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SCK•CEN-BLG-1066

Status and Perspectives of Nuclear Reactor Pressure Vessel Life Extension up to 60 Years Operation in Belgium

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September, 2009

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Executive summary

The GEMIX group is evaluating different energy policy options, including those that would require a modification of the 2003 Belgian Act, to allow extending the license of commercial Belgian Nuclear Power Plants. In this context, this document was prepared in support of the GEMIX report, in order to provide an expert opinion on the technical feasibility of extending the operating license of Belgian reactors beyond 40 years of operation.

The scope of this report is limited to a safety evaluation of the reactor pressure vessel (RPV) against neutron embrittlement, in the most severely irradiation region (beltline) and in the event of a pressurized thermal shock. The irreplaceable RPV is considered to be the most critical component for lifetime considerations of the nuclear power plant. However, an application for operation extension will also depend upon a number of additional considerations, including the technical assessment of other plant components, as well as non-technical arguments (e.g. political, environmental, economical, strategical…) that are outside the scope this report. In the hypothesis of a request for operation extension, it is the responsibility of the utilities to provide the safety authorities with an exhaustive dossier demonstrating that safe extended operation is guaranteed. The role of the safety authorities is to critically evaluate the safety dossier for eventually granting the operation extension.

Belgium is certainly not the only country in the world presently evaluating the possibility of extending the lifetime of its nuclear power plants. Indeed, this report gives a comprehensive overview of what is taking place in other countries worldwide with respect to operation extension. Moreover, a comparison is presented between Belgian reactors and similar US units that have been granted license extensions up to 60 years of operation by the American safety authorities. Among the 54 reactors that have already been granted a 60 license extension, 8 units present similarities with the Belgian plants.

Different concepts are needed to understand the regulatory aspects for the evaluation of RPV embrittlement. The assessment of reactor pressure vessel embrittlement is one of the expertise fields of SCK•CEN. All surveillance capsules that are used to monitor the vessel materials' degradation were tested at SCK•CEN. Moreover, beside the regulatory tests, for a number of years we have been applying the so-called "advanced surveillance" approach in order to better characterize the materials. Indeed, under the support of the utilities and of the Belgian government, advanced tools were developed at SCK•CEN not only to improve the physical understanding but also to enhance the safety assessment of the reactors. The analysis of all available Belgian surveillance data has also shown that the regulatory approach is often conservative, and additional safety margins can be identified.

Finally, although the RPV safety assessment can also be based on analytical embrittlement curves according to the legislation in force, by 2012 surveillance data corresponding to 60 or more years of operation will be available for all Belgian units (at the time of writing, such data are already available for Doel II). These are expected to confirm the favorable trends shown by our current regulatory analyses.

Keywords

GEMIX, nuclear power plant, license extension, reactor pressure vessel, embrittlement, safety assessment, surveillance program, advanced surveillance approach.

Nomenclature/Glossary

AGR Advanced Gas-cooled Reactor
Second generation of British gas-cooled reactors, using graphite as the neutron moderator and carbon dioxide as coolant.

ASN Autorité de Sûreté Nucléaire (French Nuclear Safety Authority)

ARN Autoridad Regulatoria Nuclear (Argentinian Nuclear Safety Authority)

ART Adjusted Reference Temperature
Ductile-to-brittle transition temperature calculated by adding to the unirradiated value of RT_{NDT} (see), *the variation* ΔRT_{NDT} *caused by irradiation* (see) *and a margin term that accounts for experimental uncertainties.*

ASTM American Society for Testing and Materials (now ASTM International)
International standards organization that develops and publishes technical standards for a wide range of materials, products, systems, and services.

Beltline *In the reactor pressure vessel of a nuclear power plant, region which is nearest to the core, and therefore most heavily irradiated.*

BR2 Belgian Reactor 2
Material Testing Reactor located in Mol and operated by SCK•CEN.

BR3 Belgian Reactor 3
Located in Mol, it was the first PWR (pressurized water reactor) in Western Europe and it was also the first to be decommissioned.

BWR Boiling Water Reactor
Type of nuclear reactor where the heat produced by nuclear fission in the reactor core causes the cooling water to boil, producing steam, which is directly used to drive a turbine.

C1L/C2L *Denomination of the two base metals of the Tihange I unit*

Charpy test *It is a standardized high strain-rate (impact) test which determines the amount of energy absorbed by a material during fracture. This absorbed energy is a measure of a given material's toughness and acts as a tool to study temperature-dependent brittle-to-ductile transition. Test specimens are small bars, 55 mm long and with square cross section (10 mm × 10 mm), with a 2 mm-deep notch on one face. This test is included in the nuclear codes and regulations currently in force.*

CNEA Comisión Nacional de Energia Atómica
Argentinian governmental agency whose mission is the development and control of nuclear energy for peaceful purposes in Argentina.

CRIEPI Central Research Institute of Electric Power Industry
Comprehensive research organization for the electric utility industry in Japan.

CRP Coordinated Research Project
Research project, launched and financed by IAEA, that brings together research institutes in both developing and developed Member States to collaborate on the research topic of interest.

CSN Consejo de Seguridad Nuclear
Spanish Nuclear Safety Authority.

DOE Department of Energy (United States)

EDF Electricité de France
The main electricity generation and distribution company in France.

ENSI Swiss Federal Nuclear Safety Inspectorate

EPR European (or Evolutionary) Pressurized Reactor
Third generation pressurized water reactor, designed and developed mainly by Framatome (now Areva NP) and Electricité de France (EDF) in France, and Siemens AG in Germany.

EU European Union

IAEA International Atomic Energy Agency
International organization that seeks to promote the peaceful use of nuclear energy and to inhibit its use for military purposes.

INAP *Brazilian test reactor*

KAERI Korean Atomic Energy Research Institute
The sole professional research-oriented institute for atomic energy in South Korea.

KKL Kernkraftwerk Leibstadt
BWR reactor located in Leibstadt (Switzerland)

KKM Kernkraftwerk Mühleberg
BWR reactor located in Mühleberg (Switzerland)

KKP1 Kernkraftwerk Philippsburg/Rhine unit 1 (Germany)

KWO Kernkraftwerk Obrigheim (Germany)

LTO *Long Term Operation of nuclear power plants*

LWR Light Water Reactor
Nuclear power reactor type that uses light water (H_2O) as coolant and moderator; includes both PWR and BWR.

Magnox *Nuclear power reactor designed and is still in use in the United Kingdom. It was exported to other countries, both as a power plant, and, when operated accordingly, as a producer of plutonium for nuclear weapons. The name Magnox comes from the alloy used to clad the fuel rods inside the reactor.*

Master Curve *Analytical approach which allows analyzing a limited number of fracture toughness results, in order to derive the statistical distribution of the fracture toughness for a steel in the ductile-to-brittle transition region. It is presently standardized in the ASTM E1921 test standard.*

NA-SA Nucleoeléctrica Argentina S.A.
Argentinian government-owned company that manages the Atucha I and Embalse nuclear power plants and is responsible for the finalization of the Atucha II plant.

NPP *Nuclear Power Plant*

NRC National Regulatory Commission

US governmental commission that regulates commercial nuclear power plants and other non-military uses of nuclear materials, through licensing, inspection and enforcement of its requirements.

OECD Organization for Economic Co-operation and Development
International organization of 30 countries that accept the principles of representative democracy and free-market economy.

ORNL Oak Ridge National Laboratory
Multiprogram science and technology national laboratory managed for the United States Department of Energy by UT-Battelle.

PHWR Pressurized Heavy Water Reactor
Nuclear power reactor that uses heavy water (deuterium oxide D_2O) as its coolant and moderator. The heavy water coolant is kept under pressure in order to raise its boiling point, allowing it to be heated to higher temperatures without boiling, much as in a PWR.

PM1/PM2 *Denomination of the two base metals of Doel I and Doel II units*

PSR Periodic Safety Review
Systematic safety reassessment of a nuclear power plant, to assess the cumulative effects of plant ageing and plant modifications, operating experience, technical developments and siting aspects.

PTS Pressurized Thermal Shock
It is the shock experienced by a thick-walled vessel due to the combined stresses resulting from a rapid change of temperature (cooling) and/or pressure.

PWR Pressurized Water Reactor
Type of nuclear reactor where the primary coolant loop is superheated water under high pressure to prevent film boiling within the reactor.

R&D Research & Development

RBMK High Power Channel-type Reactor (in Russian)
Light water graphite-moderated nuclear power reactor built in the Soviet Union for use in nuclear power plants to produce nuclear power from nuclear fuel.

RPV Reactor Pressure Vessel
In a nuclear power plant, it contains the fuel and is made of thick steel plates or forgings that are welded together.

RT_{NDT} Reference Temperature for Nil Ductility Transition
For a steel, the ability to absorb energy during impact loading decreases with temperature. At a specific temperature, called RT_{NDT}, the ductility may suddenly decrease to almost zero and the material behaves in a fully brittle (fragile) manner.
In the unirradiated condition, RT_{NDT} is determined from impact (drop-weight and Charpy) test results. In the irradiated condition, RT_{NDT} is obtained by adding to the unirradiated value the variation of T_{41J} (ΔT_{41J}, see below) caused by irradiation.

ΔRT_{NDT} *Shift (variation) of the Reference Temperature for Nil Ductility Transition, caused by neutron irradiation*

RT_{To} *Alternative definition of RT_{NDT} (see), obtained by using direct fracture toughness measurements instead of Charpy impact test results*

SCK•CEN Studie Centrum voor Kernenergie / Centre d'Etudes Nucléaires
Belgian Nuclear Research Centre (http://www.sckcen.be)

STUK Radiation and Nuclear Safety Authority in Finland

T_0 *Temperature which corresponds to a fracture toughness reference level of 100 MPa√m, as determined using the Master Curve procedure*

ΔT_0 *Shift (variation) of the T_0 temperature, cause by neutron irradiation*

T_{41J} *Temperature which corresponds to an absorbed energy level of 41 J during Charpy impact testing. It is used for nuclear reactor pressure vessel steels, as an index of the transition from ductile to brittle behavior.*

ΔT_{41J} *Shift (variation), caused by neutron irradiation, of the temperature which corresponds to an absorbed energy level of 41 J during Charpy impact testing*

VVER-440 *Older type (before 1970) water-cooled, water-moderated energy reactor developed in the Soviet Union and used by Armenia, Bulgaria, China, Czech Republic, Finland, Hungary, India, Iran, Slovakia, Ukraine and the Russian Federation. (Also: WWER-440)*

VVER-1000 *Newer (after 1975) and larger type of VVER reactor. (Also: WWER-1000)*

1 Introduction and Scope

The objective of this report, prepared in support of the GEMIX group, is to provide an expert opinion on the technical viability of the reactor pressure vessels (RPV's) of the Belgian units for extended operation up to 60 years. It is important to emphasize that although safe operation of a Nuclear Power Plant (NPP) does not rely solely on this single component, the RPV is considered to be the most crucial component in the determination of the lifetime.

Historically, a 40 years license term for NPP's was selected by the American congress not on a technical basis but because this time period corresponds typically to the amortization period for an electrical power plant. Within the USA the 40 years license therefore represents the so-called end-of-life definition for the NPP [1-1].

In Belgium and in most European countries, end-of-life is not defined as such, and the license for plant exploitation is granted every time given for 10 years after being subjected to a Safety Assessment by the regulatory body [1-2]. As such, in the European perspective, the actual lifetime of the power plant can be potentially much higher than 40 years, on condition that safety can be guaranteed. This implies that all maintenance, parts replacement, periodic inspections, and any other required action are taken into account to ensure the safety of the plant.

As will be shown in this report, the development of an extended operation scheme for a nuclear power plant is a large project that is currently undertaken by several utilities around the world. Such projects, that have to meet the standards imposed by the safety authorities, generally include:

- a comprehensive evaluation of all components that can be potentially affected by ageing or obsolescence (including the RPV),
- the demonstration of the long-term safety of the nuclear power plant,
- the assessment of the impact of new requirements from the regulation,
- an investment plan including economical and social aspects.

In such projects, it is the role of the utilities to submit an exhaustive dossier to the safety authorities who have the responsibility to evaluate it and eventually grant extended operation.

In this report we focus on the integrity and safety assessment of the reactor pressure vessel, on account of the following three fundamental reasons.

- The vessel plays a major role in the safety of the plant as a confinement barrier and for ensuring the cooling of the core.
- It suffers degradation (embrittlement) due to neutron irradiation and thermal ageing. This degradation is monitored by means of a "surveillance program" (see Section 4) using mechanical specimens made of the vessel beltline[1] materials, that were inserted in the RPV when the reactor was put in operation.
- The replacement or damage recovery of the vessel is not feasible for technical and economic reasons, and therefore this component actually determines the life of the plant.

Indeed, the only part of the vessel that can be replaced is the vessel head (top cover). Although this part is not subjected to neutron embrittlement, stress corrosion problems have been reported and have led to its replacement (in Belgium for Tihange I and in the US for the Davis-Besse power plants, see [1-1]). These aspects should also be carefully considered for the Belgian NPP's, in the framework of extension of operation time, and a specific follow-up program for the Belgian vessel heads is in place.

In general, it should be stated that the operation extension is coupled to an investment plan which has to ensure the safety of the plant. Logically, the longer the extension period is expected to be, the more substantial the long-term investments will be.

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[1] Materials from the most heavily irradiated region of the vessel (see also definition of "beltline" in the Nomenclature section on page 3).

Based on the previous considerations, it is clear that the operating life conditions of the plant are intrinsically linked to the justification of the use of the vessel.

More specifically, the aspect investigated in this report is the region of the RPV subject to the most intense neutron irradiation, the so-called "beltline", and in particular its safety in case of the most severe accidental condition, the Pressurized Thermal Shock (PTS). In this accidental scenario, cold water is injected in the core, which is located at the center of the RPV. This cools the vessel, while the pressure inside is maintained. The combined effect of internal pressure in the vessel containment, thermal stresses on postulated flaws and low temperatures in the RPV wall, combined with material degradation (embrittlement) due to long term neutron irradiation, could lead to the brittle rupture of the vessel [1-3]. This situation has to be avoided at all costs, and for this reason specific codes and regulations have been issued to deal with these circumstances.

In Section 2 we present an overview of the international situation on power plant operation extension. As will be shown, Belgium is not a particular case with respect to nuclear power plant operation extension. It is also interesting to identify some of the US units that are similar to Belgian reactors and examine their situation with respect to license renewal and extension; this will be treated in Section 3.

The basic principles of a surveillance program, which monitors the evolution of the material properties of the RPV, are outlined in Section 4.

Since SCK•CEN is very active in the domain of RPV integrity assessments, its internationally recognized expertise will be presented in Section 5.

Such expertise is available for our government as well as for our safety authorities, and is applied for an assessment of the Belgian reactor pressure vessels in Section 6.

Finally, conclusions are provided in Section 7.

1.1 References

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[1-2] U.S. Nuclear Regulatory Commission, Office of the Inspector General, Semiannual Report to Congress, October 1, 2006 – March 31, 2007, NUREG-1415, Vol.19, No.2 (http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1415/v19n2/sr1415v19n2.pdf)

[1-2] R. Gérard, "Survey of National Regulatory Requirements," AMES Report No.4, EUR 16305 EN, 1995, European Commission DG XI/C/2.

[1-3] Department of Energy Fundamentals Handbook, Material Science, "Pressurized Thermal Shock", MS-03, page 6, DOE-HDBK-1017/2-93. (http://www.tpub.com/content/doe/h1017v2/css/h1017v2_24.htm)

2 Worldwide scenario with respect to NPP operation extension and license renewal

2.1 Introduction: current outlook of nuclear power energy in the world [2-1,2-2,2-3]

In the world, there are currently 436 commercial nuclear power reactors operating in 30 countries, with a total capacity of 372 GW, providing about 15% of the world's electricity.

Sixteen countries depend on nuclear power for at least one quarter of their electricity. France gets around three quarters of its power from nuclear energy, while Belgium (58%), Hungary, Lithuania, Slovakia, South Korea, Sweden, Switzerland, Slovenia and Ukraine get one third or more. Japan, Germany and Finland get more than a quarter of their power from nuclear energy, while the USA gets almost one fifth.

As of 30 June 2009, 48 additional NPP's are in construction in 15 countries (13 of which in China, 8 in Russia and 6 in India) with an installed capacity of 42 GW. Additionally, some 16 countries with existing nuclear power programs have plans to build new power reactors (beyond those already under construction).

The International Atomic Energy Agency (IAEA) has significantly increased its projection of world nuclear generating capacity [2-4], anticipating now at least 70 new plants in the next 15 years, totaling 470 to 750 GW in place by 2030 - 27% to 103% more than actually operating in 2008. OECD estimates range up to 680 GW in 2030. The change is said to be based on specific plans and actions in a number of countries, including China, India, Russia, Finland and France, coupled with the changed outlook due to the Kyoto Protocol, available reserves in terms of fossil fuel and energy security. This would give nuclear power a 17% share in electricity production in 2020, with the fastest growth taking place in the Asian continent (China, India, South Korea, Japan).

In some countries, increased nuclear capacity is resulting from the uprating (i.e. increase of power) of existing plants: examples are several operating reactors in USA, Belgium, Sweden, Spain, Switzerland, Finland and Germany.

To date, 122 commercial power reactors have been retired from operation [2-5], based on economical, political or safety (technical) reasons. Many of these reactors are of very specific design, and cannot be directly compared with the Belgian units.

2.2 Situation of plant operation extension

The world's fleet of nuclear power plants is, on the average, more than 20 years old. Originally, most nuclear power plants had a nominal design lifetime not exceeding 40 years, but engineering assessments of many plants over the last decade have established that many can operate longer in a completely safe manner. The original 40-year term for reactor licenses was not imposed by limitations of nuclear technology, but rather determined by economic considerations. It can therefore be expected that many plants will be able to operate in excess of their design lives, provided that nuclear power plant operating companies demonstrate that the plant will operate safely, by analysis, trending, equipment and system upgrades, increased vigilance, testing and ageing management.

The technical and economic feasibility of replacing major reactor components, for example steam generators in Pressurized Water Reactors (PWR's) and pressure tubes in CANDU heavy water reactors, has been demonstrated [2-6]. However, if replacing the vessel became necessary, owners would have to weigh the project costs, including the time the reactor is out of service, against other generation options. Indeed, it could be more cost efficient to order a new reactor, and for this reason the RPV is considered irreplaceable: it represents the component that determines the lifetime of the whole plant. Other structures can be considered equally as critical as the RPV for its "survival", such as the concrete containment building or the loop piping. Moreover, in some cases seismic considerations (risks to the stability of the building and safety shut-down of the reactor in case of severe earthquakes, as in the case of several Japanese plants) have justified the definitive closure and decommissioning of a plant.

The following subsections will briefly describe the status of NPP license renewal plans in different countries, with particular emphasis on the two cases which are most relevant to the Belgian situation: the United States (some reactors have identical or similar design as the Belgian units) and France (some Belgian RPV's were fabricated according to French specifications).

2.2.1 *United States* [2-7]

The USA has 104 nuclear power reactors distributed in 31 states and operated by 30 different power companies. Of these, 69 are Pressurized Water Reactors (PWR's) like the Belgian units and the remaining 35 Boiling Water Reactors (BWR's). In 2008, NPP's were responsible for almost 20% of the total energy generated. Of the 104 reactors, 69 are of PWR type and 35 of BWR type. They were all built between 1967 and 1990.

In 1974, the Nuclear Regulatory Commission (NRC) was established as a government agency responsible for the regulation of the nuclear industry, notably reactors, fuel cycle facilities, materials and wastes, as well as other civil uses of nuclear materials.

In March 2000, the NRC renewed the operating licenses of the two Calvert Cliffs units for an additional 20 years. As of June 2009, the NRC has extended from 40 to 60 years the licenses of 54 reactors, more than half of the US total. Currently, the NRC is examining license renewal application for 16 more units, while more than 15 additional applications are expected to be submitted by 2013. All or nearly all the operators of current reactors are expected to apply for 20-year extensions, by submitting to NRC general and technical information in compliance with specific prescriptions given in 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants".

Recently, the NRC has launched a new oversight and assessment procedure for nuclear plants, yielding publicly-accessible information on the performance of plants in 19 key areas: 14 indicators on plant safety, 2 of radiation safety and 3 on security. Performance against each indicator is reported every three months on the NRC website (http://www.nrc.gov/) according to whether it is normal, attracting regulatory oversight, provoking regulatory action, or unacceptable (in which case the plant will probably be shut down).

Overall, the situation of the US nuclear fleet at the time of writing can be summarized as follows:

- 54 reactors have been granted 20-year license extension;
- 21 reactors are currently filed for license renewal;
- 23 reactors are expected to apply for license renewal.

Only 6 plants are currently expected to close down after 40 years of operation, without applying for license renewal, based on grounds of economical viability.

Currently, the Department of Energy (DOE), the NRC and industry are starting to consider what research needs to be conducted to determine the feasibility of keeping reactors operating beyond the 60 years timeframe ("life beyond 60"), for example up to 80 years operation [2-8]. Technical results of the research will indicate what the risks are for a reactor vessel of a given age, which would then trigger a policy decision on whether those margins are acceptable or not.

2.2.2 *France* [2-9, 2-10]

France has 59 nuclear reactors, all of PWR type like the Belgian units and operated by Electricité de France (EDF), with total capacity of over 63 GW, supplying 78% of the total electricity generated in the country.

The present situation stems from the decision of the French government in 1974 (just after the first oil crisis) to expand rapidly the country's nuclear power capacity. This decision was taken in the context of France having substantial heavy engineering expertise but few indigenous energy resources. Nuclear energy, with the fuel cost being a relatively small part of the overall cost, made good sense in minimizing imports and achieving greater energy security.

In France there is no regulatory lifetime and a NPP can operate indefinitely, as long as all safety requirements are met. The National Safety Authority (ASN) can stop the operation of a reactor in case of danger to public safety or for protecting the environment.

French nuclear power plants were designed for an operating life of 40 years, that was considered the operating limit for critical (irreplaceable) components, such as the pressure vessel and the containment wall.

In the mid-90's, EDF launched, in collaboration with ASN, a project for the management of ageing phenomena concerning the reactors having reached their third 10-year inspection, or 30 years of operation.

For all 34 French reactors of the 900 MW class, which went into operation between 1977 and 1987, the periodic safety review (PSR) was launched in 2002 and concluded at the end of 2008; as a result, all the reactors had their lifetimes extended by 10 years beyond their third 10-year outage (30 years of operation). The oldest 18 will reach the 40-year mark between 2015 and 2020. EDF recently announced its plans for a further operation extension to 60 years [2-11].

The younger French reactors include 20 units of the 1300 MW class and 4 units of the 1500 MW class. In October 2006, ASN cleared the 1300 MW units to run for another 10 years, provided some modifications are made during their 20-year outages, which are planned in the period 2005-2014.

In France, 12 power reactors have been shut down and are being decommissioned. Among them, only the small prototype 305 MW Chooz-A unit was a PWR reactor. The remaining 11 were either Gas-Cooled Reactors (9), Gas-Cooled Heavy Water Reactors (1) or Fast Breeder Reactors (1).

2.2.3 Other European countries [2-12, 2-13]

In the *Czech Republic*, a 10-year extension to 40 years is under consideration for the four units of the Dukovany power station, which were commissioned between 1985 and 1987.

In *Finland*, the two units located in Loviisa have an expected operating lifetime of 50 years, though they were originally licensed for 30 years only. The Loviisa units are both of VVER-440 type (design typical of Eastern European reactors – not representative of Western-type units), which are known to suffer from material embrittlement problems due to neutron exposure in combination with high phosphorous content. They started operation in 1977 (Loviisa 1) and 1981 (Loviisa 2). A 20 year license extension was recommended by the Radiation and Nuclear Safety Authority (STUK) and granted in mid 2007, taking them to 2027 and 2030, subject to safety evaluation in 2015 and 2023.

The other two 870 MW reactors at Olkiluoto, which started up in 1979 and 1982, have had their license extended to 60 years, subject to safety evaluations every 10 years. In Olkiluoto, a third unit is currently under construction for a start-up in 2012; this will be a Generation III EPR reactor, which typically has a 60 year design lifetime.

Germany has 17 operating nuclear power plants, supplying about one quarter of the electricity. They came into commercial operation between 1975 and 1989. In October 1998, the coalition government changed the law to establish the eventual phasing out of nuclear power by closing all 17 reactors by 2021.

In May 2007, the International Energy Agency warned that Germany's decision to phase out nuclear power would limit its full potential to reduce carbon emissions "without a doubt." The agency urged the German government to reconsider the policy in the light of "adverse consequences".

Fuelling the dispute within the grand coalition government, a January 2007 report by Deutsche Bank warned that Germany will miss its carbon dioxide emission targets by a wide margin, face higher electricity prices, suffer more blackouts and dramatically increase its dependence on gas imports from Russia as a result of its nuclear phase-out policy, if it is followed through. Meanwhile, the utilities expressed their intention to extend the lifetimes of all 17 reactors, first to 40 years and then individually seeking extensions to 60 years as in the USA.

Currently, life extension for nuclear plants hinges on whether Chancellor Angela Merkel's conservatives win a majority in the general elections which are due in September 2009.

In 2005, the Parliament of Hungary endorsed plans to extend the Paks 1-4 lifetimes by 20 years, up to 2032-37. License renewal is being sought accordingly. The four reactors, all of VVER-440 type, would have otherwise closed after 30 years [2-14,2-15].

The Netherlands have only one nuclear power plant in Borssele. It is a Siemens design plant, identical to many German reactors, it was connected to the grid in 1973 and supplies about 4% of the country's electricity. In 1994, the Dutch parliament voted to phase out the Borssele nuclear power plant by 2003. The government however ran into legal difficulties to implement that decision and in 2003, the ruling conservative government coalition moved the closure date back to 2013; in 2005 the phase-out decision was abandoned. Having been granted a license extension from 40 to 60 years, the reactor is now allowed to operate until 2034 on certain conditions: it would be maintained to the highest safety standards, and the stakeholders (Delta and Essent) agreed to invest EUR 250 million towards sustainable energy projects [2-16].

Russia, that currently has 31 operating reactors, is moving steadily forward with plans for a much expanded role of nuclear energy, doubling output by 2020. Nuclear electricity output has been rising strongly due to better performance of the nuclear plants, with capacity factors leaping from 56% to 76% 1998-2003 and then on to 79.5% in 2008. All current Russian reactors are of VVER-440 or VVER-1000 type, and are fundamentally different from typical Western-type LWR reactors.

Generally, Russian reactors are licensed for 30 years from first power. Late in 2000, plans were announced for operation extensions of twelve first-generation reactors of VVER-440 type, with an extension period between 15 to 25 years, necessitating major investment in refurbishing them. Many of these reactors have been thermally annealed, in order to recover at least part of the irradiation-induced embrittlement induced by the high phosphorous levels in the vessel materials. Generally, the VVER-440 and RBMK (light water graphite reactors) units will get 15-year life extensions and the nine VVER-1000 units 25 years. So far, 15-year extensions have been granted to Novovoronezh-3 & 4, Kursk-1, Kola-1 & 2 and Leningrad-1 & 2. Bilibino 1 & 2 have been given 5-year license extensions. Kola 3 & 4, Novovoronezh 5 and Beloyarsk 3 are next in line, together with six of the RBMK units.

In *Spain*, the license renewal for the Santa Maria de Garoña plant (in operation since 1968) came up for review in 2009. In June, the Nuclear Safety Council (CSN) recommended that a 10-year extension be granted up to 2019, stating that plant owner and operator Nuclenor had implemented a comprehensive work program to keep the 40-year old reactor fully serviceable. The Socialist government, that presently endorses a policy of closing down Spanish nuclear plants as early as possible, granted only a four-year license extension, up to 2013. In general, the commitment of the present government to the future of nuclear energy in Spain is still uncertain.

Sweden has 10 nuclear reactors, providing over 40% of its electricity. Earlier plans to shut down all Swedish reactors by 2010, largely in response to the Three Mile Island accident in the US, have been shelved as concerns for climate changes and security of supply grow. However, no extension to the 40-year operating lifetimes have been granted, and the reactors are due to close between 2012 and 2025. In February 2009, the Swedish coalition government announced plans to abolish the act banning construction of new nuclear reactors.

Switzerland has 5 operating nuclear power plants, that generate about 40% of its electricity. In 2003, by a two-thirds majority in a popular referendum, Swiss voters rejected two anti-nuclear proposals which were originally put forward in 1998, aimed at phasing out nuclear power by 2014.

The Swiss government announced early in 2007 that the existing five nuclear power reactors should be replaced in due course with new units. The currents units could close between 2019 and 2034, after an operational lifetime of 50 years.

The Nuclear Installations Inspectorate (NIL) of the *United Kingdom* granted in March 2009 the permission to run the Oldbury 1 plants for another two years; in December 2008, the Oldbury 2 plant was also approved to operate for another two years, instead of being closed down at the end of 2008. In June 2009, approval was granted to run the 1,000-MW Wylfa nuclear power station in Wales for at least nine months beyond its planned closure date of March 2010. It must be noted that the Oldbury and Wylfa reactors (4 units in total) are of Magnox type, that are cooled with CO_2, moderated with graphite, use natural (non-enriched) uranium as fuel and Magnox alloy (hence the name) as cladding. These reactors are therefore very different from the typical PWR reactor design of all the Belgian units.

British Energy's twin reactor Hartlepool power station, which began operation in 1984-85 and two Heysham 1 reactors, which opened in 1985-86, have been cleared by the safety regulator to run for another five years beyond their scheduled closure in 2014. These units are all of AGR type (advanced gas-cooled reactors using graphite as moderator), which is also significantly different from the PWR design of the Belgian reactors.

British Energy's new French owner EDF plans to decide by 2011 whether it wants to run the plants until 2019.

2.2.4 *Japan* [2-17]

As Japan has few natural resources of its own, it depends on imports for some 80% of its primary energy needs. The country's 53 reactors provide some 30% of the country's electricity and this is expected to increase to at least 40% by 2017. As at today, 3 units are under construction and 13 more are planned. Although detailed information about license renewal is not publicly available, it is known that several of the existing nuclear power plants are currently considering an extension of their operating lifetime up to 60 years or more.

2.2.5 *South Korea* [2-18]

In South Korea, the Ministry of Education, Science & Technology's third comprehensive nuclear energy development plan for 2007-11, projected that South Korea should develop its nuclear industry into one of the top five in the world, with about 60% of electricity from nuclear by 2035 (currently it's almost 40%). In the country's 2008 Energy Plan to 2030, the increase was quantified as ten or eleven new nuclear power units.

Currently, utilities are negotiating license renewals to extend 30-year operating lifetimes by ten years, starting with the oldest units (Kori-1 and Wolsong-1). This was successful for Kori-1, a Westinghouse type unit where a six-month upgrading and inspection outage in the second half of 2007 concluded a major refurbishment program and enabled its relicensing for a further ten years.

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3 Similarities between US and Belgian units

Within the US fleet of nuclear reactors, we have identified several plants which bear substantial similarities to the 7 Belgian NPP's. Such commonalities were identified based on the following characteristics [3-1], listed in order of priority:

- chemical composition of the pressure vessel beltline materials, in terms of the alloying elements which most influence the sensitivity to radiation embrittlement (copper, nickel and phosphorous);
- plant age, as given by the date of grid connection;
- overall reactor design (Westinghouse, Framatome, Siemens etc).

Note that a similar exercise would have also been useful for the French reactors, considering the similarities with some of the most recent Belgian units. However, detailed and updated information on the French reactor fleet is not publicly available as in the case of the US plants.

3.1 Doel I (grid connection: 2/1975)

The base metals (PM1 and PM2) of Doel I are similar to those of three US reactors: R.E. Ginna, Turkey Point 3 and Prairie Island 2. These reactors are all from the late 60's (1969 – R.E. Ginna) or early 70's (1972 – Turkey Point 3 and 1974 – Prairie Island 2); for the third one, the vessel manufacturer is the same as for Doel I (Societé des Forges et Ateliers du Creusot). R.E. Ginna and Turkey Point 3 have been granted by NRC a 60y license renewal up to 2029 and 2032, respectively. For Prairie Island 2, license renewal has been requested in April 2008 and the application is under examination.

For the weld metal, similarities were identified with Sequoyah 2, which being relatively younger (1981) is currently licensed until 2021 and has not yet requested a license renewal from NRC.

All these US reactors share the same reactor design as Doel I (Westinghouse).

3.2 Doel II (grid connection: 12/1975)

The base metals (PM1 and PM2) of Doel II are similar to the base metal of Catawba 1 (grid connection in 1985). NRC granted this plant a 60y license renewal until 2043.

The low copper weld metal resembles that of Calvert Cliffs 1 (similar age also: grid connection in 1974), which was granted the 60y license renewal until 2034.

As far as the high copper weld metal is concerned, the closest match is Sequoyah 1, that started operation in 1980 and has not yet requested a license renewal.

Both Catawba 1 and Calvert Cliffs 1 have the same reactor design as Doel II (Westinghouse).

3.3 Doel III (grid connection: 10/1982)

The base metal has similar chemical composition to that of 5 US reactors, namely:

- Turkey Point 3 (grid connection: 1972)
- Kewaunee (grid connection: 1973)
- Point Beach 2 (grid connection: 1973)
- Turkey Point 4 (grid connection: 1973)
- Braidwood 2 (grid connection: 1988)

The first 4 have all been granted 60y license renewals by NRC, while the last one has not formulated an official request yet.

The weld metal of Doel III is similar to the one of Catawba 1, which has similar age (1985) and has been granted 60y license renewal until 2043.

3.4 Doel IV (grid connection: 7/1985)

The two base metals (core shell and transition ring) of Doel IV bear similarities to the base metals of Braidwood 1 (1987), Byron 1 (1985) and Byron 2 (1987). The three US units have not yet submitted an application for license renewal, and all share the same reactor design as Doel IV (Westinghouse).

The weld metal of Doel IV has no direct equivalent among the US reactors.

3.5 Tihange I (grid connection: 10/1975)

For the base metals C1L and C2L, the closest matches are R.E. Ginna (1969) and Prairie Island 2 (1974); the former has obtained the 60y license renewal, while an application for the latter was submitted to NRC in April 2008. The C1L material is also similar to the base metal of Turkey Point 3 (1972 – renewal granted until 2032), Turkey Point 4 (1973 – renewal granted until 2033), Point Beach 2 (1973 – renewal granted until 2033), Kewaunee (1973 – renewal requested in August 2008), and Braidwood 2 (1988 – renewal not requested yet).

The weld metal of Tihange I can be compared to that of Prairie Island 2 (1974 – renewal requested in April 2008) and North Anna 2 (1980 – license renewal granted until 2040).

3.6 Tihange II (grid connection: 2/1983)

The base metal of Tihange II can be compared with the base metal of 7 US reactors, namely:

- R.E. Ginna (1969): license renewed for 60y until 2029;
- Turkey Point 3 (1972): license renewed for 60y until 2032;
- Turkey Point 4 (1973): license renewed for 60y until 2033;
- Kewaunee (1973): 60y license renewal requested in August 2008;
- Point Beach 2 (1973): license renewed for 60y until 2033;
- Braidwood 2 (1988): no application submitted yet.

As far as the weld metal is concerned, similarities were found with Vogtle 1, which started operation in 1987 and was granted a license renewal for 60y until 2047.

3.7 Tihange III (grid connection: 9/1985)

We were unable to find any close match among the US reactors for either the base or the weld metals of Tihange III.

3.8 Summary

The overall situation is summarized in Table 1.

Table 1 - Comparison between Belgian NPP's and similar US reactors.

Belgian unit	Grid connection	Beltline material	Similar US units	Grid connection	License status for US units
Doel I	02/1975	Base	Ginna	1969	60y license granted
			Turkey Point 3	1972	60y license granted
			Prairie Island 2	1974	60 y license requested (4/08)
		Weld	Sequoyah 2	1981	Renewal not requested yet
Doel II	12/1975	Base	Catawba 1	1985	60 y license granted
		Weld low Cu	Calvert Cliffs 1	1974	60 y license granted
		Weld high Cu	Sequoyah 1	1980	Renewal not requested yet
Doel III	10/1982	Base	Turkey Point 3	1972	60y license granted
			Turkey Point 4	1973	60y license granted
			Point Beach 2	1973	60y license granted
			Kewaunee	1973	60 y license requested (8/08)
			Braidwood 2	1988	Renewal not requested yet
		Weld	Catawba 1	1985	60 y license granted
Doel IV	07/1985	Base (core shell)	Braidwood 1	1987	Renewal not requested yet
			Byron 2	1987	Renewal not requested yet
		Base (trans.ring)	Byron 1	1985	Renewal not requested yet
			Braidwood 1	1987	Renewal not requested yet
Tihange I	10/1975	Base (C1L)	Ginna	1969	60y license granted
			Turkey Point 3	1972	60y license granted
			Turkey Point 4	1973	60y license granted
			Point Beach 2	1973	60y license granted
			Kewaunee	1973	60 y license requested (8/08)
			Prairie Island 2	1974	60 y license requested (4/08)
			Braidwood 2	1988	Renewal not requested yet
		Base (C2L)	Ginna	1969	60y license granted
			Prairie Island 2	1974	60 y license requested (4/08)
		Weld	Prairie Island 2	1974	60 y license requested (4/08)
			North Anna 2	1980	60 y license granted
Tihange II	2/1983	Base	Ginna	1969	60y license granted
			Turkey Point 3	1972	60y license granted
			Turkey Point 4	1973	60y license granted
			Point Beach 2	1973	60y license granted
			Kewaunee	1973	60 y license requested (8/08)
			Braidwood 2	1988	Renewal not requested yet
		Weld	Vogtle 1	1983	60 y license granted

Table 1 shows that, among the 54 US reactors that have already been granted for license renewal up to 60 years operation, eight units (Ginna, Turkey Point 3, Turkey Point 4, Point Beach 2, Calvert Cliffs 1, North Anna 2, Vogtle 1 and Catawba 1) are similar to the Belgian reactors.

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4 General principles of reactor pressure vessel surveillance

4.1 Introduction

The previous Sections presented an overview of the international situation with regard to operation extension of nuclear power plants and a comparison between Belgian reactors and similar US units.

It must be emphasized that one of the basic conditions for requesting and eventually being granted an extension of the operating license, is to demonstrate the safety of the reactor pressure vessel up to the projected lifetime.

The reactor pressure vessel surveillance program, which will be outlined in this Section, provides engineers, plant operators and safety authorities the necessary information about the future integrity of the pressure vessel by anticipating the evolution of the mechanical properties of the pressure vessel beltline materials as they change (and typically degrade) due to neutron exposure. This allows bringing into play in a timely fashion solutions for possible future problems, for example by adopting mitigation measures such as neutron fluence reduction. Reactor pressure vessel surveillance programs form a legal part of the safety evaluation of NPP's.

The "conventional" surveillance outlined in Section 4.2 consists in tests and measurements prescribed by the current codes and regulations in order to exclude brittle failure of the vessel even under the most severe accidental condition. This approach mainly reflects the state of scientific knowledge during the 70's and 80's, when the discipline of fracture mechanics was still in its prime. The "conventional" surveillance is part of the legislation currently in force.

The "advanced" surveillance approach addressed in Section 4.3 takes advantage of the most recent advances in the analysis and interpretation of mechanical test results, with specific emphasis on the direct measurement of the fracture toughness of the RPV beltline materials. The main advantage is that fracture toughness properties are not inferred from Charpy data, as in the case of the "conventional" approach.

These methodologies, although still not fully adopted by the current legislation, represent the direction that RPV integrity assessments will take in the years to come. As such, they have been consistently applied by SCK•CEN in the last decade in the analysis of surveillance capsules from all Belgian units.

Note that corrosion-related considerations are not addressed in this section.

4.2 Regulatory approach: "conventional" surveillance [4-1,4-2,4-3]

The surveillance program of a nuclear power plant consists of inserting capsules into the RPV, containing test specimens of the same materials (plate, forging, weld) used for the vessel, in order to monitor the materials' degradation under neutron exposure. Temperature and neutron flux monitors are also inserted in the capsules.

In order to anticipate any action that would become necessary, the capsules are inserted in a region closer to the core than the vessel wall itself. This way, the irradiation dose at the surveillance capsule position is substantially higher (generally by a factor of 2-3) than at the vessel wall and allows predicting material behavior and if needed anticipating mitigation measures. For instance, assuming an anticipation factor of 3, specimens contained a surveillance capsule extracted after 13 years in the reactor, would be representative of the mechanical behavior of the vessel beltline material after 39 years of reactor operation. The capsules are regularly retrieved and analyzed in order to follow up the materials' degradation.

Surveillance capsules contain a number of Charpy impact specimens, as well as tensile specimens to be tested for measuring the materials' mechanical strength.

The Charpy impact test consists of testing at different temperatures small bars with a V-shaped notch on one side and plotting the absorbed energy (i.e. energy needed to break the specimen) as a function of test temperature.

At low temperatures, the fracture is brittle and therefore requires little energy; as temperature increases, the fracture changes from brittle to mixed ductile-brittle and finally fully ductile, with the absorbed energy increasing accordingly. A ductile-to-brittle transition temperature can thus be obtained, to be used to characterize the material's fracture resistance. However, this simple test does not allow a direct access to structural integrity parameters such as fracture toughness, but relies on empirical correlations with actual fracture toughness parameters. Note that the operating temperature of PWR's such as the Belgian units is in the range 285-300°C, which implies that the materials are operating in fully ductile conditions under normal operating conditions.

Under irradiation, the material becomes harder and the ductile-to-brittle transition temperature increases (the energy vs. temperature curve shifts to higher temperatures). It is essential to verify that this "new" transition temperature does not jeopardize the safe operation of the reactor and remains safely below a limiting value prescribed by the legislation.

Charpy specimen results are used to draw the so-called transition curve. An example is given in Figure 1, which shows the absorbed energy transition curve before and after irradiation.

Due to space limitations inside the pressure vessel, the surveillance capsules cannot contain large samples or a large number of test specimens. Therefore, the determination of the RT_{NDT}, the nil ductility transition reference temperature which indexes the material's fracture toughness curve before and after irradiation, is indirectly derived from the Charpy impact transition curve. The legislation, based on the analysis of a large database of experimental results [4-4], assumes that the shift of the Charpy impact transition curve at an absorbed energy level of 41 J (T_{41J}) is equal to the shift of the RT_{NDT} (Figure 2).

In other words, the fracture toughness of the RPV beltine materials is not measured directly, but inferred from existing lower bound curves (i.e. the most conservative curves) based on the values of RT_{NDT} after irradiation. These are obtained by adding to the values of RT_{NDT} for the unirradiated materials, the increase of T_{41J} measured on the irradiated materials by means of Charpy tests (as illustrated in Figure 2) [4-5].

The safety of the RPV, in case of PTS event, is ensured as long as RT_{NDT} remains below a limiting value prescribed by the applicable legislation (PTS screening limit, represented by the black curve in Figure 2) [4-6].

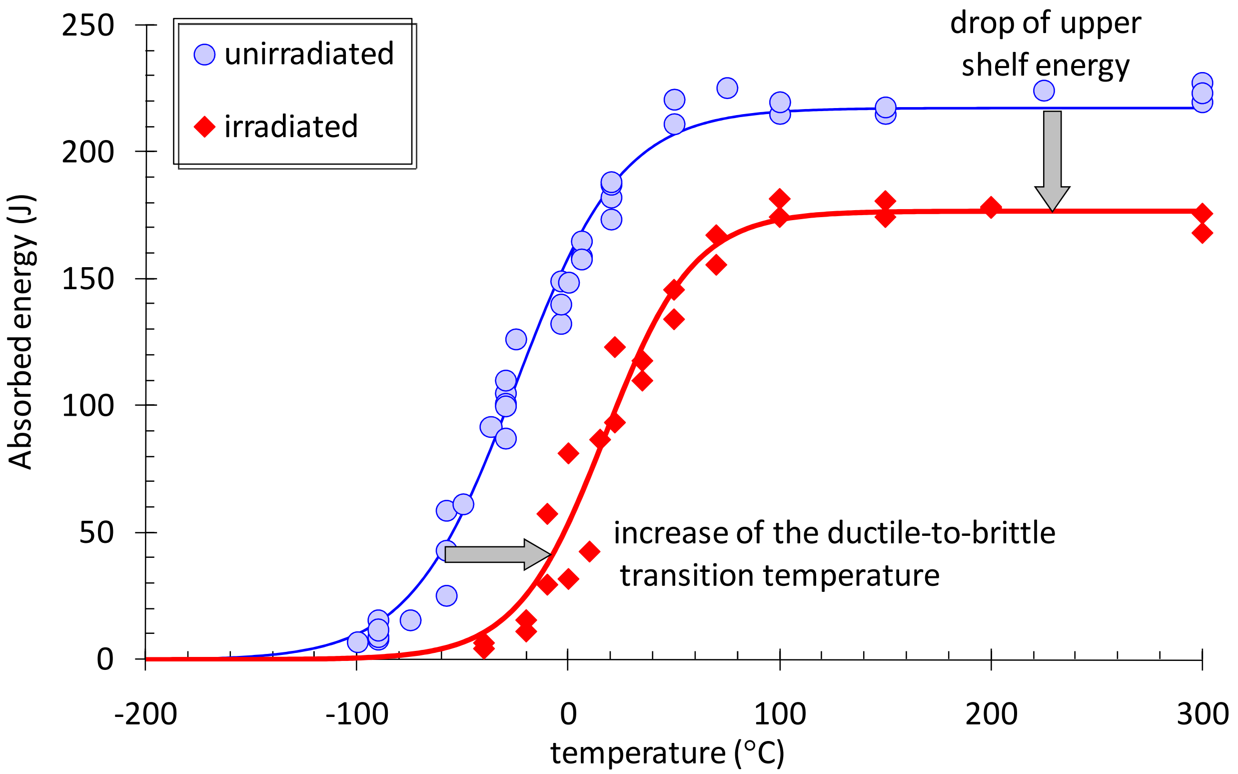

Figure 1 - Illustration of the effect of irradiation on the absorbed energy transition curve.

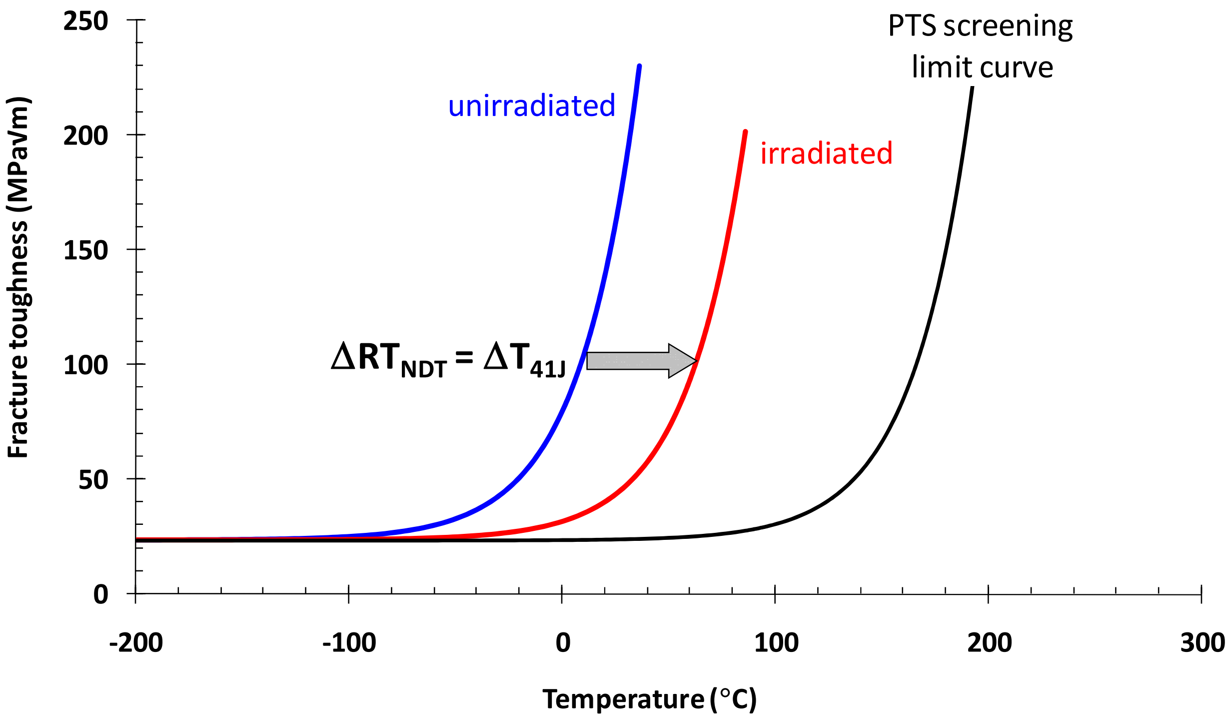

Figure 2 - Illustration of the effect of irradiation on the fracture toughness according to the current legislation. The limit curve given by the legislation, that corresponds to the PTS screening limit, is also shown.

4.3 "Advanced" surveillance [4-7 to 4-15]

"Conventional" surveillance to monitor RPV degradation relies on knowledge and technology available at the time second generation NPP's entered operation (late 80's). As the safe exploitation of the reactor pressure vessel is a major concern in nuclear power plant life management and could be a limiting factor for extended operation, a large amount of research activities were undertaken worldwide to verify and improve the regulatory methodology.

Indeed, there is a worldwide effort towards improving nuclear safety regulations through the improvement of current understanding. Consequently, research and development should be pursued at a high scientific level in research institutes and regulatory bodies should be able to follow up the scientific developments.

In Belgium, the general philosophy of the utilities has always been to anticipate any potential problem and this is the reason why the regulatory surveillance program is nowadays on a regular basis coupled to additional investigations ("advanced" surveillance) to support the concept of "defense in depth". This approach has proven very effective in improving the quality of RPV assessments based on more physical principles, although it has not yet been adopted by the legislation currently in force.

SCK•CEN has contributed to and keeps actively participating in related R&D activities, ranging from fundamental to applied research. These techniques allow:

- generating additional data to consolidate the surveillance program (e.g. hardness, instrumented Charpy test, tensile test on miniature specimens and reconstitution);
- improving the understanding of RPV degradation mechanisms to complement embrittlement trend curves which are empirical models currently used in the regulation (e.g. microstructural investigations, micro-mechanical modeling and multi-scale modeling using computer simulation tools);
- directly measuring the fracture toughness of the material using small specimens, thus avoiding some of the empiricism entailed by the current regulation and relying on physically based data (such as the Master Curve technique, see Section 4.3.3 below).

This so-called "advanced" (or "enhanced") surveillance strategy was developed at SCK•CEN more than a decade ago to improve understanding of the materials' evolution under neutron irradiation, and has since been internationally acknowledged.

As a consequence, rather than limiting the test program to the regulatory requirements, additional tests are performed together with specific interpretation tools to achieve a better knowledge and quality control of all the available data. As a result, usually, the number of tensile tests is increased to cover a larger temperature range. Moreover, thanks to specimen reconstitution, additional samples are tested for fracture toughness evaluation. Finally, it is important to notice that a number of tools were developed to help understanding the relation between the various properties, which further enhanced the quality control of the various parameters. An extensive effort was also devoted to modeling irradiation effects to rationalize the experimental observations.

It should be emphasized that the "conventional" approach prescribed by the regulation is semi-empirical based and is intentionally associated to a high level of conservatism. The advanced surveillance program, which is based on a physics approach, provides a better insight on the surveillance results. Indeed, should the results of the "conventional" surveillance be non-conservative, the "advanced" approach would also demonstrate it.

Some aspects of the advanced surveillance approach which are bound to have a significant impact on the current and near-future developments of the legislation are described in the following subsections.

4.3.1 Load diagram approach [4-16]

One of the key mechanical tests used in the regulation is the Charpy impact test. However, the "simplified" approach adopted by the regulation in the framework of the "conventional" surveillance has sometimes proven to be inaccurate and therefore SCK•CEN has developed, under the partial sponsorship of Electrabel/Tractebel, physically-based tools allowing a better analysis and interpretation of the test results.

The load diagram approach takes benefit from the available instrumented Charpy data to build a consistent picture where tensile and instrumented Charpy information are combined in a single physically-based diagram. These tools do not only allow the extraction of additional information on the tested material but also provide a quality control method of the test results. Although the current regulation does not prescribe this type of analyses, this is nowadays performed for all Belgian capsules mainly as a support to the conventional approach.

Nowadays, Belgium is the only country worldwide where such an in-depth analysis of the experimental data with physically-based tools is available.

4.3.2 Specimen reconstitution [4-17,4-18]

The amount of available specimens in the surveillance capsules is usually limited. Therefore, in order to optimize material consumption, we developed at SCK•CEN a reconstitution technique that allows obtaining new test specimens from previously broken ones.

Reconstitution not only allows to increasing the number of Charpy specimens if needed, but more importantly to perform fracture toughness tests in order to directly measure the actual fracture toughness transition curve as well as the crack resistance curve under fully ductile conditions (see below).

4.3.3 Direct fracture toughness measurements [4-19]

Structural integrity calculations rely on parameters such as the fracture toughness of the material, i.e. the property which describes the ability of a material containing a crack to resist fracture.

For a given material, if the temperature dependence of the fracture toughness is known, one can determine the region of safe operation where the loading conditions are such that the structure remains below a "critical" level of toughness (i.e. at one temperature, the maximum value of loading that can be sustained before fracture occurs).

To determine such a curve, the specimen size that is required is too large to be inserted in a surveillance capsule. Consequently, according to the legislation, this curve is indirectly determined from testing Charpy impact specimens and then applying a semi-empirical procedure described in the regulatory codes.

However, advances in fracture toughness testing and evaluation allow nowadays establishing this curve directly, by measuring the fracture toughness as a function of temperature using small size specimens. Specific tools have been established and validated on a large experimental database, allowing to correlate between small-size samples and real structures and components.

Moreover, thanks to the reconstitution technology, fracture toughness samples can be manufactured using previously broken Charpy specimens. Such direct fracture toughness measurements on reconstituted Charpy specimens are routinely performed at SCK•CEN.

Once experimental results are obtained, the so-called Master Curve (MC) concept [4-20] allows the determination of the median toughness curve as a function of temperature, taking into account the inherent scatter which is typical of the ductile-to-brittle transition regime. This curve can be directly used in structural integrity calculations.

Not only these measurements provide a more direct determination of the ductile-to-brittle transition temperature, but it was also found that usually this approach provides additional safety margins with respect to the regulation limits (see also Figure 6 on page 35) [4-15].

An illustration is given in Figure 3 where it can be seen that often, both the initial RT_{NDT} and the shift of the transition curve can be overestimated. Note that, in the case of direct fracture toughness measurements, the temperature corresponding to a reference toughness level of 100 MPa√m is used as the equivalent of the Charpy-based reference temperature T_{41J}.

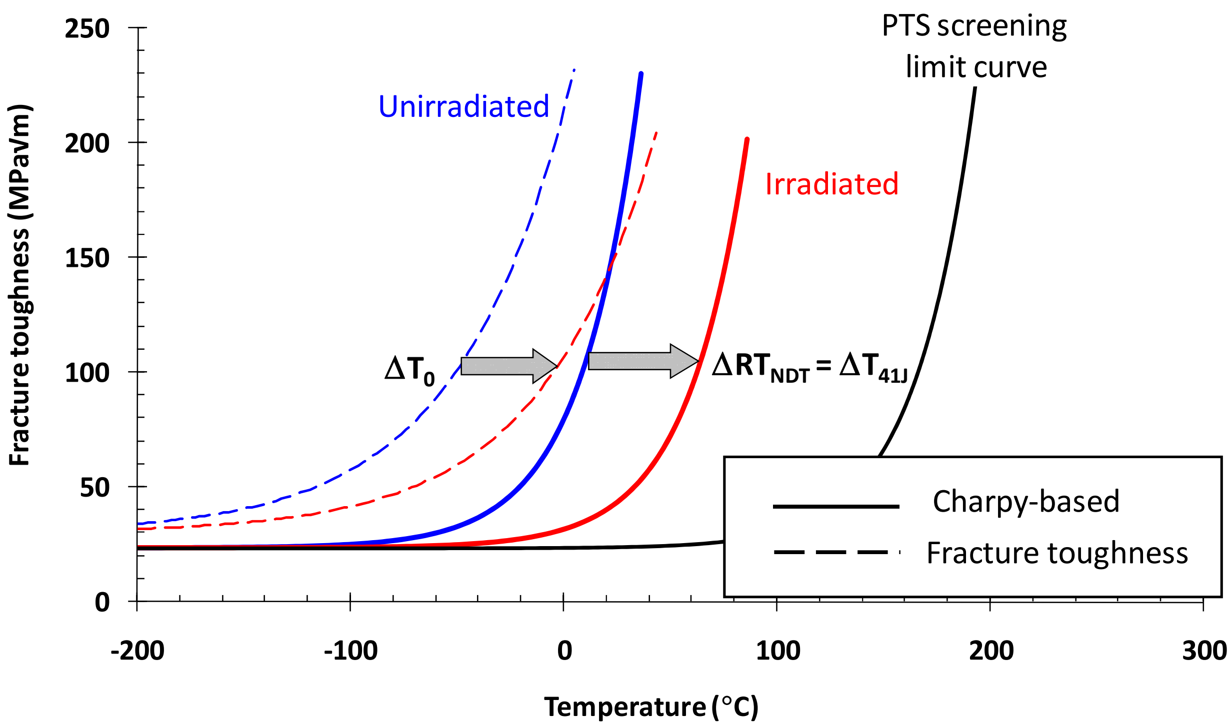

Figure 3 - Illustration of the effect of irradiation on the fracture toughness transition curve using the conventional (Charpy-based) and the advanced (fracture toughness) approach, showing the additional margin allowed by the latter.

4.3.4 Modeling support

Regardless of the available experimental data, it is important to verify that the effects of irradiation are well accounted for and that they do not deviate from trends that were obtained on similar materials.

The regulatory guides simple semi-empirical formulas to evaluate the expected embrittlement of vessel materials. The proposed formulas are all based on the regression of databases consisting of specific experimental data (US, French etc).

SCK•CEN has developed a model that relies on the physics and on our current understanding of radiation damage, therefore avoiding the use of empirical trend curves and with an extended application domain. The model also provides additional support to the obtainment and interpretation of experimental data and therefore contributes to the quality control of the results.

4.4 Concluding remarks

In Belgium, the application of both the "conventional" (legislation-based) and the "advanced" (research-based) surveillance to monitor RPV degradation provides a better insight in the phenomena associated with material embrittlement and offers furthermore a quality control tool for the experimental data. The combination of all available information, usually not implemented in current practices, provides an integrated package with solid physical basis in view of a more accurate interpretation of the results.

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5 Overview of SCK•CEN expertise in the analysis of surveillance capsules and the assessment of RPV integrity

SCK•CEN, and more specifically the Institute of Nuclear Material Science, has a multi-decennial experience in the testing, analysis and interpretation of the surveillance capsules of all Belgian nuclear power plants and several foreign reactors. Additional contracts related to RPV integrity assessments were executed in the past for other countries and nuclear industries (Argentina, Brazil, South Korea, US, Switzerland, Germany).

Specific information is provided in the following subsections.

5.1 Belgian NPP's

In Belgium, there are 7 nuclear power reactor units in operation, 3 are located in Doel and 4 in Tihange.

From the RPV material point of view, but not only, one can classify the Belgian units as a function of their start-up time. Indeed, Doel-I, Doel-II and Tihange-I units were connected to the grid in the 70's (1975) while Doel-III (1982), Tihange-II (1983) and Tihange-III and Doel-IV (1985) were connected during the 80's.

With respect to the earlier units, the materials of the later generation reactors have largely benefitted from the knowledge that was available in the early/mid 80's on the mechanisms of irradiation damage, in particular the detrimental effect of copper (Cu) and phosphorous (P) on radiation embrittlement and the improved manufacturing techniques that allowed reducing the concentration of impurities. As a result, Cu and P contents were significantly reduced.

Note also that while the reactor design of Doel-I, Doel-II, Doel IV and Tihange-I is of the Westinghouse type, for Tihange-II, Tihange-III and Doel-III the design is of Framatome type. Therefore, although the Belgian nuclear legislation is based on the US codes and regulations, the applicable embrittlement models for all reactors, except Doel I and Doel II, are those prescribed by the French code [5-1].

For the Belgian units, the first surveillance report was issued by SCK•CEN in 1978, following the analysis of the first surveillance capsule of Doel I. Since then, SCK•CEN has authored 72 reports (all propriety of Electrabel/Tractebel) addressing the testing and analysis of surveillance capsules removed from the four units of Doel and the three units of Tihange, the last one at the time of writing issued in April 2009.

Further details are given in Table 2 below. Note that information concerning the retrieval of the forthcoming surveillance capsules, corresponding to irradiation times beyond 60 years, are also provided in the Table (in bold italics).

Table 2 - SCK•CEN reports published from 1978 to 2009 on Belgian NPP surveillance capsules.

Unit	Surveillance capsule	Retrieval date	Equivalent RPV irradiation time (y)	No. of SCK•CEN reports issued
Doel I	First	1977	5.12	4
	Second	1980	13.37	2
	Third	1989	25.73	2
	Fourth	1993	34.07	3
	Fifth©	2000	48.26	4
	Sixth	***2010***	***65.74***	-

Unit	Surveillance capsule	Retrieval date	Equivalent RPV irradiation time (y)	No. of SCK•CEN reports issued
Doel II	First	1977	5.19	3
	Second	1982	20.22	2
	Third	1982	11.28	1
	Fourth	1991	29.45	2
	Fifth	1996	41.41	4
	Sixth©	2008	72.69	2
Doel III	First	1986	11.75* - 8.39†	3
	Second	1988	18.97* - 14.41†	1
	Third	1996	47.22* - 32.82†	3
	Fourth	***2010***	***89.59* - 63.35†***	-
Doel IV	First	1989	11.72‡ - 9.72◊ - 17.08†	1
	Second	1994	23.56‡ - 19.91◊ - 32.85†	1
	Third©	2003	44.90‡ - 37.33◊ - 62.93†	4
	Fourth	***2011***	***72.28‡ - 60.00◊ - 101.52†***	-
Tihange I	First©	1979	4.2	6
	Second©	1985	12.3	3
	Third	1992	20.8	3
	Fourth©	2001	33.3	4
	Fifth©	2002	40.7	3
	Sixth	***2009***	***63.6***	-
Tihange II	First	1986	5.64* - 7.50†	2
	Second	1989	13.08* - 19.55†	2
	Third©	1997	38.41* - 55.15†	1
	Fourth	***2011***	***60.32* - 85.56†***	-
Tihange III	First	1988	6.16‡ - 5.08◊ - 8.26†	2
	Second	1995	22.47‡ - 15.87◊ - 35.53†	1
	Third©	2004	50.35‡ - 38.62◊ - 73.51†	3
	Fourth	***2012***	***85.50‡ - 67.23◊ - 121.58†***	-

LEGEND - * Base metal - † Weld metal - ‡ Core shell (base metal) - ◊ Transition ring (base metal).
©Surveillance capsules for which the advanced surveillance program has been executed.

Although the regulatory assessment of RPV integrity can be based only on the application of analytical embrittlement correlations and experimental data are not strictly required, nevertheless surveillance capsules corresponding to 60 years of operation will be tested by 2012 for every Belgian unit. Note that this is not the case for most US reactors, whose license renewal is or will be based on data extrapolation and application of trend curves.

A typical surveillance report includes a description of the capsule contents, dimensional controls of the mechanical specimens, results of the mechanical tests (hardness, tensile and Charpy impact), dosimetry measurements and calculations, comparisons with unirradiated materials and previous surveillance capsules.

Starting in 2002, for every Belgian surveillance capsule examined in the framework of the current legislation ("conventional" surveillance), an additional report has been issued containing additional advanced research to be eventually used for the safety assessment of the RPV, in a "defense-in-depth" perspective ("advanced" surveillance, see Section 4.3). 14 of the 72 reports mentioned in Table 2 fall into this category.

The contents of an "enhanced" surveillance report include additional tensile test results, in-depth analysis of the instrumented Charpy impact data ("Load Diagram approach") and fracture toughness test results.

On top of the 72 reports mentioned in Table 2, SCK•CEN issued seventeen additional reports for the Doel and Tihange reactors, concerning various investigations on the surveillance materials (mechanical and microstructural investigations, characterization of the unirradiated condition, revision of the dosimetry results etc).

Finally, six research reports have been produced detailing investigations conducted on the materials of BR3 (Belgian Reactor 3), which was the first PWR reactor in Western Europe (first criticality in 1962) and that was shut down in 1987 and subsequently decommissioned. In the case of BR3, SCK•CEN investigations based on direct fracture toughness measurements, performed after the closure of the reactor, demonstrated that large margins still existed before a real concern would be justified for the pressure vessel integrity.

5.2 Foreign NPP's

5.2.1 Spain

From 2002 to 2006, SCK•CEN tested and analyzed according to the current US legislation[2] surveillance capsules for several Spanish nuclear power plants (Almaraz, Ascò, Cofrentes, Santa Maria de Garoña, Vandellòs). In total, eleven reports (mechanical testing and dosimetry) have been published.

Additionally, SCK•CEN participated in the CUPRIVA program [5-2,5-3,5-4], in which surveillance materials from Santa Maria de Garoña and Ascò were used for reconstituting fracture toughness specimens, that were subsequently tested for an advanced assessment of the integrity of the reactor pressure vessels. Material was also irradiated in the BR2 test reactor in Mol.

In 2007, a study was conducted and documented in order to consolidate the surveillance program of Santa Maria de Garoña, the oldest Spanish power plant, in view of its license renewal (which was subsequently granted – see Section 2.2.3 on page 11).

5.2.2 Switzerland

In 2006 and 2009, SCK•CEN performed testing and analysis on the surveillance materials of the Swiss nuclear power station KKM (Kernkraftwerk Mühleberg), including an assessment of the KKM beltline materials based on the current legislation and on more advanced embrittlement models. The final report, which was accepted by the Swiss Safety Authorities (ENSI), is currently being used to support the request for license renewal of the Mühleberg plant.

At the time of writing, SCK•CEN has just been awarded the contract for the testing and analysis of three surveillance capsules from KKL, Kernkraft Leibstadt.

5.2.3 Argentina

The Argentinian pressurized heavy water reactor (PHWR) Atucha 1 entered commercial operation in 1974. In the framework of a Belgo-Argentinian Cooperation Agreement on Nuclear Safety and Plant Life Management, in 2002 SCK•CEN signed a collaboration agreement with the Argentinian National Atomic Energy Commission (CNEA) in support of the safety evaluation of the Atucha 1 vessel. The project was given the name TANGO.

An irradiation program in the BR2 reactor was carried out, aimed at confirming past results obtained by CNEA in the 90's from available surveillance sets, extending the available surveillance database and preparing future RPV surveillance beyond design life. In parallel, a study focusing on

[2] Like Belgium, Spain also follows the US legislation.

possible spectrum and dose rate effects on the results of the TANGO experiment was conducted and reported. Finally, an intercomparison of neutron dosimetry measurements was jointly documented by CNEA and SCK•CEN.

The surveillance capsules of Atucha 1 were recognized to be non-representative of the actual irradiation conditions of the pressure vessel, due to the significantly different neutron energy spectrum. In 2006, SCK•CEN led an international team of experts from Belgium, Argentina, UK, US and Finland in a project denominated INTEGRITY. Its aim was to evaluate the results of complementary irradiation programs that had been performed in Germany, Belgium (BR2) and Finland using the beltline materials of Atucha 1. In the final report of the INTEGRITY project, an integrated analysis was presented of all the existing data with respect to the integrity assessment of the Atucha 1 pressure vessel at end-of-license conditions and beyond. The evaluations presented, which also covered possible spectrum and dose rate effects, showed that the integrity of the pressure vessel could be ensured for the limiting surveillance materials. The final report was accepted by the Argentinian Safety Authority (ARN).

A second PHWR reactor, Atucha 2, was ordered in 1979 and its construction started in 1981. However, due to lack of funds the works were suspended in 1994 with the plant 81% complete. Plans to complete the reactor were presented in 2003 to the government and in 2006 a strategic plan was announced which included completion of Atucha 2 by 2010.

In 2008, NA-SA (Nucleoeléctrica Argentina), who manages the Atucha NPP's, contracted SCK•CEN for setting up the surveillance program of Atucha 2, including the design and realization of a remote handling tool for the insertion and removal of the surveillance capsules at beltline position (the original surveillance program envisaged capsules located at the bottom of the vessel, in a non-representative position).

At the time of writing, the contract is in execution.

5.2.4 Other foreign NPP's

In 2001, SCK•CEN designed the surveillance program of the *Brazilian* test reactor INAP, in accordance with the current legislation at the time (ASTM E185-82) and including advanced surveillance concepts such as specimen reconstitution and fracture toughness testing. Within the same contract, SCK•CEN also characterized the mechanical properties (tensile, Charpy impact, fracture toughness, drop-weight and hardness) of the surveillance materials in the unirradiated condition.

In the late 90's, SCK•CEN provided training to personnel of the *Bulgarian* NPP of Koslodui, specifically on the topics of specimen reconstitution and mechanical testing. Part of the work was conducted in the framework of the EU-sponsored international project RESQUE, which was co-ordinated by SCK•CEN. Specific problems of the Koslodui six units that were addressed at the time were the unavailability of a surveillance program for the older units (available material was in the form of broken specimens machined from boat samples extracted from the vessels) and the non-representativity of the original location of the surveillance capsules, in terms of both irradiation temperature and neutron energy spectrum, for the newer units.

Fracture toughness specimens from the weld metal of the *South Korean* plant Kori 1 were reconstituted, prepared and tested at SCK•CEN in 2005/2006 in the framework of the life management of the plant, following a request of collaboration from KAERI (Korean Atomic Energy Research Institute). The activities performed at SCK•CEN significantly contributed to the granting of a license extension for the Kori-1 reactor (see Section 2.2.5 on page 13).

In recent years (2005-2009), SCK•CEN irradiated in BR2 and characterized materials from actual *American*[3] and *Japanese* operating reactors, in the framework of collaborative research programs with Oak Ridge National Laboratory (ORNL, *US*) and CRIEPI (*Japan*). All these programs have as ultimate goal the plant life management of several US and Japanese nuclear reactors.

In the course of the past 25 years, several reports and scientific papers have been published by SCK•CEN describing investigations conducted on pressure vessel materials of other international reactors, such as KKP1 (Kernkraftwerk Philippsburg) and KWO (Kernkraftwerk Obrigheim) in *Germany*, Yankee Rowe and Davis-Besse in the *US* (respectively decommissioned and in operation), Chooz-A in *France* (decommissioned) and Balakovo Unit 1 in *Russia* (in operation).

5.3 Additional international expertise of SCK•CEN

SCK•CEN has participated to several Co-ordinated Research Projects (CRP) established by the International Atomic Energy Agency (IAEA) in areas of common interest to a number of Member States [5-5]. More specifically, a leading role has been played by SCK•CEN in the following CRP's:

- CRP-3, "Optimizing of Reactor Pressure Vessel Surveillance Programmes and Their Analyses" (1984-1992),
- CRP-4, "Assuring Structural Integrity of Reactor Pressure Vessels" (1996-1999),
- CRP-8, "Master Curve Approach to Monitor Fracture Toughness of RPV Steels" (2004-2008),

as well as in the IAEA Round Robin Exercise on "WWER-440 RPV Weld Material Irradiation, Annealing and Re-Embrittlement" (1996-2004) [5-6].

SCK•CEN research activities related to the life management of nuclear power plants and more specifically to the integrity assessment of reactor pressure vessels, have also been conducted in the framework of numerous EU-sponsored international projects within the so-called Framework Programmes (FP), such as:

- FP4 (1994-1998) [5-7]: *Reconstitution Techniques Qualification and Evaluation* (RESQUE, where SCK•CEN played the role of coordinator) and *Relation between Different Measures of Exposure-Induced Shifts in Ductile-Brittle Transition Temperatures* (REFEREE);
- FP5 (1998-2002) [5-8]: *AMES Thematic Network on Ageing* (ATHENA) and *Fracture Mechanics Based Embrittlement Modelling* (FRAME);
- FP6 [5-9]: *Nuclear plant life prediction* (NULIFE, 2006-2012) and *Prediction of Irradiation Damage Effects on Reactor Components* (PERFECT, 2002-2006).

Within the current Framework Project (FP7, 2007-2013) [5-10], SCK•CEN contributes to the following projects:

- PERFORM-60 (*Prediction of the effects of radiation for reactor pressure vessel and in-core materials using multi-scale modelling - 60 years foreseen plant lifetime*)
  Relying on the existing PERFECT Roadmap, this 4-year collaborative project has mainly the objective of developing multi-scale tools aimed at predicting the combined effects of irradiation and corrosion on internals (austenitic stainless steels) and also improving existing ones on RPV (bainitic steels). The existing, predominantly empirical, approach can now be complemented and improved thanks to advanced tools. Indeed, continuous progress in physical understanding of radiation damage and in computer technology has made it possible to develop multi-scale numerical tools capable of

[3] None of which corresponds to any of the reactors which have been identified in Section 3 as similar to the Belgian units.

simulating the effects of neutron irradiation on mechanical and corrosion properties of reactor materials.

- LONGLIFE (*Treatment of long term irradiation embrittlement effects in RPV safety assessment*)
This EU sponsored project aims at investigating RPV materials in terms of improved understanding and prediction of irradiation embrittlement effects connected with long term operation (LTO) up to 80 years. The overall objective is to enhance the knowledge on LTO phenomena relevant for European Light Water Reactors, to assess prediction tools, codes and standards including proposals for improvements, and to elaborate best practice guidelines for RPV irradiation surveillance. The proposed work will improve the RPV safety assessment of existing European LWR's under long-term operation conditions, also of Generation-III reactors under construction to support ageing management and plant operation extension.

At the time of writing, SCK•CEN is negotiating with NRC (Nuclear Regulatory Commission in the US) and Oak Ridge National Laboratory (ORNL) a research contract by the name of MISS-WIFE (*Development of Predictive Models of Irradiation Hardening in Western RPV Steels with a Focus on Flux Effects*). This project aims at developing a physically-based theoretical foundation for a model that describes the effects of various irradiation variables on the degree of hardening and embrittlement experienced by the steels that have been and will be used in the construction of light water PWR's in western countries, taking also possible flux effects into account and thereby enabling the confident use of test reactor data to predict embrittlement trends for power reactor conditions.

As well as regularly participating to international conferences, often as invited lecturers or session chairpersons, SCK•CEN members actively participate to the work of the American Society for Testing and Materials (ASTM) standardization committees [5-11].

Within subcommittee E10.02 on the *Behavior and Use of Nuclear Structural Materials*, we have been in charge of the latest two revisions (2002 and 2009) of the E636 standard, "Standard Guide for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels". Significant contributions were also given to other relevant standards, such as:

- E185-02, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels";
- E900-02, "Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials";
- E1253-07, "Standard Guide for Reconstitution of Irradiated Charpy-Sized Specimens";
- E2215-02, "Standard Practice for Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels".

Inside ASTM committees E28 (*Mechanical Testing*) and E08 (*Fracture and Fatigue*), SCK•CEN researchers contribute to the development and revision of numerous test standards, as well as holding the chairmanship or co-chairmanship of the subcommittees on Numerical Methods, Crack Arrest and Instrumented and Miniature Charpy Testing.

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6 Assessment of Belgian reactor pressure vessels

For historical reasons, all 28 surveillance capsules of the 7 Belgian reactors were tested at SCK•CEN. This has the advantage of having a well documented record of all tests in a single place and an additional expertise that is provided based on all additional tools that were developed for a more extensive evaluation of the materials properties and better quality assurance (see Section 4.3 on page 21).

The "conventional" surveillance can be used in the current regulatory context to evaluate the safety of the RPV under PTS accidental conditions. As the Belgian surveillance data can be considered reliable, the analysis presented here is based on actual surveillance test results and not on embrittlement correlations which depend on the chemistry of the investigated materials.

The US regulation applicable to Belgian units is based on the PTS screening criteria, which ensure sufficient safety against brittle failure of the vessel provided the projected RT_{NDT} remains below the PTS screening criteria. If this is not the case, the NPP needs to either shut down or to provide additional plant-specific analyses demonstrating additional safety margins against brittle rupture of the vessel.

It is important to emphasize that the PTS screening criterion is not, per se, a border separating no-failure from failure. Actually, this limit/curve was established using intentionally significant safety margins. As a result, the current regulation is associated to a significant degree of conservatism; in spite of this, as will be shown later, the Belgian units are well within these regulatory requirements.

The result of the analyses for the Belgian plants is presented in Figure 4 and Figure 5 for the first and second generation of Belgian NPP's respectively; the diagrams show the curves of ART (adjusted reference temperature, which includes a margin term related to experimental uncertainties), calculated according to the current legislation (Regulatory Guide 1.99 Rev.2) using the surveillance results obtained from all the base and weld metals of the Belgian reactors. Due to material degradation (embrittlement), ART increases with increasing years of operation, but in no circumstances data corresponding to 60 years of operation lie above the PTS screening criteria. Sufficient safety margins with respect to the regulatory limits (PTS screening criteria) exist for all Belgian NPP's, for both first () and second generation units. For the older units (Figure 4), the margins are in the worst case of the order of 30°C, while for the newer reactors (Figure 5) the margins are always larger than 100°C.

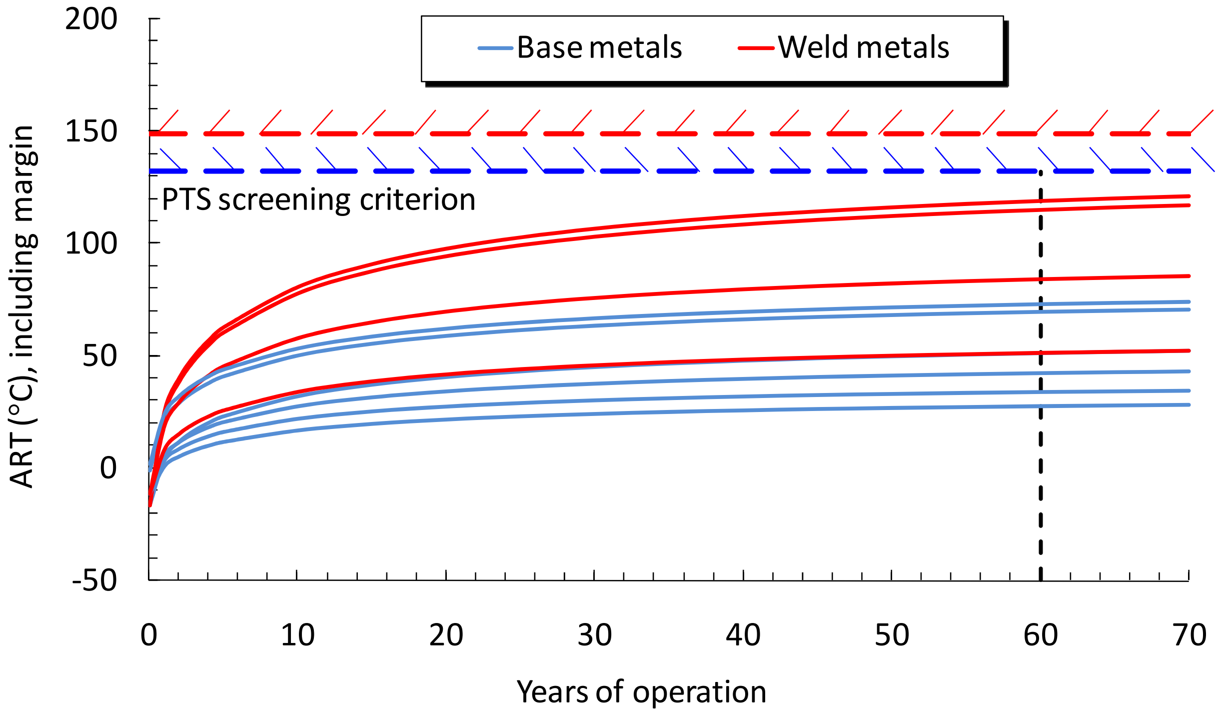

Figure 4 - Application of regulatory PTS screening criteria to materials from Doel I (two base and one weld metals), Doel II (two base and two weld metals) and Tihange I (two base and one weld metals). NOTE: curves should be compared to dashed lines of the same color (blue for base metals, red for weld metals).

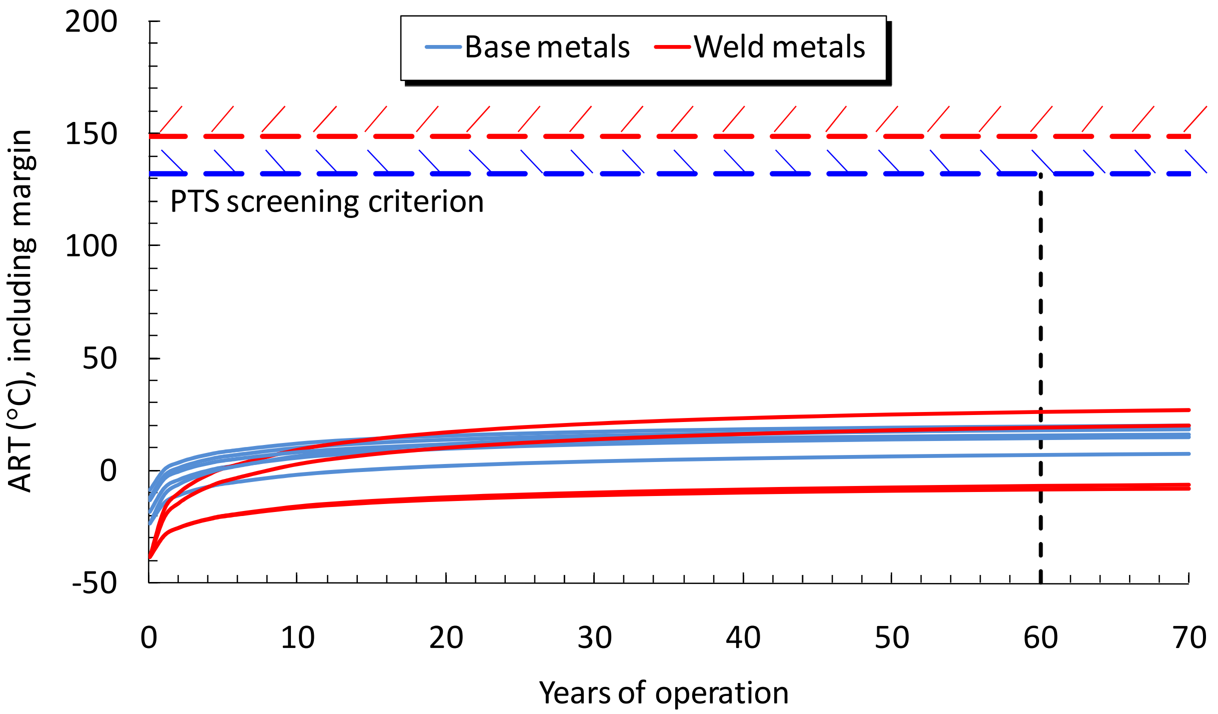

Figure 5 - Application of regulatory PTS screening criteria to materials from Doel III (one base and one weld metal), Doel IV (two base and one weld metals), Tihange II (one base and one weld metal) and Tihange III (two base and one weld metals). NOTE: curves should be compared to dashed lines of the same color (blue for base metals, red for weld metals).

For some of the surveillance capsules (see Table 2 on page 26), fracture toughness tests were performed on reconstituted precracked Charpy specimens, in order to obtain direct values of the transition temperatures based on fracture toughness measurements (RT_{To}).

From Figure 6 it appears that in most cases, an additional safety margin is obtained with respect to the conventional, Charpy-based RT_{NDT} value: this is demonstrated by the fact that most data points in Figure 6 lie below the 1:1 line. It is important to emphasize here that while RT_{NDT} is obtained indirectly through a correlation using the Charpy impact data, RT_{T0} is a direct and reliable measure of the transition temperature obtained from actual fracture mechanics tests in accordance with the Master Curve methodology.

It was also shown [6-1] that additional safety margins are larger for data points which correspond to first generation units (70's).

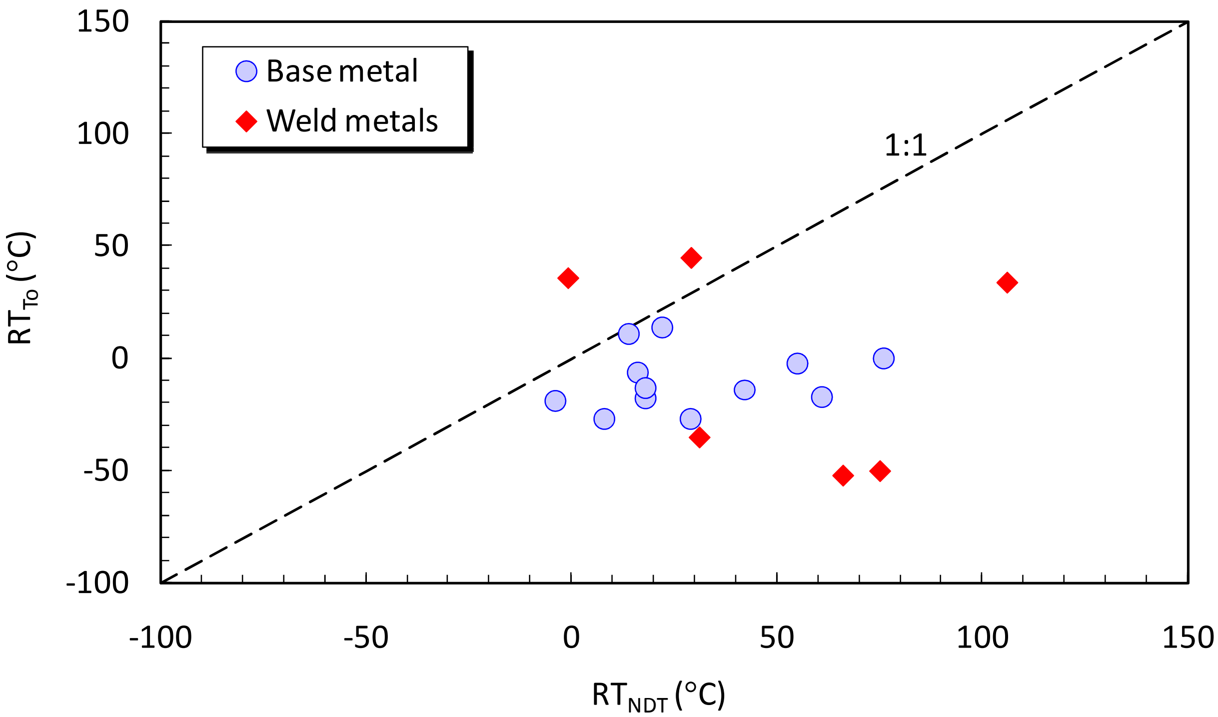

Figure 6 – Comparison between Charpy-based (RT_{NDT}) and fracture toughness-based (RT_{To}) RPV evaluation approach for several Belgian units.

Previous evaluation was performed within the current regulatory context. Although the change of the legislation is a slow process, the NRC in the US is actively working on updating the regulation related to RPV assessments. The impact of the proposed changes that are under current evaluation by the NRC have also been evaluated for the Belgian NPP's.

There are three aspects that can affect our evaluations.

(a) The use of the Master Curve on a plant-by-plant justification basis [6-2,6-3]. There is still a worldwide effort devoted to developing and extending the Master Curve concept, and future developments will certainly affect the procedure for the determination of the fracture toughness/temperature curve. As we have mentioned before, using direct fracture toughness measurements instead of the Charpy-based methodology can often reduce unjustified overconservatism.

(b) The change of the PTS screening criteria [6-4]. Current criteria are based on predominantly empirical assumptions and are extremely conservative. The change of the PTS screening criteria will also allow deriving additional safety margins, as the generic safety assessment performed by the NRC in [6-4] demonstrates that, using more realistic hypotheses, the limits can be relaxed without reducing the safety level.

(c) The revision of the embrittlement correlations [6-5]. The embrittlement trend curves that are used today within the current regulations have shown their limitations. On one hand, they do not satisfactorily represent a number of experimental data (such as low copper steels or high neutron fluences). On the other hand, they often overestimate the actual embrittlement. A number of international studies are in progress to reduce such overconservatism and allow a more accurate evaluation of embrittlement and a reduction of the associated uncertainties. The effects of these new correlations were evaluated for the Belgian plants and were found not to substantially affect the safety margins [6-6].

It can therefore be stated that the possible changes in the current regulation that are under evaluation by the NRC, will provide additional safety margins for the Belgian NPP's.

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7 Conclusions

In this report, we aimed at providing the GEMIX working group a well-documented expert opinion on the technical feasibility of extending the operating license of Belgian nuclear power plants, based on the integrity and safety of the most fundamental and irreplaceable component: the reactor pressure vessel. Obviously, a full justification for plant life extension has to rely on a thorough safety assessment of all the components of the plant and not just the RPV.

The most important remarks emerging from our analyses can be summarized as follows.

- In the United States (where several reactors have identical or similar design to the Belgian units), 54 of the currently 104 operating reactors have already been granted an operating license extension from 40 to 60 years of operating, while 21 have already applied and 23 are expected to file an application in the coming years. US safety authorities (NRC) are also seriously considering the possibility of extending the lifetime of several reactors beyond 60 years (project "Life beyond 60").
  In France (some of the Belgian RPV's were fabricated according to French specifications), all 34 plants of the 900 MW class have been granted a 10-year operation extension beyond 30 years. EDF has also announced plans to operate all 59 French reactors up to 60 years at least.
  Many other countries worldwide have seen, or are considering, an extension of the operating lifetime of some of their reactors.

- Among the fleet of US nuclear power plants, we have identified 16 reactors which are similar to the Belgian units. Among those, 8 have been granted a 60-year license extension, 2 have requested it and 6 (all relatively recent plants) are expected to apply for an extension in the coming years.

- At SCK•CEN, the evolution of the mechanical properties of the RPV beltline materials of the Belgian units is monitored both from a strictly regulatory point of view ("conventional" surveillance), and using advanced techniques and methodologies. To date, SCK•CEN has issued 72 technical reports on the analysis of the 28 surveillance capsules that have been extracted from the seven Belgian units. 17 additional reports have been written to detail supplementary investigations conducted on several materials from the Doel and Tihange reactors, as well as 6 documents concerning materials from the now-decommissioned BR3 unit. More confidential technical publications from SCK•CEN are available which address the characterization and integrity assessment of several foreign RPV's (Spain, Switzerland, Argentina, Brazil, Bulgaria, South Korea, etc). The advanced surveillance allows extracting additional information from test results and gaining more insight on the phenomena induced by neutron irradiation.

- Using the strictly regulatory approach based on the results of Charpy impact tests, we have shown that sufficiently safety margins exist with respect to the most severe accident scenario (pressurized thermal shock) for all the pressure vessels of the 7 Belgian nuclear power plants, and considering an operation extension to 60 years of service. A more advanced analytical approach, primarily based on direct fracture toughness measurements, as well as consideration of the current and near-future developments of the legislation, provide additional safety margins at the 60-year mark or even beyond.

As a final remark, we would like to note that, irrespective of the future energy scenario in our country and based on a number of strategic considerations, it will be important to have a strong safety regulatory body and to preserve the level of research and technical expertise which can assist all policy-making decisions in Belgium.

Additional general references on plant life management and extension

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