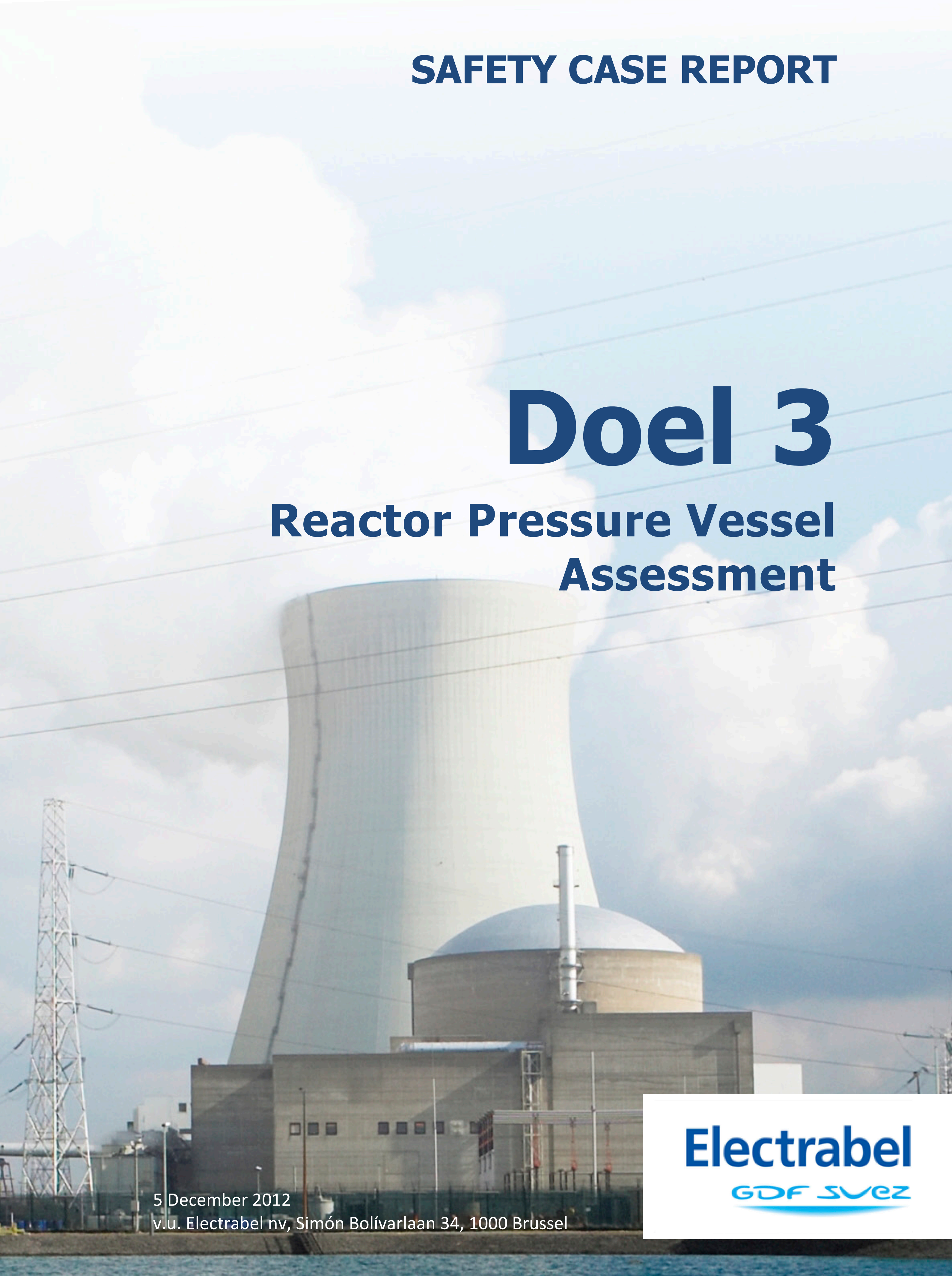

SAFETY CASE REPORT

Doel 3

Reactor Pressure Vessel Assessment

5 December 2012

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Electrabel
GDF SUEZ

Content

1 Summary of the Doel 3 and Tihange 2 Safety Cases

In the Summer of 2012, Electrabel found indications in the reactor pressure vessels of the Doel 3 and Tihange 2 nuclear power units. Guided by its in-house safety culture, Electrabel informed the Authorized Inspection Agency (AIA), Bel V, and the Belgian Federal Authority for Nuclear Control (FANC). It was decided to keep both reactors in cold shutdown, core unloaded, until the issue was resolved.

Subsequently, Electrabel, assisted by external experts, started a thorough investigation of the nature, origin, and possible evolution of the indications and their potential impact on the structural integrity of the reactor vessels.

Having finished this comprehensive investigation, Electrabel is convinced that the structural integrity of the reactor vessels has been demonstrated, allowing for an immediate restart and safe operation of the two reactors. In parallel, operational measures will be implemented and a series of short and mid-term actions carried out.

1.1 Findings

Safety culture triggers additional inspections

At Electrabel, safety is the number one priority. The safety culture is firmly integrated in all day-to-day activities. Operational personnel are trained extensively in safety measures. Furthermoe, periodic in-service inspections of the power generation units are organized systematically. In addition, operating experience is systematically picked up and exchanged with other power plants worldwide.

Based on this international exchange of operating experience, Electrabel agreed with AIA to check the reactor pressure vessels of all its nuclear power units for underclad cracks after such defects were detected at similar nuclear power plants in France. As a result, during the scheduled in-service inspection of Doel 3 in June 2012, additional ultrasonic inspections of the first 30 mm from the inner vessel surface of the vessel shell beltline were executed, in addition to the standard ultrasonic inspection. Doel 3 was the first Belgian unit to undergo this additional inspection.

Immediate action after unexpected indications found

During the additional ultrasonic inspection of the Doel 3 reactor pressure vessel, Electrabel did not find any underclad cracks. However, the inspection revealed a number of other unexpected and inexplicable indications of nearly laminar orientation, located in the core shells. Therefore, Electrabel scheduled further inspections on the entire height and thickness of the reactor pressure vessel wall. This second ultrasonic inspection took place in July 2012 and confirmed the presence of a large number of indications in various parts of the shell.

Based on these findings, Electrabel informed the FANC, AIA, and Bel V. It also decided to investigate the Tihange 2 reactor pressure vessel, as its casting (Krupp) and forging (RDM/RN) history is similar to that of the Doel 3 vessel. The inspections at Tihange 2 were performed during the planned shutdown in August 2012 and showed similar indications, but to a lesser extent. These new findings were also communicated to the FANC, Bel V, and AIA.

First diagnosis: hydrogen flaking, a known metallurgical phenomenon

Electrabel called upon the AREVA metallurgy experts for an initial diagnosis. Based on the findings of the in-service inspections, the manufacturing documentation review, and their vast experience with heavy forging manufacturing, the AREVA experts came to the preliminary conclusion that hydrogen flaking in the macro-segregation zone of the shells was the most likely cause of the indications and that they had appeared during the manufacturing of the reactor shells. Hydrogen flaking is a known metallurgical phenomenon that may occur during the casting and forging process and causes flaws in steel under specific circumstances. This diagnosis was confirmed later, after thorough investigation.

ASME XI definitions

Indication = the response or evidence from the application of a non-destructive examination

An indication is an elementary record or set of records indicating the possible presence of a flaw. The definition of an indication is directly linked to the type of flaw (nature and size) being targeted in a given examination.

Flaw = an imperfection or unintentional discontinuity that is detectable by a non-destructive examination

1.2 Roadmap

Roadmap developed to build a unique safety case

Electrabel developed a roadmap to confirm the first diagnosis and carry out a comprehensive safety assessment of the Doel 3 and Tihange 2 reactor pressure vessels. Creating a roadmap to achieve these goals was extremely challenging.

The subject of the safety case is first-of-a-kind. Therefore, the roadmap had to be developed completely. In addition, it needed to be comprehensive and technically robust. The roadmap also had to achieve the following:

- Confirm the origin and stability of the indications. The first diagnosis had to be evaluated against other possible causes during a thorough root cause analysis.
- Validate the UT inspection technique.
- Obtain the necessary material properties needed for the structural integrity assessment. A material testing program had to be put in place. This program should make use of the appropriate archive materials.
- Develop a methodology for the structural integrity assessment of the vessels that was solid and conservative.

During the development of the safety case roadmap, special attention was given to the appropriate definition of the safety issue, so that it could be addressed and solved adequately and with a high level of confidence. In addition, the safety case roadmap also took into account the expectations and requirements communicated by the FANC

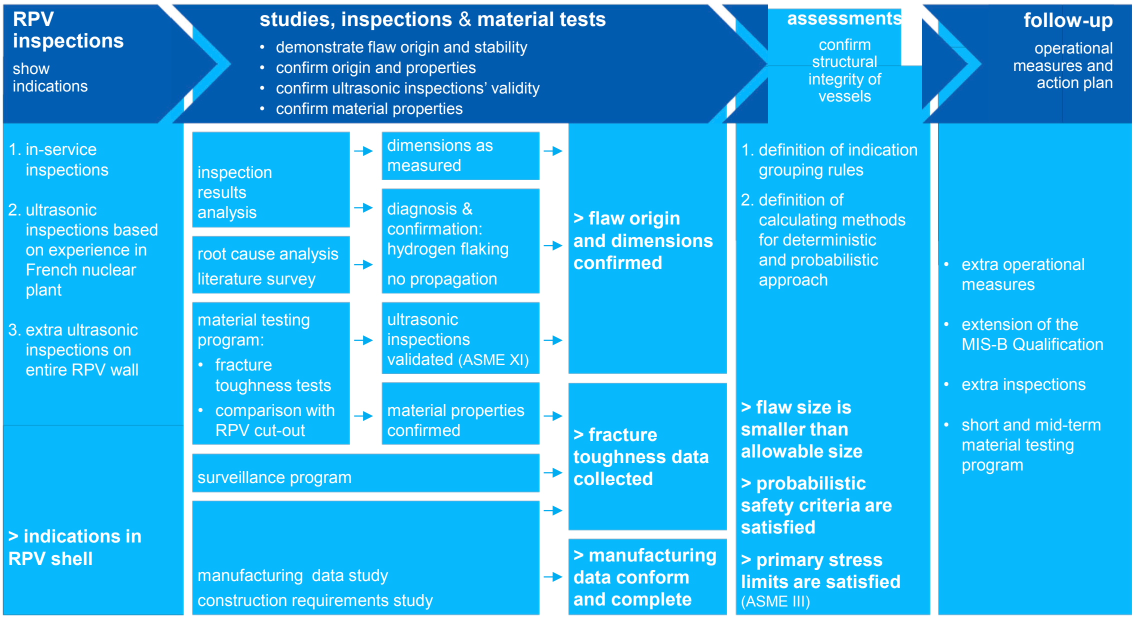

Figure 1.1 The safety case roadmap at a glance

A multidisciplinary team, assisted by experts and academics

The roadmap was executed by a multidisciplinary team of GDF SUEZ experts and external experts from the nuclear industry and academic organizations. The team was placed under the leadership of Electrabel, the nuclear license holder, and included:

- Electrabel, for its nuclear power-related operational and safety expertise
- Laborelec, for its specialized knowledge of non-destructive testing techniques and metallurgy
- Tractebel Engineering, for its know-how on nuclear power plant design and engineering and its expertise in structural integrity, materials, and safety

External specialists and laboratories such as AREVA, SCK.CEN, and Oak Ridge National Laboratory were called upon to develop the safety case and conduct the material testing program.

Additionally, external experts and academics contributed to a robust safety case and reviewed its completeness and soundness (see Table 1.1 below).

Independent analysis and review

The Nuclear Safety Control Department of Electrabel conducted an independent analysis and review. It relied upon a panel of international academics and experts from British universities and the Sandia National Laboratories (US Department of Energy), as well as its own resources on the corporate (Electrabel Corporate Nuclear Safety Department) and on-site (CARE Doel and Tihange) levels.

The roadmap: studies and material tests with positive results

To demonstrate the integrity of the reactor pressure vessel under all conditions, the safety case roadmap specified an extensive phase of documentation research, studies, and material tests. This resulted in the following conclusions:

- **Construction in accordance with international codes and standards**. A close review of all of the original manufacturing data and documentation revealed that both the Doel 3 and Tihange 2 reactor pressure vessels were manufactured in accordance with the prevailing international codes and standards, in particular the ASME Boiler & Pressure Vessel Code. All manufacturing inspections required by the construction code were performed and witnessed by the customer and regulatory body and concluded in the acceptance of all parts of the reactor pressure vessels.

  The manufacturing data and documentation proved to be complete, traceable, and in accordance with international codes and standards.

- **Hydrogen flaking confirmed and stable**. The first diagnosis of hydrogen flaking was evaluated based on:

  - An extensive literature study
  - A root cause analysis of all potential causes
  - An evaluation of the possible flaw formation mechanisms
  - A detailed evaluation report of the AREVA metallurgy experts based on the construction files and the shape and size of the indications

  This report was challenged and completed by external experts. As a result, the first diagnosis was confirmed. It was also concluded that the identified indications were stable.

- **UT inspection technique is valid.** The ultrasonic inspection was performed with the automated MIS-B (Machine d'Inspection en Service Belge) equipment, which has been used for over thirty years to inspect the reactor vessels of all Belgian units. The ultrasonic inspection technique that was used to characterize the indications at Doel 3 and Tihange 2 is state-of-the-art and is used in many nuclear power plants worldwide. It has been qualified for all mandatory inspections and underclad crack detection and sizing against international standards, prior to its use and under the control of the AIA.

  For this specific safety case, the chosen inspection technique was evaluated through cross-checking and extensive destructive tests on a reference block. This block was known to have hydrogen flakes and was taken from an available archive forged shell of equivalent material and size to the reactor vessels' shells.
  The testing program demonstrated that the applied inspection technique is valid and appropriate for characterizing the types of indications found in the Doel 3 and Tihange 2 reactor pressure vessels. Moreover, it was shown that the applied inspection technique tends to oversize the dimensions of the indications, making it more conservative.

- **Affected material is sound and with good properties.** In addition to a material-related literature survey, a comprehensive testing program was launched. Many mechanical and metallurgical tests (on more than 400 test samples) were performed in different laboratories (AREVA, Laborelec, SCK.CEN) on archive materials, including a piece of 1.2m diameter originating from the Doel 3 vessel itself. These tests showed that there is no significant effect of orientation or macro-segregation on fracture toughness. All results confirmed that the curves that the ASME code requires to be used in the assessment are conservative. The destructive tests performed on steel samples with hydrogen flaking also showed that the material between and around the flaws is sound and of a normal metallurgical structure.

- **Structural integrity is confirmed.** After studies and testing, the multidisciplinary team developed detailed methodologies for assessing the structural behaviour of each flaw detected in the vessels shell, in all possible operational modes and transients. These methodologies have been validated after research and were challenged by external experts specialized in fracture mechanics and structural analysis, who confirmed the conservativeness of the methods.

  Based on these methodologies, detailed calculations were made using state-of-the-art modelling and computing techniques, in order to verify the applicable structural integrity requirements. Calculations were performed using conservative data: in particular, very conservative fracture toughness data were used compared to actual material test results. These calculations included the following:

  - Deterministic calculations according to ASME Section III, to assess general stresses in the vessel
  - Deterministic calculations according to ASME Section XI, to demonstrate that the dimensions of every flaw and group of flaws are well below the allowable dimensions, in all operating conditions
  - Probabilistic safety analyses based on the US regulation

  All studies and calculation results have been thoroughly reviewed internally and by external experts and academics. The calculations confirm that the acceptance criteria of the deterministic studies are met with a significant safety margin. The criterion of the probabilistic safety analysis is widely satisfied as well, even under the conservative assumptions.

1.3 General Conclusions

As documented and further demonstrated in this safety case report, all studies and calculations are solid and verify that all safety criteria in the structural integrity assessment of the reactor vessels are met, for each detected flaw, with significant margins.

The applied methodologies include demonstrated conservativeness at each step and margins on the used data. In particular, conservative fracture toughness data were used compared to actual material test results.

All studies and calculations were subject to a rigid review process. They have also been validated by external experts.

Consequently, having thoroughly assessed the roadmap results, Electrabel is convinced that the integrity of the reactor vessels has been demonstrated, allowing for an immediate restart and safe operation of Doel 3 and Tihange 2.

1.4 Action Plan

1.4.1 Additional Operational Measures

The following actions will be implemented before restart, on top of the existing operational measures:

- Electrabel will reduce the authorized heat-up and cool-down gradients during start-up and shut-down operations. This will reduce the thermal and pressure loading on the reactor pressure vessel even more during normal operation.
- Guided by its nuclear safety culture, Electrabel decided to implement a permanent preheating of the safety injection water reservoirs of Doel 3 to 30°C. This measure is not necessary, given the results of the structural integrity assessment; nevertheless, it will add a 20 % margin to the acceptable flaw size close to the vessel's inner surface.
- All operators of the Doel 3 and Tihange 2 units had a refresher training session on the full scope simulator in the last quarter of 2012. An extended briefing will be given to all shift personnel about the start-up and changes in the operational parameters and specifications.

1.4.2 Short and Mid-Term Actions

Future inspection program

At the end of the first cycle, the same inspection of the entire reactor pressure vessel wall thickness will be performed. Therefore, a program for extending the qualification of the MIS-B equipment will be launched under the supervision of the licensee's qualification body and the AIA. The basic objective of the qualification extension is to document the flaw detection and characterization capability on a block of an AREVA shell known to contain hydrogen flakes.

Material testing

The very comprehensive material and metallurgical literature research and testing program, especially on archive materials of the Doel 3 reactor vessel, demonstrated that the material between the flaws is sound and of a normal metallurgical structure. There are no effects of orientation and segregation.

Consequently, the conservative material properties used in the structural analysis are more than adequate to cover any local effect or peculiarity and do not need to be further investigated.

In addition, Electrabel will launch a confirmatory testing program on materials from the block of the AREVA shell that contains hydrogen flakes. This program, still to be finalized with the safety authorities, will encompass two phases:

- **In the short term** (about 4 months), Electrabel, together with SCK.CEN, will perform a test program on a series of small-scale tensile and fracture toughness specimens located in two zones: one in material out of the segregated zone, and the other in material located between flakes (ligaments) in the segregated zone. The objective is to assess the conservativeness of the ligament's mechanical properties that were used in the structural assessment.
- **In the medium term** (about one year), Electrabel will conduct a test of large tensile specimens containing hydrogen flakes in an orientation comparable to the orientation of the indications in the Doel 3 and Tihange 2 reactor pressure vessels. The objectives are:

- To confirm that the flakes in a nearly laminar orientation do not significantly affect the load-bearing capacity of the specimen
- To assess the conservativeness of the structural integrity assessment method
- In order to be representative, the specimens need to be of large dimensions, which requires more time for preparation and execution.

The work of the project team was reviewed by a team of external experts, in order to ensure the completeness and reliability of the project. The following experts were involved:

Name	Organization	Domain of expertise
Dr. Bernard Marini	CEA (France)	Mechanical properties of materials
Prof. George Robert Odette	UCSB Department of Mechanical Engineering (USA)	Mechanical properties of materials
Prof. Emeritus Dr. Gerard Lesoult	Ecole Polytechnique de Lorraine (France)	Phenomenology of defects in materials
Dr. Kurt Maile	MPA Stuttgart / Department Material Testing (Germany)	Phenomenology of defects in materials
Dr. Clarisse Poidevin	CEA (France)	Non-destructive testing
Dr. Russ Booler	Inspection Validation Centre AMEC (UK)	Non-destructive testing
Dr. Russel Cipolla	Aptech Engineering Services, Inc. (USA)	Structural integrity
Prof. Dr. Bob Ainsworth	Manchester University (UK)	Structural integrity
Dr. Greg Wilkowski	Emc² (USA)	Structural integrity
Terry L. Dickson	Oak Ridge National Laboratory (USA)	Probabilistic fracture mechanics
B. Richard Bass	Oak Ridge National Laboratory (USA)	Probabilistic fracture mechanics

Table 1.1 International experts

2 Introduction

During the in-service inspection conducted in June 2012, indications were found inside the shells of the Doel 3 reactor pressure vessel (RPV). In order to fully analyze these indications and their potential safety impact, a strategy was carefully and fully developed. The investigation was independently reviewed by the Electrabel Physical Control Service. In addition, different teams of external experts and academics were involved.

The present document meets all the requirements stipulated by the Belgian Federal Agency for Nuclear Control (FANC). It describes the roadmap that was followed during the safety case investigation and focuses on the conclusions.

The results of the safety case investigation show that the indications that have been found will not impact the integrity of the Doel 3 RPV, no matter what operating mode, transient, or accident condition. This document provides the evidence that RPV safety is guaranteed in all circumstances and that Doel 3 can be safely restarted.

2.1 The Context

The Royal Decree of Authorization requires periodic safety reviews of each Belgian Nuclear Power Plant (NPP). The principal general objectives of these periodic safety reviews are:

- To demonstrate that the unit has—at the very least—the same level of safety as when the licence was granted to operate at full power and when the most recent periodic safety review was completed
- To inspect the condition of the unit, with a special emphasis on ageing and other factors which may affect the safe operation over the next ten years
- To justify the unit's current level of safety, taking into account the most recent safety regulations and practices

The decision was taken at the time of the construction of the Belgian nuclear power plants to follow the US rules, codes, and standards, in particular the ASME code. This code (especially Section XI) stipulates regular inspections of the reactor pressure vessel (RPV).

Ultrasonic examination

During the periodic in-service inspection of June 2012, the RPV's beltline region was subjected to an ultrasonic examination. The intent was to check for the presence of underclad cracking, which had been detected in two French nuclear power plants.

Nearly laminar indications

The ultrasonic tests (UT) at the Doel 3 nuclear unit did not reveal any underclad crackings. However, a large number of nearly laminar indications (see glossary next page) were detected, mainly in the lower and upper shells of the RPV.

In July, a second inspection with UT probes was performed to investigate the entire thickness of the vessel shell. This inspection identified the same type of indications deeper in the material.

The results of these UT examinations were immediately reported to the Federal Agency for Nuclear Control (FANC), Bel V, and the Authorized Inspection Agency (AIA).

Similar inspection at Tihange 2

The investigations at Doel 3 showed that the indications in the RPV shell could be associated with a zone of macro-segregations, originating from the fabrication process. As the Tihange 2 RPV was built by the same manufacturer at the same time and under the same technical specifications, Electrabel decided to conduct similar inspections at Tihange 2. These UT inspections, conducted during the planned August outage, demonstrated similar indications, but to a lesser extent. As was the case for Doel 3, Electrabel also immediately reported the results of the Tihange 2 inspections to the FANC.

Consequences

- Doel 3 and Tihange 2 remain in a cold shutdown, core unloaded, until evidence is given that they can be safely operated
- Electrabel immediately set up a dedicated project team to fully investigate the indications and their potential safety impact
- The FANC sent the requirements to be fulfilled prior to restarting Doel 3 (August 2012), later confirmed to be applicable for Tihange 2 as well
- A strategy and accompanying roadmap were lined out in order to develop the present safety case (see Chapter 4)

Glossary

Indication. The response or evidence from the application of a non-destructive examination. An indication is an elementary record or set of records indicating the possible presence of a flaw. The definition of an indication is directly linked to the type of flaw (nature and size) which is aimed at in a given examination.

Flaw. An imperfection or unintentional discontinuity which is detectable by a non-destructive examination.

Defect. A flaw (imperfection or unintentional discontinuity) of such size, shape, orientation, location, or properties as to be rejectable. A defect is a flaw that is defined as rejectable by a Code or Specification. While all defects are flaws, not all flaws are defects.

Crack. A crack is a given type of flaw which is specifically being studied in the branch of solid mechanics called 'fracture mechanics'. The definition of a crack is given by the fracture mechanics theory (not by ASME Code).

2.2 The Safety Concern

The integrity of the RPV must be maintained at all time and in all circumstances in order to guarantee the pressure boundary and the cooling of the nuclear fuel.

The Safety Concern regarding the indications can be summarized as follows: it needs to be demonstrated that the structural integrity of the RPV shall not be jeopardized during normal operating conditions as well as during incidental and accidental circumstances. In other words, it must be proven that Doel 3 still conforms to its design basis and achieves the same level of safety as stipulated in the licence.

In the case of the nearly laminar indications for the Doel 3 and Tihange 2 RPVs, the major safety concern applies to brittle initiation and propagation of defects in the RPV wall in case of pressure and thermal transients.

2.3 The Safety Case Roadmap

In order to prove the integrity of the RPV, a rigourous and comprehensive Safety Case Roadmap was developed (see Chapter 4). This roadmap consists of the following chapters:

- Safety Framework
- Ultrasonic inspections
- Origin of the indications
- Evolution of the indications
- Material properties
- Deterministic Structural Integrity assessment
- Probabilistic Structural Integrity assessment
- Operational measures
- Short- and mid-term action plan

The Safety Case Roadmap is in line with the expectations issued by the FANC in its letter of 2 August 2012. That letter indicates that the safety evaluation needs to be executed in three steps:

1. Interpretation of identified indications and collection of available information (historical production data, results of the carried out checks, et cetera)
2. Explanation of the indications' origin and evaluation of their possible evolution during operation
3. Demonstration of the RPV's structural integrity

2.4 Organization and External Review

The complexity of the Safety Case was such that the Doel 3 licence holder Electrabel took the decision to install a multidisciplinary project team. This team is composed of experts from:

- Electrabel: the license holder with expertise in nuclear operations and safety
- Laborelec: knowhow in non-destructive testing techniques and material properties
- Tractebel Engineering: specialized knowledge of nuclear engineering and design as well as structural integrity, materials, and safety

External organizations

The project team relied upon the expertise of external organizations and laboratories in building the Safety Case:

Organization	Domain of expertise
AREVA (France)	Material testing, fracture mechanics, and metallurgy
SCK•CEN (Belgium)	Material testing and fracture mechanics
EMC² (USA)	ASME calculation
Reedy Engineering (USA)	ASME code expert
CRM, Centre de Recherches Métallurgiques (Belgium)	Metallurgy

Table 2.1 International experts

The work of the project team was reviewed by a team of external experts, in order to ensure the completeness and reliability of the project. The following experts were involved.

Name	Organization	Domain of expertise
Dr. Bernard Marini	CEA (France)	Mechanical properties of materials
Prof. George Robert Odette	UCSB Department of Mechanical Engineering (USA)	Mechanical properties of materials
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Terry L. Dickson	Oak Ridge National Laboratory (USA)	Probabilistic fracture mechanics
B. Richard Bass	Oak Ridge National Laboratory (USA)	Probabilistic fracture mechanics

Table 2.2

2.5 Independent Analysis and Review

One of the requests formulated by the FANC in its letter of 2 August, 2012 involves an independent analysis and review by the Electrabel Physical Control Service.

They relied upon a panel of external international academics and experts such as Sandia National Laboratories (US Department of Energy) as well as its own resources (both on corporate and site level).

Although it will be issued in a separate report, this independent review is part of the Safety Case File and will be presented to the Belgian Safety Authorities.

3 Description of the Unit

The Doel Nuclear Unit 3, which is at the scope of this Safety Case Report, is a PWR type. Its RPV is constructed in accordance with the ASME Code. All international safety requirements have been integrated and respected in the design basis as well as in the maintenance and operations of the unit.

3.1 Doel Nuclear Power Plant (NPP)

Belgium comprises seven pressurized water reactor (PWR) nuclear power units, spread over two sites, one in the south of the country (Tihange) and the other in the north (Doel).

The Doel site consists of four nuclear units, a water and waste treatment facility and a number of storage buildings, including one building for the dry storage of the spent fuel casks.

	Doel NPP	Type and supplier	Start-up
1	Nuclear Unit 1	PWR Westinghouse	1975
2	Nuclear Unit 2	PWR Westinghouse	1975
3	Nuclear Unit 3	PWR Framatome	1982
4	Nuclear Unit 4	PWR Westinghouse	1985

Table 3.1

Figure 3.1 The Doel Nuclear Power Plant is situated along the Schelde River downstream Antwerp

3.2 Focus on Doel Nuclear Unit 3

The Doel Nuclear Unit 3, which is at the scope of this Safety Case Report, is an electric power generating unit with an initial thermal capacity of 2,785 megawatts (MW). Like all Belgian nuclear power plants, its reactor is of the PWR type. It was designed by Framatome (now AREVA).

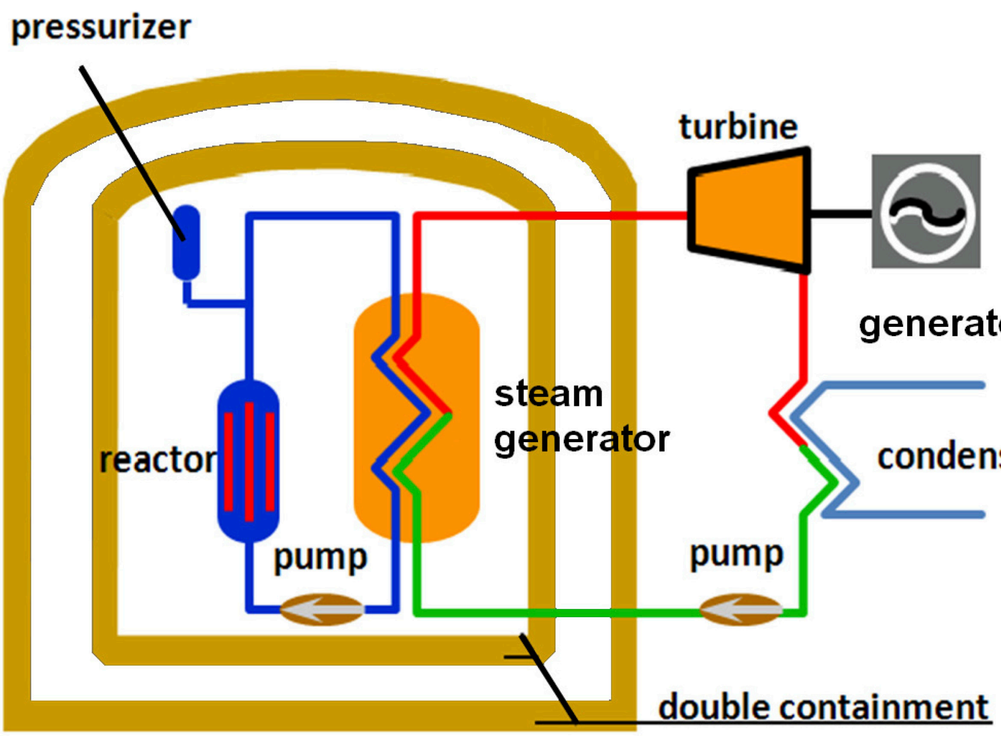


Figure 3.2 Schematic representation of a pressurized water reactor (PWR)

Double containment structure

The nuclear heart of the Doel Nuclear Unit 3 is located inside a double containment structure:

- The first containment, made of prestressed concrete, with a steel liner on the inside for leak-tightness. It is designed to withstand internal accidents such as a loss of coolant accident or a steam line break.
- The second containment, made of reinforced concrete. It aims to protect the installation against external hazards including man-made hazards such as an aircraft crash.

Protection against Accidents

Electrabel operates the Doel Nuclear Unit 3 in accordance with the GDF SUEZ Nuclear Safety Culture and its overall dedication to Nuclear Safety. From the start safety has been the number one priority at the Electrabel Nuclear Units. All international safety requirements have been respected.

In case of an accident, two types of protection systems (first-level and second-level) are available:

Pressurized water reactor (PWR)

The nuclear steam supply system is composed of a reactor vessel and three cooling systems, which form the primary system. Each of these primary loops comprises a centrifugal pump and a steam generator with vertical inverted U-tubes. The pressuriser is installed on one of the three hot legs of the primary loops.

- Safeguard systems (first-level protection) operate in case of an internal accident. They are mainly housed in the nuclear auxiliary building and in the reactor building.
- Emergency systems (second-level protection) are designed to perform the emergency functions aimed at mitigating the consequences of an external accident scenario (for example an aircraft crash, gas cloud explosion, or major fire) affecting the installations all in combination with certain first-level protections. Monitored and controlled from within the Bunker, automatically in the initial phase (i.e. during the first three hours after an event) and manually after this first automatic phase.

Years to remember

The construction of the Doel Nuclear Unit 3 dates back to 1975. In 1982 it was connected to the grid. The unit has undergone three periodic safety reviews up to the present.

1975	Start of construction
1982	First criticality – first connection to the grid
1992	First periodic safety review
1993	Replacement of the steam generators, increasing the capacity to 3,064 MWth, 1,020 MWe
1996	Replacement of the three low pressure turbine rotors, increasing the capacity to 1,075 MWe
2002	Second periodic safety review
2012	Third periodic safety review

Table 3.2

3.2.1 The Primary Circuit

The primary circuit (also called the reactor coolant system) consists of the reactor pressure vessel (RPV), the steam generators, the reactor coolant pumps, a pressurizer, and the connecting piping. The RPV is the place where the nuclear reactions are initiated and the energy is released. This energy is transferred to the secondary circuit by the steam generators through the reactor coolant pumps. The pressurizer controls the pressure in the primary circuit.

The integrity of the primary system has to be maintained under all normal and transient operating conditions. Any major water inventory loss through leaks or even a pipe rupture, will be compensated for by the safeguard systems in order to ensure core cooling.

Nominal operation pressure (at pressurizer)	155 bara
Temperature at core inlet/in the vessel	282.5 °C
Temperature at vessel outlet	324.7 °C
Average temperature of the loops	303.6 °C
Mean power per-unit-length of the core	195.5 W/cm
Flow per loop (thermal design flow)	21,190 m^3/h

Table 3.3 Main operational parameters at 100 % nominal load

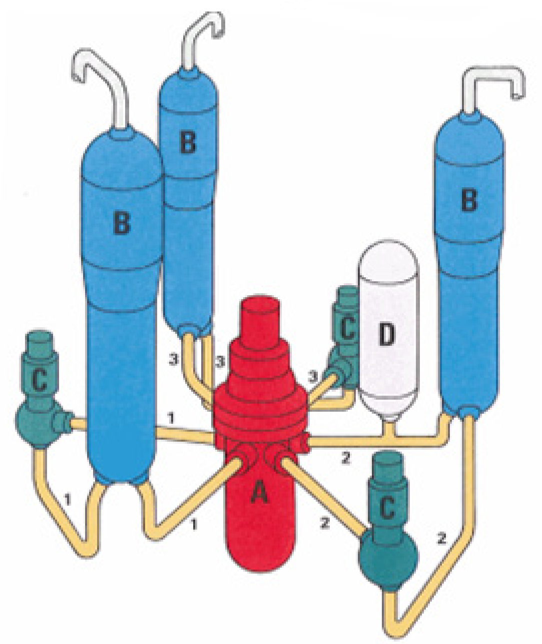


Figure 3.3 Schematic representation of the primary circuit

A = Reactor pressure vessel C = Primary pump

B = Steam generator D = Pressurizer

3.2.2 Reactor

The reactor is an assembly consisting of the following equipment:

- The RPV and the reactor vessel head (pressure boundary)
- The lower reactor internals, a barrel inside the reactor pressure vessel
- The reactor core, consisting of 157 nuclear fuel elements, supported by the lower reactor internals
- The upper reactor internals, on top of the fuel elements, to guide the control rods
- The control rods to control the nuclear chain reaction

Reactor fuel

The fuel is sintered uranium oxide (UO_2) presented in the form of pellets stacked in zirconium-alloy tube sleeves that confine the fission products. The so formed rods are assembled and maintained in a 17 x 17 lattice. The core comprises in all 157 fuel assemblies, each containing 264 fuel rods. The active height of the fuel is 3,657 mm. The fuel cycle is a 12 months cycle.

Reactor reactivity control is achieved through control rods and through the boron present in the form of a boric acid solution in the cooling water.

As a result of the nuclear chain reaction, high radiation fields are present in the area close to the core. It is known that irradiation causes ageing of the RPV material. Therefore, a surveillance program is in force to predict RPV material ageing by periodic testing of material samples that are in place in areas closer to the core, and hence, subject to higher radiation fields than the RPV itself. According to the latest results, ageing of the RPV material will not cause any problems for a life span of 60 years.

The RPV is designed to have the core covered with water under all circumstances, and particularly in the case of a primary circuit pipe rupture. Hence, the integrity of the reactor must be guaranteed at all time, no matter what condition.

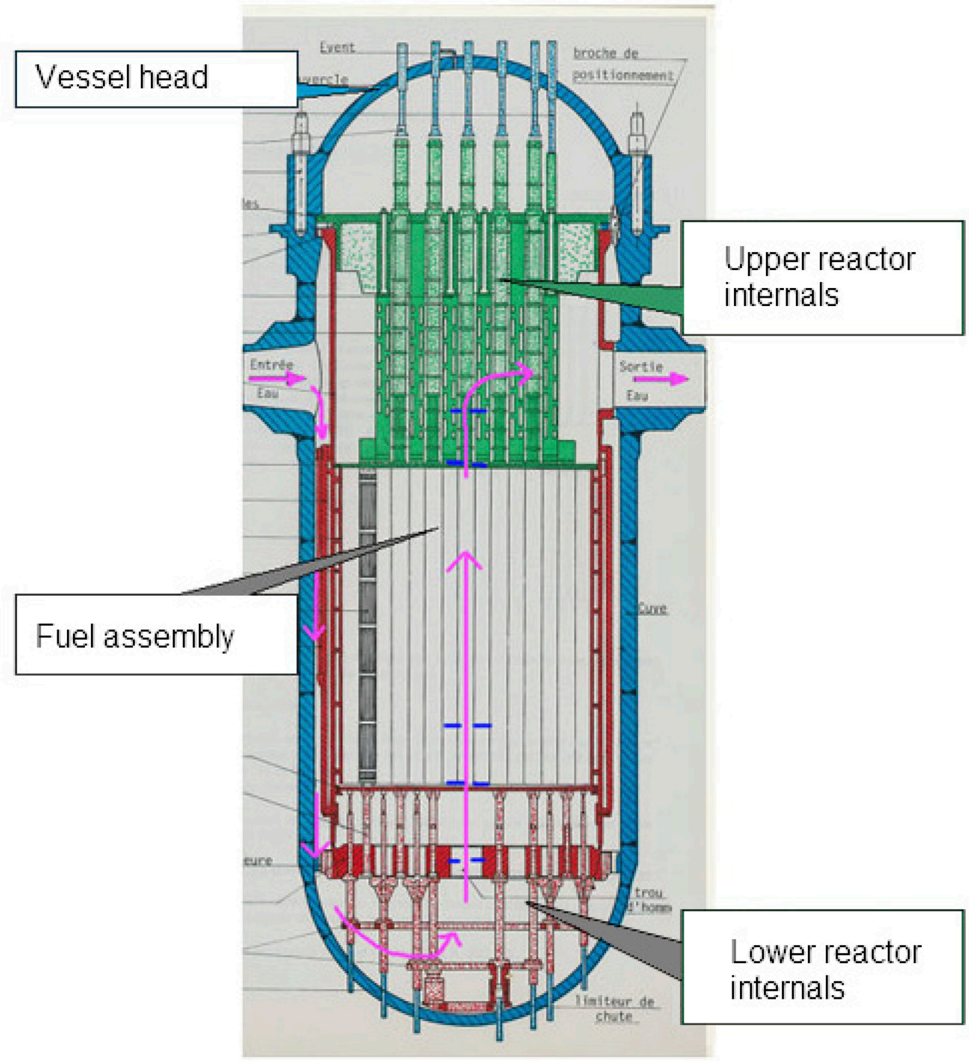


Figure 3.4 Schematic representation of the fuel in the reactor pressure vessel

3.2.3 Reactor Pressure Vessel

The Doel 3 RPV and the vessel head are composed of six forged components and six nozzles in low-alloy steel (SA-508 Class 3). The vessel bottom cap and vessel head top cap consist of two plates in low-alloy steel (SA-533 Grade B Class 1). See drawing below.

The Doel 3 RPV is constructed in accordance with the ASME Code, Section III for Class 1 components. It has the following characteristics:

Total height	13.2 m
Inner diameter	4.0 m
Weight	330 tons
Operating pressure	155 bar
Design pressure	171 bar
Hydraulic test pressure	215 bar
Wall thickness (without cladding)	200 mm
Cladding thickness	7 mm

Table 3.4

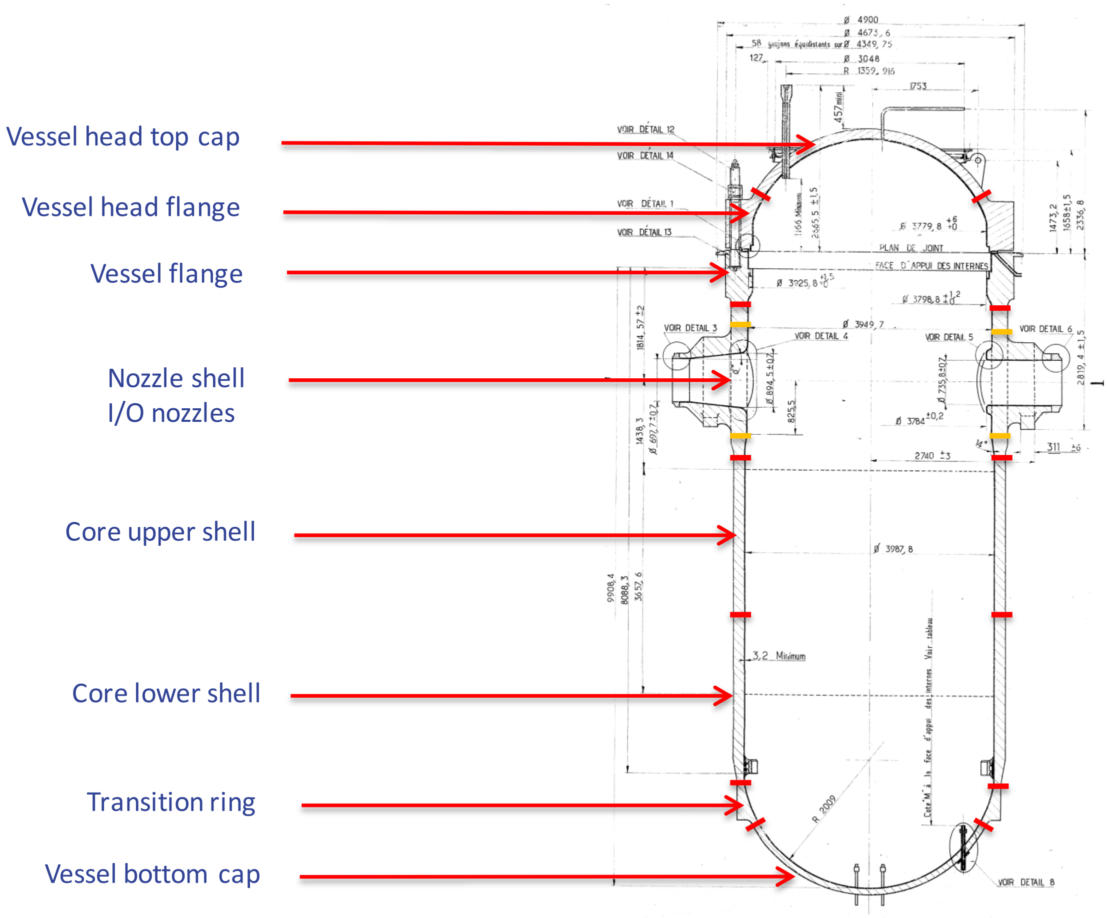


Figure 3.5 Components of the Doel 3 reactor pressure vessel and vessel head

In addition to the reactor surveillance program (see Chapter 3.2.2) the RPV is subject to many controls and examinations:

- Camera inspections
- Ultrasonic testing
- Dye penetrant testing and eddy current inspections for studs and nuts

Ultrasonic testing has been carried out for more than 30 years by INTERCONTROLE, a subsidiary of AREVA, by means of the MIS-B (Machine Inspection en Service – Belge), a robot for RPV inspection.

4 The Safety Case Roadmap

4.1 Description of the Roadmap

After the first diagnosis of hydrogen flaking, Electrabel built a rigorous and comprehensive safety case to confirm the structural integrity of the reactor pressure vessels (RPVs) of Doel 3 and Tihange 2. Therefore, a technically robust roadmap was developed. This roadmap also takes into account the requirements and expectations of the Federal Agency for Nuclear Control (FANC).

4.1.1 Requirements for the Roadmap

The subject of the safety case is a first-of-a-kind, even unique concern. Therefore, developing a safety case roadmap was extremely challenging. It needed to be both comprehensive and technically robust, and meet the following requirements:

- To confirm the origin and stability of indications, the first diagnosis had to be evaluated against other possible causes during a thorough root cause analysis.
- To obtain the necessary material properties needed for the integrity assessment, a material testing program had to be put in place. This program should test the appropriate archive materials.
- The methodology for the structural integrity assessment of the vessels should be solid and conservative.

During the development of the safety case roadmap, special attention was given to an appropriate definition of the safety concern, so it could be addressed and solved adequately and with a high level of confidence. In addition, the safety case roadmap also took into account the expectations and requirements communicated by the FANC.

4.1.2 Roadmap Overview

The following illustration is a high-level overview of the roadmap. More detail about each phase is given in the following paragraphs.

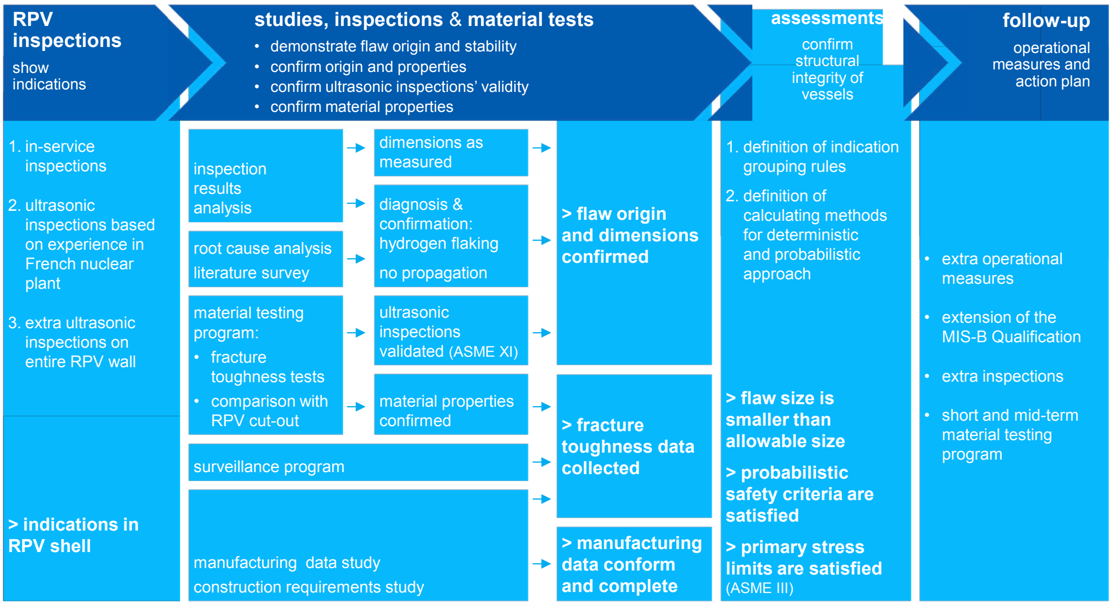
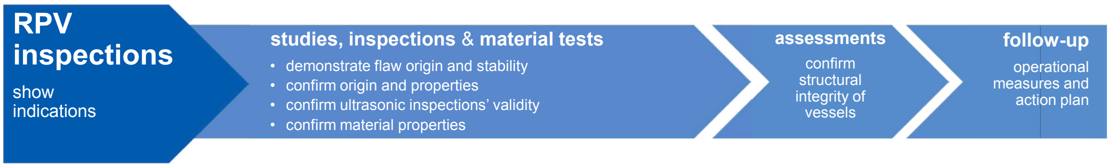


Figure 4.1 The safety case roadmap at a glance

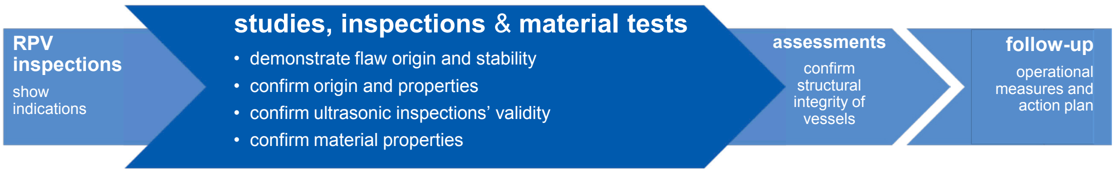
4.1.3 Phase 1: RPV Inspections



The roadmap started with the additional inspection of the Doel 3 RPV shell. Both the data of the initial in-service inspection (June 2012) and the additional inspection (July 2012) were investigated to determine the number, location, and dimensions of the indications.

As the Tihange 2 RPV was built by the same manufacturer, at the same time, and under the same technical specifications, Electrabel conducted similar inspections at Tihange 2, during the scheduled August outage.

4.1.4 Phase 2: Studies, Inspections, and Material Tests



Based on the findings of the in-service ultrasonic inspection, a team of experts, including metallurgy experts from AREVA, diagnosed the indications: the most likely origin was identified as hydrogen flaking. This is a known metallurgical phenomenon that occurs during the production process and causes flaws in the steel. The second phase of the safety case roadmap therefore is comprised of a number of studies, inspections, and tests to evaluate this diagnosis and the validity of the UT inspections.

Confirming the origin of the indications

As the indications were most likely formed during the manufacturing process, Electrabel scheduled a thorough review of all construction files, documentation, manufacturing data and requirements to investigate whether or not:

- The manufacturing conforms with the applicable international codes and standards
- The manufacturing documentation is complete, traceable, and in conformance with the international codes and standards
- The indications were already reported during inspections carried out at the time of fabrication

Confirming the nature and stability of the indications

Hydrogen flaking confirmed?

To confirm the hypothesis of hydrogen flaking, further investigation into this phenomenon was necessary. Literature reviews on hydrogen flakes in steel were held by a team of experts in metallurgy. AREVA made a detailed evaluation report based on the construction files. Also, the outcome of the detailed characterization of flaws in a forged shell with known hydrogen flakes demonstrated that the indications found in the Doel 3 and Tihange 2 RPV core shells are associated with a zone of macro-segregations, and that hydrogen flakes are the likely origin of the indications.

Other potential causes?

This argumentation was completed through a root cause analysis: an examination of the plausibility of other potential causes and flaw formation mechanisms.

From the above studies and analyses, also the potential in-service evolution was addressed. To this end, the following actions were taken:

- A literature review on propagation mechanisms

  Several mechanisms were considered, such as fatigue crack growth, hydrogen-induced growth, irradiation-induced growth and the combination of several of these mechanisms. In relation with hydrogen-induced growth, the possible sources of hydrogen were investigated.

- A comparison of the sizes of the UT indications found in Doel 3 and Tihange 2 with the typical sizes of hydrogen flakes found in fabrication
- A mapping of the indications' distribution versus the neutron fluence at the indications' location (= correlation check)
- An analysis of the fatigue crack growth according to Appendix A of the ASME XI code

Confirming the validity of the UT inspections

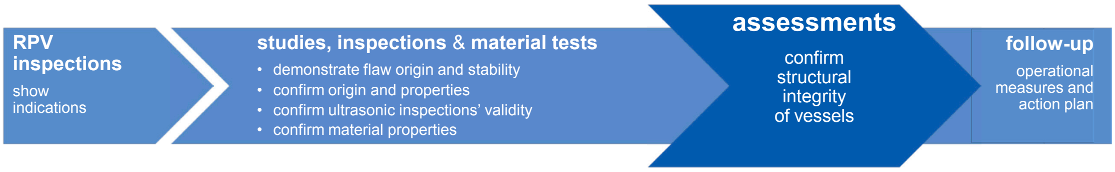
A final but essential element in confirming the origin and dimensions of the found indications, was the validation of the applied ultrasonic inspection techniques in this safety case. This was necessary, because the applied probes and techniques were formally qualified, but not for the type of indications and the type of application in this specific safety case.

The validity of the ultrasonic inspection technique that was used to characterize the indications was evaluated based on destructive tests on a specimen taken out of the AREVA shell and known to contain hydrogen flakes.

Confirming material properties

The material testing program was also executed to determine the material properties, and in particular fracture toughness. The fracture toughness data for the structural integrity assessment are documented in the ASME XI, which contains the usual fracture toughness curves applicable for RPV steels. The orientation of the flaws (nearly laminar) and their location in a zone of macro-segregation could both affect the fracture toughness.

4.1.5 Phase 3: Assessments



The structural integrity assessment was composed of a deterministic and a complementary probabilistic assessment.

Deterministic Structural Integrity Assessment (SIA)

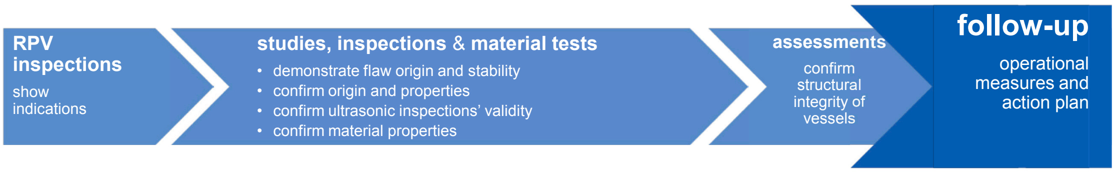
The deterministic SIA is divided into three separate assessments:

- **Flaw Acceptability Assessment**: inspired on the ASME XI procedure but adapted to the nature and number of the indications found in the Doel 3 and Tihange 2 RPVs. For the indications' grouping a particular methodology has been developed. The Flaw Acceptability Assessment consists of three parts:
    - **ASME III Primary Stress Re-evaluation**: the primary stresses are re-evaluated according to the ASME III rules, taking into account the presence of flaw indications in the RPV shell.
    - **Fatigue Crack Growth Analysis**: assessment of potential crack growth due to fatigue, in accordance with the rules of ASME XI Appendix A.
    - **Flaw Acceptability Analysis**: based on the ASME XI criteria but adapted to the nature of the flaws.
- **Fracture Toughness Requirements**: an analysis of the fracture toughness requirements, as stipulated in 10 CFR 50 Appendix G, and performed according to the rules of ASME XI Appendix G.
- **Deterministic PTS Analysis**: a deterministic pressurized thermal shock (PTS) analysis as requested by 10 CFR 50.61.

Complementary Probabilistic SIA

The complementary probabilistic SIA covers a probabilistic PTS analysis according to 10 CFR 50.61a.

4.1.6 Phase 4: Follow-up



If the above analyses meet the corresponding acceptance criteria, the RPV's fitness for continued operation until the end of its service lifetime is confirmed. Electrabel will define the appropriate operational measures and formulate proposals for a future test and inspection program.

4.2 The Safety Framework

The Safety Framework, used for the evaluation of the Doel 3 and Tihange 2 reactor pressure vessel (RPV) Safety Case, is fully in line with Belgian legislation.

Like all Belgian nuclear power plants, the Doel 3 and Tihange 2 Nuclear Units were constructed in accordance with the applicable national and international rules and standards, such as the worldwide accepted ASME Code. The same applies to all the review and inspection programs that have been carried out over the years.

The application of the most severe safety requirements is the number one priority in the GDF SUEZ Nuclear Safety Policy.

4.2.1 In Line with Belgian Legislation

The Safety Framework used for the evaluation of the Doel 3 and Tihange 2 RPV Safety Cases is fully in line with Belgian legislation.

The safety rules and standards which apply to the Doel 3 and Tihange 2 Nuclear Units are defined in the Safety Analysis Report (SAR). That report refers to the US Code of Federal Regulations and is updated at each Periodic Safety Review (PSR).

Specifically for the pressure equipment, the US Code of Federal Regulations is completed by the Belgian Ministerial Decree of Derogation (V.4.087 dated 11 June 1993). This Decree requires:

- The application of the ASME Code (see sidebar) for safety classified pressure equipment in all Belgian nuclear power plants
- The intervention of the Belgian Mandated Organism for pressure equipment, acting as Authorized Inspection Agency (AIA), as defined in the ASME code

ASME (American Society of Mechanical Engineers)

ASME, founded in 1880, develops codes and standards for the industry, in particular for pressure equipment.

In the 1960s, ASME started writing sections specifically dedicated to the nuclear industry. These ASME Sections are regularly updated (based on experience) and are used worldwide, often with a local transposition. The nuclear sections are incorporated by reference in the regulation of different countries (e.g. USA, Belgium).

4.2.2 Conforming to International Rules and Standards

These are the international rules and standards which are the most relevant for the Doel 3 and Tihange 2 RPV Safety Cases:

- ASME Section III
- ASME Section XI
- Supporting ASME Sections
- European Methodology for NDT qualification
- 10CFR50 Appendix G
- 10CFR50.61
- 10CFR50.61a

ASME Section III

ASME, Section III—Rules for Construction of Nuclear Facility Components, in particular its Belgian transposition:

- Contains not only the rules for construction, but also general requirements and restrictions applicable to other sections of the Code
- Covers design, material, fabrication, examination, and testing, but only in the manufacturing stages
- Discusses in particular the responsibilities and duties of the owner, owner's agent and contractors, the quality assurance requirements, and the duties of the AIA

ASME Section XI

ASME, Section XI (1992)—Rules for In-service Inspection of Nuclear Power Plant Components, in particular its Belgian transposition (see sidebar):

- Describes among others the sharing of the activities between the authorities (Mandated Organism acting as AIA and FANC)
- Contains not only the rules for In-Service Inspection (ISI), but also the requirements for Repair and Replacement (RR) activities
- Defines which areas (mostly welds) of an equipment must be examined in service, with which method and with what frequency
- Provides acceptance criteria

> The **ASME Section XI** is applicable only once all of the requirements of the construction code ASME III or any other are met.
>
> As soon as the nuclear power plant has started operation, Section XI is the main entry into the ASME Code. For a new plant or for delivering a replaced item, Section III is the entry key.

Supporting ASME Sections

Both the ASME Section III and Section XI refer to the following supporting sections:

- Section II (Materials): lists all types of base and filler (weld) materials with their properties and associated requirements
- Section V (Non-Destructive Examination): gives the requirements for the examination techniques and procedures (type of examination and acceptance criteria are defined in Section III)
- Section IX (Welding and Brazing): gives the requirements for writing WPS (Welding Procedure Specification) and PQR (Procedure Qualification Record) for qualifying procedures and welders

ENIQ (European Network for Inspection and Qualification)

ENIQ is driven by the nuclear utilities in the European Union and Switzerland and managed by the European Commission's Joint Research Centre (JRC).

ENIQ operates in the field of in-service inspections (ISI) for nuclear power plants using non-destructive testing (NDT), and works mainly in the areas of qualification of NDT systems and risk-informed in-service inspections (RI-ISI).

European Methodology for NDT qualification

The European Methodology for NDT qualification, developed by ENIQ (see sidebar), was used for the UT inspection of the RPV. The methodology serves as an alternative for the ASME requirements for NDT qualification, concerning equipment, procedure and staff.

10CFR50 Appendix G (fracture toughness requirements)

The appendix G of the 10CFR50 standard specifies the fracture toughness requirements for materials of pressure-retaining components of the Reactor Coolant Pressure Boundary (RCPB). As such, it provides adequate safety margins during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the RCPB may be subjected during its service lifetime. Some references are made to the equivalent appendix of ASME XI.

10CFR50.61 (Fracture toughness requirements for protection against pressurized thermal shock events)

The 10CFR50.61 standard specifies the requirements in order to guarantee the RPV integrity in case of pressurized thermal shock (PTS) events. The main requirements address the screening criterion for the expected Reference Temperature for Nil Ductility Transition (RT_{NDT} – the reference temperature for the transifion from brittle to ductile behaviour) at the expiration date of the license. It allows the licensee to perform safety analyses based on probabilistic fracture mechanics, without providing any methodology.

10CFR50.61a (alternate fracture toughness requirements for protection against pressurized thermal shock events)

The 10CFR50.61a standard may be implemented as a probabilistic alternative to the deterministic requirements of 10CFR50.61 for license holders of a PWR that is designed and manufactured according to the ASME Boiler and Pressure Vessel code, 1998 edition or earlier. It allows the licensee to perform, when necessary, safety analyses based on probabilistic fracture mechanics in order to demonstrate that the probability of RPV failure as a result of postulated PTS (Pressure Thermal Shock) events is acceptable.

4.3 Ultrasonic Inspection

During the 2012 outage of Doel Nuclear Unit 3, specific ultrasonic in-service inspections were performed to check for underclad cracks in the reactor pressure vessel (RPV). No underclad cracks were found, however, a number of unexpected and unexplained nearly laminar indications were detected, mainly in the lower and upper core shells. Through a second ultrasonic inspection of the entire thickness of the RPV was then conducted. This subsequent examination confirmed the presence of a large number of nearly laminar indications.

4.3.1 In-Service Inspection (ISI)

The ASME Code Section XI, Edition 1992—including its acceptance criteria—is fully applicable to the regular in-service inspection of all Belgian units. The current in-service inspection program starts from this ASME framework, but in addition, national and international operating experience (OE) is constantly being integrated in order to continuously improve the inspection program. This implies that the inspection planning and methods are re-evaluated at regular intervals.

The in-service inspection program of a ten-year interval covers the volumetric examination of all welds, studs, and bolts as well as the visual inspection of the inner surface of the vessel. It must be completed by the end of the interval. The inspection procedures are qualified by the licensee's qualification body (EQB). The work is supervised by the AIA.

Equipment and techniques

Since the 1982 Pre-Service Inspection of the Doel 3 RPV, regulatory ultrasonic examinations have been carried out by INTERCONTROLE (France) using the automated MIS-B (Machine d'Inspection en Service Belge) equipment (see figure and sidebar).

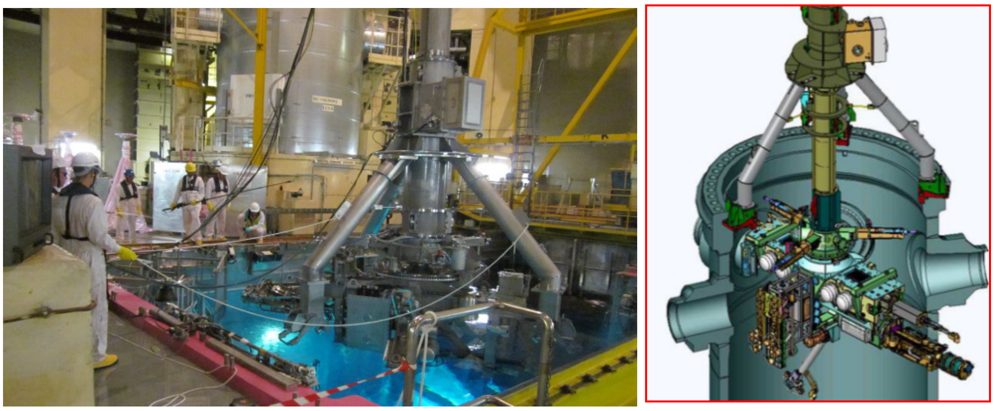


Figure 4.2 The automated MIS-B equipment (left), its operational configuration…

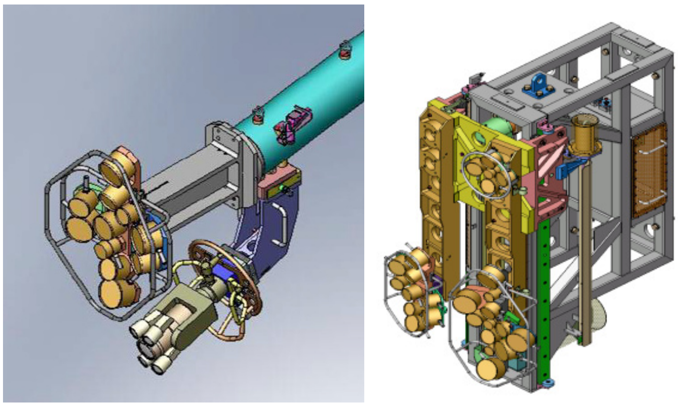


Figure 4.3 … and its specific tools

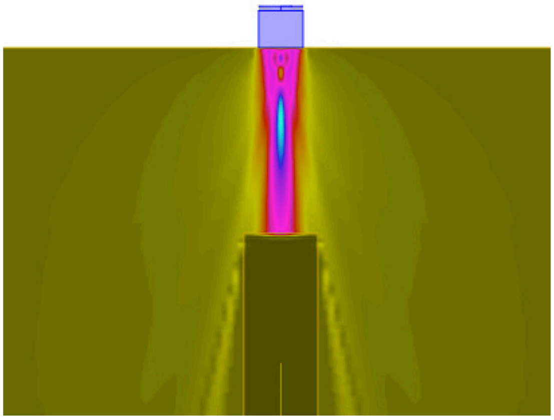


Figure 4.4 Detection and sizing of the flaws was performed using 0° transducers

The **MIS-B equipment** uses the acoustic beam focusing technique, concentrating sound energy in a restricted volume. For this purpose the RPV wall is divided into three depth ranges, to be insonified by different ultrasonic transducers. One 0° probe and four 45° shear wave transducers (orthogonally oriented) are dedicated to each depth range.

All transducers are immersed within the RPV and are actuated in pulse-echo mode without contact with the examination surface.

The acoustic beam focusing technique generates narrow acoustic beams, which require, in order to assure full material coverage, small scanning steps and increments between emissions of acoustic pulses. Scanning steps of 2 mm and increments of 2 or 5 mm (depending on flaw indication density) have been used.

Indication dimensions were measured by the standard 6 dB amplitude drop technique.

Manual ultrasonic inspection—relying essentially on the same 0° contact probe as that prescribed for manufacturing inspections (see Chapter 4.3.2)—were carried out on the vessel head flange. However, the equipment settings and scanning modalities, clearly made the inspection clearly more sensitive than the inspections that were executed at the end of manufacturing.

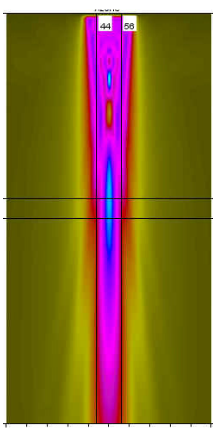


Figure 4.5 Manual UT inspection was performed by means of the 0° contact probe

4.3.2 The 2012 Inspection Results

Taking into account the operating experience with underclad cracks in French nuclear power units, a voluntary additional inspection was qualified to be applied in all units during the ten-year interval inspection. It was applied for the first time in Belgium on the Doel 3 RPV in 2012. This inspection covered the first 30 mm in depth of the core shells. No underclad defects were found, but numerous nearly laminar indications were detected mainly in the lower and upper core shells. A second inspection performed with UT probes able to examine the entire thickness of the vessel, identified the occurrence of the same type of indications deeper in the material.

The detection and sizing of the nearly laminar flaws was performed using 0° transducers (three in the RPV shells, two in the transition ring, one in the flange) in order to gain the ful benefit from specular reflection on the flaw surface. It was verified that the angle beam transducers available on the MIS-B equipment do not offer valuable additional information on the flaws on large zones of the RPV.

The ultrasonic inspections revealed indications with the following characteristics:

- Nearly laminar orientation
- Round shaped
- Typical size of 10 mm
- Large number in the lower core shell
- In areas starting at the inner wall and extending through half of the wall
- A grouping pattern that could be associated with the typical thermo-mechanical history of the shell
- No resemblance to other planar flaws

The number of indications observed in each of the Doel 3 RPV components:

Vessel head flange	3
Vessel flange	2
Nozzle shell	11
Upper core shell	857
Lower core shell	7,205
Transition ring	71

Table 4.1

The figure below displays a typical example of data recorded in the lower core shell. Left: an axial section, with indications appearing as colour spots. Right: the indications, all detected in a 20° sector of the shell, are cumulated on the figure plane. They appear to be nearly laminar in nature and form a cluster sinking with increasing altitude, from the inner surface up to a depth of approximately 120 mm.

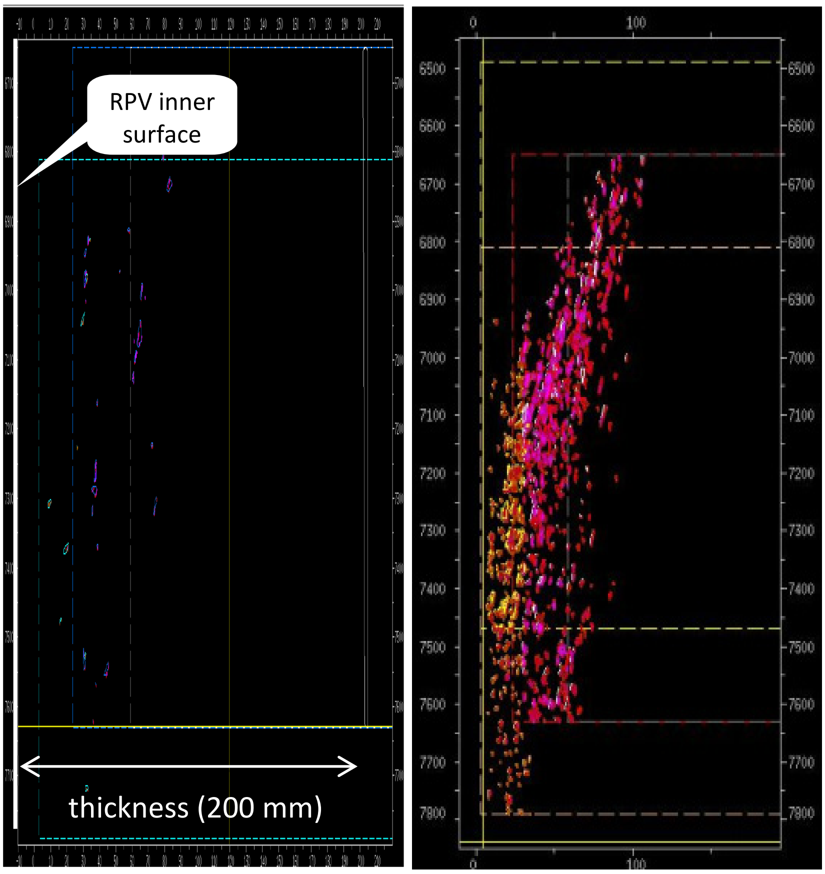


Figure 4.6 Indications recorded in the lower core shell of the Doel 3 RPV

The figure below shows the distribution of indications as a function of depth (measured from the inner surface).

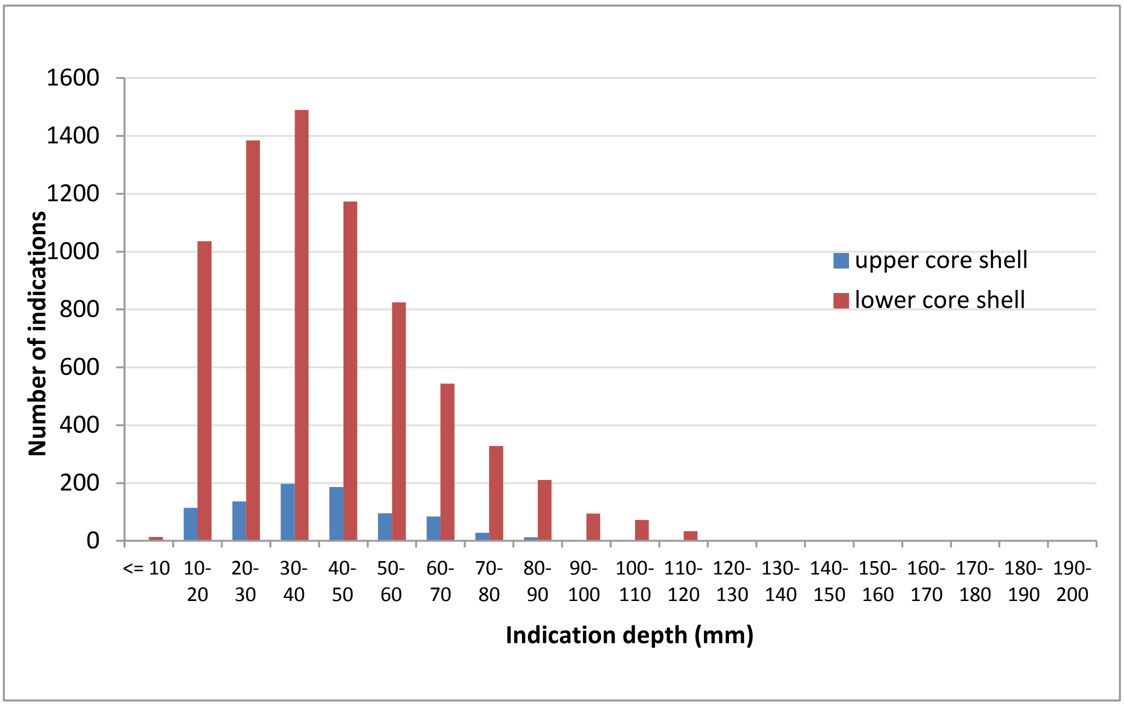


Figure 4.7 Distribution of indications as a function of depth

Only a few indications were detected in the transition ring. They are laminar as well but do not form a cluster.

The nozzle shell and the reactor vessel head flange can be regarded as unaffected.

The results of the UT inspections give the location, the sizing along an X and Y-axis and the depth for each indication. This constitutes the input for the structural integrity assessments (see Chapter 4.8).

4.3.3 Investigations

All required manufacturing examinations were duly performed and documented. All RPV components were accepted in compliance with the applicable regulations. No lack of adequate skill or workmanship was in evidence in the construction files.

Inspection and reporting during manufacturing
ASME III (see sidebar) prescribes that the entire manufacturing process should be subjected to a non-destructive testing (NDT) program. As such, NDT was performed on the Doel 3 RPV components in two ways:

ASME III

The reporting and acceptance criteria—mainly defined by ASME III—are detailed in the manufacturer's examination documents. The documents, validated by all parties, rely mainly on the fact that the ultrasonic signal reflected by the backwall surface of a sound component, decreases where a reflector disrupts UT propagation. Taking into account the acceptance criteria of ASME III NB 2542 edition 74, a defect is unacceptable if it leads to a complete loss of backwall echo.

The ultrasonic response coming from the component backwall, considered as an infinite reflector, remains predominant compared to the response obtained on a reflector of limited size (the flaw). This fact is disadvantageous for the rejection of small dimension indications.

- By magnetic testing (surface examination)
- By ultrasonic testing (volumetric examination)

Magnetic testing is intrinsically unable to detect the type of flaws as found in the Doel 3 RPV, due to their depth and orientations.

UT angle beam and straight beam transducers were used for the inspection. The most appropriate technique to reveal the presence of laminar flaws, is straight beam ultrasonic testing. The component was inspected manually from its outer surface with the straight beam transducer.

Both magnetic and ultrasonic inspection reports conclude that the components are acceptable.

- One document reports an ultrasonic inspection of the Doel 3 upper core shell, conducted internally by RDM/RN. A 6 m long area is declared to contain acceptable indications, mainly in the inner half of the shell's thickness.
- Nine indications were reported by successive inspections on the Doel 3 transition ring. This inspection report establishes the acceptability of the component.
- All other Doel 3 RPV forged parts, as well as the vessel head flanges, were inspected during manufacturing. No flaws were detected.

Comparison between manufacturing and in-service inspection

The identification of the indications, triggered investigations seeking to understand the discrepancy between past and current outcomes.

It can be observed that the 2012 in-service inspection was undoubtedly much more sensitive than the manufacturing inspection in 1975.

Ultrasonic capability assessment

The acoustic beam focusing technique that was used during the 2012 in-service inspections generates finer beams than standard contact transducers, thus enhancing examination sensitivity and lateral resolution. Consequently, focused beam transducers provide a more detailed image of the material's condition and detect smaller flaws than conventional pulse-echo ultrasonic techniques.

In addition, the most important difference between manufacturing and in-service inspection techniques as applied on Doel 3, comes from the very different sensitivity levels applied for these inspections. For the manufacturing inspection, the sensitivity level is based on the UT response of an infinite reflector (the backwall) while for in-service inspection, the sensitivity level is based on the UT response of a side drilled hole with a 2 mm diameter.

The 6 dB (50 %) amplitude drop technique is a quite common flaw sizing method, particularly effective in case of planar flaws perpendicular to the sound propagation direction. Straight beam (0°) transducers are therefore an ideal tool to measure the dimensions of the observed laminar flaw indications. Limitations of the 6 dB drop technique notoriously appear on flaws, that cannot be sized when they are smaller than the UT beam and are then given a conventional size equal to the beam dimension. The enhanced lateral resolution offered by beam focusing leads to such oversizing on smaller flaws than with conventional transducers.

Therefore, some generic oversizing is incorporated in the inspection results, particularly on small flaws, since it is a common practice in RPV inspection to attribute the dimension of the sound beam to any very small flaw detected.

Correlation between past and new indications

Both the manufacturing and in-service inspections reported indications in the Doel 3 transition ring. Most of these indications could be correlated in terms of position.

Legitimacy of component acceptance during manufacturing

Even though the capabilities of the techniques used during the in-service inspection surpass the capability of the manufacturing inspection, there is still question of why these indications were not highlighted at the time.

Independent expertise was solicited to gain insight in the plausibility of not reporting the flaws and not rejecting the components at the time of manufacturing. Its conclusion acknowledged that the Doel 3 components were duly accepted.

Computerized modelling of ultrasonic propagation investigated the response of the 0° transducer unit used for fabrication examinations. The study concluded that the RPV components were correctly accepted, during and at the end of manufacturing, in compliance with the procedures and criteria applicable at that time.

AREVA conducted experimental measurements on a component known to contain hydrogen flakes, to compare the reporting and rejection outcomes of the examination procedure used during the Doel 3 RPV manufacturing and of their own procedure applied at their shop. The comparison showed that the application of the former one shows good performance in terms of flaw detection and reporting but leads to the acceptance of flaws of very large sizes.

All of these investigations agree that the acceptance of the various elements of the Doel 3 RPV was in accordance with the applicable procedures and regulations.

4.3.4 UT validation

Flaw Type Diagnosis

Diagnoses of the flaw nature and origin cannot be inferred based solely on ultrasonic testing data. Examination data can often provide helpful insights. In relation to the identified nature of the indications, it must be pointed out that hydrogen flakes are known, from the metallurgical experience and from literature survey, to concentrate in segregated zones of large forgings. This fits very well with the spatial flaw distribution detected in the Doel 3 RPV shells.

The AREVA experience also reports on typical flaw orientation and size that are fully consistent with the outcome of the 2012 in-service inspection of the Doel 3 RPV. Ultrasonic testing data can clearly be seen as an additional argument to point out hydrogen flaking as the cause of the observed material condition.

Ultrasonic Validation on AREVA Specimen

The UT validation took place on a specimen (VB395/1) known to contain hydrogen flakes and taken out of the AREVA reference shell (VB395). The hydrogen flakes of the specimen generate ultrasonic responses that are very similar to those recorded from the Doel 3 RPV. Some degree of conservatism arises from the slight generic tendency of ultrasonic testing to overestimate flaw dimensions and undersize sound material ligaments between neighbouring flaws. No sign of poor detection capability of the ultrasonic examination technique was observed. Using angle beams in addition to straight beam examination does not enhance the flaw detection or sizing capability on laminar or nearly laminar indications.

The availability of a reference forged shell (VB395) known to contain numerous hydrogen flakes, offered the opportunity to conduct experimental ultrasonic tests in order to document:

- The typical ultrasonic response of the flaws detected in the Doel 3 RPV
- The flaw detection and sizing capability of the ultrasonic testing equipment
- The presence of a sound ligament between two adjacent flaws

The decision was taken to conduct a validation based on the capability of a UT phased array transducer (see sidebar) to create a UT beams of given sizes, the target beams dimensions being those of the MIS transducers.

Test conditions

Based on exploratory ultrasonic testing of the VB395 reference shell, a part containing a high number of hydrogen flakes was selected. From that part, a 500 x 500 mm sample was cut out (block VB395/1) and machined to a 200 mm thickness, similar to the thickness of the RPV walls. The specimen, however, does not have a stainless steel clad layer on the inner surface.

Ultrasonic results

The 0° beams designed to represent the MIS-B transducers detected 372 indications in specimen VB395/1. The generated images are quite similar to the records obtained for Doel 3, although the indications appear to be slightly larger. The distribution depth does not exactly match those observed in the Doel 3 unit, this being due to the choice of the specimen location within the initial AREVA shell.

Assessing ultrasonic capability via destructive examination

The capability of the ultrasonic inspection was validated based on a destructive examination of 18 flaw indications of the specimen VB395/1. These indications were selected as representing a wide variety of flaws in terms of size, location and slope, or as relevant examples of neighbouring indications.

Regarding the ultrasonic flaw detection capability, destructive testing did not show any material discontinuity that was not reported by the ultrasonic inspection.

Comparison of the actual flaw dimensions and those measured by the ultrasonic straight beam test showed on average a slight oversizing, and no occurrence of significant undersizing. Such results are commonly seen as being of a high standard. It was also observed on three pairs of neighbouring indications that the ultrasonic evaluation of the sound material ligament between them is slightly underestimated, i.e. conservative.

Finally, the ultrasonic measurements carried out with refracted and tilted beams spreading from -20° to 20° along two orthogonal axes concluded that, even on the most inclined hydrogen flakes, no flaw is missed and the flaw sizing quality is not degraded when only straight beam examination is implemented.

Phased array tool

The strict timing prevented the use of the MIS-B equipment on specimen VB395/1. Consequently, the ultrasonic transducers used in typical RPV inspections have been re-created with the help of phased array tool. This technology allows the electronic shaping of the UT beam.

Each beam is determined by a series of acoustical parameters, called focal laws. Defining different groups of laws enabled the creation of different UT beams, focussed at different depths in the material, just as they are on the MIS through the focused transducers.

The work plan incorporated, in addition to the reproduction of MIS-B transducers, varying the tilt and skew angles of the sound beam in order to analyze the possible impact of flaw misorientation on the inspection capability.

A 2 MHz matrix array probe was selected to perform the UT validation. Approximately 700 flaws, determined with the help of the CIVA modelling software, were needed to carry out the tests described here below. All ultrasonic measurements were mechanized and all transducer positions and ultrasonic signals were digitized and recorded for off-line data analysis.

The flaw indication reporting threshold was selected so as to reproduce the field settings of the MIS equipment, and the indication sizing method was the -6 dB amplitude drop, as applied in Doel 3.

Figure 4.8 AREVA VB395 shell (left) and VB395/1 sample

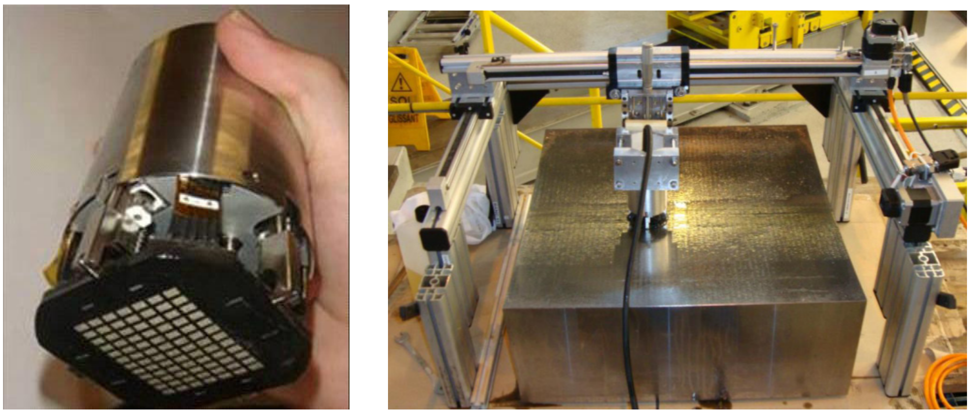


Figure 4.9 UT PA transducer (left) and UT inspection system

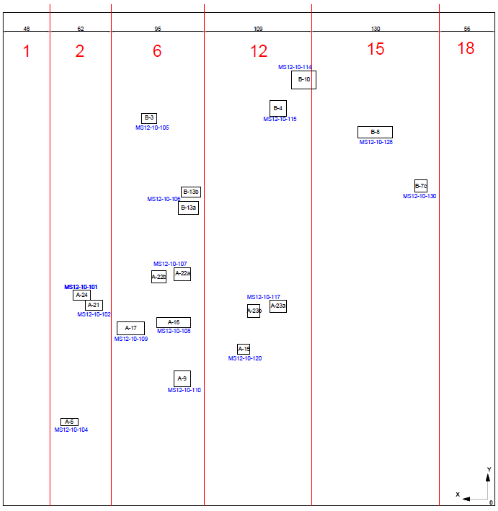


Figure 4.10 Location of the cut indications within the VB395/1 block

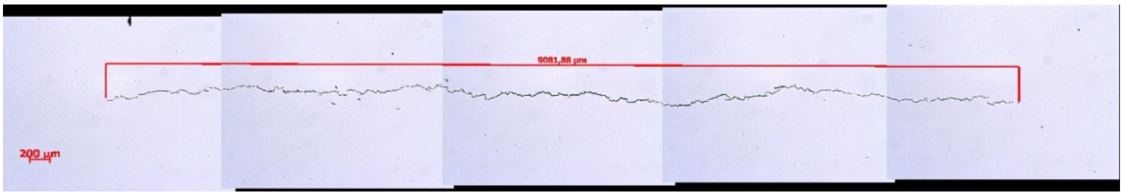


Figure 4.11 Profile of a cut indication in the VB395/1 specimen (9,081.88 µm)

4.4 Manufacturing History

All of the required reactor pressure vessel (RPV) manufacturing documentation was available at Electrabel. A thorough analysis revealed that the construction was executed in conformance with the requirements specified in the international codes and standards. In addition, the documentation proved to be complete, traceable, and in accordance with the international codes and standards.

All information was gathered regarding the manufacture of the forged components of the Doel 3 reactor pressure vessel and its vessel head. The vessel and its vessel head consist of six forged components (see figure below), six nozzles in SA-508 Class 3 low-alloy steel and two plates in SA-533 Grade B Class 1 low-alloy steel for the vessel bottom cap and vessel head top cap.

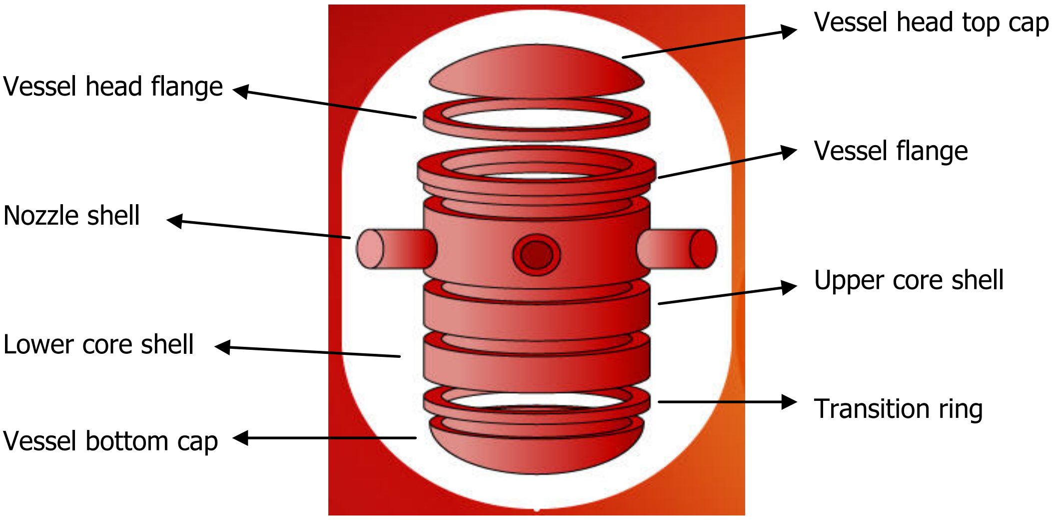


Figure 4.12

4.4.1 Owner and Engineering

As part of the nuclear steam supply system (NSSS), the Doel 3 RPV was ordered under the contract between project owner Sociétés Réunies d'Energie du Bassin de l'Escaut SA (EBES, Belgium) and contractor Framaceco (see sidebar).

The project owner EBES charged the engineering company Traction & Electricité with the daily follow-up of the construction of Doel 3. At the same time, the utility firm Intercom ordered Tihange 2 and charged the engineering company Electrobel with the daily follow-up of the construction of Tihange 2. Both engineering companies joined forces as the temporary company Trabel.

4.4.2 Owner Requirements

The owner requirements regarding the Doel 3 RPV are defined in a technical specification, which stipulates that the RPV must be manufactured in accordance with:

- The US Code of Federal Regulations 10 CFR 50, Section 50.55a 'Codes and Standards'
- The ASME Code, Sections II, III, V, IX and XI
- The ASME Material Specification SA-508 Class 3 (base material of the forgings)
- The ASME Code, Section III for Class 1 components, Edition 1974 up to and including the Summer 1974 Addenda

- The additional Belgian requirements included in the Conditions Complémentaires d'Application regarding the ASME Code Sections III, V, IX, and XI

4.4.3 Early Manufacturing Steps

The manufacturing steps most relevant to the Doel 3 Safety Case, are the early steps in the manufacturing process. These steps were identified based on the following information sources:

- The Cockerill construction files of the forged components
- The archives of Friedrich Krupp Hüttenwerke and Rotterdamsche Droogdok Maatschappij/Rotterdam Nuclear (RDM/RN) (see sidebar)
- The mail traffic between the owner's engineer (Trabel) and the contractors
- The mail traffic between Cockerill and RDM/RN

The various manufacturing steps were (see also figure below):

- The plate material for the vessel bottom cap and vessel head top cap was provided by Marrel Frères (France)

- The caps were formed in hot condition by Rotterdamsche Droogdok Maatschappij/Rotterdam Nuclear (RDM/RN)
- The base material for the forged components was provided by Friedrich Krupp Hüttenwerke (Germany): the ingot of the transition ring was delivered directly to RDM/RN, where it was cropped; the ingots for the remaining components were pre-forged by Krupp and cropped (i.e. removal of top and foot part of the ingot) by flame cutting; the resulting blooms were delivered to RDM/RN
- For all components, the main forging operations as well as heat treatment and various examinations were carried out by RDM/RN
- Cockerill (Belgium) applied the cladding on the components making up the lower portion of the vessel and performed the assembly of that part of the vessel
- Cockerill also cladded and assembled the vessel head
- At the same time, Framatome applied the cladding on the nozzle shell, vessel flange, and the inlet and outlet nozzles, and assembled all those components into the upper vessel subassembly

Rotterdamsche Droogdok Maatschappij / Rotterdam Nuclear (RDM/RN)

In addition to the Doel 3 and Tihange 2 reactor pressure vessels, RDM/RN was involved in the manufacturing of another 26 reactor pressure vessels. RDM/RN played a variety of roles in the manufacturing of the different reactor pressure vessels.These roles ranged from the forging of individual components, over assembly of an entire or only part of the reactor pressure vessel up to complete manufacturing, assembly, and erection at the site.

Framaceco regroups three companies: Framatome (Franco-Américaine de Constructions Atomiques, France), Acec (Ateliers de Constructions Electriques de Charleroi SA, Belgium) and Cockerill (Cockerill-Ougrée-Providence et Espérance Longdoz, Belgium).

In 1990, EBES merged with UNERG and Intercom to form Electrabel, which is the actual operator of Doel 3. Electrabel is now part of GDF SUEZ.

- The final circumferential weld joining lower and upper vessel subassemblies was carried out by Framatome
- Finally, the reactor vessel and the vessel head were subjected to hydrostatic testing by Framatome

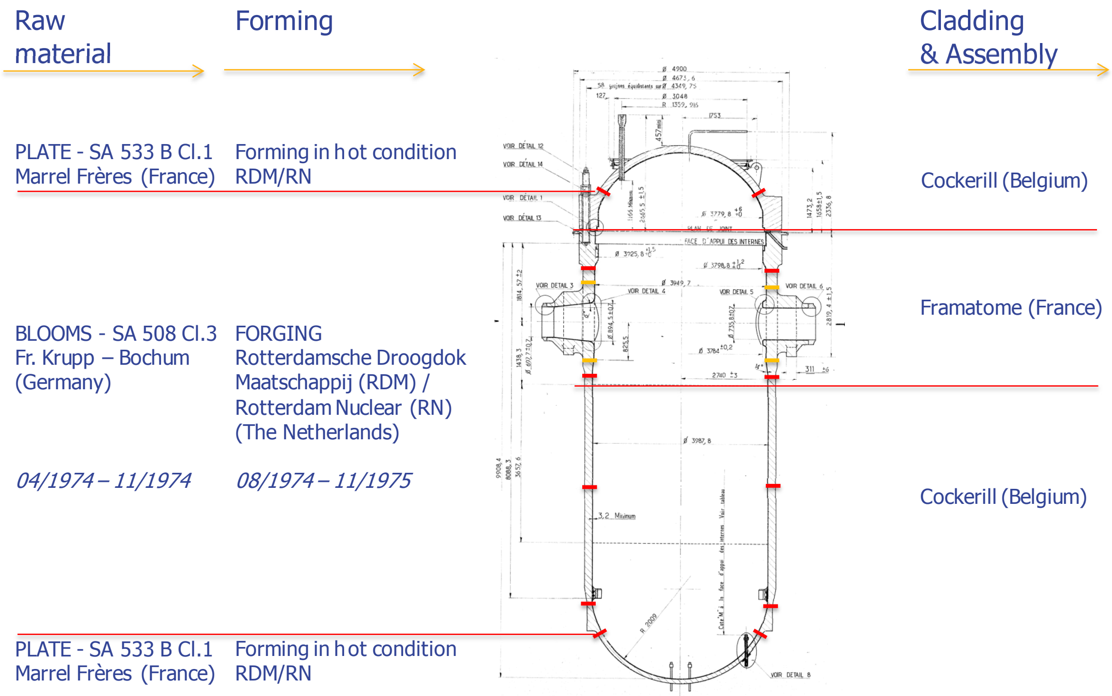


Figure 4.13 The figure above shows the manufacturing steps of the Doel 3 vessel and vessel head, including the companies involved

4.4.4 Outcome of the Documentation Review

The outcome of the documentation review is to be situated in the following areas:

Construction code and material

- The construction files prove that the Doel 3 RPV is constructed **in accordance with all requirements** mentioned in Chapter 4.4.2.
- The Doel 3 and Tihange 2 RPVs are fabricated from **type SA-508 Class 3 steel.** The base material of the forgings conforms to ASME Material Specification SA-508 Class 3.
- All components were delivered at their **required dimensions**, either to Cockerill in Belgium or to Framatome in France.

Manufacturing of ingots and blooms

The ingots were poured using a state-of-the-art vacuum casting technique. The hydrogen content for all six forged components is **below the 1.7 ppm (parts per million) limit** of the RDM/RN specifications, and also at or below the target hydrogen content of 1.5 ppm set forth by RDM/RN (see table).

Component	C [%]	S [%]	Mn [%]	H [ppm]
Vessel head flange	0.21	0.009	1.27	1.170
Vessel flange	0.22	0.008	1.28	1.450
Nozzle shell	0.23	0.010	1.26	1.000
Core upper shell	0.23	0.012	1.34	1.400
Core lower shell	0.21	0.010	1.25	1.500
Transition ring	0.23	0.011	1.30	1.500

Figure 4.14 The most important chemical parameters for hydrogen flaking (see Chapter 4.5)

Heat treatments of the ingots and blooms

- Because the ingots were poured by Krupp and the forging mainly carried out by RDM/RN, the ingot/blooms were cooled down to ambient temperature before being transported to RDM/RN. This cool-down does not generally take place when forging is performed by the same company as the one that pours the ingot: in that case the ingot arrives in hot condition from the steel works. The Krupp procedure regarding the manufacturing of blooms states that the **cool-down is completed in a controlled manner**, but no details are provided in the manufacturing documentation.
- The RDM/RN Master Traveller documents—listing the subsequent manufacturing and inspection steps—refers to a **first heat treatment** as 'Austenitizing and Tempering' and a **second heat treatment** as 'Hardening-Quenching-Tempering'.

The **chemical composition** of the forged components was determined at two different times. The first analysis, when pouring the ingot (ladle analysis)—was made by Krupp. The hydrogen content was also determined from a sample taken at the top of the ingot. The second time an analysis was made, it was based on two specimens taken from the foot end of the forging (product analysis).

Early inspections and reports

- For all Doel 3 RPV components, the final examination reports conclude with **status 'Accepted'**.
- The mandatory inspection reports make clear that the forged components have been subjected to various and **different types of inspection** in the RDM/RN shop, including such items as dimensional controls, ultrasonic testing (UT), and magnetic testing (MT). In addition micrographic, mechanical, and chemical tests were performed on specimens taken from the forged components.
- There is a corresponding procedure and there are **component-specific reports** for each inspection and test. All reports are included in the construction file, with the exception of:
  - The final UT inspection report of the nozzle shell. Instead, the construction file contains a number of telex messages between RDM/RN and Cockerill on the UT inspections. These messages confirm the acceptance of the component.
  - The first UT inspection report of the lower core shell. However, this inspection was not required according to the UT inspection specification. Nevertheless, the final mandatory UT inspection report is available.

Acceptance criteria

According to the component-specific reports, the ultrasonic examinations did not reveal any indications in the core lower shell. In the core upper shell, the transition ring, and nozzle shell on the other hand, some indications were found, but all of them fell within the acceptance criteria.

- For the **upper core shell** indications were found during UT inspections. All of the indications fell within the acceptance criteria and the component was accepted.
- Initially, the intention was to manufacture the transition ring for Doel 3 and Tihange 2 from a single ingot. UT examination on the Doel 3 transition ring revealed minor indications. The component was accepted. Inspection of the transition ring of Tihange 2 revealed unacceptable indications that were due to hydrogen flaking, according to RDM/RN. The component was rejected and a new transition ring was manufactured for Tihange 2.
- The nozzle shell initially made for Doel 3, was rejected because of a RT_{NDT} value (the reference value for the transition from brittle to ductile behaviour) that was too high. It was decided to install the Tihange 2 nozzle shell (with a RT_{NDT} value that had been accepted) on the Doel 3 vessel instead. A new nozzle shell was subsequently constructed for Tihange 2.

4.5 Origin of the Indications

The origin of the indications found during the 2012 in-service inspection of the Doel 3 and Tihange 2 reactor pressure vessels (RPVs) can be attributed to hydrogen flaking during fabrication.

This conclusion is based on:

- **A profound evaluation by AREVA**
- **A root cause analysis evaluating all possible flaw formation mechanisms**
- **A detailed comparison with a reference block with known hydrogen flakes**

Based on the indications' morphology (elliptical and flat), their number and position in relation to the residual metallurgical features in the original ingot (positive macro-segregations) and the associated detrimental elements (e.g. relatively high carbon content, small piercing diameter of the ingot), it can be assumed that the indications found are caused by hydrogen flaking. The full screening of all potential forming mechanisms confirms the hydrogen flaking as the most likely origin of the indications.

4.5.1 Evaluation by AREVA

AREVA carried out a detailed evaluation of the origin of the indications detected during the 2012 in-service inspection. This evaluation was based on a study of the Doel 3 RPV construction files, on the size and shape of the indications' distribution, and on AREVA vast experience with large forged components and the associated metallurgical phenomena. AREVA concluded that:

- The size and shape strongly suggest that the indications are linked to the thermo-mechanical manufacturing of the shells. More precisely, their pattern corresponds to the expected shape of the positive macro-segregations in the shells.
- It is very unlikely that the indications could correspond to metallurgical features such as 'A' segregates, macro-segregations, et cetera. The main argument is that such features are invariably present in all large forged parts and that indications similar to those detected in the Doel 3 RPV were never identified by AREVA in any of the large numbers of forged components they produced.

Hydrogen flakes

The hydrogen flaking hypothesis was considered to be the most plausible after concluding the July 2012 inspection of the Doel 3 RPV.

Doubt remained after the June 2012 inspection (initial UT inspections focused on the first 30 mm of the RPV shell) because AREVA had no experience with the appearance of hydrogen flakes close to the inner surface. This could be explained by the fact that AREVA components receive a dehydrogenation heat treatment during manufacturing. Even if this treatment was not sufficient to totally avoid the apparition of hydrogen flakes, it would at least can prevent the regions close to the surface from being affected by hydrogen flakes.

> **Hydrogen flakes**
>
> Hydrogen flakes are caused by insufficient dehydrogenation at the time of ingot pouring and the absence of a compensatory heat treatment just after forging.

The extra July inspections on the full thickness of the shell provided additional information supporting the hydrogen flaking hypothesis. The main arguments are:

- The large number of indications
- The pattern of the indications: flat and separated from each other
- A better understanding of the root causes leading to hydrogen flaking during fabrication (see sidebar)

Distribution profile and expected profile

A clear link was established between the indication distribution profile in the component and the expected positive macro-segregation profile. This expected profile is based on an extensive research program performed by AREVA in the 1990s.

This program makes it possible to predict the shape of macro-segregations in a forged component based on the forging process and the segregation distribution in the ingot. It was validated by an extensive experimental program that included the destructive characterization of several large forged components.

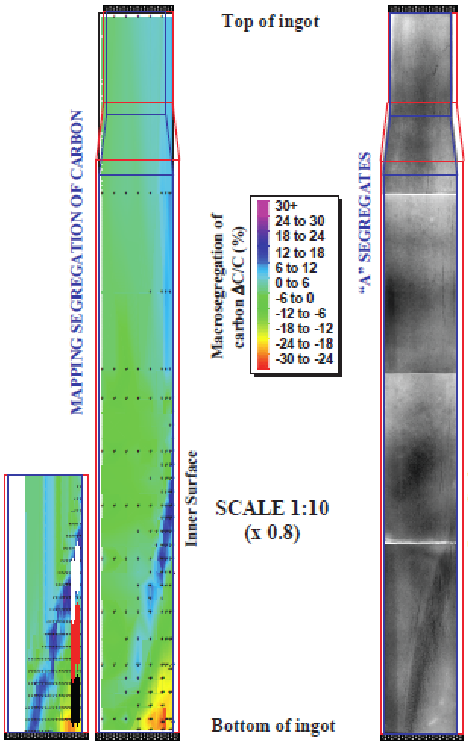


Figure 4.15 Metallurgical features revealed in a forged shell (right) and result from carbon mapping (left)

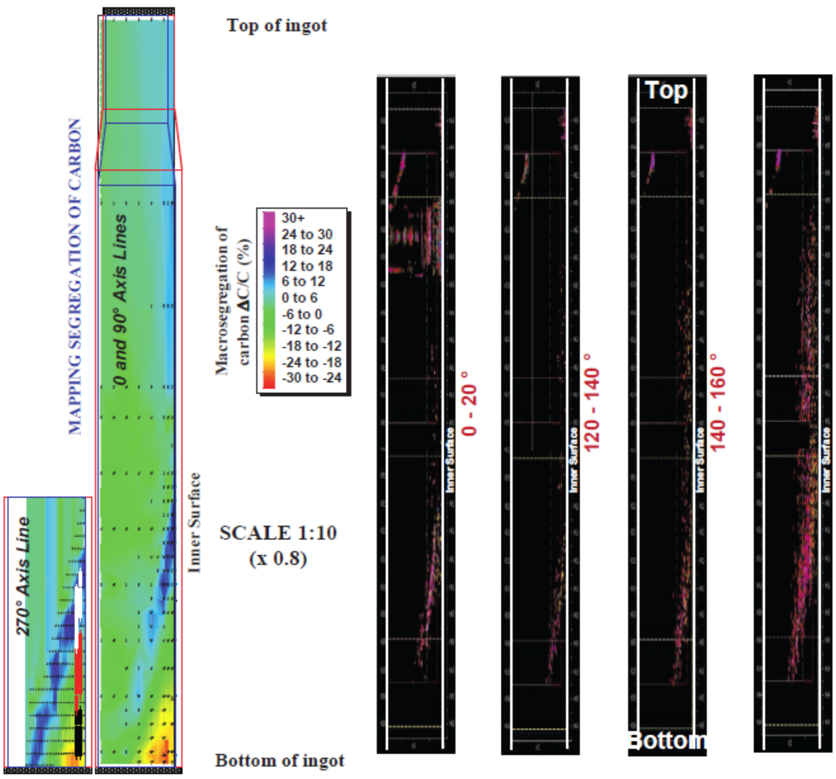


Figure 4.16 This figure shows a comparison with the profile of the indication envelope in the Doel 3 RPV

In the figure above, the shape of the distribution suggests that the positive macro-segregation has an influence on the mechanism by which the flaws are formed. The general shape of the envelope of the indications is similar in both the Doel 3 core shells and the Tihange 2 upper shell.

Flaking mechanism

Hydrogen flaking is the term used for a certain type of hydrogen damage, caused by hydrogen that is picked up during fabrication and for which segregation and differences in hydrogen solubility play a major role. There is a consensus in current literature that three factors are necessary in order to create hydrogen flakes:

- A sufficient amount of hydrogen
- Stresses
- A sensitive microstructure

It is well known that the larger the forged component, the more sensitive it is to hydrogen flaking during manufacturing. Hydrogen is much more soluble in its liquid than in its solid phase and much more soluble in a γ structure (face cubic centered) such as austenite than in an α structure (body cubic centered) such as ferrite.

As the material cools down after forging, the austenite to ferrite transformation occurs first in the zones of negative segregation and in segregation-free zones (due to the shape of the Fe-C phase diagram). This leads to an increased hydrogen concentration in the zones of positive segregation which are still in an austenitic state. If no precautions are taken to remove this excess hydrogen, flaking will occur at ambient temperature around traps formed by the discontinuities in the material: carbides, non-metallic inclusions, and grain boundaries, where the excess hydrogen gathers.

The ability of the material to withstand hydrogen flaking depends on the hydrogen content and on the density and hydrogen absorption capacity of these traps, as well as on the local microstructure influencing the local mechanical properties of the material. Laboratory studies have shown that a residual hydrogen content below 0.8 ppm (before cooling to ambient temperature for the first time after forging) will not cause hydrogen flaking.

Possible Root Cause of Hydrogen Flaking in the Doel 3 and Tihange 2 RPVs

It can be asserted with a high level of confidence that hydrogen flaking occured in the zone of positive macro-segregation during cooling after forging.

The following elements related to the fabrication history (see Chapter 4.4) or a combination of them can lie at the origin of the hydrogen flaking:

- The fact that the hydrogen content of the Doel 3 and Tihange 2 components was between 1.4 and 1.5 ppm, after the vacuum casting process executed by Krupp Bochum according to the common practice at the time of construction.
- There is no trace whatsoever of a dehydrogenation treatment, performed by RDM/RN. Although this is no proof that none has been applied, it could explain (a) a hydrogen content above the threshold for hydrogen flake formation and (b) the presence of flakes close to the metal surface.
- The carbon content of the ingots is high, compared to modern steels, which favors a structure more sensitive to flaking.
- The central part of the bloom that was removed by piercing is smaller than that typically found in comparable vessels produced in France. This can lead to a larger residual part of the area of positive massive macro-segregation in the vessel wall. According to AREVA experience based on practical observation and on modelling, these residual areas of positive segregation are the ones that are most prone to hydrogen flaking after forging.

German research
A German research program, performed in the 1970s and 1980s, reports that, despite a relatively low hydrogen content (0.8 to 1.2 ppm) during manufacturing, segregation cracks may occur when a dehydrogenation heat treatment is not carried out immediately after forging.

4.5.2 Root Cause Analysis

Electrabel conducted a full screening of all potential origins and flaw formation mechanisms independently from the AREVA evaluation (see table below).

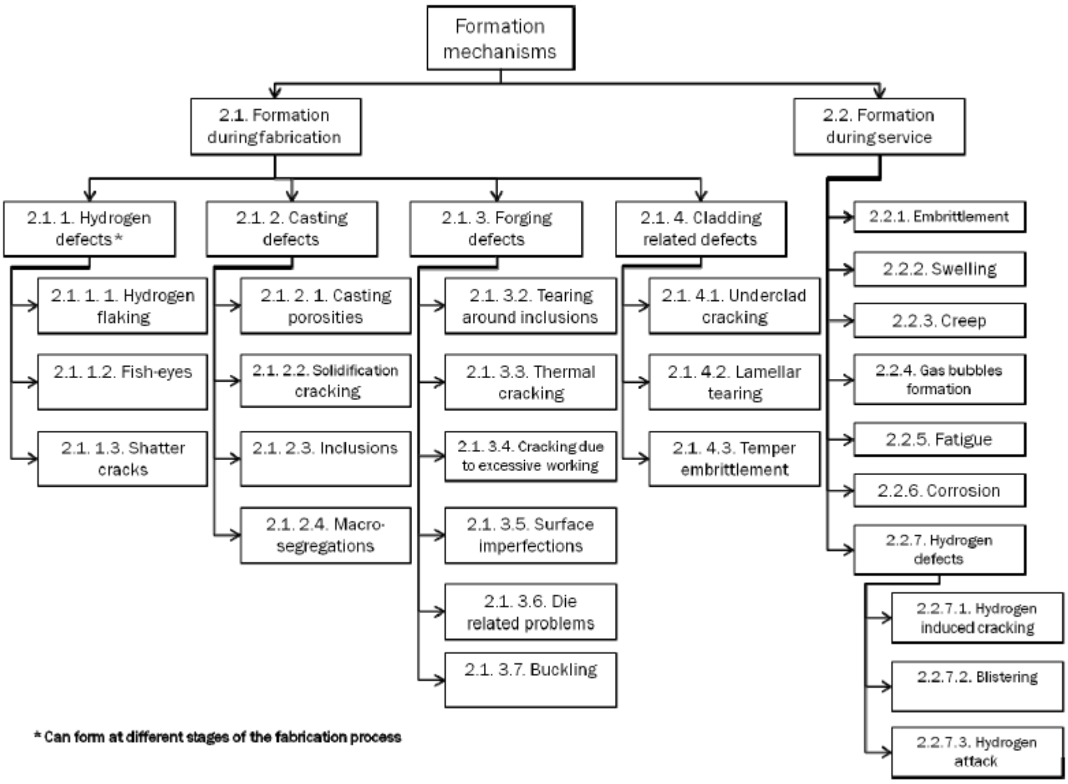


Figure 4.17

Formation during fabrication
All formation mechanisms during fabrication have been evaluated during the root cause analysis. This analysis revealed that hydrogen flaking is the most likely. All other formation mechanisms are excluded or are very unlikely:

Hydrogen
- **Fish-eyes** are perpendicular to the direction of maximal tensile stresses, which is not the case for the indications found in the Doel 3 and Tihange 2 RPVs.
- **Shatter cracks** are characterized by their multidirectionality, while the indications found in the Doel 3 and Tihange 2 RPVs are planar.

Casting
- **Casting porosities.** The forging ratios of the forging process are sufficiently high to get rid of the majority of the present casting porosities.

- **Solidification cracking**. The two main causes of this type of crack (a complicated mould design and/or excessive cooling rates), are not considered to be present. Moreover, the crack orientation and position are incompatible with the planar indications that were found.
- **Macro-segregation – Inclusions in the casting**:
  - The size of the indications (a few mm) is much larger than the typical size of inclusions (a few µm) and smaller than the usual size of inclusion clusters. The shape and the separation between the indications is clearly defined, which should not be the case for inclusion clusters.
  - The complementary use of the angle beam probe during the 2012 inspection should have detected volumetric inclusions, which was not the case.

Forging

- **Tearing around inclusions during forging**:
  - Only a few brittle inclusions (e.g. Al oxides) of very limited size (around 100µm), compared to the total thickness of the piece (200mm), were observed in the investigated material of the Doel 3 nozzle shell cut-out. The position of the indications does not fully correspond to the expected position of harmful inclusions.
  - Most of the UT indications have a more or less circular shape, while tearing around inclusions would cause more elongated indications.
- **Thermal cracking** during forging and **cracking due to excessive working** of the parts: both mechanisms will mainly result in a cracking pattern perpendicular to the reported indications.
- **Surface imperfections** during forging do not lead to a similar indication pattern as that which was detected on the Doel 3 and Tihange 2 RPV parts.
- **Die misalignment**, which is a typical feature on closed-die forgings and **buckling** during upsetting are both excluded because they are not considered relevant for the Doel 3 and Tihange 2 RPVs (open die forging and ratio length on diameter too small).

Cladding

- **Underclad cracking**. The orientation and position of the underclad cracks do not correspond to the indications that were found during non-destructive examinations.
- **Lamellar tearing** is a type of cracking that is generally located directly under the weld bead heat-affected zone (i.e. up to maximum 20 mm) and is not expected at a depth of e.g. 40 to 60 mm.
- **Temper embrittlement.** During manufacturing, the exposure of the shells to the critical temperature range is too short.

Formation during service

In addition to these findings, none of the potential in-service formation mechanisms proved to be plausible for the following reasons:

- **Embrittlement.** Embrittlement affects the mechanical properties of the material but does not cause cracking and is not detectable by UT inspection.
- **Swelling**. Temperature and irradiation doses are too low; bainitic and ferritic steels have a high resistance to swelling.
- **Creep**. Combined dose, stresses, and temperature are too low for significant creep effect; bainitic and ferritic steels have a high resistance to irradiation creep.
- **Gas bubbles formation**. The dose is too low and there are no significant quantities of elements susceptible to transmutation (inducing gas formation) in the RPV steel.
- **Fatigue-induced cracking** should be oriented according to the highest stresses, but is not consistant with the orientation of the indications in this case.
- **Corrosion**. The indications are internal, with no contact with primary water or any other electrolyte, which therefore impedes corrosion.

- **Hydrogen defects**:
  - **Hydrogen-induced Cracking** is a stress-induced mechanism but the indications are not perpendicular to the direction of the main stresses in service.
  - **Blistering**. The hydrogen concentration in the metal is too low.
  - **Hydrogen attack.** The indications are located inside the vessel wall, while hydrogen attack occurs mostly at the surface of the materials.

Hydrogen flaking confirmed

Hydrogen flaking is very likely for the following reasons:

- It is a known problem frequently reported in steel making.
- Planar morphology of the indications, i.e. parallel to the reactor pressure vessel surface, is typical for the morphology of hydrogen flakes, and is parallel to the direction of principal deformation.
- The position of the indications is compatible with the expected position of the positive segregation areas, which are the last areas that solidify/transform and are the ones most prone to hydrogen flaking.
- The measured hydrogen level in the liquid metal of around 1.5 ppm, could be above the threshold for hydrogen flaking, since the sulphur level is relatively low. AREVA recommends a conservative maximum allowed hydrogen content of 0.8 ppm.
- The hydrogen content of the bloom could still be high, since its diameter is 2,050 mm and diffusion at room temperature is limited. This will of course depend on the temperature/time cycle before cooling to room temperature. Also some additional hydrogen could have been picked-up from the environment during further fabrication steps at high temperature.
- The shells with a lower initial hydrogen content (about 1 ppm) have no or few indications.
- No records of any anti-flake or dehydrogenation heat treatment after forging, as applied by AREVA.
- The relatively small piercing diameter is expected to have left a larger part of the positive segregation area, being richer in hydrogen and hence more sensitive to flaking, inside the final piece.
- One shell (transition shell) was rejected in fabrication because of flaking. The twin shell, fabricated from the same ingot, with the same forging process also showed defects.
- Forging stresses can provide the stress necessary to produce hydrogen flakes (depending on the forging conditions). Also, the presence of high transformation stresses can not be excluded since there is no information about how the cooling down after forging was carried out.
- Based on the chemical composition (presence of elements favouring the quenchability of the steel, such as carbon and manganese), the local formation of martensite (which is brittle in untempered condition and sensitive to hydrogen flaking) is possible.
- The UT indications of the Doel 3 and Tihange 2 RPV shells are very similar to those of the AREVA shell, known to contain hydrogen flakes.

4.5.3 Flake Study on Reference Sample

In 2011, a forged shell, identified as VB395, meant for a 1,300 MW plant steam generator, was rejected at AREVA (France) because of hydrogen flaking. As a result of a loss of reliability of the hydrogen measurement, which led to an insufficient dehydrogenation heat treatment, the forged shell showed a large number of hydrogen flakes in the center part of the forging.

The thickness of the VB395 forged shell is comparable to the Doel 3 and Tihange 2 RPV shells (maximum 264 mm), though it has a slightly smaller internal diameter (3,564 mm). Because of its hydrogen flaking, the block has been used as a reference.

Figure 4.18 The VB395 forged shell

The indications in the VB395/1 sample have larger dimensions than those in the Doel 3 and Tihange 2 RPVs, but the sample was selected in order to have a high density of indications and a large range in dimensions. A perfect correspondence of both size distribution and number density of hydrogen flakes in different components is not possible since both parameters depend on the specific distribution of macro-segregations, hydrogen content and fabrication conditions.

The tests conducted by Laborelec confirm the nature of the defects as hydrogen flakes. More particularly the tests on the fracture surface of an open specimen provide strong evidence of hydrogen flaking, excluding all other defect mechanisms.

The hydrogen flakes are round, flat flaws. They have typically a diameter of 10 to 15 mm and are very thin (1 to 4 µm). They are laminar or slightly inclined.

Some flakes were precisely measured and characterized by destructive measurements: 18 indications were cut, 12 of them are isolated indications and 6 correspond to 3 groups of 2 indications. The main findings, which confirm the modelling assumptions made in the structural integrity analyses, are:

- Most hydrogen flakes are laminar with a maximum inclination of 14°.
- The 3 groups of two successive indications were chosen to evaluate the ligament between them along the different axes. For those 3 cases, the results of the cuttings confirm the ultrasonic conclusions and the existence of this ligament.
- The material is perfectly sound between the successive hydrogen flakes.

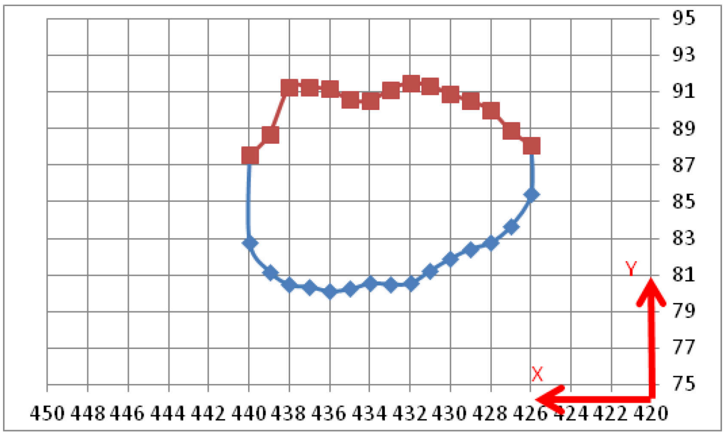


Figure 4.19 The typical shape of a hydrogen flake established by destructive measurements

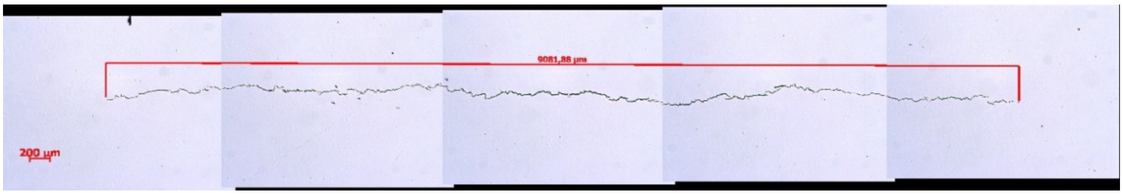


Figure 4.20 This figure shows a typical cross-section of the flake (9,081.88 µm)

4.6 Evolution of the Indications

A rigorous assessment of the flakes' potential growth over time has been carried out by Electrabel. The results of this analysis show that the only possible propagation mechanism is fatigue crack growth, the evolution of which is calculated to be less than 2.4% over 40 years. Therefore, there is no risk of ligament cracking between the flakes.

4.6.1 Study of the Propagation Mechanisms

Since there is a high density of hydrogen flakes in the Doel 3 RPV—and in particular in the lower core shell—the potential propagation of these flakes as well as the risk of *linking flakes by ligament cracking* needed further consideration. Consequently, a literature survey followed by an analysis of all possible propagation mechanisms of hydrogen flakes, was performed:

- Defect growth by a mechanism similar to flake formation
- Hydrogen-induced cracking
- Fatigue crack growth
- Corrosion
- Swelling
- Irradiation creep
- Gas bubble formation

Ligament between flakes

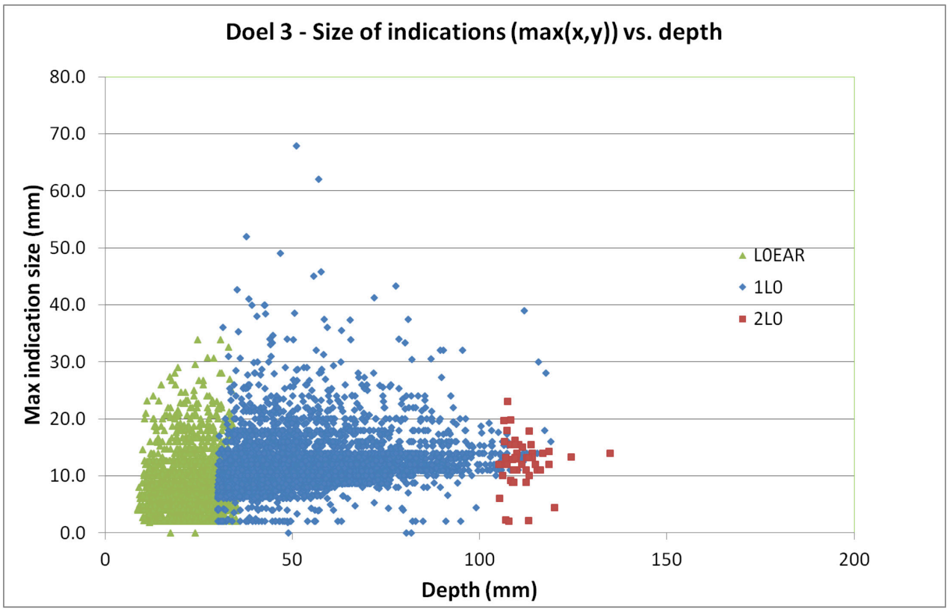
A ligament is a bond of sound matter between flakes. Sound means with no mechanical discontinuities such as cracks, cavities, et cetera. The ligament can be evaluated along different directions. When not specified, the direction along which the ligament is the shortest is usually assumed.

Based on this thorough analysis, fatigue crack growth has been identified as the only propagation mechanism possible in the specific situation of the Doel 3 and the Tihange 2 RPVs.

Defect growth by a mechanism similar to flake formation

Creating the same conditions as those occurring during flake formation requires a massive hydrogen feeding. This is impossible:

- After the various heat treatments during the manufacturing process and after 30 years of operation at 300 °C, all of the diffusible hydrogen that stems from the fabrication process will have left the material.
- The in-service uptake of hydrogen from the primary water or from corrosion processes is very limited (well below 0.1 ppm). This is confirmed by the absence of operating experience feedback on hydrogen-induced cracking in reactors which are or have been in operation. This is especially noticeable for first generation VVER reactors (the Russian version of PWR reactors), which do not have a stainless steel cladding and for which the potential hydrogen intake from the primary environment is higher.
- There would be a concentration gradient of hydrogen from the inner to the outer surface, where no hydrogen is present. If this mechanism were active, hydrogen flakes should be larger in size close to the surface than any found deeper in the vessel. This has not proved to be the case. The largest indications are clearly not close to the surface but rather at a depth of 40 to 60 mm (see figures below).



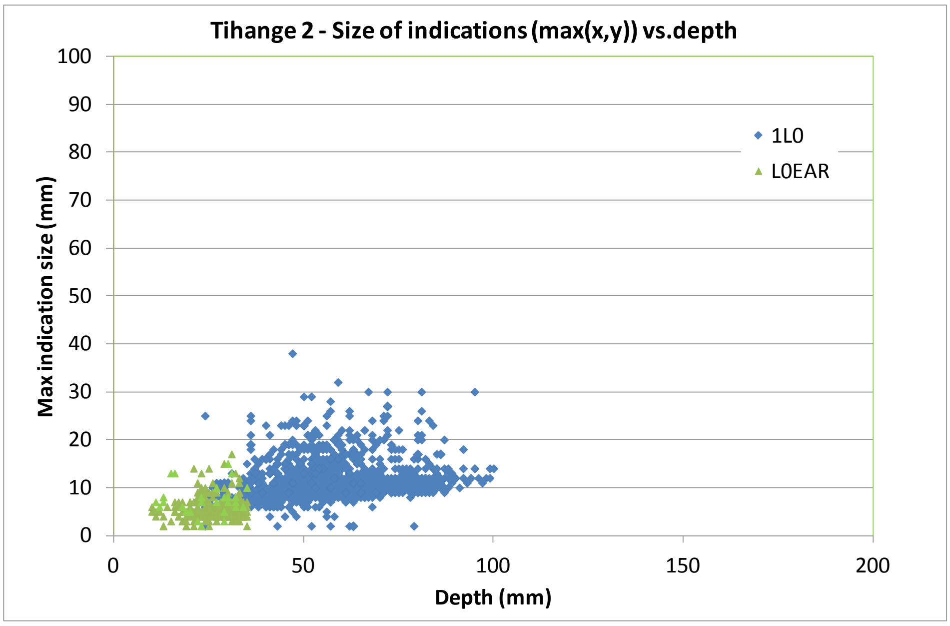


Figure 4.21 Size of indications versus depth. The L0 EAR, 1L0, and 2L0 are the transducers used to inspect the different depth zones in the shell (7 to 35 mm, 30 to 125 mm, and 100 to 200 mm respectively, the distance being measured from the cladding inside surface).

Hydrogen-induced cracking

Hydrogen-induced cracking would require a considerable amount of hydrogen to be present in the steel. This is not the case, as is explained for defect growth by a mechanism similar to flake formation.

In addition to a sufficient hydrogen intake in the material, sufficient stresses are also required to propagate the flaws. In operation, the stresses (in the direction perpendicular to the flaw) on nearly laminar flaws are low.

Corrosion

Corrosion occurs at the surface of the material, whereas the flakes are located inside the material.

Fluence is the flux integrated over time. It is the energy delivered per unit area. It is considered one of the fundamental units in dosimetry.

In the present case, the key parameter is fast neutron fluence (> 1 MeV).

Swelling

Swelling can result from irradiation at high temperatures:

- The temperature, as well as the irradiation dose (neutron fluence) during operation is too low in this case. Swelling is observed at temperatures higher than 300 °C.
- Ferritic steels are known to be highly resistant to swelling.

Irradiation creep

Ferritic steel has a high resistance to creep. Moreover, the irradiation dose, stresses, and temperature are all too low to elicit a significant creep effect.

Gas bubble formation

Gas bubble formation can be ruled out, as the irradiation dose is too low. In addition, there is an insufficient amount of elements present that are susceptible to transmutation.

4.6.2 Fatigue Crack Growth

Negligible Impact of Hydrogen on Crack Propagation

Fatigue crack growth is a very understood crack propagation mechanism, when no hydrogen is present. Since no significant hydrogen uptake can have occurred in service (see Chapter 4.6.1), the influence of hydrogen on crack propagation due to fatigue can be considered to be negligible. Hence the rules of ASME XI can be applied. This is addressed in the structural integrity assessment (see Chapter 4.8).

Fatigue is one of the degradation mechanisms for crack propagation in the reactor pressure vessel (RPV) taken into consideration at the design stage. The design is based on the ASME reference curve for RPV steels, taking into account the potential ageing effects. The fatigue which is considered for flake propagation in RPV conditions is low cycle fatigue resulting from transients such as stop-start cycles.

The fatigue crack growth evaluation has shown the following outcome:

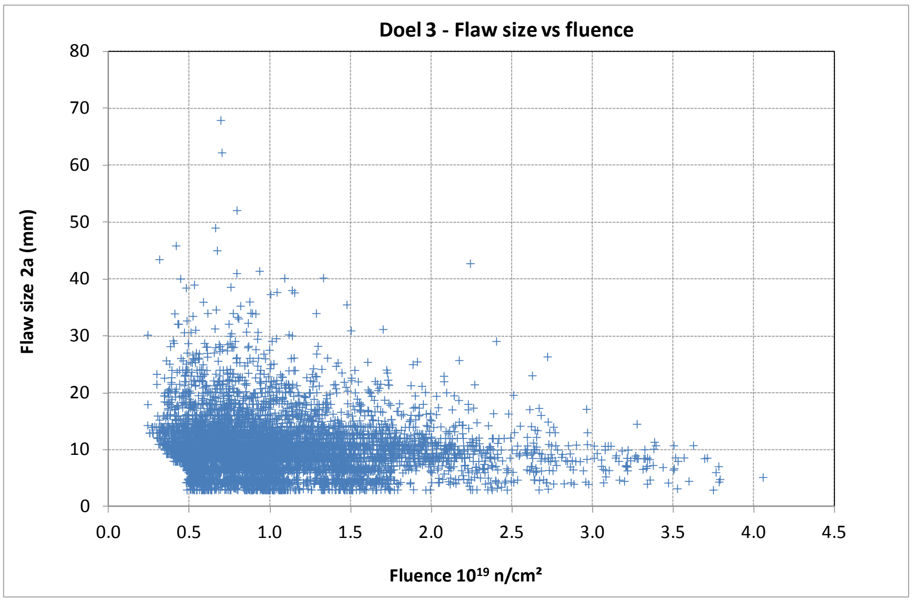
- The flaws propagate with less than 2.4 % of the flaw size, over a period of 40 years. For instance, for the largest flaw, the growth would be less than 1.1 mm in 40 years.
- The flaws that were present in the RPV from the beginning of service could not have propagated significantly due to fatigue crack growth over the past 30 years (1982-2012).

Since the computations are very conservative in terms of stress intensity factor (consideration of the axial projection instead of the actual orientation), and since the vast majority of flaws are much smaller size than the largest flaws, this propagation is considered as negligible and no significant evolution of the flaw sizes is to be expected.

4.6.3 Irradiation Effect

Since specialized literature offers no information about the possible irradiation effect on hydrogen flakes, the issue called for a pragmatic approach. This approach consists in identifying a potential trend in the size of the indications versus the fluence.

A 3D map of the neutron fluence (see sidebar)—radial, circumferential, and axial direction—is available and the specific fluence at the position of each indication is known. A plot of the indication size versus fluence should indicate a definite trend if irradiation induced a growth of the indications. The observed distribution confirms that there is no correlation between indication size and fluence (see figures below).



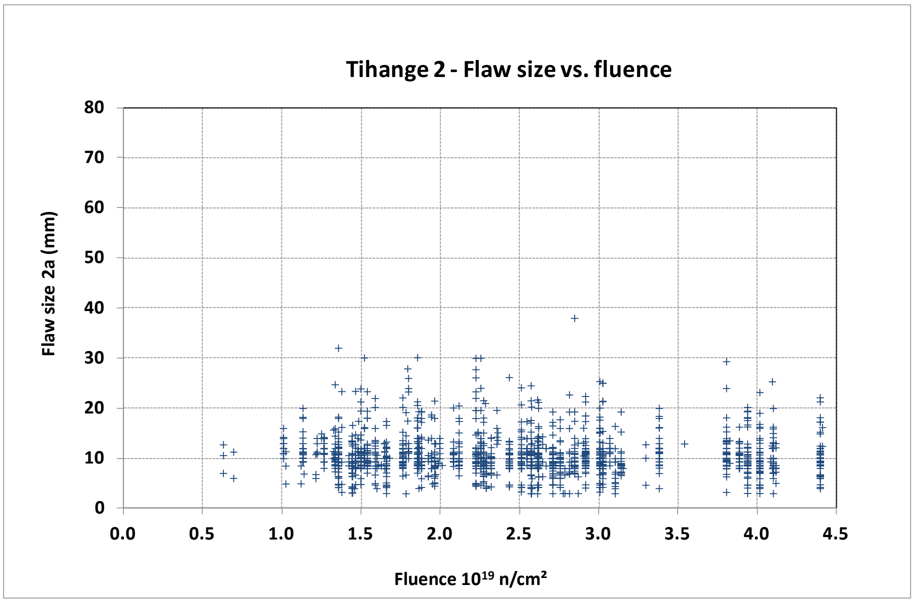


Figure 4.22 Distribution of flaw size versus fluence. The observed distribution confirms that there is no systematic increase in size of the indications with increasing fluence (E > 1MeV).

4.7 Material Properties

To determine the actual material properties, Electrabel carried out a materials testing program during which more than 400 specimens were tested. Moreover, the tests were carried out on highly representative material. Compared to the reactor pressure vessels, the samples had the same chemical composition, mechanical characteristics and origin. The samples also contained similar segregation zones.

After conclusion of the tests, no effect of orientation nor segregation was found on the samples. This means that no additional margin needs to be considered to cover the orientation and segregation effects on fracture toughness during the structural integrity assessments of Doel 3 and Tihange 2 vessels. There is a slightly higher sensitivity to irradiation embrittlement of the segregated zone.

Nevertheless, it has been decided to apply an additional shift of 50°C on the RT_{NDT} on top of the irradiation effect. This approach has shown to be very conservative.

4.7.1 Summary

In the course of this chapter, the following conclusions are reached:

- The justification of the structural integrity is based on the **fracture toughness** curves of the ASME Code indexed on the RT_{NDT}.
- No additional margin needs to be considered to cover the **orientation and segregation effects** on fracture toughness.
- The **irradiation effect** is evaluated by the FIS formula (irradiation embrittlement correlation). To take the effect of the composition of the macro-segregation into account, the contents in copper, nickel and phosphorus is increased (by 25% for Cu, 35% for P and 8% for Ni).

  The RT_{NDT} to be used in structural integrity analyses is summarized as follows:
  $RT_{NDT} = RT_{NDT\ init} + \Delta RT_{NDT\ FIS,\ segregation}$
- The **structural integrity assessments** of Doel 3 and Tihange 2 RPVs started at the same time as the materials testing program. Therefore, the results reported here were not available. A conservative approach was followed to determine the material properties to be used in these assessments. They were also performed on the basis of the RT_{NDT} evaluated by the FIS formula (for the nominal composition), with an additional shift of 50°C.

  This approach has shown to be very conservative given the fact that the additional shift of RT_{NDT} due to effect of orientation, segregation and irradiation was shown not to exceed 17°C at the most.
  The conservatism of the initial value of $RT_{NDT\ init}$ was shown through the Master Curve approach.

4.7.2 Material Properties of Interest

To analyze fracture mechanics, fracture toughness is key

The justification of the integrity of a structure with flaws such as the reactor pressure vessels (RPV) of Doel 3 and Tihange 2 is to a large extent based on fracture mechanics analyses. A key input for these analyses is a material property called fracture toughness.

Determining fracture toughness, as defined in ASME

In the ASME code, the fracture toughness of RPV steels is represented as an exponential curve that is indexed on a conventional temperature called Reference Temperature for Nil Ductility Transition (RT_{NDT}). RT_{NDT} characterizes the brittle-ductile transition of low alloy steels and is closely related to fracture toughness behaviour.

According to ASME NB 2331, the RT_{NDT} is usually determined by combining the results of drop-weight tests (ASTM E208) and Charpy tests. For reactor pressure vessel regions that have been affected by neutron irradiation, the effect of irradiation embrittlement is taken into account by predictive formulas that calculate the increase of RT_{NDT} under irradiation, in function of the fluence and the chemical composition of the material.

This results in the following:

$$RT_{NDT} = RT_{NDT\ init} + \Delta RT_{NDT\ irradiation}$$

Fracture toughness

The fracture toughness represents the resistance of the material to crack initiation. This material property can be directly related to the solicitation of defects expressed in terms of stress intensity factor.

Stress intensity factor

The stress intensity factor (K) is a fracture mechanics concept, used to characterize the stress field that exists near the tip of a crack and that is due to a remote stress field in the component.

Determining fracture toughness, for this safety case

For this specific safety case of Doel 3 and Tihange 2, it had to be confirmed whether or not the normal way of determining the RT_{NDT} (see above) to index the ASME fracture toughness curves is relevant.

The following items had to be addressed:

- the fracture toughness properties in the orientation of the indications (nearly laminar), as compared to the orientation in which fracture toughness is usually assessed (plane perpendicular to the inner surface)
- the specific properties of the zone of macro-segregations, which could be different from the properties of the bulk material outside of the zone where the RT_{NDT} was evaluated
- the fact that the specific composition in the zone of segregations could be more detrimental than the nominal composition, regarding irradiation embrittlement sensitivity

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4.7.3 Methodology for Obtaining Fracture Toughness Curves

The applied methodology for obtaining the fracture toughness properties is summarized in this flowchart:

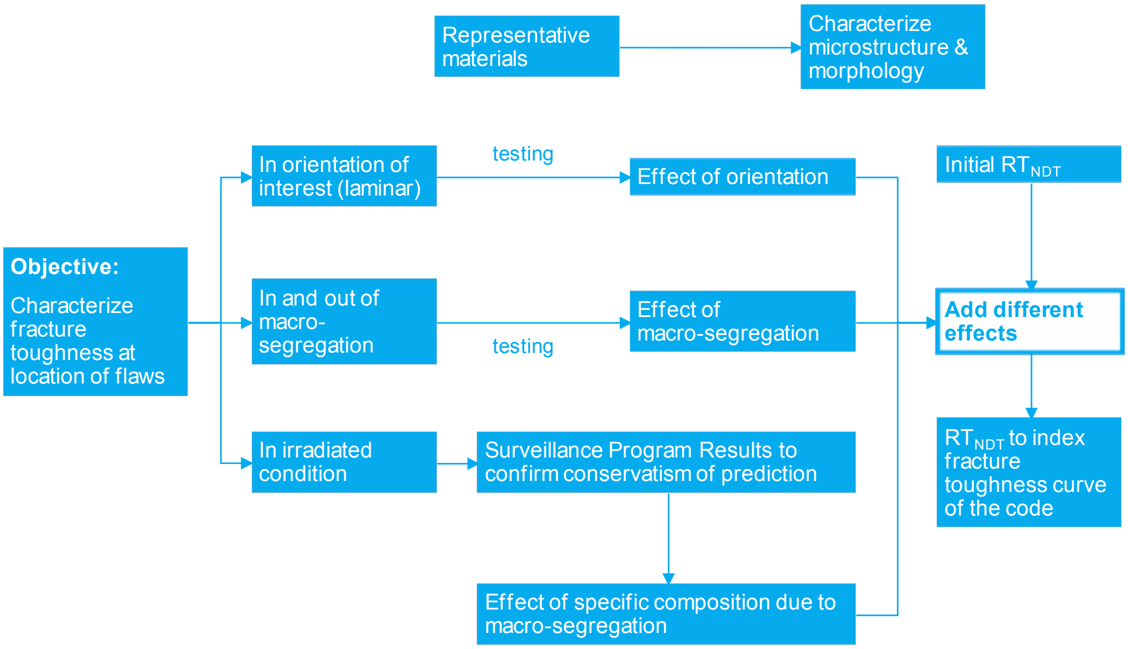


Figure 4.23: methodology

Sum of all factors affecting fracture toughness

All factors that affect or influence the fracture toughness properties of the material are added up in the following way:

$$RT_{NDT} = RT_{NDT\ init} + \Delta RT_{NDT\ irradiation} + \Delta RT_{NDT\ segregation} + \Delta RT_{NDT\ orientation}$$

Material investigation

The influence of the orientation and of the macro-segregations on the fracture toughness properties was assessed based on the results of an extensive test program executed in 2012 on representative materials. The conclusions of several large research programs carried out in Germany during the late seventies and early eighties were also considered. The irradiation embrittlement effects are covered by the RPV embrittlement surveillance results and some results from the German programs.

The representative materials tested in 2012 are the following:

- **Archive material from the Doel 3 core shells (SCK.CEN Doel 3 block)**: an unirradiated weld coupon with the base material of the upper core shell remaining from the RPV embrittlement surveillance program was available at the Belgian nuclear research centre SCK.CEN, with dimensions of 245 x 200 x 610mm.
- **An AREVA nozzle cut-out**: the nozzle cut-out is the circular part removed from the nozzle shell to insert the inlet and outlet nozzles. One such nozzle cut-out from a French unit was available at AREVA. The cut-out was already partially characterized and it was known to have macro-segregations.

- Also at AREVA, **a nozzle cut-out from Doel 3** (Figure 4.24) was available. In this cut-out the presence of a zone of macro-segregations was identified by chemical etching

Figure 4.24: Doel 3 nozzle shell cut-out

The following table shows a summary of the mechanical tests and metallurgical investigations:

Available materials	**Mechanical properties**			**Defect morphology**
	Irradiation effects	**Orientation effects**	**Segregation effects**	
D3 irradiation surveillance program	x	-	-	-
SCK.CEN Doel 3 block	-	x	-	-
Doel 3 Nozzle shell cut-out H1	-	x	x	-
AREVA nozzle shell cut-out H2BQ3	-	x	x	-
AREVA 395 shell – PREL block	-	-	-	x
German research programs 1970-1980	x	x	x	-

Table 4.2: mechanical tests and metallurgical investigations

Number of material samples used in tests

In total the following number of specimens were tested during the material properties program:

Tests	**Number of samples tested**
Fracture toughness tests on pre-cracked Charpy specimens	188
Fracture toughness tests on compact-tension (CT12.5mm) specimens	138
Charpy impact tests	42

Tensile tests	64

Table 4.3: number of samples tested

Master curve

Recent developments in elasto-plastic fracture mechanics make it possible to obtain valid static initiation fracture toughness data from small specimens. The results are interpreted using the Master Curve technique, first standardized in ASTM E-1921 in 1997 (the latest version, edited in 2012, was used in the present program). It is applicable to all sizes of specimens.

The Master Curve is an analytical formulation of the fracture toughness curve in the transition range, indexed in function of the Reference Temperature T_0 which represents the temperature at which the median master curve is at 100 MPa √m.

As an alternative to RT_{NDT} as the indexing reference temperature for the toughness curves in Appendix G of Section III and XI respectively, ASME Section III Code case N-631 and Section IX Code case N-629 allow the use of:

Consequently:

The influence of the orientation and of the macro-segregations on the RT_{NDT} were directly evaluated by the fracture toughness tests using the Master Curve approach.

This means that the general methodology can be summarized as follows:

$RT_{NDT} = RT_{NDT\ init} + \Delta RT_{NDT\ irradiation} + \Delta T_{0\ -segregation} + \Delta T_{0\ -orientation}$

4.7.4 Effects on Fracture Toughness

Orientation effect

The orientation effects on the fracture toughness were tested on the Doel 3 surveillance block (SCK.CEN Doel 3 block), on pre-cracked Charpy specimens tested in three point bending (see Figure 4.25).

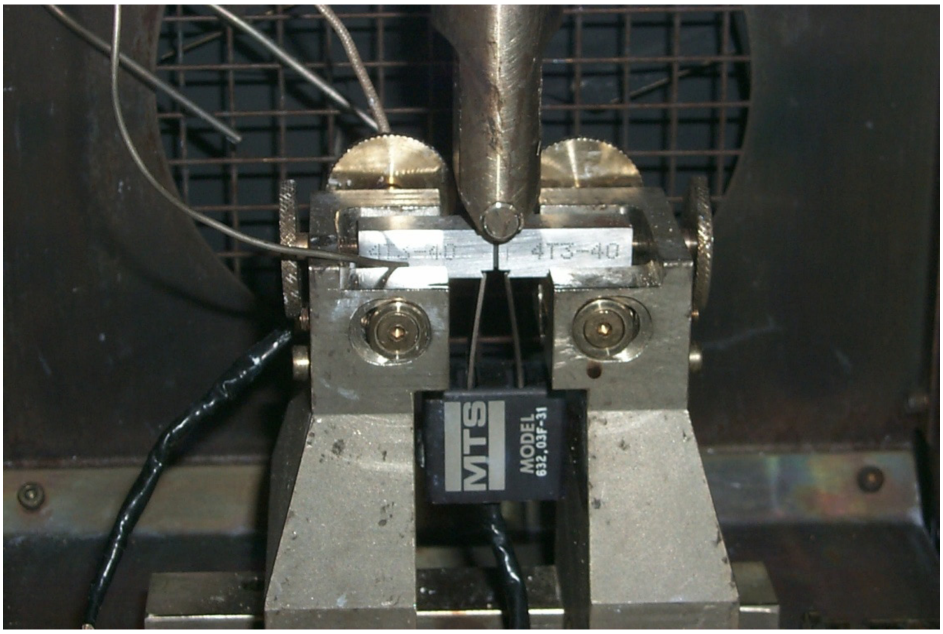


Figure 4.25: Pre-cracked Charpy specimen tested in three-point bending

A total of 129 valid test results were obtained. The results are shown in Table 4.4. No significant orientation effect was observed. The T_0 temperatures were comparable in all orientations tested. The differences were compatible with the normal scatter of this kind of tests and were not systematic.

These results (see Table 4.4) were confirmed by additional tests on Compact Tension specimens (CT 12.5) (see Figure 4.26) from the AREVA and Doel 3 nozzle cut-outs, in orientations representative of propagation of laminar cracks as well as of cracks perpendicular to the surface. This led to the conclusion that no additional margin needs to be considered to cover the orientation on fracture toughness.

$\Delta T_{0\ -orientation} = 0$

Figure 4.26: CT 12.5 specimen

Segregation effect

To quantify the effect of the segregations on the fracture toughness, CT 12.5 specimens were taken in the zone of macro-segregation in the AREVA cut-out, in an orientation representative of laminar flaws. The same number of CT12.5 specimens were taken in the same orientation, but outside of the zone of macro-segregation. 12 valid tests (according to ASTM E-1921) were obtained in each case.

The same exercise was done on the Doel 3 cut-out, both in the SCK.CEN and in the AREVA laboratories. The SCK.CEN repeated the experiment with specimens taken in an orientation representative of flaws perpendicular to the surface. A minimum of 12 valid tests (according to ASTM E-1921) were obtained in each case.

The results (see Table 4.4 and Figure 4.27) show that there is no difference between specimens sampled in the laminar orientation (propagation of a laminar defect in a plane parallel to the vessel surface) in the segregated zone and outside of the segregated zone. The difference does not exceed the usual experimental scatter in T_0 temperature. Similar results were obtained in the perpendicular orientation on the Doel 3 cut-out.

Material	Orientation	Segre-gation	Specimen type	T_0 (°C)	RT 1/2CT (°C)	RT_{NDT} (°C)	RT_{NDT} -RT
Doel 3 upper core shell (Surveillance block)	Laminar	No	PCCV	-87.9	-53.5	-22	31.5
Doel 3 upper core shell (Surveillance block)	Perpendicular	No	PCCV	-87.5	-53.1	-22	31.1
Doel 3 upper core shell (Surveillance block)	Perpendicular	No	PCCV	-82	-47.6	-22	25.6
AREVA cut-out	Perpendicular	No	½ CT	-84	-64.6	-25	39.6
AREVA cut-out	Laminar	Yes	½ CT	-92	-72.6	-25	47.5
AREVA cut-out	Laminar	No	½ CT	-92.5	-73.1	-25	48.1
Doel cut-out	Perpendicular	No	½ CT	-67.3	-47.9	-12	35.9
Doel cut-out	Perpendicular	Yes	½ CT	-76	-56.6	-12	44.6
Doel cut-out	Laminar	No	½ CT	-74	-54.6	-12	42.6
Doel cut-out	Laminar	Yes	½ CT	-75.6	-54.2	-12	42.2

Table 4.4: orientation and segregation test results

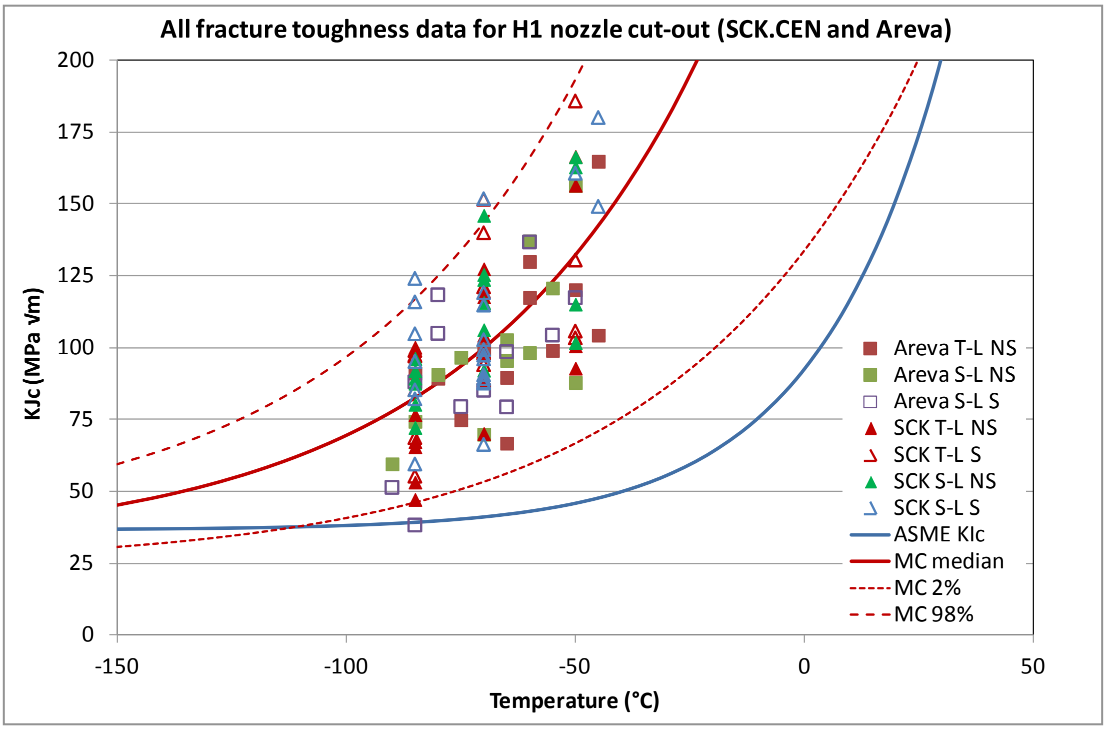


Figure 4.27 all fracture toughness test results from the nozzle cut-out of Doel 3, showing that all specimens can be considered as belonging to a same population

There is a visible effect of the segregation in laminar orientation in the fracture toughness properties in the ductile domain. However, these properties are of no concern to the present subject.

The German programs show some effect of the segregation on the fracture toughness (up to 20°C difference in T_0). However, all fracture toughness results are bound by the regulatory curve (the German KTA curve is practically identical to the ASME fracture toughness curve) and no additional margin is required to take this effect into account.

$\Delta T_{0\ -segregation} = 0$

Irradiation effect

The RPV embrittlement due to neutron irradiation is monitored with the help of capsules containing representative specimens of the most irradiated materials (base metal, weld and heat affected zone) and dosimetry, inserted at locations in the vessel where they experience higher exposure compared to the vessel shell (see Figure 4.28). The capsules are withdrawn at regular intervals and are representative of the vessel at certain times in the future.

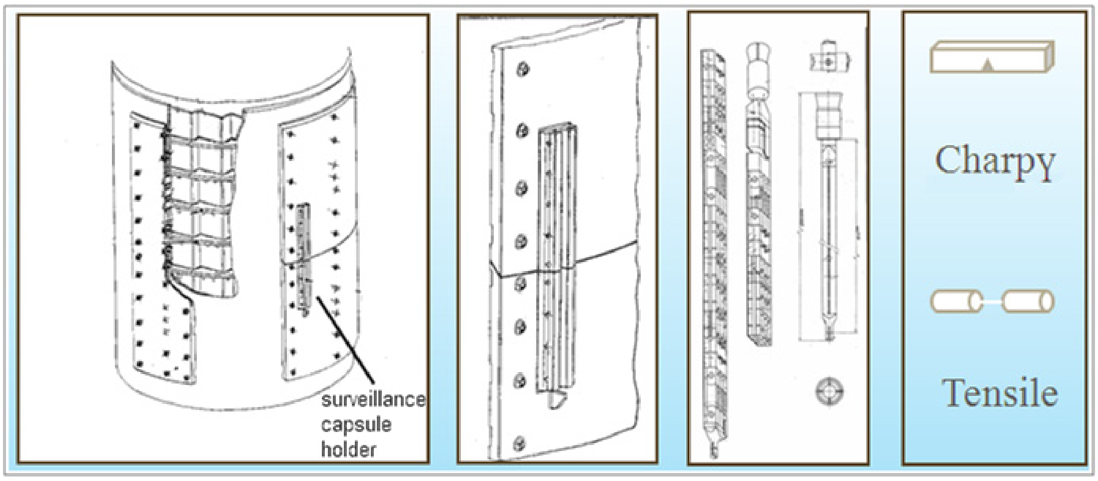


Figure 4.28: position of the surveillance capsules on the thermal shield

The 4 surveillance capsules initially in reactor have all been withdrawn from Tihange 2 and Doel 3. Experimental results are available for a fluence corresponding to at least 60 years of operation (the fluence at 40 years is 6.2 10^{19} n/cm^2 in Doel 3 and 6.7 10^{19} n/cm^2 in Tihange 2). They are illustrated by Figure 4.29 and Figure 4.30 below, where they are compared to the FIS predictive formula.

The FIS formula predicts the irradiation embrittlement on the basis of the copper, nickel and phosphorus content and of the neutron fluence. It was developed for the French RPVs, very similar to Doel 3 and Tihange 2 in terms of composition. In both cases, the FIS prediction is conservative, for example, a minimum of 10°C difference for forty years of operation (again, see Figure 4.29 and Figure 4.30, full lines).

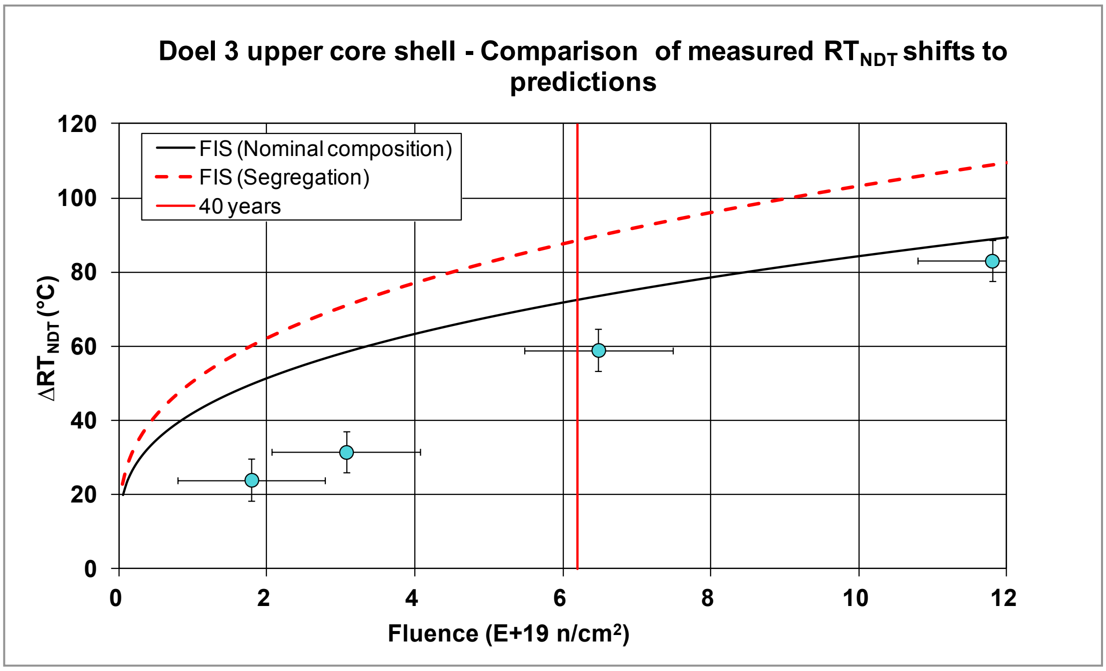


Figure 4.29

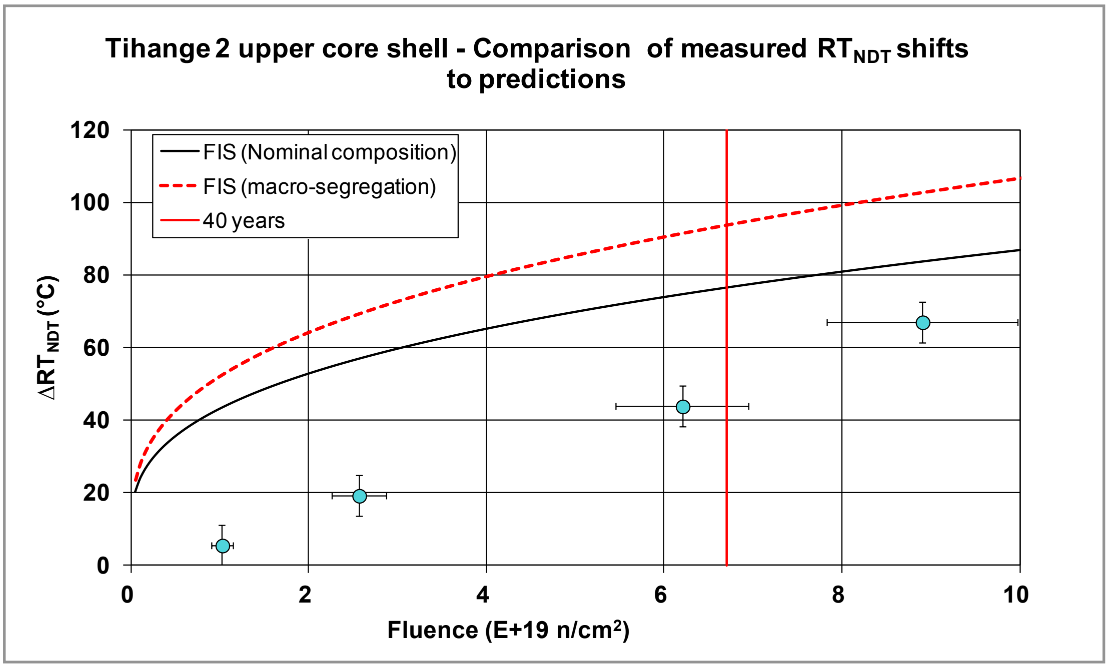


Figure 4.30

Irradiation effect on segregated zones

To take into account the chemical composition of the segregation zones on the irradiation embrittlement, the following enrichment factors will be considered in the FIS predictive formula:

$\%Cu_{segregation} = 1.25 \times \%Cu_{bulk}$

$\%P_{segregation} = 1.35 \times \%P_{bulk}$

$\%Ni_{segregation} = 1.08 \times \%Ni_{bulk}$

These values are the upper bounds resulting from a German program on the subject referred to as the BMI-TB SR 76 program in which several large forged components were characterized. They are conservative as compared to those measured on the Doel 3 nozzle cut-out.

The effect on the FIS predictions is illustrated in Figure 4.29 and Figure 4.30 (red dashed line) for Doel 3 and Tihange 2 respectively.

The maximum additional RT_{NDT} shift due to these enrichment factors, corresponding to the material with the most unfavourable composition (upper core shell of Tihange 2: Cu=0.05 %; Ni=0.73 %, P=0.012 %) at the peak fluence location is of the order of 17°C, thus relatively small.

The irradiation experiment performed in the German programs showed that specimens taken in the segregation zone or out of the segregation zone behaved in a very similar way under irradiation. Although the detailed composition of the segregation zone was not reported, this shows that the consideration of the enrichment factors described above on the Cu, Ni and P contents is conservative.

$\Delta RT_{NDT\ irradiation} = \Delta RT_{NDT\ FIS,\ segregation}$

Fracture load on large-scale specimens

It is worth noting that besides the small-scale fracture toughness tests considered above, also tests on large-scale tensile specimens containing hydrogen flakes were performed in the framework of the German research program FKS. The results show that, although the fracture load of the specimens is affected when the flakes occupy a significant proportion of the surface, the stress on the net section corresponding to the load at fracture is still above the minimum tensile stress of the material. However, there is a significant reduction of the ductility (elongation at break and reduction of area). This qualitatively confirms the validity of the net section approach used to verify the ASME III criteria on the primary membrane stress on the section affected by defects (see Chapter 4.8).

4.8 Structural Integrity Assessment

The Doel 3 RPV is fit to continue its safe operation. This is supported and substantiated by deterministic Structural Integrity Assessments (SIA) and an additional probabilistic SIA. All acceptance criteria set forth in the Safety Framework (see Chapter 4.2) are fulfilled.

4.8.1 Summary

The Structural Integrity Assessment (SIA) consists of a deterministic SIA and a complementary probabilistic SIA. The entire Assessment is performed using a shift of 50 °C in RT_{NDT} (the reference temperature fot the transition from brittle to ductile behaviour) that accounts for the influence of macro-segregations on the fracture toughness properties of the base metal. This value is conservative as compared to the one of 17 °C resulting from the experimental program. Moreover, the SIA covers 40 years of operation of Doel 3. The main conclusions of both SIAs, deterministic and probabilistic, can be summarized as follows:

- The verification of the primary stress limits, considering a reduced wall thickness that accounts for the presence of flaws, leads to the conclusion that the criteria regarding the primary stresses are met at all reactor pressure vessel locations.
- Mechanical interactions between geometrically close flaws are taken into account by defining grouping criteria. These criteria are applied to the whole set of detected flaws before assessing their acceptability using fracture mechanics.
- A conservative assessment shows that fatigue crack growth over the service lifetime of the Doel 3 RPV is very small. The maximum crack growth amounts to 1.72 % in the core shells and 2.32 % in the vessel flange, the transition ring and the nozzle shell. This means that the detected flaws could not have grown significantly due to fatigue during Doel 3's operation (1982-2012) and will not do so in the future. Therefore, fatigue crack growth is not a concern and need not be considered in the Flaw Acceptability Analysis.
- Given the number of detected flaws, it is relevant to define acceptable flaw sizes that cover the entire range of flaw characteristics (inclination, RT_{NDT}, ligament) according to 'Acceptance Criteria Based on Applied Stress Intensity Factor' in ASME XI. Each flaw or group of flaws in the Doel 3 RPV has a size that fully meets the applicable ASME XI acceptance criteria:
    - Lower shell: the margin to acceptable size is at least 22 % for individual flaws and 10 % for grouped flaws. Moreover, for the latter a detailed 3D analysis shows an increase of the margin to acceptable size to more than 50 %.
    - Upper shell: the margin to acceptable size is at least 50 % for individual flaws and 48 % for grouped flaws.
    - Vessel flange, transition ring and nozzle shell: a very conservative assessment shows that the few flaws detected in these components are acceptable.
- Prevention against brittle failure of the pressure boundary of primary components is re-assessed to take into account the presence of segregated zones in the reactor pressure vessel (RPV) material. This analysis of the Fracture Toughness Requirements shows that protection against low temperature overpressure is guaranteed.
- As the Pressure-Temperature limits and the Protection Against Cold Overpressure by Mass Input and Energy Input are impacted by the re-assessment, the plant's Technical Specifications have to be adapted accordingly.
- The protection of the RPV against pressurized thermal shock (PTS) events is confirmed as the criteria on the RT_{NDT} is met by all vessel components.

- The complementary probabilistic SIA concludes that the Frequency of Crack Initiation of Doel 3 RPV amounts to 2.3×10^{-8} per reactor year, which is two orders of magnitude below the acceptance criterion of 10^{-6} defined in 10CFR50.61a.

4.8.2 Methodology

The Structural Integrity Assessment (SIA) procedure is presented in the scheme below.

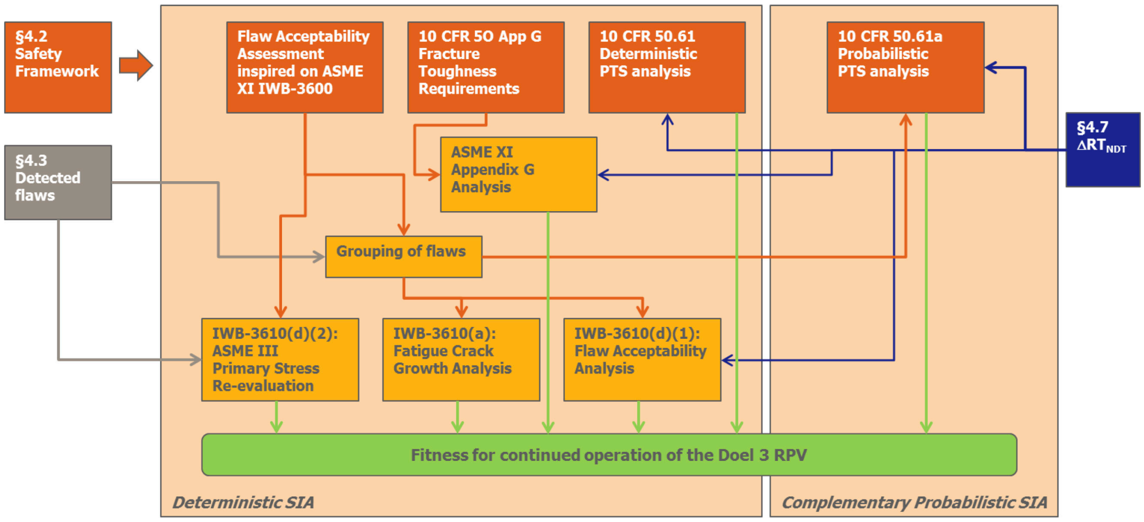


Figure 4.31 Schematic presentation of the Doel 3 RPV Structural Integrity Assessment (SIA)

All evaluations, both deterministic and probabilistic, consider an additional shift in RT_{NDT} of 50°C, which is conservative with respect to the results of the material investigations (see Chapter 4.7).

Deterministic SIA (see 4.8.3)

The deterministic SIA is divided into three separate assessments:

- **Flaw Acceptability Assessment**: inspired by the ASME XI procedure but adapted to the nature and number of indications found in the Doel 3 RPV. For the indications' grouping, a particular methodology has been developed. The Flaw Acceptability Assessment consists of three parts:
  - **ASME III Primary Stress Re-evaluation**: the primary stresses are re-evaluated according to the ASME III rules, taking into account the presence of flaw indications in the RPV shell.
  - **Fatigue Crack Growth Analysis**: assessment of potential crack growth due to fatigue, in accordance with the rules of ASME XI Appendix A (see also 4.6.2).
  - **Flaw Acceptability Analysis**: based on the ASME XI criteria but adapted to the particular nature of the flaws.
- **Fracture Toughness Requirements**: an analysis of the fracture toughness requirements, as stipulated in 10 CFR 50 Appendix G, and performed according to the rules of ASME XI Appendix G.
- **Deterministic PTS Analysis**: a deterministic pressurized thermal shock (PTS) analysis as requested by 10 CFR 50.61.

Complementary Probabilistic SIA (see 4.8.4)

The complementary probabilistic SIA covers a probabilistic PTS analysis according to 10 CFR 50.61a.

4.8.3 Deterministic SIA

Flaw Acceptability Assessment

Electrabel decided to perform a Flaw Acceptability Assessment in line with ASME XI. The conclusions of this assessment can be summarized as follows:

- **The most critical criterion regarding primary general membrane hoop stress is still met in all Doel 3 RPV components.**
- **The results have shown that fatigue crack growth is not a concern.**
- **All indications encountered in the Doel 3 RPV core shells, transition ring, nozzle shell, vessel flange, and vessel head flange fully meet the acceptance criteria.**

From measurements to calculations

Every single flaw detected by UT inspection is known in terms of its exact position in the RPV wall and its maximum dimensions in three orthogonal directions. This means that each flaw may be represented by a rectangular box surrounding the flaw, with the same dimensions in three orthogonal directions as the flaw. This principle is illustrated in the picture below. For visualization purposes, the detected flaw is schematically represented by an ellipse in this figure.

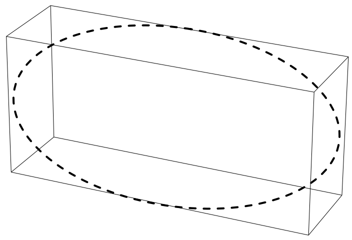


Figure 4.32 Illustration of rectangular box surrounding one detected flaw

For each flaw, i.e. each rectangular box, the corresponding circular flaw characteristics used in the calculations need to be defined. This is illustrated in the figure below:

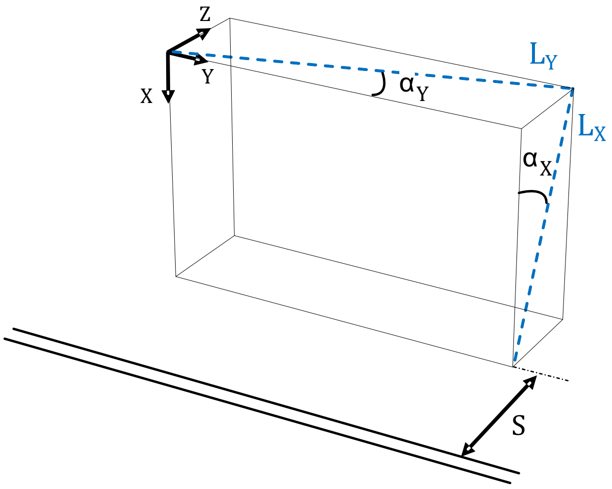


Figure 4.33 Illustration of definition resulting flaw characteristics

- The diameter of the circular flaw is taken as the maximum length of the two diagonals (L_X, L_Y) depicted in the figure above.
- The flaw inclination is taken as the maximum of the angles of the diagonals (α_X, α_Y) depicted above.
- The ligament S of the circular flaw is taken as the minimum distance between the box and the interface between the cladding and base metal.

The definition of the resulting circular flaw is illustrated in the figure below. The dashed line depicts the largest flaw that may be included in the rectangular box. In general, this flaw is elliptical. The resulting flaw that will actually be considered in the Flaw Acceptability Analysis is the circular flaw (indicated by the solid line) that envelopes the elliptical flaw.

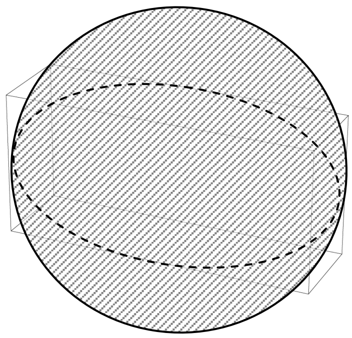


Figure 4.34 Illustration of resulting flaw to be assessed

Flaw grouping rules

The flaws found in the Doel 3 RPV fall outside of the ASME XI in-service framework. Therefore, before the Flaw Acceptability Assessment could start, the ASME XI assessment procedures had to be adapted to the very specific indications found in the Doel 3 RPV, i.e. present in large numbers and nearly laminar in shape. Two specific rules have been developed:

- One for the grouping of closely spaced flaws, based on ASME XI IWB-3300 (flaw characterization).
- One for demonstrating the flaw acceptability, based on ASME XI IWB-3600 (analytical evaluation of flaws).

Before assessing the acceptability of flaws by fracture mechanics, closely spaced flaws must be grouped first. Mechanical interactions between flaws tend to increase when the distance between the flaws decreases. Rules are developed that enable the experts to decide whether two closely spaced flaws can be considered as individual flaws or have to be combined into one single flaw. This principle is applied for each flaw and all of its surrounding flaws. This sequence continues until no interaction with surrounding flaws is found anymore.

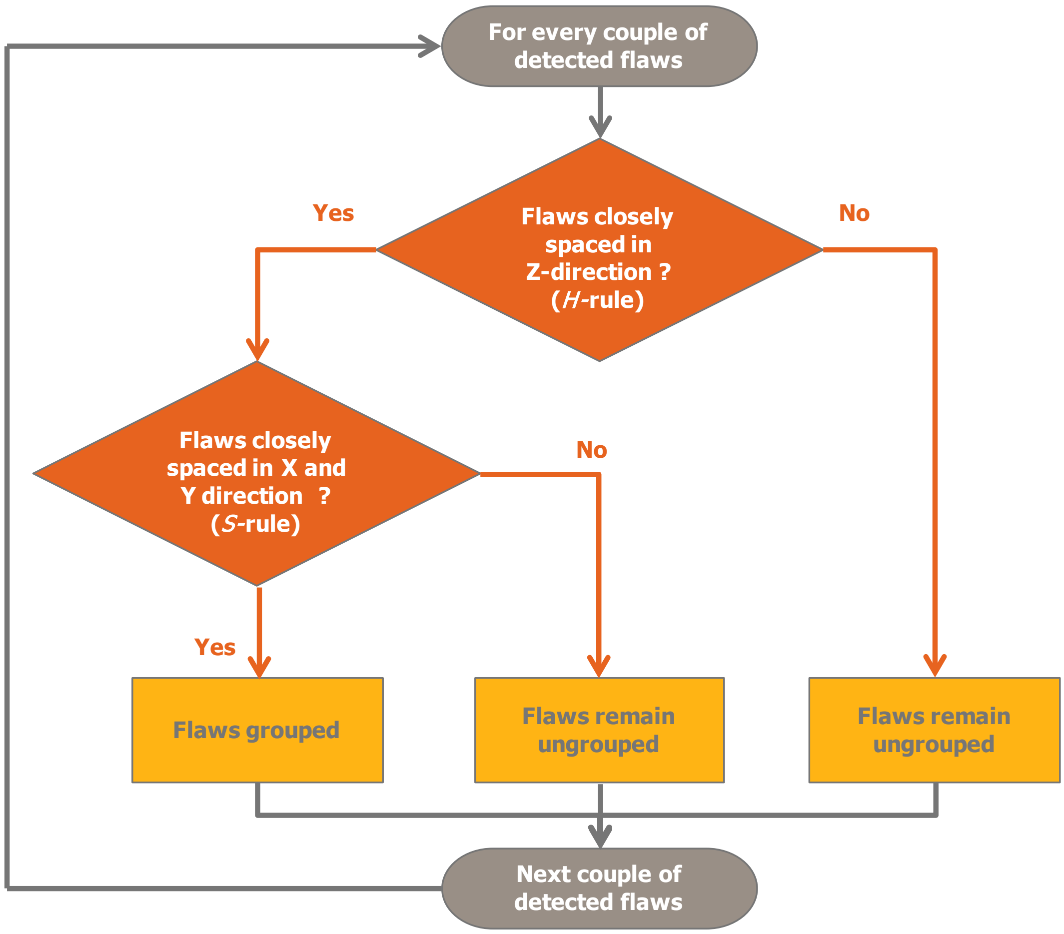


Figure 4.35 Application of grouping rules to detected flaws

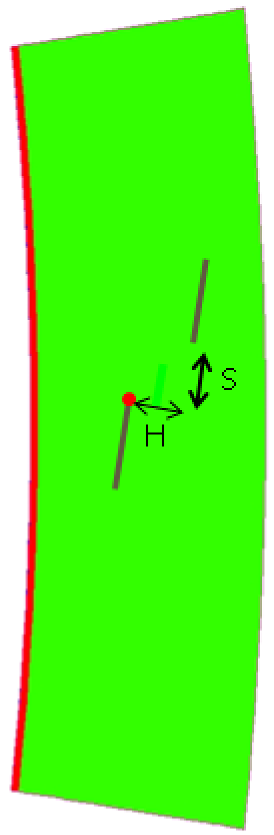


Figure 4.36 Any flaw or group of flaws is screened based on its H and S values

Flaw grouping rules

In fracture mechanics, each individual flaw may be characterized by its Stress Intensity Factor (SIF, see definition in Chapter 4.7). When the distance between flaws decreases, the interaction between both the flaws increases and hence so do their respective SIF. Experiments show that interactions between flaws may be considered as significant when the increase of a flaw's SIF due to the presence of another flaw, is larger than 6 %, i.e. the interaction factor is larger than 1.06. This observation is the basis for the development of the rules on grouping of flaws.

In fact, the spatial situation of any pair of flaws may be characterized by **two geometrical parameters:**

- **H:** used to check the spacing between the two flaws in the direction of the RPV wall thickness (Z-direction).

- **S:** used to check the spacing between the two flaws in the axial (X) and circumferential (Y) direction of the RPV wall.

Any pair of flaws in the Doel 3 RPV forgings is screened based on its H and S value (see scheme left). The values of H and S below which flaws have to be grouped are determined based on 2D finite element calculations. These calculations cover all possible configurations of flaw pairs that may be encountered in the Doel 3 RPV forgings. For each configuration, the interaction factor related to the SIF is calculated and is compared to the threshold value of 1.06.

The grouping rules have been applied to all possible pairs of flaws detected in the Doel 3 RPV forgings as follows: as explained above, every single flaw detected by UT inspection is represented by a rectangular box surrounding the flaw, and with the same dimensions in three orthogonal directions as the flaw. In addition, this box has a well defined position in the RPV wall. This principle is illustrated in the picture below for a case in which three flaws have to be grouped.

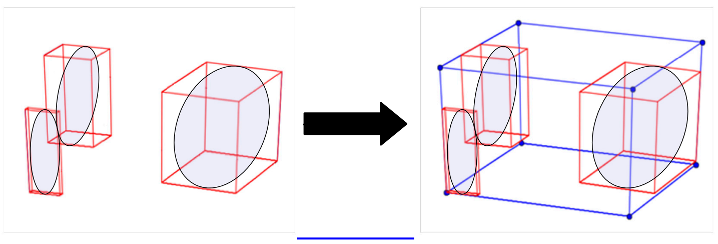


Figure 4.37 Illustration of flaw grouping

On the left hand side of this figure, the three flaws are schematically depicted as ellipses (for visualization purposes) together with the three red surrounding rectangular boxes. If the flaws are to be grouped according to the H and S rule (see sidebar), this means that the red boxes will be replaced by the blue box depicted on the right hand side. It is the smallest box that surrounds the three red boxes.

Thus, for each forging, the result of this operation is a combination of a list of individual flaws that will remain individual ones in the acceptability assessment phase and a list of flaws that have to be grouped. For the Doel 3 RPV as a whole, 18.6 % of the detected flaws have been grouped. For the lower core shell in particular, 19.7 % of the detected flaws have been grouped.

Definition of combined flaw

In exactly the same way as explained above for the individual flaws, the corresponding circular flaw for which the actual Acceptability Assessment will be performed needs to be defined. This time however, the starting point is the blue rectangular box that envelopes the red boxes surrounding the flaws that have to be grouped. The result of this step is illustrated in the figure below:

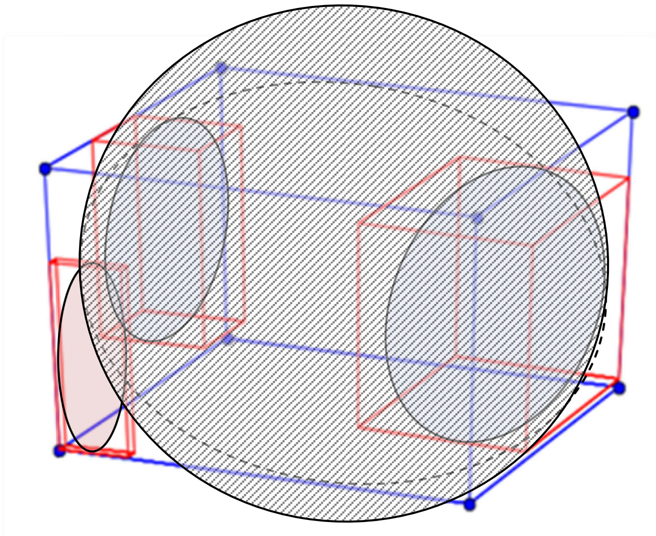


Figure 4.38 Illustration of resulting flaw to be assessed in case of flaw grouping

The dashed line depicts the largest flaw that may be included in the rectangular box. In general, this flaw is elliptical. The resulting flaw that will actually be considered in the Flaw Acceptability Analysis, is the smallest circular flaw (indicated by the solid line) that envelopes the elliptical flaw.

The figures above also illustrate the conservatism of the grouping approach: the flaw size that is finally considered in the Flaw Acceptability Analysis, will be much larger than the size of each of the individual flaws in the red boxes. Among the grouped flaws of the Doel 3 RPV, inclinations up to 40° are encountered, while the detected individual flaws have inclinations between 0° and 20°. The higher the inclination of the flaw, the more the flaw moves away from the relatively favourable orientation that pure laminar flaws have with respect to the governing stress fields in cylindrical vessels, and, hence, the more conservative the analysis becomes.

ASME III Primary Stress Re-evaluation

The Doel 3 RPV has been designed in accordance with ASME III for Class 1 components. As such, the different stress and stress range criteria given under Subsection NB-3000 ‘Design by Analysis’ have been verified at the design stage. These criteria aim to prevent a number of pressure vessel failure modes, such as excessive plastic deformation, plastic instability leading to incremental collapse, and fatigue, among others.

The flaw indications revealed during the 2012 in-service inspection, are addressed in ASME Section XI. One ASME XI Article, IWB-3610(d)(2), refers to the 'Design by Analysis' Subsection NB-3000 as it requires the verification of the primary stress limits of that Subsection, assuming a local area reduction of the pressure retaining membrane that is equal to the area of the detected flaws.

Membrane hoop stress analysis

In each stress criterion the calculated primary stress is compared to the allowable stress. The original Doel 3 RPV stress analysis has shown that the criterion regarding the primary general membrane hoop stress in the circumferential direction is the one where the calculated stress comes closest to the allowable stress, i.e. 95.1 % of the allowable value. The calculated stress is proportional to the internal design pressure (171.6 bar) and the diameter of the reactor vessel and inversely proportional to its wall thickness (200 mm). The allowable stress is a material characteristic defined in the ASME Code and has to be taken at the design temperature of the vessel.

Membrane hoop stress re-evaluation

As requested by IWB-3610(d)(2), the calculated stress must be re-evaluated, thereby considering a reduced wall thickness that accounts for the presence of flaws in a particular location of the vessel wall. As long as this thickness reduction does not exceed 4.9 % (i.e. 100 - 95.1 %), the criterion on the primary general membrane hoop stress will still be met. This verification has been done explicitly for the lower and upper core shells of the Doel 3 RPV, where numerous flaws have been detected. It also automatically covers the transition ring, the nozzle shell and the vessel flange where much smaller numbers of flaws have been detected.

Doing so, it appears that the most critical ASME III NB-3000 criterion regarding primary general membrane hoop stress is still met in all reactor pressure vessel locations.

Fatigue Crack Growth Analysis

As requested by ASME XI IWB-3610(a), Fatigue Crack Growth is evaluated by the analytical procedures described in ASME XI Appendix A 'Analysis of Flaws', based on linear elastic fracture mechanics. The objective is to calculate the growth of the nearly laminar flaws until the end of service lifetime of the Doel 3 RPV.

The results show that even when applying the very conservative approach (see sidebar), the maximum growth of the flaws in the forgings over the whole service lifetime of the RPV is limited to 2.32 % of their size, and for the core shells in particular, the maximum growth is limited to 1.72 %. **This means that fatigue crack growth is not a concern and does not need to be further considered in the Flaw Acceptability Analysis.**

As required by the ASME Code, the **flaws** in the forgings are subjected to the ASME Level A and B transients with their maximum allowable number of occurrences over the whole service lifetime, as specified in the Final Safety Analysis Report of the unit. This is a **conservative approach**, since in reality the transients appear less frequently than supposed in the analyses.

In addition, conservative heat transfer coefficients for the heat transfer from primary water to the reactor pressure vessel wall are considered during the transients. The temperature-time and stress-time histories at the level of the reference flaws are obtained from Finite Element modelling, and the crack growth is evaluated by the fracture mechanics software TEEPAC.

The fatigue crack growth analysis is performed on the axial and circumferential projections of the flaws rather than by considering their real orientations. Comparison between both approaches has shown that the approach based on axial and circumferential projections is much more conservative.

Flaw Acceptability Analysis

Since the Fatigue Crack Growth Analysis has shown that the growth of the flaws is negligible, the Flaw Acceptability Analysis can consider the flaws with their dimensions derived from the 2012 ultrasonic inspections (as described above).

This analysis has made clear that all indications encountered in the Doel 3 RPV core shells are acceptable. The few indications found in the transition ring, nozzle shell, vessel flange, and vessel head flange were accepted as well.

As can be seen in the diagram below, the vessel shells (transition ring, core shells and nozzle shell) are addressed in a different way than the vessel flanges. This is because of the number of flaws in the vessel shells and because the stress field in the vessel flanges is differs from the stress field in the vessel shells (due to the presence of holes for the RPV bolting).

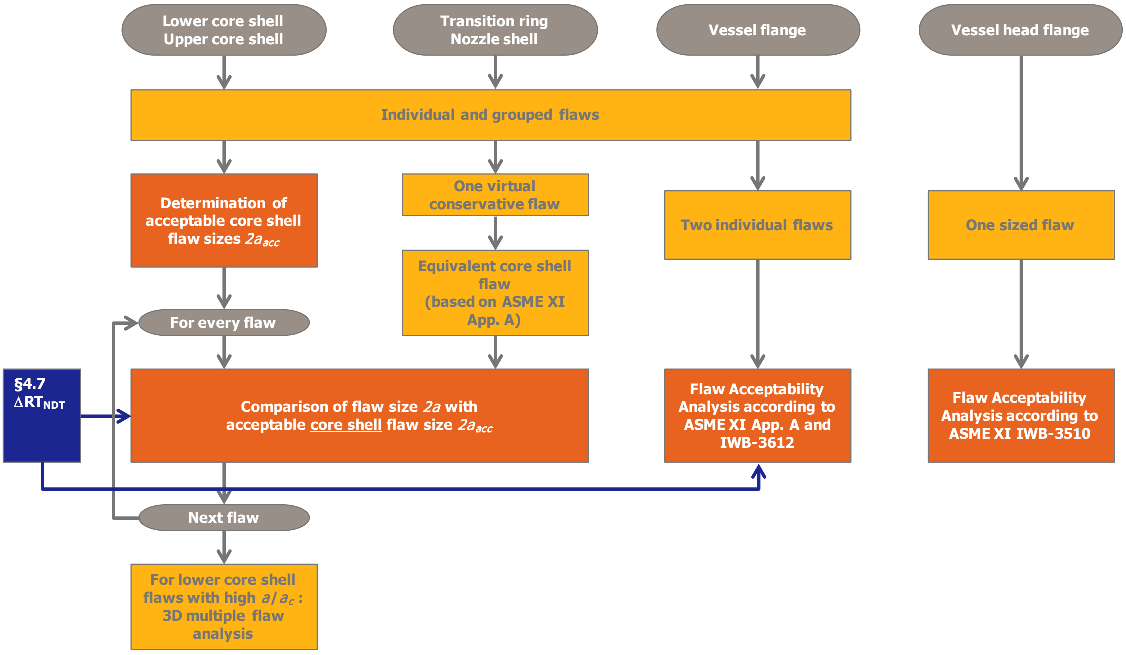


Figure 4.39 Schematic overview of Flaw Acceptability Analysis for Doel 3 RPV

Core shells

For the Flaw Acceptability Analysis of the core shells, a methodology inspired by ASME XI has been developed to determine acceptable sizes of large numbers of nearly laminar flaws.

The first step was to develop a set of curves that enable the assessment of the acceptability of all possible configurations of individual flaws or flaw groups detected in the Doel 3 RPV. As such, parametric analyses were made to cover each parameter that may take any value within a particular range. These parameters are:

- The inclination of the individual flaw or group of flaws with respect to the cylindrical surface of the RPV.

> **A ligament between flakes** is a bond of sound matter between flakes. By sound is meant: with no mechanical discontinuities such as cracks, cavities, etc. It can be evaluated along different directions, when not specified, the direction along which the ligament is the shortest is usually assumed.
>
> A ligament can also stand for the position of the flaw in the direction of the wall thickness of the RPV and with respect to the cladding-base metal interface. In that sense it corresponds to the ASME variable S.

- The RT_{NDT} of the base metal at the location of the flaw. This temperature varies with the position in the RPV in a similar way as the neutron fluence responsible for this temperature shift.
- The ligament (see sidebar) of the flaw or group of flaws, being the position of the flaw in the direction of the RPV wall thickness, measured from the interface between the cladding and base metal.

Each curve gives the acceptable flaw size as a function of the ligament and is valid for a particular combination of inclinations and RT_{NDT} values.

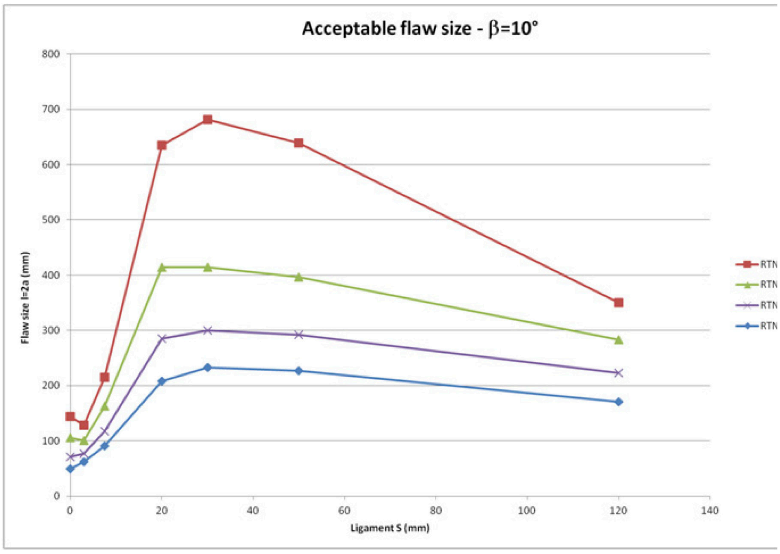


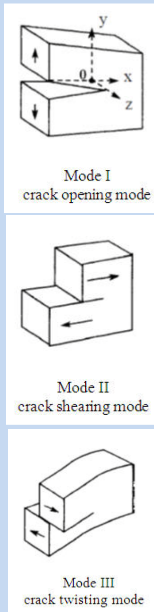
Figure 4.40 Four acceptable flaw size curves for 4 different RT_{NDT} values, and for inclinations up to 10° (ß)

The second step consists of comparing the size (2a) of each individual flaw and of the grouped flaws to its acceptable flaw size ($2a_{acc}$). This acceptable flaw size was determined based on the acceptable flaw size curves, and by considering the inclination and ligament of the flaw and the RT_{NDT} at the flaw's location, evaluated at the end of the RPV service lifetime. As mentioned earlier, the RT_{NDT} at the end of the service lifetime includes an additional shift of 50 °C that covers the effect of macro-segregations on the fracture toughness properties of the base metal.

Finally, the $2a/2a_{acc}$ ratio for all flaws in a particular forging were represented in one single plot showing this ratio as a function of the flaw ligament. The figures below give the results for the lower and upper core shells of the Doel 3 RPV.

How to determine the acceptable flaw size curves? The points of the curves are determined by numerous 3D finite element simulations of the flaw behaviour in the RPV wall. The key parameter is the Stress Intensity Factor (SIF) at the flaw tip. For the most penalizing load cases the RPV may be subjected to, an equivalent SIF Ke is calculated at the flaw tip. This equivalent SIF Ke considers three different modes of deformation that may be triggered by the stress field in the case of nearly laminar flaws: crack opening, crack shearing, and crack twisting (common ASME XI analysis only considers a SIF KI related to the crack opening mode I).

Comparison between the equivalent SIF and the acceptance criteria given in ASME XI IWB-3612 Acceptance Criteria Based on Applied Stress Intensity Factor indicates whether the analyzed flaw is acceptable or not.



Modes of deformation considered in the equivalent Stress Intensity Factor Ke

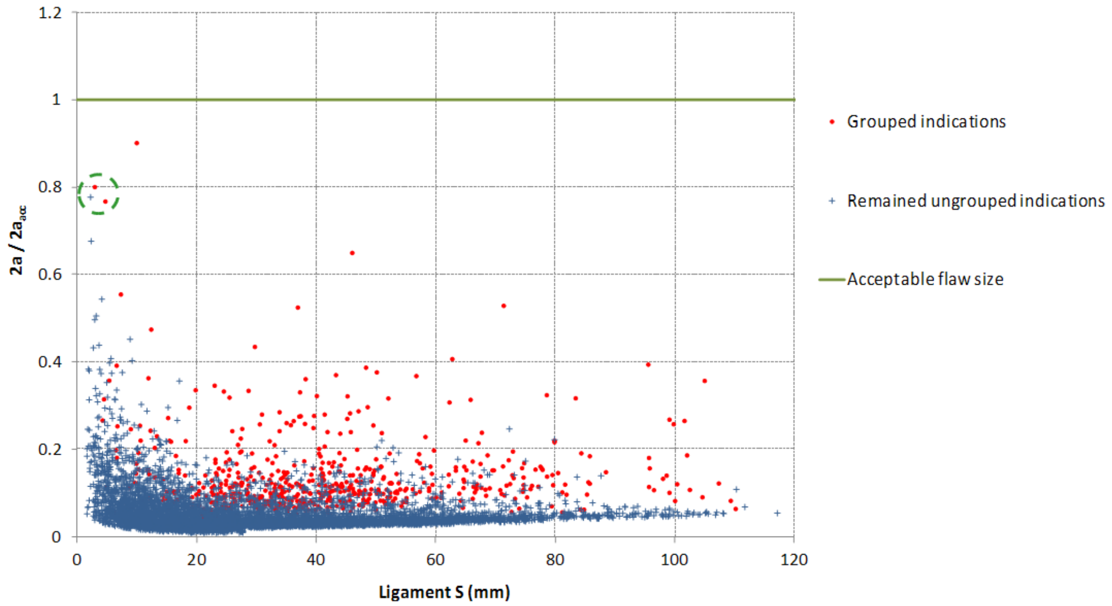


Figure 4.41 Comparison of actual flaw size to acceptable flaw size for lower core shell of Doel 3

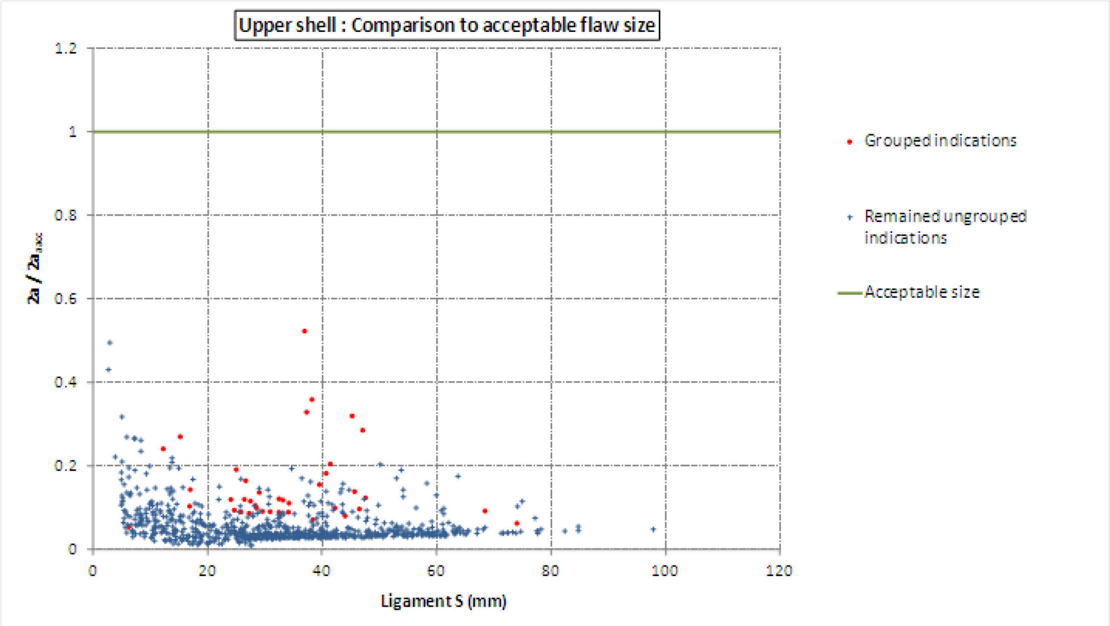
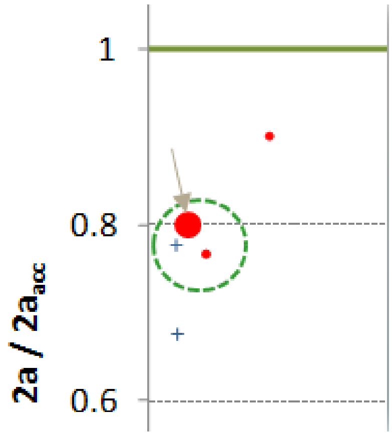


Figure 4.42 Comparison of actual flaw size to acceptable flaw size for upper core shell of Doel 3

Because all flaws have a $2a/2a_{acc}$ ratio below 1, it can be concluded that all flaws encountered in the lower and upper core shell are acceptable.

In order to assess the conservatiness of the grouping methodology, a number of grouped flaws were ungrouped, and a 3D Finite Element analysis was performed covering all individual flaws in the group and their mutual interactions. A typical result of such an analysis for one of the grouped flaws of the lower core shell is given in the figure below. The illustration below shows that the 20 % margin of the grouped flaw increases to more than 70 % when considering a 3D analysis of the multiple flaw configuration.



Detail from figure 4.41

This **correction factor** was obtained by comparing two classical Stress Intensity Factor analyses performed according to Appendix A of ASME XI and based on a 2D Finite Element model of the entire RPV. In the first analysis the virtual conservative flaw was supposed to be located in the core shell, and in the second analysis in the transition ring or nozzle shell. As such, a correction factor of 1.48 was obtained for the transition ring, and 1.18 for the nozzle shell.

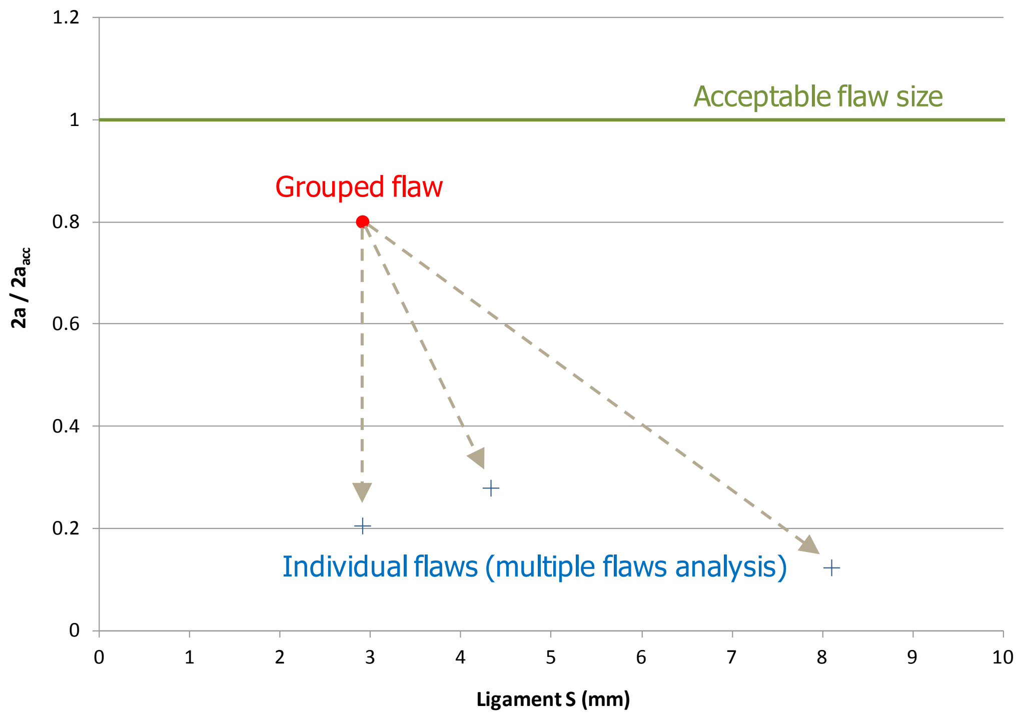


Figure 4.43 Comparison of grouped flaw versus multiple flaw analysis for the Doel 3 lower core shell

Transition ring and nozzle shell

The Flaw Acceptability Analysis developed for the core shells can also be applied to the transition ring and nozzle shell. However, because there are only a few flaws in both components and because both components do not have the purely cylindrical geometry of the core shells, the methodology had to be adapted:

- Instead of analyzing all individual and grouped flaws separately, a virtual conservative flaw was defined enveloping all individual and grouped flaws.

- In order to account for the geometrical differences between the transition ring and nozzle shell on the one hand, and the core shells on the other, the size of the virtual conservative flaw was multiplied by a correction factor (see sidebar) to obtain the equivalent core shell flaw size.

The equivalent core shell flaw sizes for the transition ring and nozzle shell were compared to the acceptable core shell flaw sizes. Both flaw sizes are accepted.

The **p-T curves** are updated to include both the embrittlement of the RPV at the end of its service lifetime and the additional shift in RT_{NDT} of 50°C that covers the effect of macro-segregations on the fracture toughness.

The update is performed based on Appendix G of ASME XI, Edition 1992 and Code Case N-640 Alternative Reference Fracture Toughness for Development of p-T Limit Curves.

The update of the p-T curves leads to more restrictive curves than those that are applicable today.

Vessel flange

Only two flaws were detected in the Doel 3 RPV flange. These flaws remained ungrouped after applying the flaw grouping methodology. Since the vessel flange geometry is quite different from the core shell geometry (due to the presence of holes for the RPV bolting), a classical Flaw Acceptability Analysis according to Appendix A of ASME XI was performed.

In this approach, the axial and circumferential projections of the flaws were assessed separately, which is much more conservative than an analysis that considers the actual orientation. Both flaws are accepted.

Vessel head flange

Three indications were detected in the Doel 3 RVH flange. Two of them were too small to be sized and hence are not addressed as flaws. The third indication is a volumetric indication with a diameter of 5 mm. This indication is acceptable through application of the ASME XI IWB-3500 Acceptance Standards.

Fracture Toughness Requirements

Appendix G of 10CFR50, which is applicable to the Doel 3 RPV, defines the fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors. The goal is to provide sufficient safety margins in any conditions of normal operation, as well as anticipated transient and accident conditions, to which the pressure boundary may be subjected over its service lifetime. As such, 10CFR50 Appendix G refers to the analysis method included in ASME XI Appendix G, which covers the requirements regarding Pressure-Temperature Operating Limits and Low-Temperature Overpressure Protection.

Pressure-Temperature Operating Limits

The pressure-temperature domain in which the reactor can be operated safely, is characterized by the pressure-temperature operating limits given in the form of pressure-temperature (p-T) curves.

The p-T curves have been updated to include both the embrittlement of the RPV at the end of its service lifetime and the additional shift in RT_{NDT} of 50 °C that covers the effect of macro-segregations on the fracture toughness. The update is performed based on Appendix G of ASME XI, Edition 1992 and Code Case N-640 Alternative Reference Fracture Toughness for Development of p-T Limit Curves that allows using KIc (crack initiation fracture toughness) instead of KIa (crack arrest fracture toughness) in the criteria of Appendix G.

The updated p-T curves (see sidebar) and the effects of the update on the plant's operation will be fully integrated into the plant's Technical Specifications (see Chapter 4.9).

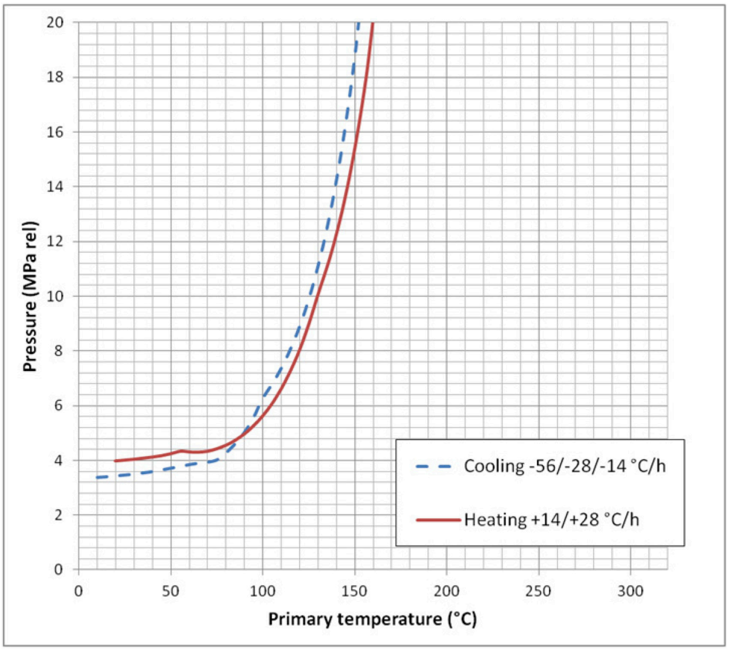


Figure 4.44 Updated p-T curves for Doel 3

Low-Temperature Overpressure Protection

The effect of the updated p-T curves on the Low-Temperature Overpressure Protection is addressed through the analysis of typical Mass Input and Energy Input transients that may lead to overpressures in the Reactor Coolant System (including the RPV) and Residual Heat Removal System at low temperature:

- The analysis of Mass Input transients leads to more stringent operating limits for the Residual Heat Removal System and to a change of the maximum Safety Injection Accumulator's pressure. Both effects will be integrated into the plant's Technical Specifications.
- The analysis of Energy Input transients leads to smaller allowable temperature differences between primary and secondary side temperatures in the steam generators that will be integrated into the plant's Technical Specifications.

Deterministic PTS Analysis

The Deterministic PTS Analysis is restricted to the RPV core shells, since they are the only forgings exposed to neutron irradiation. The analysis has shown that the RT_{NDT} of the base metal will remain below 132 °C at the end of its service lifetime.

The assessment prescribed in 10CFR50.61 'Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events' aims to verify that the (irradiated) RT_{NDT} of the base metal of the Doel 3 RPV forged components will remain below 132 °C at the end of its service lifetime, and that the RT_{NDT} of the circumferential welds at the end of the RPV's service lifetime will remain below 149 °C (300 °F). This assessment was performed only for the two core shells, as they are the only forgings exposed to neutron irradiation.

The RT_{NDT} of the core shells at the end of the RPV's service lifetime was calculated as the sum of the RT_{NDT} evaluated in the framework of the Doel 3 RPV Surveillance Program and the additional RT_{NDT} shift of 50 °C that covers the effect of macro-segregations on the fracture toughness. The maximum RT_{NDT} obtained for the upper core shell is 106.5 °C, which is still below the acceptable value of 132 °C.

For the circumferential weld material of the Doel 3 RPV, there was no additional shift of RT_{NDT} due to macro-segregations to be considered. Hence, the RT_{NDT} at the end of the RPV's service lifetime evaluated in the framework of the Doel 3 RPV Surveillance Program (48.7 °C), remains valid. This temperature is well below the allowable value of 149 °C.

4.8.4 Complementary Probabilistic SIA

During plant operation, the walls of reactor pressure vessels (RPVs) are exposed to neutron radiation, resulting in localized embrittlement of the vessel steel and weld materials in the core area. If an embrittled RPV had a flaw of critical size and certain severe system transients were to occur, the magnitude of the tensile stresses (pressure and thermal stresses) could be sufficient to initiate a through-wall crack, thus challenging the integrity of the RPV. The severe transients of concern, known as pressurized thermal shock (PTS), are characterized by a rapid cooling of the internal RPV surface in combination with repressurization of the RPV. In such a case, the potential existing flaws near the inner surface of the RPV wall could:

- In a first stage, initiate in cleavage fracture, i.e. the so-called crack initiation phase
- In a second stage, propagate through the RPV wall, thus introducing the possibility of RPV failure, i.e. the so-called through-wall crack stage

NUREG-1806

The technical basis for the revised set of PTS rules given in 10CFR50.61a (see chapter 4.2) is described in the NUREG-1806: Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61).

Mass Input transients are related to operational events that result in an attempt being made to increase the water inventory in the reactor coolant system. The compressibility of water is such that pressure increases suddenly. Examples of such transients are inadvertent actuation of High Pressure Safety Injection pumps or inadvertent opening of Safety Injection Accumulator valves.

Energy Input transients are related to operational events during which energy (heat) is suddenly transferred to the reactor coolant system, causing a pressure increase. Such a phenomenon may occur for instance when the average reactor coolant temperature is lower than the temperature of the secondary side of the steam generators before starting the first primary water circulation pump. When the first pump is started, there is not only an immediate pressure peak in the reactor coolant system due to the start-up of the pump, but also a sudden heat transfer from the secondary side of the steam generator to the primary water, which generates a pressure increase in the reactor coolant system.

In order to quantitatively assess the risk of the loss of RPV integrity when subjected to PTS, NUREG-1806 proposes a methodology to calculate two values, each of them characterizing the probability that existing flaws in a RPV can either initiate or become through-wall cracks when a RPV is subjected to a PTS event. These two values are defined respectively as:

- The frequency of crack initiation (FCI)
- The through-wall cracking frequency (TWCF)

The approach for calculating TWCF proposed in the NUREG-1806 is illustrated below (figure extracted from the NUREG 1806). The same methodology is applied to calculate the FCI.

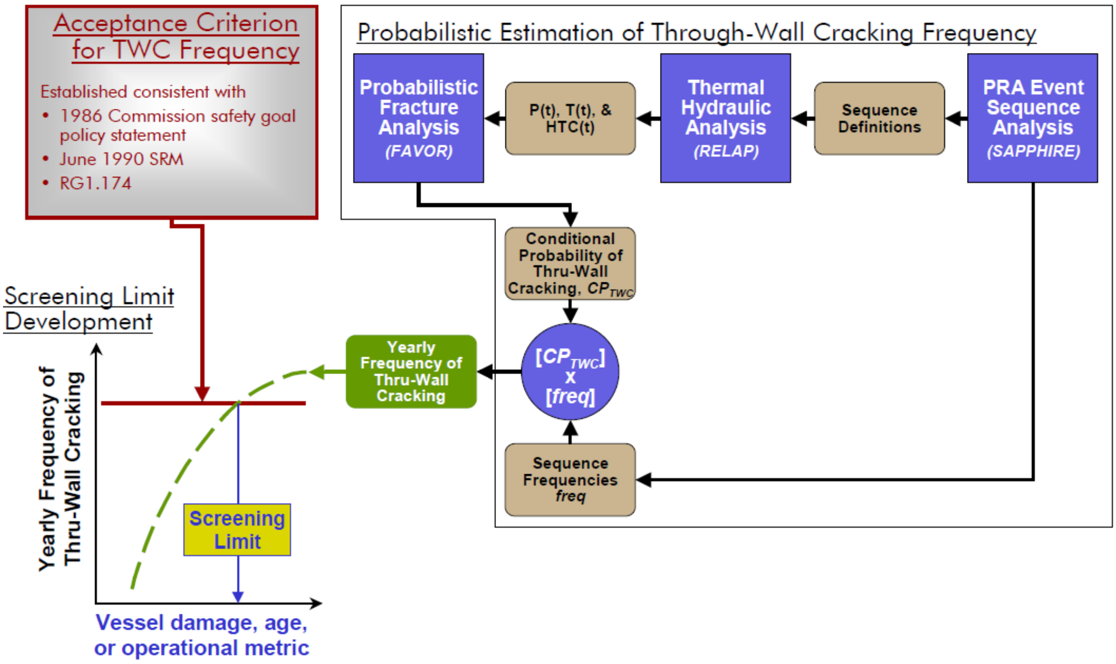


Figure 4.45 Diagram of combination of probabilistic estimate of through-wall cracking frequency (TWCF) and a TWCF acceptance criterion to develop a proposed revision of the PTS screening limit

The probabilistic approach is based on realistic inputs and an explicit treatment of uncertainties. The principle is the following:

- Identify all the sequences of events leading to a PTS-type transient, namely, a transient leading to a rapid cooldown, or a low final temperature, or a high pressure after cooldown. Typical examples are primary breaks of various sizes, spurious opening of a safety valve that recloses after an extended cooldown, or steam line breaks. These transients are defined by their expected occurrence frequency.
- Thermal-hydraulic analyses are then performed to determine the time evolutions of pressure, temperature, and the heat transfer coefficient for each of the scenarios identified in the previous step.
- For each transient defined by pressure, temperature, and heat transfer coefficient history, fracture mechanics analyses are performed with the FAVOR code developed by Oak Ridge National Laboratory (ORNL), considering a statistical defect distribution present in the vessel wall and the material properties distribution. The FAVOR code calculates the conditional probabilities for the initiation (CPI) and for the Through-Wall Cracking (CPF) of postulated existing flaws based on the statistical flaws distribution, for each of these transients. These probabilities are called conditional because they are calculated assuming that the transients effectively occur.
- The total FCI and the TWCF are then obtained by first multiplying the frequency of occurrence of each given transient by its conditional probability of crack initiation and of through-wall cracking and then by summing these results for all considered transients.

The total TWCF is then compared to the 10CFR50.61a criterion, i.e. it must remain below $1x10^{-6}$ per reactor year.

The NUREG-1806 methodology has been developed based on a complete analysis of three specific plants (Beaver Valley Unit 1, Oconnee Unit 1, Palisades). The responses of five additional plants have also been investigated in NUREG-1806 to demonstrate the generalization of the NUREG-1806 conclusion to all PWRs.

Probabilistic assessment for Doel 3

The probabilistic analyses of the Doel 3 RPV have been performed based on the NUREG-1806 report and on the 10CFR50.61a. For the probabilistic fracture mechanics methodology, described in NUREG-1806 and applied in the framework of the PTS evaluation of the Doel 3 RPV, the following data were used:

- The actual materials characteristics and RPV geometry of the Doel 3 RPV. Instead of using the generic flaws distribution database of NUREG-1806, the actual flaw characteristics and distribution in Doel 3 was used. Those flaws are represented by using the same methodology of grouping as used in the deterministic analysis.
- The probabilistic assessment of the Doel 3 RPV structural integrity is based on the PTS events of Beaver Valley Unit 1. These transients were thoroughly developed and detailed in the NUREG-1806. These PTS transients of Beaver Valley Unit 1 are representative of the Doel 3 case for the following reasons:
  - Beaver Valley Unit 1 is a 3-loop Westinghouse 900-type unit, as was the original reference for Doel 3.
  - The reactor coolant loops of Beaver Valley Unit 1 and Doel 3 have the same layouts (one hot leg, one U-tubes SG, one cross-over leg, one primary pump and one cold leg).
  - Concerning the safety injection system which plays a major role in the PTS events, the main characteristics of the Safety Injection (SI) relevant for the PTS scenarios are more penalizing for Beaver Valley Unit 1 than Doel 3 regarding the PTS issue (namely High Pressure Safety Injection pumps with higher shut-off head in Beaver Valley Unit 1 than in Doel 3, which produce higher pressure in the primary circuit during some specific PTS scenarios which in turn lead to more challenging PTS conditions in Beaver Valley Unit 1 than in Doel 3).
- The frequency of PTS events in the Beaver Valley Unit 1 were replaced by the specific frequency of such events in Doel 3.

Moreover, a positive shift of 50 °C in RT_{NDT} is applied to the irradiated RT_{NDT} for Doel 3. This shift of 50 °C is the upper bound of the actual situation of the Doel 3 RPV as evidenced by the experimental investigations (see above). In the framework of this Safety Case, the acceptance criterion of the present assessment is conservatively applied on the FCI instead of the TWCF and becomes: FCI < 10^{-6} per reactor year.

In other words, each crack that initiates is assumed to become through-wall. The result in terms of Frequency of Crack Initiation at End Of Life of Doel 3 amounts to 2.3×10^{-8}, which is far below the acceptance criterion of 10^{-6} defined in 10CFR50.61a.

The probabilistic assessment shows that the Doel 3 RPV with its actual flaw distribution and characteristics exhibits a satisfactory level of structural integrity regarding the PTS solicitation.

4.9 Operational Measures

The safety case assessment confirms that all of the criteria have been met for a safe restart of the Doel 3 reactor pressure vessel (RPV). Nevertheless, Electrabel has commited itself to taking a number of additional operational measures. While some are mandatory, others are voluntary and are being taken to increase the safety margins.

4.9.1 Mandatory Operational Measures

Some modifications must be made to the plant's Technical Specifications (see sidebar) and the corresponding bases to account for:

- The RPV fluence and embrittlement that may have occurred over its 40 years of operation, using the last results of the surveillance programme.
- An additional RT_{NDT} (reference temperature for the transition of brittle to ductile behaviour) shift of 50 °C.

The **Technical Specifications** are a set of operating limits and conditions. The operator must comply with these rules, in order to operate the plant in accordance with the boundary conditions that have been used to validate all transients and accidents as described in the Safety Analysis Report.

10CFR50, Appendix G describes the requirements for the prevention of brittle failure for the primary components' pressure boundary (see Chapter 4.8.2). The re-assessment of the sensitivity to brittle failure is made based on the requirements of the ASME XI, Appendix G (1992) and the Code Case N-640.

The analysis methods mainly concern the pressure-temperature (p-T) operating limits and the Low Temperature Overpressure Protection (LTOP). The p-T curves define the allowable reactor coolant system p-T domain in which the RPV is protected from non-ductile fracture.

Based on the experimental programme of the safety case, an additional 50 °C RT_{NDT} shift is taken into account due to the specific composition of the macro-segregations zone. This decision shows the conservative nature of the calculations, since the maximum additional shift (at the peak fluence location) has been evaluated to be only 17 °C (see Chapter 4.8).

The p-T curves result in a (small) adaptation of the maximum cooling and heating gradients as a function of the average reactor coolant temperature and on the minimum temperature under which the Residual Heat Removal System (RHRS) must be connected:

- Maximum cooling gradient as a function of the average reactor coolant temperature:
  - Tm > 100 °C: -56 °C/h
  - 70 °C < Tm < 100°C: -28 °C/h
  - Tm < 70 °C: -14°C/h (previously -28 °C/h)
- Maximum heating gradient in function of the average reactor coolant temperature:
  - Tm < 50 °C: +14 °C/h
  - Tm > 50 °C: +28°C/h (previously 56 °C/h)
- The limiting reactor coolant system cold leg temperature for the connection of the RHRS is 157 °C. The temperature currently defined in the Technical Specifications is 160 °C.

4.9.2 Voluntary Operational Measures

Electrabel has decided to increase the temperature of the water in the Doel 3 Refueling Water Storage Tanks (RWST) to 30 °C. This modification includes the implementation of a circulation heating system in the safety injection system. As a result, a supplementary safety margin on the acceptable flaw size of about 20 % is obtained (as shown in the figure below), although it is not necessary in order to comply with the criteria.

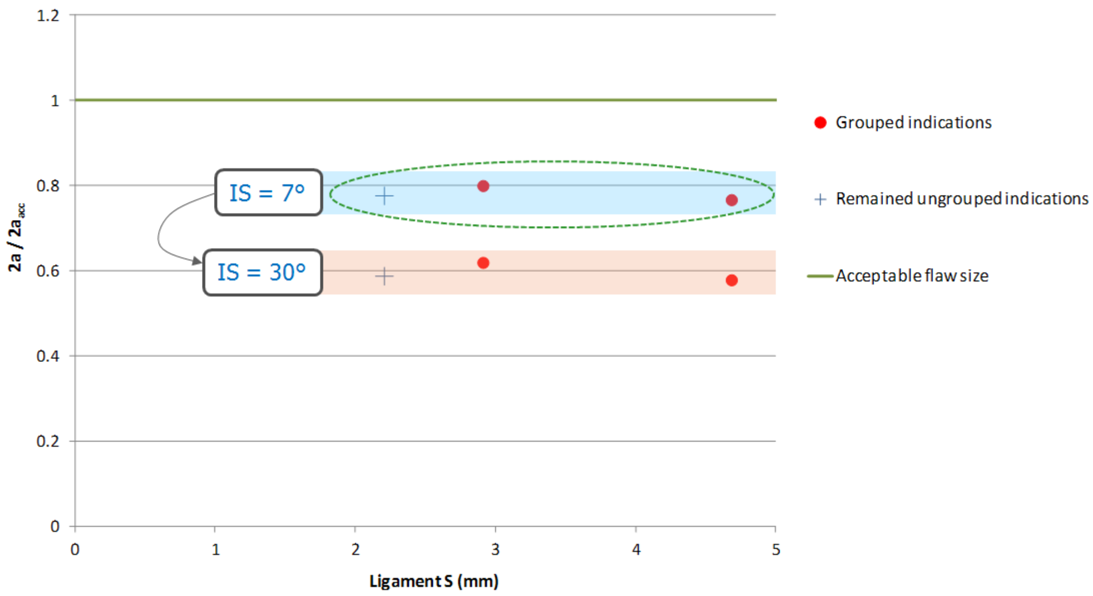


Figure 4.46 This figure shows the impact of temperature of the safety injection water.

One of the transients, known as pressurized thermal shock (PTS), is characterized by a rapid cooling of the internal RPV surface in combination with repressurization of the RPV. This can only be the result of an injection of water from the safety injection system. The temperature gradient of the inner surface is mainly determined by the temperature of the water from the safety injection system (RWST). In the current analysis, a conservative value of 7 °C for the safety injection water is taken into consideration. By increasing the temperature to 30 °C, a supplementary safety margin can be obtained, which is especially useful for the flaws closely located to the RPV's inner wall surface.

5 Conclusions

Electrabel is convinced that the structural integrity of the reactor vessel has been demonstrated, allowing for an immediate restart and a safe operation of Doel 3.

This chapter regroups the main conclusions from this safety case report.

5.1 Detailed Conclusions from the Roadmap

The roadmap: studies and material tests with positive results

To demonstrate the integrity of the reactor pressure vessel under all conditions, the safety case roadmap specified an extensive phase of documentation research, studies, and material tests. This resulted in the following conclusions:

- **Construction in accordance with international codes and standards**. A close review of all of the original manufacturing data and documentation revealed that the Doel 3 reactor pressure vessel was manufactured in accordance with the prevailing international codes and standards, in particular the ASME Boiler & Pressure Vessel Code. All manufacturing inspections required by the construction code were performed and witnessed by the customer and regulatory body and concluded in the acceptance of all parts of the reactor pressure vessels.

  The manufacturing data and documentation proved to be complete, traceable, and in accordance with international codes and standards.

- **Hydrogen flaking confirmed and stable**. The first diagnosis of hydrogen flaking was evaluated based on:

  - An extensive literature study
  - A root cause analysis of all potential causes
  - An evaluation of the possible flaw formation mechanisms
  - A detailed evaluation report of the AREVA metallurgy experts based on the construction files and the shape and size of the indications

  This report was challenged and completed by external experts. As a result, the first diagnosis was confirmed. It was also concluded that the identified indications were stable.

- **UT inspection technique is valid.** The ultrasonic inspection was performed with the automated MIS-B (Machine d'Inspection en Service Belge) equipment, which has been used for over thirty years to inspect the reactor vessels of all Belgian units. The ultrasonic inspection technique that was used to characterize the indications at Doel 3 is state-of-the-art and is used in many nuclear power plants worldwide. It has been qualified for all mandatory inspections and underclad crack detection and sizing against international standards, prior to its use and under the control of the AIA.

  For this specific safety case, the chosen inspection technique was evaluated through cross-checking and extensive destructive tests on a reference block. This block was known to have hydrogen flakes and was taken from an available archive forged shell of equivalent material and size to the reactor vessel shell.

The testing program demonstrated that the applied inspection technique is valid and appropriate for characterizing the types of indications found in the Doel 3 reactor pressure vessel. Moreover, it was shown that the applied inspection technique tends to oversize the dimensions of the indications, making it more conservative.

- **Affected material is sound and with good properties.** In addition to a material-related literature survey, a comprehensive testing program was launched. Many mechanical and metallurgical tests (on more than 400 test samples) were performed in different laboratories (AREVA, Laborelec, SCK.CEN) on archive materials, including a piece of 1.2 m diameter originating from the Doel 3 vessel itself. These tests showed that there is no significant effect of orientation or macro-segregation on fracture toughness. All results confirmed that the curves that the ASME code requires to be used in the assessment are conservative. The destructive tests performed on steel samples with hydrogen flaking also showed that the material between and around the flaws is sound and of a normal metallurgical structure.
- **Structural integrity is confirmed.** After studies and testing, the multidisciplinary team developed detailed methodologies for assessing the structural behaviour of each flaw detected in the vessels shell, in all possible operational modes and transients. These methodologies have been validated after research and were challenged by external experts specialized in fracture mechanics and structural analysis, who confirmed the conservativeness of the methods.

  Based on these methodologies, detailed calculations were made using state-of-the-art modelling and computing techniques, in order to verify the applicable structural integrity requirements. Calculations were performed using conservative data: in particular, very conservative fracture toughness data were used compared to actual material test results. These calculations included the following:

  - Deterministic calculations according to ASME Section III, to assess general stresses in the vessel
  - Deterministic calculations according to ASME Section XI, to demonstrate that the dimensions of every flaw and group of flaws are well below the allowable dimensions, in all operating conditions
  - Probabilistic safety analyses based on the US regulation

  All studies and calculation results have been thoroughly reviewed internally and by external experts and academics. The calculations confirm that the acceptance criteria of the deterministic studies are met with a significant safety margin. The criterion of the probabilistic safety analysis is widely satisfied as well, even under the conservative assumptions.

5.2 Conservativeness

A conservative approach has been taken in each step of the safety case, to ensure high confidence in its conclusions, which were documented in great detail. As a consequence of this conservativeness, additional margins exist between the assessments and reality. Some of them are quantified, some are assessed through sensibility studies, and others remain qualitative.

Conservative approaches are present at different steps of the assessment, namely in:

- The ultrasonic examination technique
- The deterministic structural integrity assessment
- The probabilistic structural integrity assessment

Conservativeness can be categorized in:

- Conservativeness on input data
- Conservativeness of the applied methodology

There are also conservative approaches inherent to and part of applicable codes, norms, and common penalizing assumptions.

The main notable examples of conservativeness are as follows:

- Tests achieved on specimen VB395/1 indicate an over-sizing tendency of UT sizing through the 6 dB drop method. Moreover, a compulsory practice for RPV examination in Belgium attributes to any indication at least the dimension of the sound beam that detected it. This over-sizing results in an under-sizing of the ligaments between flaws.
- The structural integrity assessment has been performed on the basis of the RT_{NDT} evaluated by the FIS formula, with an additional shift of 50°C.
- No irradiation effect was taken into account for the yield strength. By using a lower value for the yield strength, the plastic strains in the vicinity of the crack tip are overestimated, which leads to a conservative value of the J integral and the stress intensity factor K_{eq}.
- Penalizing assumptions were used in defining the temperature and pressure transients for the Small Break and Large Break LOCA. This leads to a conservatively high cooling rate of the vessel wall, which will, in turn, increase the calculated stress intensity in the inner part of the vessel shell.
- Flaw geometry used in the deterministic structural assessment is defined from data provided by INTERCONTROLE (corresponding to the coordinates of the rectangular box that surrounds the entire indication). This definition process includes three conservative assumptions:
  - The flaw is circular with a diameter 2a taken as the largest diagonal of the box faces
  - The angle ß of this circular flaw is taken as the highest inclination of these diagonals
  - The ligament (distance to the inner Clad-base metal interface) is taken as the shortest distance from the box
- Several conservative assumptions are also present in the deterministic structural integrity assessment methodology:
  - Conservative formula for calculating J and K_{eq}.
  - The grouping criteria are established on 2D analyses only. 3D calculations performed for some challenging configurations demonstrate that the ratio of each flaw size to its acceptable size is at least a factor of 3 smaller than the ratio of the considered configurations

5.3 General Conclusion

As documented and further demonstrated in this safety case report, all studies and calculations are solid and verify that all safety criteria in the structural integrity assessment of the reactor vessel is met, for each detected flaw, with significant margins.

The applied methodologies include demonstrated conservativeness at each step and margins on the used data. In particular, conservative fracture toughness data were used compared to actual material test results.

All studies and calculations were subject to a rigid review process. They have also been validated by external experts.

Consequently, having thoroughly assessed the roadmap results, Electrabel is convinced that the integrity of the reactor vessel has been demonstrated, allowing for an immediate restart and safe operation of Doel 3.

5.4 Action Plan

The following actions will be implemented before restart, on top of the existing operational measures:

- Electrabel will reduce the authorized heat-up and cool-down gradients during start-up and shut-down operations. This will reduce the thermal and pressure loading on the reactor pressure vessel even more during normal operation.
- Guided by its nuclear safety culture, Electrabel decided to implement a permanent preheating of the safety injection water reservoirs of Doel 3 to 30°C. This measure is not necessary, given the results of the structural integrity assessment; nevertheless, it will add a 20 % margin to the acceptable flaw size close to the vessel's inner surface.
- All operators of the Doel 3 unit had a refresher training session on the full scope simulator in the last quarter of 2012. An extended briefing will be given to all shift personnel about the start-up and changes in the operational parameters and specifications.

Operators training

Operations staff have a bi-annual operator license. This license needs to be renewed after extensive classroom courses, simulator sessions, and on-the-job training. During this four-week training, plant operation under normal conditions and during transients and accidents are covered. On the simulator, the operators are trained on starting and shutting down the reactor and managing transients, incidents, and accidents.

Since the shutdown of the unit, operations are limited to:

- Monitoring of the spent fuel pools
- Monitoring of the safety systems
- Continue the normal surveillance program, as required by the Technical Specifications
- Day-to-day maintenance activities

After the long period of reduced operations, and conforming to Electrabel's safety culture, all Doel 3 operators had refresher courses on the simulators. This one-week training focused on start-up / shutdown and normal operations.

Prior to starting up the reactor, a comprehensive plant status briefing will be organized, emphasizing possible start-up transients and including a thorough explanation of the adapted operational measures.

Qualification process

Qualification is required by the ASME XI Code.

The qualification process follows the methodology developed by the European Network for Inspection & Qualification (ENIQ), an organization of utilities established in the 1990s under the auspices of the European Commission.

The process is governed by the rules set out by the licensee's qualification body (EQB) and is supervised by the Authorized Inspection Agency (AIA).

Future inspection program

At the end of the first cycle, the same inspection of the entire reactor pressure vessel wall thickness will be performed. Therefore, a program for extending the qualification of the MIS-B equipment will be launched under the supervision of the licensee's qualification body and the AIA. The basic objective of the qualification extension is to document the flaw detection and characterization capability on a block of an AREVA shell known to contain hydrogen flakes.

Qualification of Ultrasonic Testing Procedures

The ultrasonic testing procedures applied to examine the Doel 3 RPV are formally qualified for their original scope, i.e. RPV weld examination and the underclad cracks examination.

Since the nature of the flaws of interest was not known prior to the 2012 in-service inspection, no formal qualification could have been organized before. These examinations should therefore be formally regarded as expertise interventions. The extension of qualification is planned to take place in the next future, still in the spirit of the ENIQ methodology. According to the licensee's procedures, this will be done by the licensee's qualification body (EQB) and the supervision of the Authorized Inspection Agency (AIA). The objective of the qualification extension is to document the detection and sizing capability of laminar or nearly laminar flaws.

Practically, a second sample, referred to as VB395/2, will be extracted from the VB395 shell. The sample location was chosen to contain a large and representative number of hydrogen flakes over 1.5 m in the circumferential direction and over 1 m in the shell axis direction.

As the manufacturing process of the VB395 shell was interrupted following the discovery of flaws, some preparatory works are needed to fully reproduce the manufacturing sequence and the thickness of the Doel 3 RPV components:

- Quenching & tempering (to be preceded and followed by standard manual ultrasonic testing)
- Machining of inner and outer surfaces to the relevant thickness (200 mm)
- Stainless steel clad deposit on the inner surface, with a final surface finish similar to that of Doel 3

Sample VB395/2 will be subjected, among others, to an ultrasonic inspection carried out under the same conditions as during the Doel 3 in-service inspection. Collected data will be analyzed in the same way as during the 2012 inspections.

As in the case of sample VB395/1, a number of flaw indications will be selected for extraction and detailed destructive analysis. Comparison of actual and ultrasonic results will determine quantitatively the field capability of the examination method and equipment. All ultrasonic testing conditions and results will be included in the final qualification file.

Material testing

The very comprehensive material and metallurgical literature research and testing program, especially on archive materials of the Doel 3 reactor vessel, demonstrated that the material between the flaws is sound and of a normal metallurgical structure. There are no effects of orientation and segregation.

Consequently, the conservative material properties used in the structural analysis are more than adequate to cover any local effect or peculiarity and do not need to be further investigated.

In addition, Electrabel will launch a confirmatory testing program on materials from the block of the AREVA shell that contains hydrogen flakes. This program, still to be finalized with the safety authorities, will encompass two phases:

- **In the short term** (about 4 months), Electrabel, together with SCK.CEN, will perform a test program on a series of small-scale tensile and fracture toughness specimens located in two zones: one in material out of the segregated zone, and the other in material located between flakes (ligaments) in the segregated zone. The objective is to assess the conservativeness of the ligament’s mechanical properties that were used in the structural assessment.
- **In the medium term** (about one year), Electrabel will conduct a test of large tensile specimens containing hydrogen flakes in an orientation comparable to the orientation of the indications in the Doel 3 reactor pressure vessel. The objectives are:
    - To confirm that the flakes in a nearly laminar orientation do not significantly affect the load-bearing capacity of the specimen
    - To assess the conservativeness of the structural integrity assessment method
    - In order to be representative, the specimens need to be of large dimensions, which requires more time for preparation and execution