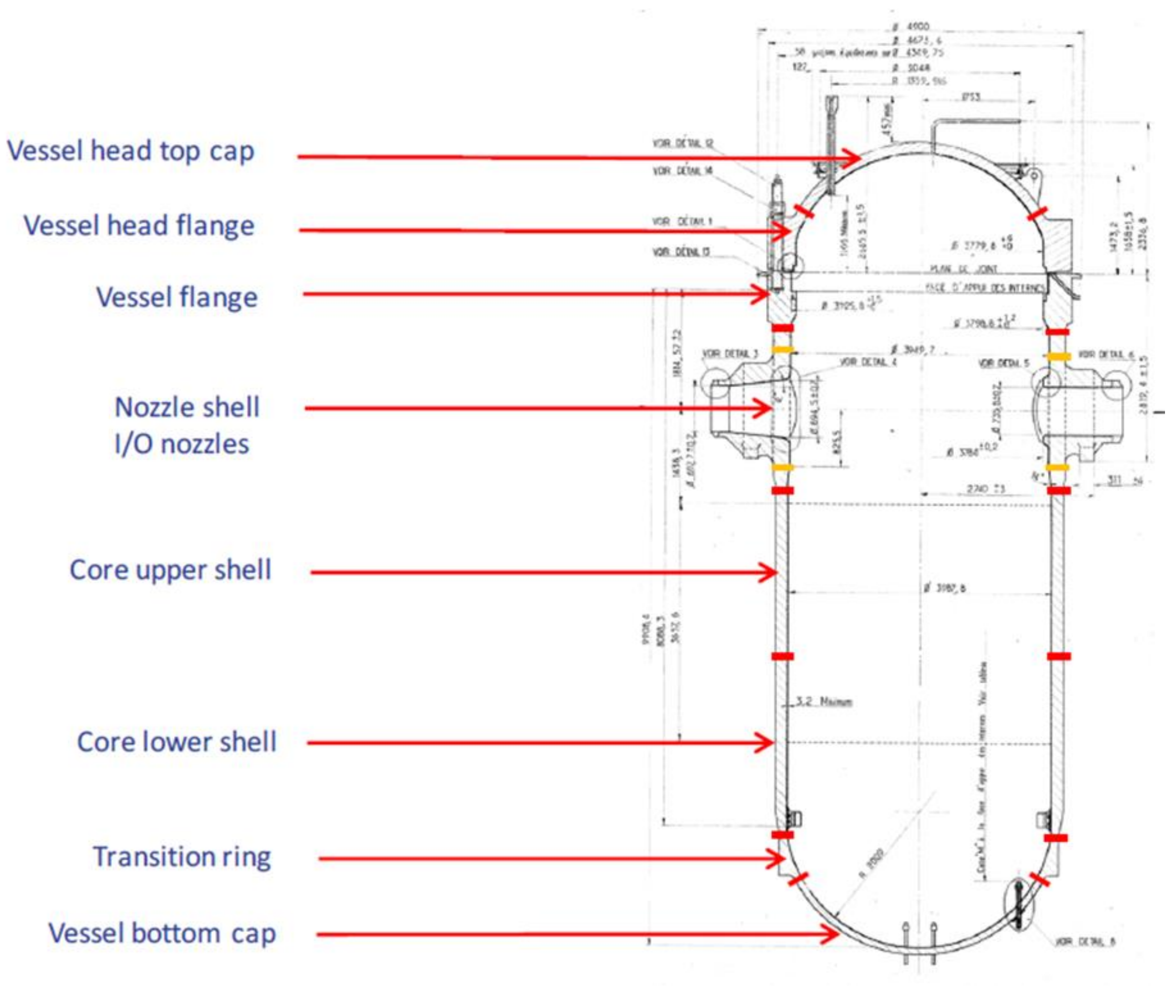

Figure App A5-2: Tihange 2 – Doel 3 reactor pressure vessels

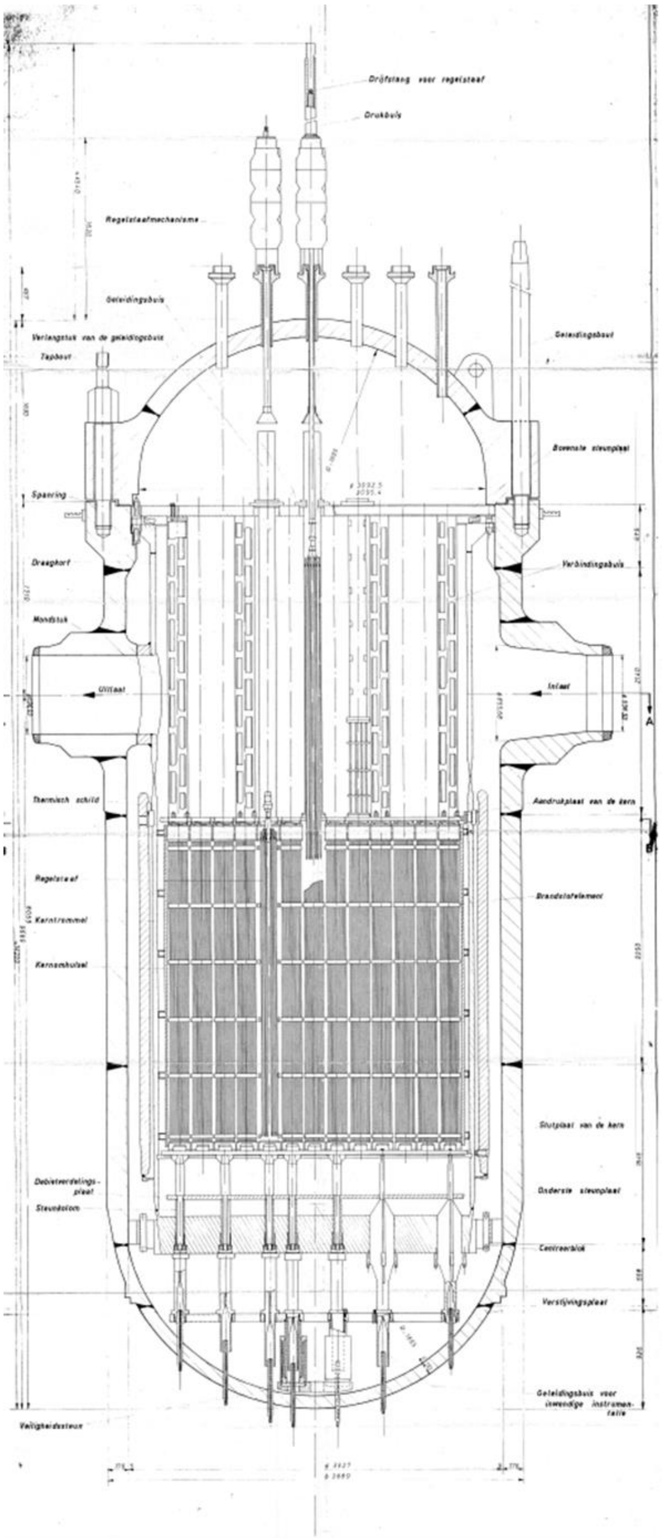

Figure App A5-3: Doel 1-2 reactor pressure vessels

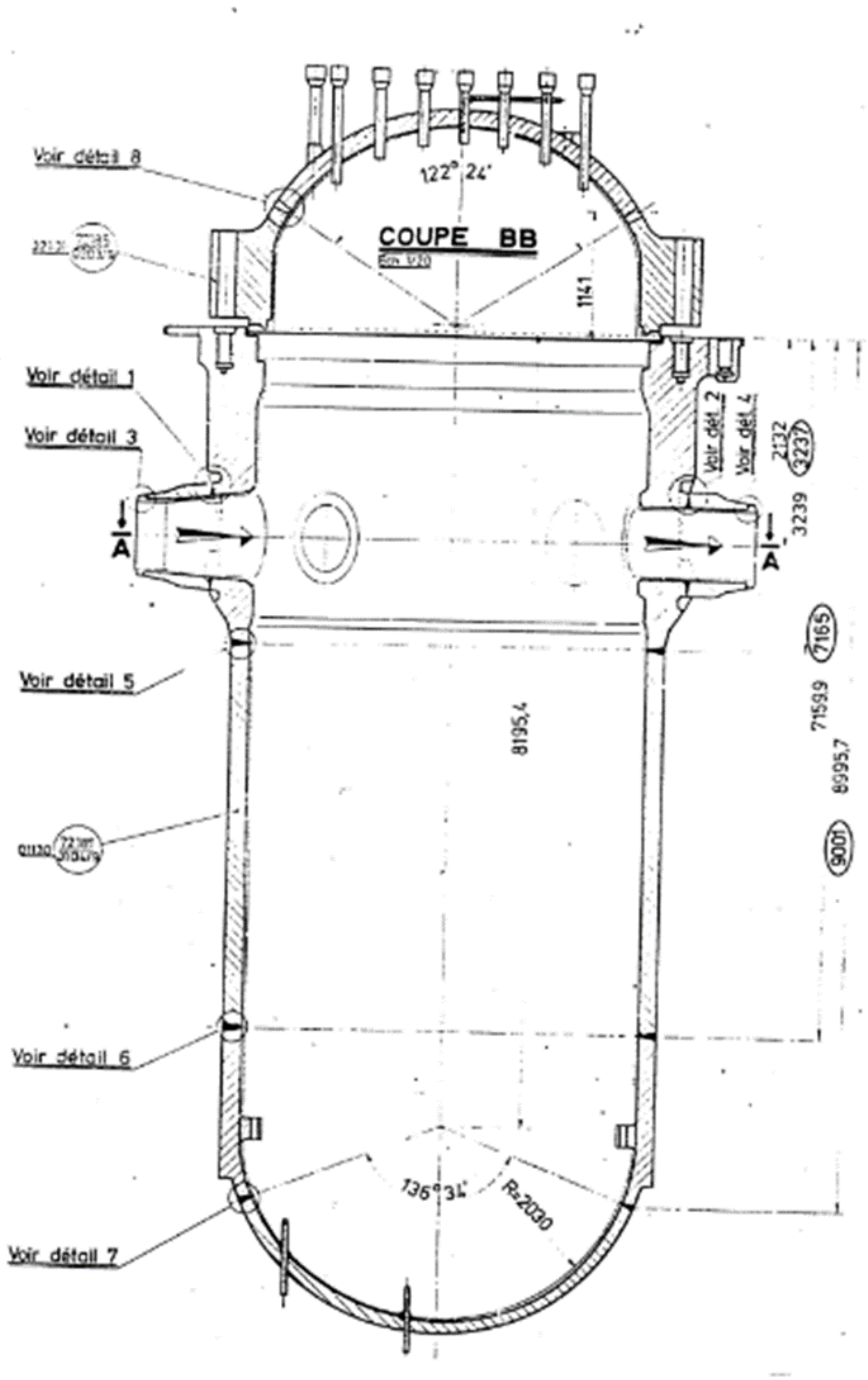

Figure App A5-4: Tihange 3 – Doel 4 reactor pressure vessels