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Summary

A state of the art safety assessment of the KCB RPV has been performed for Long Term Operation. The reason is that originally it was foreseen that NPP Borssele was going to operate from 1973 until the end of 2013, 40 years of operation. The original brittle fracture assessment was based on this operation period. Because of an agreement with the government in 2006 a new operation time came up: further operation until the end of 2033 (60 years of operation). To prove that the NPP is safe for 60 years of operation new ageing assessments, i. e. embrittlement of the RPV, needed to be performed regarding the new operation time. For the new assessment EPZ decided to go for an assessment according to state of the art regulation and knowledge for this topic.

By means of extensive thermal hydraulic analyses, fracture mechanics calculations and aging assessment including irradiation embrittlement, it could be proven that the safe operation of the KCB RPV is guaranteed for all load cases with large safety margins. The new operation time until the end of 2033 (60 years of operation) could even be extended by many years with sufficient safety margins against brittle failure of the RPV. The calculated allowable reference temperatures for postulated defects in case of the most severe loading of small break loss of coolant accidents are far away from the reference temperatures for nil-ductility transition of the material and are also very much above the limits given in existing standards, like the German KTA rules, which are most relevant for the Siemens/KWU design plant.

List of revisions

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B	2010-07-12	Chapter 5.7.2	Change of withdrawal time for SOP 3 and SOP 4	Page 33

Table of Content

Abbreviations 6

1. Introduction 9

2. General description of the KCB Reactor Pressure Vessel 11
2.1 RPV Materials and Manufacture 13
2.1.1 Base Material 22NiMoCr3-7 of the Shells 13
2.1.1.1 Mechanical technological testing 13
2.1.1.2 Chemical composition 13
2.1.2 RPV Circumferential Weld W03 14
2.1.2.1 Mechanical technological testing 15
2.1.2.2 Chemical composition 15
2.1.3 Cladding of the lower and upper cylindrical shell 16

3. Safety Standards, Guidelines and Procedures 17
3.1 German Safety Standards 17
3.2 USA Safety Standards 18
3.2.1 General 18
3.2.2 ASTM E 185 and ASTM E 2215 19
3.2.3 ASTM E 1921-09a 19
3.2.4 Reg. Guide 1.99 19
3.2.4.1 Revision 1 19
3.2.4.2 Revision 2 20
3.2.5 10 CFR Part 50 20
3.3 IAEA Guideline TRS No. 429 20
3.4 Recent Changes in the Standards and their Impact on the Application 20

4. Description of RPV Aging Effects during Operation 21
4.1 Thermal Aging 21
4.2 Irradiation Induced Aging 21
4.2.1 Ferritic RPV Steels 21
4.2.2 RPV Cladding 21
4.2.3 Special Effects 22
4.2.3.1 Impact of Hydrogen 22
4.2.3.2 Impact of γ- Radiation 22
4.2.3.3 Dose Rate - / Flux Effect 22

5. Irradiation Surveillance Program of the KCB RPV 22
5.1 Introduction 22
5.2 Monitored Materials 24
5.3 Irradiation Sets SOP 0, SOP 1 and SOP 2 24
5.3.1 Materials and Specimens 24
5.3.2 Irradiation of the Specimens 24
5.3.3 Testing 24
5.3.4 Discussion and Evaluation of the Test Results 25
5.4 Irradiation Sets SOP 0a, SOP 3 and SOP 4 26
5.4.1 Materials and Specimens 26
5.4.2 Irradiation of the Specimens 27

5.4.3 Testing of SOP 0a 27
5.4.4 Discussion and Evaluation of the Test Results 27
5.5 Irradiation Data from Research Projects (CARISMA) 28
5.6 Neutron Fluence Calculation 29
5.6.1 Introduction 29
5.6.2 Neutron Fluence Calculations for the RPV and for the Irradiation Capsules SOP 2, SOP 3 and SOP 4 29
5.6.3 RPV Fluence Extrapolation to 60 Operational Years 31
5.6.4 Impact of MOX Fuel on the Fast Neutron Fluence of the RPV and the Irradiation Capsules SOP 3 and SOP 4 31
5.6.5 Discussion and Evaluation of the Results 32
5.6.6 Verification of Neutron Fluence Calculations with Scraping Samples 32
5.7 Concept for Irradiation Surveillance to demonstrate 60 Operational Years 33
5.7.1 Boundary Conditions 33
5.7.2 Withdrawal Schedule for SOP 3 and SOP 4 33
5.7.3 Alternative Measures 34

6. Safety Assessment of the KCB RPV 34
6.1 Pressurized Thermal Shock Analyses 34
6.1.1 Introduction 34
6.1.2 Concept and Procedure 35
6.1.3 Thermal-hydraulic Calculations 35
6.1.3.1 Boundary Conditions 36
6.1.3.2 Selection of Leak Sizes and Injection Configurations 36
6.1.3.3 Transients under Investigation 37
6.1.4 Fracture mechanics Analyses 42
6.1.4.1 Geometry 43
6.1.4.2 Postulated flaws 43
6.1.4.3 Material data 44
6.1.4.4 Finite Element Model 44
6.1.4.5 Loading and boundary conditions 44
6.1.4.6 Residual stresses 44
6.1.4.7 Analytical Model 44
6.1.5 Results and Recommendations 44
6.2 Evaluation of the Irradiation Effects 45
6.2.1 Reference Temperatures RT_{NDT} in the Unirradiated State 45
6.2.2 Adjusted Reference Temperatures RT_{NDTj} 46
6.2.2.1 Evaluation according to German KTA 3203 46
6.2.2.2 U.S. Reg. Guide 1.99 Rev. 2 Analysis 46
6.2.2.3 Other State-of-the-Art Trend Curve Analyses 47
6.2.3 Reference Temperatures RT_{T0} in the Unirradiated State 48
6.2.4 Adjusted Reference Temperatures RT_{T0j} 48
6.2.5 Discussion of the Results 50
6.3 Structural Integrity Assessment 50
6.3.1 According to U.S. 10 CFR 50 51
6.3.2 End of Life Prediction 51
6.3.3 Allowable Pressure during Service Conditions Level A / B (ASME) 52
6.4 Final Assessment for 60 Years LTO 53

7. Summary and Conclusion 53

8. References 54

9. List of Figures 61

10. List of Tables 62

Abbreviations

ACC	Accumulator
AF	Assessment Fluence
ART	Adjusted Reference Temperature
A_v	Absorbed impact work [J]
BM	Base Material
CRP	Coordinated Research Project
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
ECN	Energieonderzoek Centrum Nederland (Energy research Center of the Netherlands)
EFPD	Effective Full Power Days
EFPY	Effective Full Power Years
EoL	End of Life
EPZ	Elektriciteits-Produktiemaatschappij Zuid-Nederland
GHH	Gute Hoffnungshütte
HAZ	Heat Affected Zone
HTC	Heat Transfer Coefficient
KCB	Kern Centrale Borssele
KEMA	Keuring Elektrotechnisch Materieel Arnhem (Authority in energy consulting and testing & certification)
K_{Ic}	Static fracture toughness [MPa√m]
K_{Im}	Stress intensity factor corresponding to membrane tension [MPa√m]
K_{IR}	Reference fracture toughness [MPa√m]
K_{It}	Stress intensity factor corresponding to radial thermal gradient [MPa√m]
K_{JC}	Stress intensity factor [MPa√m]
K_v	Absorbed impact work [J]
KWU	Kraftwerksunion
LHSI	Low head safety injection pump
LOCA	Loss of Coolant Accident
LTO	Long Term Operation
Marrel Fr.	Marrel Freres

MC	Monte Carlo
MCL	Main Coolant Line
MHSI	Medium head safety injection pump
M_m	Correction factor of the geometry depending on flaw size and location
MOX	Mixed Oxide
NDE	Non-Destructive Examination
NDT	Nil Ductility Transition
NDT-T	Nil-Ductility Transition Temperature
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRG	Nuclear Research & consultancy Group
Orientation	
at forgings	
	T transverse to the main direction of deformation or parallel to the rotational symmetry axis (axial)
	L longitudinal to the main direction of deformation
	S in thickness direction, perpendicular or radial to the rotational symmetry axis
at welds	
	L longitudinal to the welding direction
	T transverse to the welding direction or parallel to the rotational symmetry axis (axial)
	S in thickness direction, perpendicular to the weld
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RDB	Reaktordruckbehälter
RDM	Rotterdamsche Droogdok Maatschappij
RESA	Reaktorschnellabschaltung (Fast Reactor Shut Down)
RPV	Reactor Pressure Vessel
RSK	Reaktor-Sicherheitskommission
$RT_{limit\ curve}$	Upper bound curve according to KTA 3203
RT_{NDT}	Reference Nil-Ductility Transition Temperature
RT_{NDTj}	Adjusted Reference (Nil-Ductility Transition) Temperature for the irradiated state
RT_{PTS}	Reference Temperature in the PTS case

RT_{T0}	Static fracture toughness Reference Temperature according to the Master curve concept; $RT_{T0} = T_0 + 19.4$ K
RT_{T0j}	Adjusted Reference (Nil-Ductility Transition) Temperature according to the Master curve concept for the irradiated state
SOP	Staal Onderzoek Programma
T	Temperature [°C]
T_0	Reference Temperature, related to the temperature on the Master curve at K_{Jc} = 100 MPa$\sqrt{m}$
$T_{0.9}$	Temperature at which a mean lateral extension LE,T – compensating curve determined by C_v - transversal specimens exceeds > 0.9 mm
T_{41}	Temperature at which an averaged AV,T – compensating curve determined by C_v - transversal specimens reaches an impact work of 41 J
T_{50}	Temperature at which the crystalline shearing zone of C_v - transversal specimens reaches 50 %
T_{68}	Temperature at which a mean AV,T –compensating curve determined by C_v - transversal specimens reaches an impact work of 68 J
$\Delta T_{0.9}$	Temperature margin between the mean compensating curves from the notched bar impact test for the unirradiated and irradiated state with an lateral expansion of 0.9 mm
ΔT_{41}	Temperature margin between the mean compensating curves from the notched bar impact test for the unirradiated and irradiated state with an impact work of 41 J
ΔT_{50}	Temperature margin between the mean compensating curves from the notched bar impact test for the unirradiated and irradiated state with an crystalline shearing zone of 50 %
ΔT_{68}	Temperature margin between the mean compensating curves from the notched bar impact test for the unirradiated and irradiated state with an impact work of 68 J
T_{AV} (0.9 mm)	Temperature at which a lateral expansion not less than 0.9 mm is reached with the notched bar impact test by using Charpy-V-specimens (lower bound)
T_{AV} (68 J)	Temperature at which an impact work not less than 68 J is reached with the notched bar impact test by using Charpy-V-specimens (lower bound)
TÜV	Technischer Überwachungsverein
UMD	Unstable Matrix Damage
UPTF	Upper Plenum Test Facility
USE	Upper Shelf Energy
WM	Weld Material
WPS	Warm Pre-Stress

1. Introduction

The Nuclear Power Plant Borssele (KCB) is a 2-Loop Pressurized Water Reactor (PWR) with a power of 1366 MW_{th}. It was built during 1969 to the beginning of 1973 from the nuclear power plant manufacturer Siemens/KWU and started operation in the end of 1973.

At the end of 2013 KCB will exceed an operating time of 40 years. The intention of the owner of NPP Borssele, EPZ, is to extend the operational time for further 20 years until 2034 (Borssele should be disconnected from system in the end of 2033 in accordance to the agreement with the Netherlands government), and to get a license for such Long Term Operation (LTO).

In the frame of the LTO until 2034 a safety assessment of the Reactor Pressure Vessel (RPV) including the assessment of irradiation induced ageing of the KCB RPV has to be carried out. In the 70s one irradiation surveillance program was performed on the KCB RPV with the unirradiated reference set SOP 0 and the irradiation sets SOP 1 and SOP 2. The evaluation of the fluence detectors was done in Petten/Arnhem. A second irradiation surveillance program with the unirradiated set SOP 0a and the irradiation sets SOP 3 and SOP 4 has been started in 2007.

The objective of this safety report is to prove the integrity of the KCB RPV for a Long Term Operation up to 60 operating years. Therefore, the structural integrity of the RPV with respect to operation, irradiation surveillance and PTS analysis is assessed. For the proof of the KCB RPV integrity the results of the project CARISMA, in which highly irradiated test materials similar to those of KCB are tested and evaluated with the RT_{NDT} - and the Master curve concepts, are used as well for the assessment of the KCB RPV. Moreover a withdrawal schedule for the in the RPV inserted irradiation sets SOP 3 and SOP 4 is proposed.

Finally, the RPV safety of KCB is evaluated in terms of the up-to-dateness of the assessment methods used and by a general benchmark of the KCB results with RPV safety assessment data worldwide.

2. General description of the KCB Reactor Pressure Vessel

Figure 2 shows the KCB RPV with an overview of the single connecting welds and cladding of the RPV. The respective segments are marked with numbers where the lower head has the No. 01 and the closure head dome plate Nr. 07. The height of the RPV is approx. 9500 mm and the inner average diameter in the region of the cylindrical shells is 3730 mm. The wall thickness of the cylindrical part with cladding is ca. 188 mm.

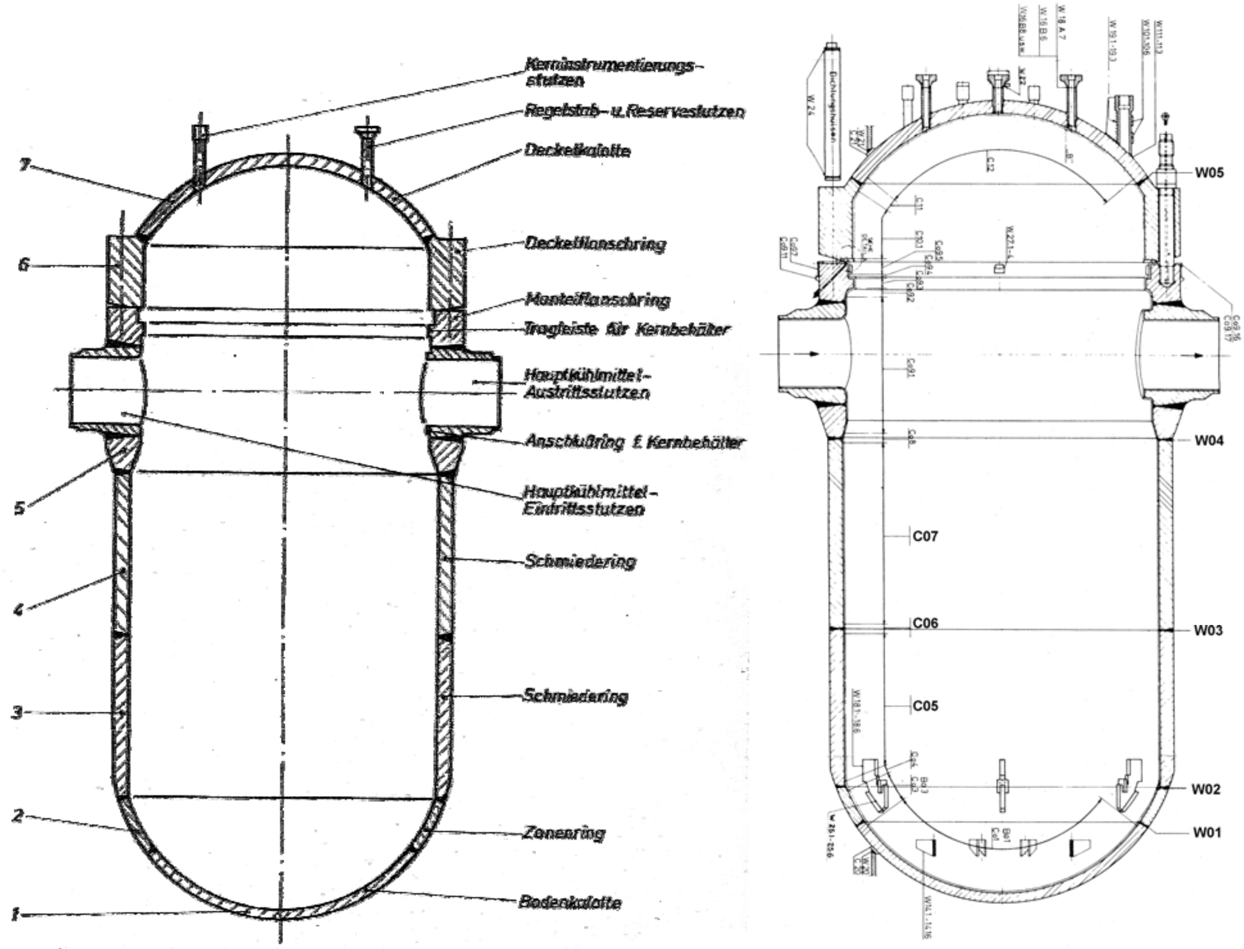

Figure 2: Reactor Pressure Vessel of NPP Borssele

2.1 RPV Materials and Manufacture

The material manufacture as well as the RPV manufacture was documented with acceptance tests. All investigations were performed with presence of the regulator TÜV. Instructions for stress-relief heat treatment of components, for pressure test, for surface crack examination, for radiographic examination and ultra-sonic examination are stipulated in RDM specifications and their compliance is documented in material, weld seam and cladding examination records.

2.1.1 Base Material 22NiMoCr3-7 of the Shells

The cylindrical shells are made of the material 22NiMoCr3-7, which is comparable to the ASTM A-508 Cl. 2.

2.1.1.1 Mechanical technological testing

The results of the tensile tests, the Charpy-V and Pellini tests, , as well as the chemical composition of the single RPV section were obtained from monitoring annealed specimens and are given more in detail . and

meet the minimum requirements on the component according to the Siemens specification

Drop weight tests according to Pellini were carried out for all forging rings of the RPV and the two dished parts, where for the two cylindrical shells 03 and 04 the NDT temperature was determined, Table 6 (page 45). For the remaining parts it was documented if specimens are broken or not. Additionally the NDT temperature for the test coupon of the weld W03 was determined, Table 6.

2.1.1.2 Chemical composition

Table 1 gives an overview of the content of the elements phosphor, nickel, copper and manganese of the core sections cylindrical shell 03 and 04 as well as for the circumferential weld W03, connecting these two shells.

Section	Section number	Heat number	P	Ni	Cu	Mn
			wt%			
Lower cylindrical shell	03	523512	0.011	0.83-0.89	0.03-0.04	0.71
Upper cylindrical shell	04	523612	0.009-0.010	0.84-0.85	0.04	0.72-0.77
Weld	W03		0.011	0.05	0.09	1.98

Table 1: P-, Ni-, Cu- and Mn- content of cylindrical shell 03, 04 and of the circumferential weld W03

2.1.2 RPV Circumferential Weld W03

The circumferential welds W01 to W09 of the RPV segments are automatic submerged flux cored arc welding. As filler material, the electrode-flux combination S3Mo/Grau L0 was used.

In the following the circumferential weld W03 is regarded more precisely since it represents the connecting weld of the two cylindrical shells 03 and 04 lying in the beltline region.

2.1.3 Cladding of the lower and upper cylindrical shell

For the double layer strip weld cladding of the lower and upper cylindrical shells 03 and 04 with a thickness of ca. 7 mm for the first layer Thermanit 22/11 ENB with the flux OP 70 Cr ELC was used, which was overcladded with CN 21/10 Nb-BS with the flux Ellira 10. Three layers were manufactured by electrode manual welding where for the first layer the over alloyed electrode FOX CN 24/13 Nb-A and for the further two layers Fox CN21/10 Nb-Kb were used, With the monitoring annealed specimen 1597-B following examinations were carried out:

Examinations	Results
Bending test specimens	No indication
Stability against intercrystalline corrosion	No indication
Examination for underclad cracks	Free from defects
Examination of surface roughness	RT 1: R_t ± 25; RT 4: R_t ± 15
Ultrasonic examination	No claims
Surface crack examination	No error indicators

3. Safety Standards, Guidelines and Procedures

3.1 German Safety Standards

In Germany, following safety standards are mainly decisive for the RPV assessment:

KTA 3201.1: Materials and Product Forms [33]

KTA 3201.2: Design and Analysis [34]

KTA 3201.3: Manufacture [35]

KTA 3203: Surveillance of the Irradiation Behavior of Reactor Pressure Vessel Materials of LWR Facilities [36]

The safety assessment is based on the RT_{NDT} concept. The determination of the K_{Ic}-T curve for the material conditions at End of Life (EoL) is done indirectly by determining a K_{Ic} reference curve with the shift of the A_v-T curve. For the material data the acceptance values and the result from the irradiation surveillance programs for the particular plant (set 1 unirradiated, sets 2 and 3 irradiated) are considered. Therewith the irradiated sets have to be around 50 % and have to cover 100 %, respectively, for the entire operating period of the RPV.

KTA 3203/01

Most of the irradiation surveillance programs for reactor pressure vessels of German plants are meanwhile tested and as far as possible evaluated [37], Thereby it became clear that the design curves of the KTA 3203/84 are not conservative at low fluences as it is particularly the case for BWR. Exceedings were not safety related in any case since conservatisms could be utilised from the RT_{NDT} determination in the unirradiated state. This fact was accommodated in the up-dated version KTA 3203/01. This version contains no longer design curves based on the transition temperature shift ΔT_{41}, instead a RT_{limit} curve is considered, see Figure 6.

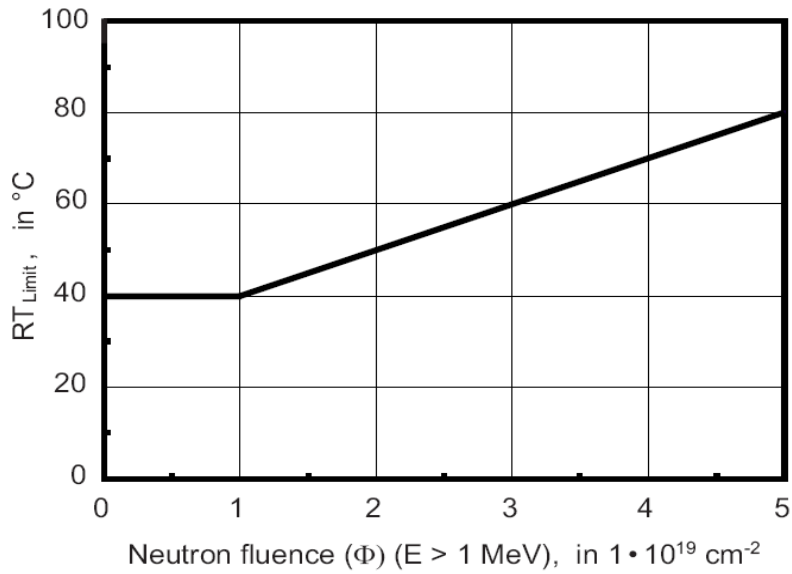

Figure 6: $RT_{NDTlimit}$ curve of the KTA 3203/01 [36]

This RT_{limit} is statistically verified for RPV materials of German nuclear power plants. It can be considered for all materials in the core region which lie in between the defined scope concerning the Cu and Ni content. The RT_{limit} has to be verified with RT_{NDTj} as result from an irradiation program. The determined RT_{NDTj} can be used for the proof of safety against brittle failure.

Additionally the possibility to validate the RPV with directly measured fracture toughness values according to the Master curve (T_0) concept was adopted to the KTA 3203/01.

3.2 USA Safety Standards

3.2.1 General

In the American safety standards a consistent separation between design and implementation guidelines exists for irradiation surveillance programs which are defined in the ASTM standard E 185 [44] (formerly also E 184). The requirements for the interpretation, transferability of the results and the determination of an adjusted reference temperature for “End of Life“ are fixed in the Reg. Guide 1.99, Rev. 2 [40] and the standard for the implementation of the safety analysis represents the 10 CFR 50 [41].

The differences between the implementation regulations of KTA 3203 and ASTM E 185 are marginal. Differences that are more significant exist between Reg. Guide 1.99, 10 CFR 50 and the adequate parts of the KTA.

3.2.2 ASTM E 185 and ASTM E 2215

The ASTM E 185 is based on a first edition from 1962, which was adjusted in the renewal years. It demands an irradiation surveillance program in which specimens of a base material and a weld material in an unirradiated and three irradiated states have to be investigated. The current valid edition [44] demands as a main change in contrast to the older edition no more HAZ specimens.

While former editions described the performance of irradiation surveillance programs the ASTM E-185 with the current edition from the year 2002 was separated in several standards due to its complexity. The current ASTM E 185-02 [44] now describes the procedures for designing an irradiation surveillance program and the minimum requirements. The ASTM E 2215-02 [45] describes the procedure for examination and evaluation of irradiation surveillance capsules. In general, the material changes due to neutron irradiation are under surveillance like the transition temperature shift, the upper shelf energy changes and the tensile test properties. Alternative methods represent the fracture mechanics examination or additional hardness testing.

3.2.3 ASTM E 1921-09a

The ASTM E 1921-09a “Standard Test Method for Determination of Reference Temperature, T_0, for Ferritic Steels in the Transition Range” [46] describes an investigation method to determine the reference temperature T_0 that characterizes the fracture toughness of ferritic steels. Basis of the ASTM E 1921 and of the Master curve concept, respectively, are the works of K. Wallin [47].

Besides the standardization of the T_0 determination in the ASTM E 1921 [46], meanwhile a code case 629 [48] exists that regulates the use of the T_0 for the proof of safety.

3.2.4 Reg. Guide 1.99

The Reg. Guide 1.99 contains guidelines for the interpretation and transformation of the results and the determination of a fluence depending Adjusted Reference Temperature (ART = RT_{NDTj}) for “End of Life“ to proof the safety of operation. There are two cases to be distinguished:

- Surveillance data not available, Position 1
- Surveillance data available, Position 2

3.2.4.1 Revision 1

The ART determination according to Reg. Guide 1.99 Revision 1 [49] Position 1 depends on the content of the residual elements copper and phosphor, which are supposed to cause material embrittlement. If credible surveillance data are available, the Reg. Guide permits to use them for the prediction of the ART at other fluences by inter- and extrapolation.

3.2.4.2 Revision 2

In the Reg. Guide 1.99 Revision 2 [40] the important residual elements responsible for the radiation-induced embrittlement and the equation for the fluence depending ART curve are updated. Due to secondary importance of phosphorus it is replaced by nickel. The ART prediction is based on a chemistry factor and a fluence function.

3.2.5 10 CFR Part 50

The toughness requirements are defined in the 10 CFR 50, App. G [41]. App. H [42] contains the specification for irradiation surveillance programs. The toughness requirements according to App. G as well as the specifications for the irradiation surveillance programs are comparable with the adequate KTA standards. Marginal differences exist, but they do not concern the entire concept. Significant differences appear for the treatment of the loss-of-coolant accident (LOCA). In this designed part (§ 50.61) of the 10 CFR 50 [43] the ”screening criterion” is adopted where a specific RT_{PTS} is defined and probabilistically justified.

3.3 IAEA Guideline TRS No. 429

The IAEA Guideline TRS No. 429 [50] is a guiding principle for the application of the master curve concept, which was created in the scope of an IAEA Coordinated Research Project (CRP). The background and the potential applications are defined in this guideline. The probabilistic procedure with considering error distributions and failure occurrence probability as well as the deterministic procedure based on a RT_{T0} are regarded. Furthermore special cases are discussed, for example the transfer of irradiation data for materials with differing chemical composition, the use of generic RT_{T0} data without available measured RT_{T0} values or the combination of unirradiated or irradiated RT_{T0} values with a transition temperature shift for inter-/extrapolation to other fluences. The combination of an unirradiated RT_{T0} with a transition temperature shift is thereby of special interest if only data in the unirradiated state are available.

3.4 Recent Changes in the Standards and their Impact on the Application

The nuclear power plant Borssele started operation in the year 1973 and was built based on the standards valid until then. Since that time following important changes in the safety standards were implemented:

- ASTM E 185-02 [44] / ASTM E 2215-02 [45] (compared to ASTM E 185-70)
    - Specimen removal direction today transverse (T-L) to the main direction of deforming, prior transverse to the main direction of loading of the vessel, which means L-S and L-T, respectively (adopted in ASTM E 185-73). (However, the T-L orientation was already required in all editions of the KTA 3203)
    - HAZ is no longer addressed (adopted in ASTM 185-93).
    - Recommendation to perform notched bar impact test instrumented (in ASTM E 2215-02 [45])
- 10 CFR 50: probabilistic approach (“screening criterion”)
- KTA 3203/01 [36] (replaces KTA 3203/84):
    - HAZ is no longer addressed
    - Performance of notched bar impact test instrumented

- o Use of an upper bound RT_{limit} for the reference temperature with respect to ART instead of design curves based on the transition temperature shift ΔT_{41}.
- o Possibility to assess the RPV with directly measured fracture toughness values according to the T_0 concept

4. Description of RPV Aging Effects during Operation

4.1 Thermal Aging

The contained copper concentration has a decisive impact on the thermal aging of materials used for the manufacture of reactor pressure vessels and thus being exposed to radiation. Copper is assumed to be the initiator for precipitation formation. The higher the copper content the more precipitations are formed and the stronger is the increase in yield strength. An increase in yield strength implies material embrittlement. With high copper content the shift in the K_v-T curve between unirradiated and irradiated material is dominated by the formation of copper precipitations. For this reason the copper content is limited for materials exposed to high thermal impact.

Thermal aging may occur at copper concentrations higher than 0.2 wt% and temperatures above 300 °C. For the KCB RPV the copper content was determined for the two cylindrical shells 03 and 04 to 0.04 wt% and for the weld metal to 0.09 wt%, Table 1. According to the information of EPZ the averaged irradiation temperature is 305 °C. Since the copper content is clearly under 0.2 wt% for the base metal as well as for the weld metal thermal aging can be excluded for the KCB RPV,

.

4.2 Irradiation Induced Aging

4.2.1 Ferritic RPV Steels

The fast neutrons (E> 1 MeV) generated at the nuclear fission, impinge on the RPV surface and produce defects during the cascade, e.g. vacancies.

A high concentration of formed vacancies leads to an increase of diffusion in the regions affected by irradiation, whereby precipitations are formed enforced. Precipitations and matrix defects like vacancies, vacancy clusters and enriched regions with impurities cause an increase in yield strength and a decrease in toughness. Thus the temperature shift ΔT_{41} increases and the upper shelf energy decreases in contrast.

The chemical element copper has besides the impact on thermal aging, see chapter 0, also a strong impact on irradiation-induced aging. Additionally also nickel and phosphor have to be considered. How the KCB RPV is affected by irradiation is described in chapter 6.2.

4.2.2 RPV Cladding

The neutron fluence of the austenitic cladding surface of the PWR RPV is slightly higher than the neutron fluence of the inner ferritic RPV wall and thus practically in its scatter band.

In examinations on a representative RPV cladding of Sizewell B, which is comparable to the RPV claddings in Siemens/KWU plants concerning chemical composition and manufacturing method, no change in fracture toughness was found, [52]. Thus, no significant changes in material behavior and resulting material properties are expected.

4.2.3 Special Effects

4.2.3.1 Impact of Hydrogen

Besides the aging of RPV-steels caused by neutrons a possible synergistic effect between hydrogen and the monodisperse matrix defects formed by irradiation has to be considered.

Irradiation induced defects represent no favored locations for a higher residence probability for hydrogen under RPV service temperature. Therefore, at service temperature there is no elevated sensitivity for embrittlement due to hydrogen.

4.2.3.2 Impact of γ - Radiation

Radiation induced matrix damage can be caused by fast neutrons as well as gamma radiation (γ).

At 290 °C the impact of γ – radiation is evaluated to be negligible since the matrix defects induced by γ – radiation show strong annealing effects at operating temperatures of 290 °C in power reactors.

4.2.3.3 Dose Rate - / Flux Effect

Since the irradiation behavior is examined by means of accelerated irradiation specimen capsules the existence of a flux effect and dose rate effect, respectively, of the neutron flux is discussed.

In the scope of the EPRI „Workshop on dose rate effects in RPV materials“ [54] it was stated consistently that for low cupper-bearing C-Mn and MnMoNi steels no flux effect can be detected as long as the threshold for formation of unstable matrix damage UMD is not exceeded [55]. This threshold is approximately at a flux density of 1E+12 n/cm^2s^1 (E> 1 MeV). For the irradiation surveillance programs of the Borssele plant this value is clearly lower. For SOP 2 the flux was 1.04E+11 n/cm^2s^1 and for SOP 3 and 4 the average neutron flux (E> 1 MeV) is 1.26E+11 n/cm^2s^1 . Moreover, the chemical composition and the lead factors of the KCB irradiation specimens meet the requirements of KTA 3203 [36] where special evaluations for considering the neutron flux density are not required for RPV materials meeting these requirements.

5. Irradiation Surveillance Program of the KCB RPV

5.1 Introduction

The RPV irradiation surveillance program serves to determine experimentally, by means of accelerated irradiation capsules containing specimens from the ferritic original materials, the strength and toughness properties of base and weld materials in the core beltline region of the RPV as a function of defined neutron irradiation [36].

In Figure 7 the region of the KCB RPV to be monitored is presented. Therewith it is about the two cylindrical forgings (cylindrical shell (l), Pos. 03; cylindrical shell (u), Pos. 04) and the weld in-between (W 03), see also Figure 2.

The change of the material properties due to neutron irradiation is determined on specimens taken from the RPV core shells to be monitored before RPV manufacturing. They are produced according to the same method using the same welding materials and applying the same heat treatment (monitored or simulative, respectively). Usually, tensile specimens, Pellini specimens, notched-bar impact specimens and fracture toughness specimens are used.

The surveillance specimen capsules are inserted between the inner RPV wall and the core internals in the region of higher flux, so that the integrated flux of fast neutrons at the location of the specimens accounts for a multiple of the integrated flux on the inner RPV wall over the same time.

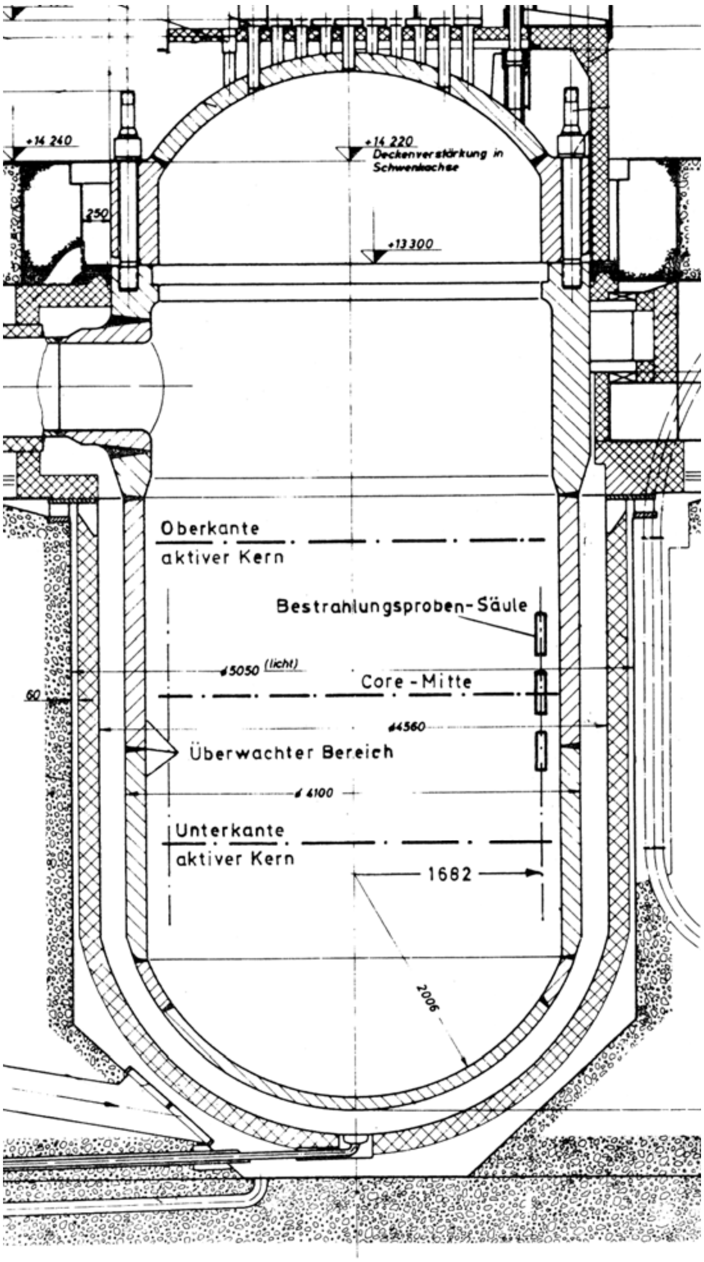

Figure 7: Position of the surveillance specimens in the RPV according to

5.2 Monitored Materials

For the KCB irradiation surveillance program original base material of the cylindrical shells 03 and 04 were used as well as weld metal and HAZ of the test coupon W03 in order to monitor the RPV core beltline region. The weld specimen W 03-1M is in accordance with the original circumferential weld W03 concerning welding material, powder, weld shape, welding conditions and heat treatment. The base material of the weld specimen stems from a plate with the same properties like the cylindrical shells, as far as possible, but with higher copper content as the base material of the cylindrical shells 03 and 04 .

5.3 Irradiation Sets SOP 0, SOP 1 and SOP 2

The objective of the first irradiation surveillance program was to proof a safe operation of the KCB RPV for 40 years.

The unirradiated reference set of the first irradiation surveillance program is called SOP 0 and the two irradiation sets are called SOP 1 and SOP 2 (SOP = Staal Onderzoek Programma).

5.3.1 Materials and Specimens

The first KCB irradiation surveillance program was designed for the two base metals Ring 03 and Ring 04, and the weld metal W03 according to the ASTM E 185-70 standard practice for RPV surveillance tests

The entire sample size of an irradiation set was stored in 6 capsules, whereof each 3 capsules were combined to one irradiation column. Thus each irradiation set consists of 2 irradiation columns.

Besides the specimens additionally flux monitors were placed in the capsules. Also temperature monitors in the form of alloys with known melting point for checking purposes were stored inside the capsules.

5.3.2 Irradiation of the Specimens

The two first irradiation sets SOP 1 and SOP 2 were inserted on July 1st 1973. The two columns of SOP 1 were irradiated in the irradiation channels 1 and 3 (azimuthal position 99°30´ and 279°30´) in the first 2 operating cycles (663.1 effective full power days (EFPD)) and withdrawn on February 16th 1976 [57]. The specimen set SOP 2 was withdrawn on October 16th 1978, hence it was irradiated during cycles 1 to 5 (1514 EFPD) in the irradiation channels 2 and 4 (azimuthal position 147° and 327°) [58].

5.3.3 Testing

For the surveillance of the radiation-induced embrittlement of the RPV for 40 operating years tensile tests, drop weight tests, Charpy-V tests and fracture toughness tests were performed. The examinations with the unirradiated materials base metal 03, 04, weld metal W03 and HAZ were performed by KWU in Erlangen. The specimens of the irradiated sets SOP 1 and 2 were examined and evaluated by ECN in Petten.

Detailed data of the examinations for the first irradiation surveillance program are given in , [57] and [58].

5.3.4 Discussion and Evaluation of the Test Results

The reference temperatures for nil-ductility transition of the monitored RPV materials were determined by evaluation of the first KCB irradiation surveillance program.
Since the testing of transverse specimens of cylindrical shells was not specified yet in 1972, the Charpy absorbed energy results of longitudinal specimens were adjusted for transverse specimens by reduction of 35 % according to [59]. The RT_{NDT}-temperatures of both cylindrical shell 03 and 04 and of the weld material W03 shows Table 2.

Material	RT_{NDT} [°C]
BM Ring 03	-10
BM Ring 04	-20
WM W03	≤ -45
HAZ	-5

Table 2: RT_{NDT}-temperature of cylindrical shell 03 and 04 and of weld material W03 from SOP 0,

The RT_{NDT}-temperature of the HAZ made from welding end material of the unirradiated specimen is -5 °C. However, the plate used for the test coupon has a clearly higher copper content compared to the forging. For this reason the HAZ examinations of the weld specimen are not representative.

The test results for the irradiation sets SOP 1 and 2 were provided by ECN to Siemens/KWU and were re-evaluated again by Siemens/KWU. The Siemens/KWU data are used in the following.

The transition temperature shifts and the changes in Upper Shelf Energy (USE), determined with the Charpy-V specimens, as well as the adjusted reference temperatures RT_{NDTj} are listed in Table 3.

Material	Set	ΔT [K]				Reduction of USE [J]	$RT_{NDT(j)}$ [°C]
		ΔT_{41}	ΔT_{68}	$\Delta T_{0.9}$	ΔT_{50}		
BM Ring 03	1	9	9	12	9	36	-1
	2	12	17	12	12	35	2
BM Ring 04	1	17	13	13	1	32	-3
	2	18	19	23	1	38	-2
WM W03	1	34	34	36	14	42	-11
	2	30	45	46	26	49	-5
HAZ	1	11	17	28	16	40	6
	2	24	33	47	39	30	19

Table 3: Changes in criteria temperature and upper shelf energy due to irradiation

According to the test results of the second irradiation set SOP 2 the HAZ is leading with an adjusted reference temperature RT_{NDTj} of 19 °C. Since the HAZ is no longer required by the KTA standard [36] the HAZ results can be rejected and the BM Ring 03 is leading with an adjusted reference temperature of 2 °C for SOP2.

5.4 Irradiation Sets SOP 0a, SOP 3 and SOP 4

The objective of the second irradiation surveillance program is to proof safe operation of the KCB RPV for 60 years. Therefore, the present irradiation surveillance program, consisting of SOP 0, SOP 1 and SOP 2, was extended by a further unirradiated reference set SOP 0a and two irradiation sets SOP 3 and SOP 4. The results of SOP 0a are used to show the influence of the different specimen orientations (longitudinal specimens for SOP 0 and transverse specimens for SOP 0a) and to obtain unirradiated data for the Master Curve analysis.

5.4.1 Materials and Specimens

Since the fabrication of the first three specimen sets SOP 0, 1 and 2, the standard was adapted to the current state-of-the-art of science and technology. The three last irradiation sets SOP 0a, SOP 3 and SOP 4 were manufactured according to the current valid safety standard KTA 3203 [36] with transversal specimen orientation for the BM.

The used base metals for this new irradiation surveillance program stems from the original material of the two cylindrical shells 03 and 04. The used weld metal comes from the monitored test coupon AW0-3.1 and thus is, like the base metal from the Ring 03 and Ring 04, directly comparable with the region of the RPV to be monitored.

The entire sample size of an irradiation set was stored in 2 capsules, which were put together in one capsule holder.

5.4.2 Irradiation of the Specimens

The two irradiation sets SOP 3 and SOP 4 were inserted in the Borssele RPV in the irradiation channels 1 at 99°30´ (column 1) and 3 at 279°30´ (column 2) on September 12th 2007 during the outage

5.4.3 Testing of SOP 0a

Tensile tests, notched-bar impact tests with specimen orientation T-L and three point bending tests for the determination of the reference temperature T_0 according to ASTM E 1921-05 [62] were performed with the unirradiated specimen set SOP 0a.

5.4.4 Discussion and Evaluation of the Test Results

The comparison of the tensile test results from SOP 0 and SOP 0a shows no significant impact of the specimen orientation for the base metals. The specimen orientation for the weld metal was not changed. As expected, there is a good agreement between the SOP 0 and SOP 0a tensile test data, .

According to KTA 3203 [36] the reference temperature RT_{NDT} is determined from the drop weight test according to Pellini and the notched-bar impact test. With SOP 0a no further drop weight tests need to be performed, since there are no orientation requirements. The values from SOP 0 are still valid. A more detailed description of the reference temperature determination and the comparison of the results from the two unirradiated sets SOP 0 and SOP 0a are given in Chapter 6.2.1. The previous RT_{NDT} values from SOP 0, see Table 2, are confirmed by SOP 0a data.

In Table 4 the transition temperatures resulting from the notched-bar impact tests are listed for the two unirradiated sets SOP 0 with the base metal specimens in L-S orientation and SOP 0a with T-L orientation of the base metal specimens.

Material	Set	T_{41} [°C]	T_{68} [°C]	$T_{0.9}$ [°C]	T_{50} [°C]	Upper Shelf Energy [J]
BM Ring 03	0	-57	-43	-48	-13	200
	0a	-58	-40	-45	-17	182
BM Ring 04	0	-57	-42	-46	1	214
	0a	-66	-44	-44	-16	185
WM W03	0	-44	-26	-34	-5	180
	0a	-42	-27	-30	-17	172

Table 4: Transition temperatures and upper shelf energy for the unirradiated sets SOP 0 with L-S orientation of the base metal and SOP 0a with T-L orientation of the base metal .

The comparison shows that the transition temperatures T_{41J} and T_{68J} for the base metals ring 03 and ring 04 of the sets SOP 0 and SOP 0a are in a good agreement, regarding the scatter of the results, in spite of the different specimen orientation. However, the specimen orientation seems to have an impact on the upper shelf energy, since a decrease of USE for the transverse specimens of the base metals was observed. The weld metal shows a small decrease in upper shelf energy, too, although the specimen orientation has not been changed.

The T_0 results from the fracture mechanics tests, determined on three point bending specimens, are listed in Table 5.

With the T_0 values the RT_{T0} values were determined according to the ASME Code Case N-629 [48] The results for the RT_{T0} are summarized in Table 5.

Material	T_0 [°C]	RT_{T0} [°C]
BM Ring 03	-106	-87
BM Ring 04	-106	-87
WM W03	-79	-60

Table 5: T_0 values for SOP 0a for the materials of the KCB RPV determined according to ASTM E 1921 [62], and reference temperatures RT_{T0} for the materials of the KCB RPV determined according to ASME Code [48]

5.5 Irradiation Data from Research Projects (CARISMA)

For the assessment of the irradiation behavior of the KCB RPV materials, the surveillance data of these materials are compared to test data from as far as possible similar materials from the research project CARISMA. The base metal P7 BM and the weld metal KS05 WM from the CARISMA project were selected for this purpose.

One possibility for the assessment is the comparison of the reference temperatures and their increase due to irradiation. the ART curves determined according to Reg. Guide 1.99, Rev. 2, Pos. 2 [40] and the RT_{NDTj} values according to KTA 3203 [36] for the KCB RPV materials are plotted with those from the two CARISMA materials together with the RT_{limit} curve from the KTA 3203 [36]. All values are enveloped by the KTA RT_{limit} curve.

The ART curve from the CARISMA weld material KS05 covers all measured data and the ART curves of the KCB materials, hence the KS05 WM shows the strongest irradiation embrittlement. The reason for it is the essentially higher nickel content of the KS05 WM compared to the KCB W03 WM, leading to a much stronger irradiation embrittlement at higher neutron fluences.

The P7 BM has a stronger irradiation embrittlement than the KCB RPV materials at a fluence of about 4.1E+19 n/cm^2. This is due to the copper and nickel content, which are both higher for the CARISMA base metal.

The selected CARISMA data is consistent with the KCB material data.

5.6 Neutron Fluence Calculation

5.6.1 Introduction

The first theoretical fluence calculations for the KCB RPV were performed in 1973 with a one dimensional transport program ANISN, . According to these calculations the design fluence of the RPV after 32 EFPY is 3.50E+19 n/cm , if in each operating cycle fresh fuel elements were inserted at the outer core region (out-in loading), .

For each fluence detector of the two irradiation sets SOP 1 and 2 from the first irradiation surveillance program the fission flux and the fission fluence were determined, but not the fluences E> 1 MeV. Thus, only theoretical estimations could be made for the fluences E> 1 MeV for SOP 1 and SOP 2. The KWU estimations led to 1.6E+19 n/cm^2 for SOP 1 and to 1.9E+19 n/cm^2 for SOP 2 The KEMA analyses for the fluences of SOP 1 and SOP 2 resulted in 1.9E+19 n/cm^2 and 2.8E+19 n/cm^2, respectively, [64].

The actual fluence after 32 EFPY is lower than 3.50E+19 n/cm^2 (E> 1 MeV) due to changing the core loading scheme after 10 operating cycles to partial low leakage core and after 19 operating cycles to full low leakage core (in-out)

Because of the uncertainties of the fluence calculations for SOP 1, SOP 2 and the RPV at that time, the fluence values had to be updated based on calculations according to the state-of-the-art. The actual state compared to that two decades ago can be described, respectively quantified, as follows:

- Use of pin-by-pin neutron source data (205 values per fuel element, formerly 9 values)
- More accurate calculation of neutron source data (CASCADE-3D)
- Use of more precise geometry models for the neutron transport calculation
- Three-dimensional performance of the neutron transport calculation due to higher processor capacity

The three-dimensional Monte Carlo calculations with the code MCNPX [68] and the two-dimensional deterministic calculations with the code DORT [69] were used for the theoretical determination of the neutron fluence E> 1 MeV. The method of calculation, the geometry input data for the 3D-model of MCNPX, the neutron source data, the neutron transport and reaction cross-sections, and the additional 2D-models for DORT are described in detail in

The calculations for the fast neutron fluence E> 1 MeV were performed for each of the operating cycles 1 to 33. The sum over these single fluence values represents the status of the RPV fluence after cycle 33, i.e. after an effective full power time of 28.6 years (28.6 EFPY).

For quality assurance independent calculations were performed by NRG [70] which lead to a minimal difference between the NRG and AREVA analyses. Also sensitivity analyses were made by NRG.

5.6.2 Neutron Fluence Calculations for the RPV and for the Irradiation Capsules SOP 2, SOP 3 and SOP 4

For each of the operating cycles 1 to 33 of the nuclear power plant Borssele, the neutron transport calculations were performed by means of the three-dimensional Monte Carlo code MCNPX [68], using the calculation method and input data described in

Inner RPV wall

The results for each single operating cycle were evaluated and summed up over all 33 cycles to obtain the accumulated neutron fluence E> 1 MeV for the RPV. The reference position for the maximum RPV fluence E> 1 MeV is located at the inner side of the pressure vessel material. That means the fluence distribution at the boundary layer between the RPV cladding and the vessel steel itself was determined. The RPV fluence distribution varies with the radial, the azimuthal, and the axial position.

The maximum of the fluence at the inner side of the RPV is located at the azimuthal position of 0°, (and the homologous positions 90°, 180°, and 270°, respectively) and the axial position z = -17 cm and was calculated to 1.91E+19 n/cm^2 after the end of operating cycle 33. The average flux over the last 3 cycles of the RPV is 1.57E+10 n/cm^{2}s^1, used to extrapolate the future RPV wall fluence.

SOP 2

The irradiation capsule SOP 2 has been inserted in the RPV for the first 5 cycles. The accumulated fluence value for cycles 1 to 5 around the reference point, i.e. the capsule centre, was determined at the capsule centre to 1.36E+19 n/cm^2, corresponding to an average neutron flux E> 1 MeV of 1.04E+11 n/cm^{2}s^1. In connection with the maximum RPV flux averaged over the cycles 1 to 5 (3.01E+10 n/cm^{2}s^1), a lead factor of 3.5 is determined for the SOP 2 capsules. The neutron fluence E> 1 MeV 1.36E+19 n/cm^2 for the irradiated SOP 2 capsules after the end of cycle 5 represents the status of the RPV material after a lifetime of 17 EFPY.

To confirm the Monte Carlo (MC) calculations for cycles 1 to 5, further fluence calculations are performed using the two-dimensional deterministic transport code DORT [69]. Besides the material specimens, the SOP 2 capsules contained a series of iron and niobium fluence detectors. The results of the activation measurements for these iron and niobium detectors are used to determine experimental fluence values for the capsules SOP 2. The experimental fluence values from the Fe detectors are in a good agreement with the theoretical fluence values, but there is an unexpected wide range of the values compared to the theory. The experimental fluence values from the Nb detectors are not reliable. For validation of the experimental values scraping samples from the RPV cladding are recommended.

SOP 3 and 4

The irradiation sets of the second surveillance program SOP 3 and 4 were inserted into the reactor at the beginning of cycle 33. For this operating cycle 33 the Monte Carlo calculation served for the determination of the neutron fluences and fluxes for SOP 3 and 4.

The SOP 3 and 4 neutron flux E> 1 MeV in the centre of the middle capsule has a value of 1.21E+11 n/cm^{2}s^1 for operating cycle 33. The neutron flux E> 1 MeV of the RPV at the position of the maximum RPV fluence is 1.57E+10 n/cm^{2}s^1. Therefore, the capsules SOP 3 and 4 of the irradiation program have a fluence lead factor of 7.7.

Since the capsules have not yet reached the target fluence values, they will remain in the RPV for some further cycles.

The future development of the SOP 3 and 4 fluence values was extrapolated using the average flux for cycles 31 to 33 (1.26E+11 n/cm^{2}s^1). Based on this extrapolation the necessary irradiation time for the SOP 3 and 4 capsules can be assessed for the planning of their withdrawal, see chapter 5.7.2.

The SOP 3 and 4 curve starts with the beginning of cycle 33.

5.6.3 RPV Fluence Extrapolation to 60 Operational Years

The future RPV fluence development was evaluated considering a long term operation to 55 EFPY. This extrapolation interval covers the extended plant lifetime until 2034 according to the expected full power time of about 52 EFPY.

Based on the neutron flux averaged over the cycles 31 to 33, the RPV neutron fluence E> 1 MeV was extrapolated to estimate the expected future development for the nuclear power plant of Borssele. The average flux over the last 3 cycles (1.57E+10 n/cm^{2}s^1) represents a typical status of the actual core loading scheme, covering minor changes from cycle to cycle.

The neutron fluence E> 1 MeV at the inner side of the RPV will reach a value of 2.08E+19 n/cm^2 after an effective full power time of 32 EFPY.

After a subsequent long term operation to 55 EFPY, covering an operating time of 60 years, a value of 3.22E+19 n/cm^2 will be reached for uranium fuel loading. These fluence values are observed at the position of the axial maximum that is located at z = -17 cm.

5.6.4 Impact of MOX Fuel on the Fast Neutron Fluence of the RPV and the Irradiation Capsules SOP 3 and SOP 4

Additional neutron fluence calculations for a core filled with a combination of MOX and uranium fuel were performed using the existing 3D-model from for the three-dimensional Monte Carlo code MCNPX [68],

The impact of MOX fuel on the future fluence development of the RPV and the irradiation capsules SOP 3 and 4 was investigated. A core loading with MOX fuel was assumed to begin in the year 2012 after the operating cycle 38.

a changing slope of the fluence development according to the changing neutron flux E> 1 MeV at the inner side of the RPV. This behavior of the neutron flux over the time is due to the changing core loading management over the lifetime of the reactor. The first time interval after the end of cycle 33 covers the cycles 34 to 38. For these cycles, still a standard core loading with uranium fuel assemblies was used. Therefore, the extrapolation of the neutron fluence was based on the average neutron flux over the cycles 31 to 33. After cycle 38 a core loading with a combination of uranium and MOX fuel is planned. Therefore, the extrapolation of the neutron fluence was based on the neutron flux for the MOX equilibrium cycle, which was set up by AREVA core physics department .

<u>Inner RPV wall</u>

The maximum fluence E> 1 MeV at the inner side of the RPV after 55 EFPY was determined. This extrapolation interval covers a long term operation until 2034 according to the expected full power time of about 52 EFPY.

At the end of the LTO at 55 EFPY a fast neutron fluence (E> 1MeV) of 3.40E+19 n/cm was obtained for the RPV, using the flux for a MOX equilibrium cycle. Using only the average flux of cycles 31 to 33, a value of 3.22E+19 n/cm^2 is expected.

<u>SOP 3 and 4</u>

In addition, the neutron fluence E> 1 MeV for the irradiation capsules SOP 3 and 4 was determined considering the impact of the MOX fuel.

value of 3.40E+19 n/cm^2 is reached for the SOP 3 and 4 capsules after 35.6 EFPY using the average flux for cycles 31 to 33 until end of cycle 38 and using the flux for a MOX equilibrium cycle starting after cycle 38. A value of 3.22E+19 n/cm^2 is reached after a plant lifetime of 35.9 EFPY, using the average flux for cycles 31 to 33 over the total irradiation time.

For the first time interval, which covers the cycles 34 to 38, an average flux of 1.26E+11 n/cm^{2}s^1 was determined for the capsule centre. Compared with the maximum flux at the RPV inner side of 1.57+10 n/cm^{2}s^1, this yields a capsule lead factor of about 8.0.

For the second time interval with MOX fuel an average flux of 1.63E+11 n/cm^{2}s^1 was determined for the capsule centre, which is about 29% higher than for the first time interval. Compared with the maximum flux at the RPV inner side of 1.83+10 n/cm^{2}s^1 for MOX fuel, this yields a capsule lead factor of about 8.9.

5.6.5 Discussion and Evaluation of the Results

the following fluence results E> 1 MeV were obtained for the RPV:

- A fluence value of 3.22E+19 n/cm^2 is reached at the inner RPV wall after 55 EFPY.
- A fluence value of 3.40E+19 n/cm^2 is reached at the inner RPV wall after 55 EFPY using the MOX fuel after cycle 38.

The increase of the expected RPV fluence after 55 EFPY due to MOX fuel is small (< 6%). The expected fluence value remains below the RPV design fluence value of 3.50E+19 n/cm^2 which is the assessment fluence used in the assessment against brittle fracture.

The fluence results E> 1 MeV for the SOP 3 and 4 irradiation capsules can be summarized as follows:

- The fluence value of 3.22E+19 n/cm^2 is reached for the SOP 3 and 4 capsules after 35.9 EFPY (after cycle 42).
- The fluence value of 3.40E+19 n/cm^2 is reached for the SOP 3 and 4 capsules after 35.6 EFPY (after cycle 41) using MOX fuel after cycle 38.

The impact of MOX fuel on the flux at the position of the SOP 3 and 4 capsules (29% increase) is bigger than the impact on the RPV flux (17% increase). This is caused by the closer position of MOX fuel assemblies to the irradiation position where the capsules are located.

5.6.6 Verification of Neutron Fluence Calculations with Scraping Samples

The state-of-the-art fast neutron fluence (E> 1 MeV) calculation will be additionally verified with scraping samples. Therefore during the outage in April 2010 scraping samples will be removed from the KCB RPV cladding at relevant positions and afterwards analyzed radiochemically for activity determination. The activity of the scraping samples will then be used for experimental fluence calculations.

5.7 Concept for Irradiation Surveillance to demonstrate 60 Operational Years

5.7.1 Boundary Conditions

The design fluence 3.50E+19 n/cm^2 of the KCB RPV has to be certified for a safe operation for 40 years, and moreover concerning a LTO to 60 operating years.

A fast neutron fluence E> 1MeV of 3.40E+19 n/cm^2, considering that after cycle 38 the MOX fuel loading starts, was obtained for the RPV after 55 EFPY, covering an operating time of more than 60 years. The expected fluence value remains below the RPV fluence design value of 3.50E+19 n/cm^2. This is especially the case, when the loading management remains like the current, which leads to a lower fluence value of 3.22E+19 n/cm^2 at EoL.

5.7.2 Withdrawal Schedule for SOP 3 and SOP 4

For experimental validation of the RPV irradiation behavior the surveillance programs were implemented. The first surveillance program is already evaluated as well as the unirradiated set SOP 0a of the second program. The two irradiation sets SOP 3 and 4 of the second surveillance program are still inserted in the KCB RPV.

For the removal time of the irradiation sets, there are two options applicable with respect to the withdrawal of SOP 3:

<u>Option 1</u>

The withdrawal of the irradiation sets SOP 3 and SOP 4 may be carried out according to KTA 3203 [36]. The KTA requires for the first irradiation set a fluence value of about 50 % of the assessment fluence (AF) and that the second irradiation set should exceed 100 % AF. The KCB RPV assessment fluence value E> 1 MeV is 3.50E+19 n/cm^2 thus 50 % corresponds to a fluence value of 1.75E+19 n/cm^2. Hence, according to the extrapolation of the updated fast neutron fluence calculations SOP 3 may be withdrawn after cycle 38 during the outage in 2012 with a neutron fluence of 2.05E+19 n/cm^2. Assuming MOX fuel loading after cycle 38, SOP 4 can be withdrawn after cycle 42 during the outage in 2016 with a neutron fluence value of 3.88E+19 n/cm^2. Without MOX fuel loading SOP 4 can be withdrawn during the outage in 2017 after cycle 43 with a neutron fluence of 3.81E+19 n/cm^2.

<u>Option 2</u>

Another possibility for the withdrawal schedule can be realized, if the fluence of SOP 3 should be more between the fluences of SOP 2 and SOP 4 in order to have more uniform distributed irradiation data versus the fluence available. That means the fluence value of SOP 3 has to be equal to the average of the fluences of SOP 2 and SOP 4. Considering SOP 2 has a fluence value of 1.36E+19 n/cm^2 and SOP 4 should have at least 3.50E+19 n/cm^2, SOP 3 can be withdrawn after exceeding a fluence value of 2.43E+19 n/cm^2. Assuming MOX fuel loading after cycle 38, the outcome of this is that SOP 3 can be withdrawn after cycle 39 in 2013 . Without MOX fuel loading SOP 3 can be withdrawn during the outage in 2014 after cycle 40. Application of this withdrawal schedule also ensures that the fluence value of SOP 3 covers the RPV fluence value until SOP 4 is withdrawn. After exceeding 100 % AF SOP 4 can be removed after cycle 42 during the outage in 2016, assuming MOX fuel loading after cycle 38. Without MOX fuel loading SOP 4 can be withdrawn during the outage in 2017 after cycle 43.

5.7.3 Alternative Measures

If material test data are needed earlier and thus the withdrawal of a set shall be carried out earlier than given in 5.7.2 and required by the standard, respectively, other approaches have to be considered.

In this case the irradiation capsules can be withdrawn at an earlier time then required and the specimens will be examined. In order to receive the required fluence at a later time inserts will be manufactured from the tested specimens afterwards. These inserts will be inserted in the irradiation column again and re-irradiated until the desired and respectively required fluence is exceeded. Then the inserts will be withdrawn and used for manufacturing the reconstituted specimens for testing. Consequently data for a third irradiated state are finally available.

6. Safety Assessment of the KCB RPV

Originally it was foreseen that NPP Borssele was going to operate from 1973 until the end of 2013, 40 years of operation. The original brittle fracture assessment was based on this operation period. Because of an agreement with the government in 2006 a new operation time came up: further operation until the end of 2033 (60 years of operation). To prove that the NPP is still safe new ageing assessments, including embrittlement of the RPV, need to be performed regarding the new operation time. For the new assessment EPZ decided to go for an assessment according to state of the art regulation and knowledge for this topic.

Therefore for the RPV a new safety analysis is performed for the most severe loading in case of a loss of coolant accident , The analysis will represent a state of the art assessment using the experience which has been gained during the renewal of the assessment for all German NPPs under operation and for some foreign plants.

6.1 Pressurized Thermal Shock Analyses

For the RPV of the NPP Borssele a fully representative spectrum of all possible large and small break loss of coolant accidents is investigated. Locations of interest are the irradiated RPV core beltline region as well as those regions with negligible irradiation but higher loading, like the RPV main coolant line (MCL) nozzles and the flange to cylinder transition area. The results derived from the thermal-hydraulic analyses are used for fracture mechanics analyses which are based on assumed bounding initial crack sizes. For each location, an admissible (allowable) ductile to brittle transition reference temperature $RT_{NDT,allowable}$ is determined and compared with the materials resistance against brittle fracture.

6.1.1 Introduction

During the design stage of the today operating German PWRs and the nuclear power plant Borssele, one major design target was to manage the materials ageing behavior of the ferritic reactor pressure vessel steel in order to prevent a brittle fracture of the RPV under all possible loads. Very conservative thermal-hydraulic input data (low injection temperature, high heat transfer coefficient, thermal shock within seconds) were used in this context for the assessment of loss of coolant accidents. Even for the very unlikely event of a guillotine break of a main coolant line (MCL), which leads to a maximum injection of emergency cooling water, a subsequent full circumferential thermal shock of the RPV was assumed for integrity analysis.

From today's point of view, this large break loss of coolant accident (LOCA) load case is highly unrealistic due to the fact that all operating German PWRs and the nuclear power plant Borssele have proven a break preclusion of their main coolant lines.

Today more realistic thermal-hydraulic analyses based on large scale experimental results are available as well as more sophisticated fracture mechanics methods to assess postulated flaws.

6.1.2 Concept and Procedure

The basic requirements of the concept are:

- No flaws before in-service (base material including welds, cladding) proven by NDE
- Crack initiation during service excluded, sub-critical crack growth during operation negligible small, NDE during operation
- Exclusion of brittle fracture initiation with realistic conservative boundary conditions:
    - Postulated flaws
    - Selection of leak sizes
    - Selection of considered regions
- Crack arrest (redundant barrier against failure).

The basics of the brittle fracture assessment concept can be found in the current valid rules of KTA and the RSK guidelines [33] - [36], [73], [74].

6.1.3 Thermal-hydraulic Calculations

The thermal-hydraulic boundary conditions for the fracture mechanics analysis are determined within two steps. In the first step the transients initiated by assumed leaks in the main coolant piping are determined by means of the system code S-RELAP5 coupled with the containment code COCO 2

S-RELAP5 is a RELAP5-based thermal-hydraulic system code for performing realistic analyses of a large break loss-of-coolant accident in pressurized water reactors. The code is also suitable for analyzing small break loss-of-coolant accidents and non-LOCA transients. RELAP5 is a light water reactor transient analysis code developed at the Idaho National Engineering Laboratory for the US Nuclear Regulatory Commission.

The S-RELAP5-Version V311-PTS used for the KCB-PTS-Study is a special version for PTS-Studies with a correct representation of the water level in a partly filled horizontal cold leg and in the vertical downcomer node connected to the cold leg.

The KCB-Version of the containment code COCO 2 is used to determine the containment pressure, the sump water temperature and the temperature of the water flowing from the flooding tanks and, after emptying the flooding tanks, from the reactor sump through the low head safety injection pumps and coolers into the primary system. The temperature on the sea side (tertiary side) of the cooler chain is assumed to be 0 °C.

The second step is performed based on the PTS-relevant results of the system code, which are the primary side pressure, the downcomer collapsed water level, the water rates injected into the primary side by the emergency core cooling system (ECC) and by the extra borating system and the temperature of the injected water. In the second step the mixing and condensation processes occurring in the hot and cold legs and in the downcomer due to the injection of subcooled water are quantified using the code KWU-MIX . KWU-MIX is an engineering model code developed by Siemens. The KWU-MIX models are based on full scale mixing and direct contact condensation tests performed in the Upper Plenum Test Facility (UPTF) and also on sub scaled mixing test results (CREARE, HDR).

The primary side pressure determined by means of S_RELAP5 is directly used as boundary condition in the fracture mechanics calculation and S_RELAP5 results only are used as thermal hydraulic boundary conditions in the fracture mechanics calculation for the period of main coolant pump coast down, that is, during the period of almost ideal fluid-fluid mixing on the primary side. KWU-MIX supplies the distributions of the fluid-temperature and the wall-to-fluid heat transfer coefficient on the inside reactor pressure vessel (RPV) wall and in the RPV nozzles after main coolant pump coast down.

6.1.3.1 Boundary Conditions

The boundary conditions for the KWU-MIX calculation, primary side pressure, downcomer water level, ECCS and extra borating system injection rates and temperatures are taken from the S-RELAP5 and COCO 2 outputs.

6.1.3.2 Selection of Leak Sizes and Injection Configurations

In principle all transients leading to a rapid global cool down of the downcomer fluid superposed by colder water stripes or colder water plumes which form in the downcomer below the cold legs with ECCS water injection are PTS relevant. A plume of colder water is formed in the water-filled downcomer in case colder water entering the downcomer via the cold leg is flowing downwards in the downcomer due to its higher water density as compared to the density of the ambient hotter water A stripe of colder water forms in the downcomer in case the downcomer water level is low The stripe touches the inner RPV wall in case the water stripe mass flow rate is below a certain threshold value. This threshold value depends on the cold leg geometry near the cold leg – downcomer interface.

Water plume width and plume center line temperature increase with increasing distance from cold leg due to the entrainment of ambient hotter water. The stripe width decreases with increasing distance from cold leg due to the acceleration caused by gravity. The stripe temperature increases with increasing distance from cold leg due to direct contact condensation of ambient steam on the water stripe.

6.1.3.3 Transients under Investigation

KWU-MIX

calculations were performed only for the leak sizes greater than 0.5 cm².

The S-RELAP5 results for all leak sizes greater than 0.5 cm² were used in the KWU-MIX calculation to determine the pressure, the fluid temperature and the wall-to-fluid heat transfer coefficient at every point on the inside RPV wall surface and on the inside walls of the RPV nozzles.

As an example S-RELAP5 and KWU-MIX results for the 200 cm² hot side leak are outlined to show the S-RELAP5 data used in KWU-MIX and the KWU-MIX results which are used as thermal hydraulic boundary conditions in the fracture mechanics analyses.

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6.1.4 Fracture mechanics Analyses

Although it is proven that the RPV is free of cracks after manufacturing and in-service inspections, it has to be demonstrated by a brittle failure safety assessment that instability of postulated cracks is excluded.

The prescribed procedure according to the standard which is applied in Germany is based on deterministic principles:

- Postulated flaws according to NDE (safely detectable flaw size multiplied by a safety factor of two).
- Selection of thermal-hydraulically justified leak sizes (every thermal-hydraulic transient will be assumed regardless their eventual low entrance probability).
- Selection of representative RPV regions (according to the standard only locations with a fluence $\Phi > 10^{17}$ n/cm^2 (E> 1 MeV)).

Based on the assumptions given above the safety assessments will comprise loading curves and material resistance curves Initiation of brittle fracture can be excluded and the safety is assessed if the loading curve may never intersect the material curve (before load maximum because the WPS effect is considered).

The material curve is the fracture toughness K_{IC}-curve according to ASME and given in the KTA standard [34]. The curve will be indexed with the reference temperature RT_{NDT} and RT_{NDTj} for irradiated materials, respectively. If it is necessary, the K_{IC}-curve may also be indexed by RT_{T0} and RT_{T0j} respectively, according to [36]. At locations of high loading and localized plasticity of the material under investigation, an additional assessment against ductile failure using crack resistance curve is performed. This is not required by the standard.

It is allowed [34] to take credit from the so-called warm pre-stress effect. For transients, which show after the load maximum a strong monotone declining load path, crack initiation may be excluded, if the crack tip of a postulated flaw has seen a warm pre-loading in the present transient. This means the load paths after load maximum will be neglected

The fracture mechanics calculations are performed using the finite element method. The loading is given from the thermal-hydraulic boundary conditions. They result in instationary temperature fields which are uncoupled to the following stress calculation. All finite element models are three-dimensional. The global analysis of the whole RPV considering symmetry delivers the temperatures and stresses for the analytical fracture mechanics calculations. Sub-models with cracks are used at the considered locations with postulated flaws. By means of these sub-models the load paths of the according transients are calculated.

6.1.4.1 Geometry

Following data are used for the fracture mechanics analyses:

Inner radius:	1863 mm
Wall thickness base metal:	181 mm
Wall thickness cladding:	7 mm

6.1.4.2 Postulated flaws

The safety assessment has to be made according to German rules for locations of the RPV which are irradiated and at which the fluence level at end of design lifetime is higher than 1.0E+17 n/cm² (E> 1 MeV) [34]. This requirement is valid for the maximum irradiated region of the ferritic inner wall of the RPV, the core region [36].

Because of the quality of the materials, the manufacturing procedures and the service conditions it turned out that the irradiation response in the core region is not very high. Therefore other regions which have a higher loading and no irradiation response might be more critical. As a consequence other regions beside the core weld are investigated. In the following the investigated locations and the according flaw postulates are summarized:

- Core weld: circumferential flaw
- Cold leg nozzle: nozzle corner flaw in 6 o´clock position
- Hot leg nozzle: semi-elliptical flaw at the highest loaded region in 6 o´clock position
- Flange weld: circumferential flaw

According to the KTA standard [36] the postulated flaws are flaws, which can be safely detected by non-destructive measurements. For relevant load cases it is stated in the code KTA standard [36] that the safely detected flaw have to be multiplied by the safety factor of 2. This size is used in the assessment. For the Borssele plant a crack depth of 10 mm is investigated, which is also used in other assessments of German plants. Because of proven absence of cracks in the cladding, semi-elliptical underclad cracks are postulated. The crack depth to length ratio is 1/6. In nozzle corners the length is adjusted to the geometry. The orientation of the flaw is perpendicular to the maximum stress.

6.1.4.3 Material data

The material properties are different for the ferritic base metal (22NiMoCr 3 7) and the austenitic cladding (CN21/10 Nb-BS). All relevant data (thermo-mechanical, thermo-physical and tensile properties) are taken into account temperature dependent.

6.1.4.4 Finite Element Model

The calculations are performed using global models and sub-models for the according postulated cracks in the regions under investigation. All analyses are conducted with the commercial FE code ABAQUS, [78]. The models are adapted to the thermal-hydraulic boundary conditions with respect to symmetry and refinement of the meshes. The sub-models contain the sub-surface cracks which are located in the regions of the maximum stresses inside the plume (center of the plume) and in circumferential direction in the core weld, because the axial stresses are higher than the tangential stresses because of additional thermal stresses of the ΔT between plume and downcomer.

6.1.4.5 Loading and boundary conditions

The boundary conditions in the finite element calculations are given by the symmetry conditions of the vessel. At the boundaries of the hot and cold legs axial forces are placed to simulate the internal pressure of the pipes. On each single node point of the inner surface the thermal loading is placed in dependence of time according to the thermal-hydraulic output and the time dependent internal pressure.

6.1.4.6 Residual stresses

According to the KTA rules [33] and [34] residual stresses need to be considered in the safety analysis.

6.1.4.7 Analytical Model

By means of analytical methods some parametric studies were performed. The study was conducted for the relevant leakage sizes . The results are based on linear-elastic stresses through the wall obtained from analyses with the global finite element model. The surface point is also calculated, but these results are very conservative, because the cladding is ignored in this case.

6.1.5 Results and Recommendations

State-of-the-art thermal hydraulics and fracture mechanics analyses have been performed to calculate the loading of the Borssele RPV under the most severe conditions of a loss of coolant accident.

All relevant locations, not only the core weld have been considered.

6.2 Evaluation of the Irradiation Effects

The irradiation effect in the core region can be assessed by using fracture toughness K_{IC} - or K_{JC} - curves, which are indexed with reference temperatures. The adjustment of the fracture toughness curve can be performed either indirectly according to the RT_{NDT} concept by comparative examinations of irradiated and unirradiated notched-bar impact specimens or directly according to the fracture mechanics concept by examination of irradiated fracture mechanics specimens (e.g. determination of T_0 using Master Curve concept according to ASTM E 1921 [46]).

RT_{NDT} concept:

$RT_{NDTj} = RT_{NDT} + \Delta T_{41}$

RT_{NDT} (for BM: T_{NDT} from drop weight test generally leading compared to T_{Av} (68J) -33K)

RT_{T0} concept:

$RT_{T0} = T_0 + 19.4$ °C

Thus, the ASME K_{IC} lower bound curve can be indexed either with RT_{NDT} or with RT_{T0}.

6.2.1 Reference Temperatures RT_{NDT} in the Unirradiated State

According to KTA 3203 [36] the determination of the RT_{NDT} results from the drop weight test according to Pellini and the notched-bar impact test. The RT_{NDT} is defined as the highest value of:

- NDT-T
- T_{Av} (68J) – 33K
- T_{Av} (0.9mm) – 33K

The NDT temperatures from the drop weight test according to Pellini for the KCB RPV materials are listed in Table 6. The change in specimen orientation of the base metal from the unirradiated set SOP 0a can cause changes in the T_{AV} (68J) – 33K and T_{AV} (0,9mm) – 33K, respectively, compared to SOP 0. Thus, also these results are presented in Table 6.

Material	NDT-T	T_{AV}(68J) – 33K		T_{AV}(0.9mm) – 33K		RT_{NDT}
	SOP 0	SOP 0	SOP 0a	SOP 0	SOP 0a	[°C]
BM Ring 03	-10	-42	-65	n.d.	-78	-10
BM Ring 04	-20	-41	-73	n.d.	-77	-20
WM W03	≤ -45	-45	-57	-54	-63	-45

Table 6: Comparison of criteria temperatures NDT-T, T_{AV} (68J) – 33K and T_{AV} (0.9mm) – 33K for the determination of RT_{NDT} (n.d. = not determined),

The table shows that for both base metals and the weld metal the NDT-T is leading. The previous RT_{NDT} values from the first irradiation surveillance program are confirmed.

6.2.2 Adjusted Reference Temperatures RT_{NDTj}

6.2.2.1 Evaluation according to German KTA 3203

The results of the adjusted reference temperatures RT_{NDTj} for SOP 1 and SOP 2 according to KTA 3203 [36] are given in the following Table 7:

Material	SOP	ΔT_{41} [K]	RT_{NDTj} [°C]
BM Ring 03	1	9	-1
	2	12	2
BM Ring 04	1	17	-3
	2	18	-2
WM W03	1	34	-11
	2	40	-5

Table 7: Adjusted reference temperatures RT_{NDTj} according to KTA 3203 [36]

All values are enveloped by the KTA RT_{limit} curve with big safety margins.

6.2.2.2 U.S. Reg. Guide 1.99, Rev. 2 Analysis

Since sufficient material data from irradiation surveillance programs exist, the Position 2 of the Reg.-Guide 1.99, Rev. 2 [40] can be applied.

The evaluation is carried out for the two base metals (ring 03, ring 04) and the weld metal for 55 EFPY, covering an operation of 60 years. Additionally the evaluation for anticipated MOX fuel loading starting after cycle 38 (32 EFPY) is carried out.

Regarding the input data a detailed description is given in This concerns the fluence values, the chemical composition, the material properties for the unirradiated initial state as well as the irradiated data from the elaboration of the irradiation surveillance sets.

The results of the Reg.-Guide analysis according to Position 2 for the base metals ring 03 and ring 04 and the weld metal are shown in Table 8:

Material	EFPY	Fluence [n/cm^2]	RT_{NDTj} [°C]
BM Ring 03	15	9.71E+18	10
	17	1.36E+19	11
	55	3.22E+19	13
	55	3.40E+19*	13
BM Ring 04	15	9.71E+18	7
	17	1.36E+19	8
	55	3.22E+19	12
	55	3.40E+19*	12
WM W03	15	9.71E+18	7
	17	1.36E+19	10
	55	3.22E+19	18
	55	3.40E+19*	18

*) with MOX fuel loading starting after cycle 38

Table 8: Results of RT_{NDTj}-determination according to Reg.-Guide 1.99 Rev.2 Position 2 [40]

The ART curves cover all measured RT_{NDTj} values determined with the temperature shift ΔT_{41}. The base metal ring 03 is the leading one of the surveillance materials until a fluence of 1.55E+19 n/cm^2 concerning the RT_{NDTj} and with it the irradiation embrittlement to be assessed, at higher fluences the weld metal leads.

The maximum RT_{NDTj} values for the leading weld metal after 55 EFPY predicted by Reg.-Guide Rev. 2 Position 2 are:

RT_{NDTj} = 18 °C at 3.22E+19 n/cm^2 (E> 1MeV) for loading without MOX fuel

RT_{NDTj} = 18 °C at 3.4E+19 n/cm^2 (E> 1MeV) for loading with MOX fuel starting after cycle 38

6.2.2.3 Other State-of-the-Art Trend Curve Analyses

Besides the Reg. Guide analysis other standards exist with analyses methods for predicting the radiation induced material embrittlement. As well as the Reg.-Guide 1.99, Rev. 2 [40] the standard guide ASTM E 900 [83] describes a method to calculate the shift in transition temperature based on the content of the chemical elements copper and nickel in the absence of surveillance data. By means of the resultant transition temperature shift the number of capsules for the irradiation surveillance program and EoL may be determined according to the ASTM E 185 [44]. However, the French standard RCC-M Section I [84] consideres in the subsection ZG 6122 the copper and the phosphorous content to calculate the transition temperature shift in dependance on the neutron fluence.

6.2.3 Reference Temperatures RT_{T0} in the Unirradiated State

According to the Master Curve Concept, T_0-values are used for the determination of the RT_{T0}. This concept has been developed during the last 15 years [46], [47] and is now settled in the ASME Code [48] and KTA [36]. This concept is based on directly measured fracture mechanics values on irradiated specimens. An essential advantage of the new approach is the direct determination of the reference temperature for brittle fracture by fracture mechanic tests and therefore a more realistic transfer to the component behavior.

The reference temperatures RT_{T0} for the core region materials of the KCB RPV is shown in Table 5.

6.2.4 Adjusted Reference Temperatures RT_{T0j}

The measured values for the adjusted reference temperatures RT_{T0j} are listed in Table 9.

Material	SOP	ΔT_{41} [K]	RT_{T0j} [°C]
BM Ring 03	1	9	-78
	2	12	-75
BM Ring 04	1	17	-70
	2	18	-69
WM W03	1	34	-26
	2	40	-20

Table 9: Adjusted reference temperatures RT_{T0j} values

The prediction of the RT_{T0j}-values for the two base metals (ring 03, ring 04) and the weld metal for 55 EFPY with and without the MOX fuel loading starting after cycle 38 was carried out using the IAEA Guideline TRS No. 429 [50].
he results of the IAEA Guideline TRS No. 429 analysis for the base metals ring 03 and ring 04 and the weld metal are shown in Table 10.

Material	EFPY	Fluence [n/cm^2]	RT_{T0j} [°C]
BM Ring 03	15	9.71E+18	-67
	17	1.36E+19	-66
	55	3.22E+19	-64
	55	3.4E+19*	-64
BM Ring 04	15	9.71E+18	-60
	17	1.36E+19	-59
	55	3.22E+19	-55
	55	3.4E+19*	-55
WM W03	15	9.71E+18	-8
	17	1.36E+19	-5
	55	3.22E+19	3
	55	3.4E+19*	3

*) with MOX fuel loading starting after cycle 38

Table 10: Results of RT_{T0j}-determination according to IAEA Guideline TRS No. 429 [50]

The weld metal is leading for 55 EFPY with following maximum RT_{T0j} values:

RT_{T0j} = 3 °C at 3.22E+19 n/cm^2 (E > 1MeV) for loading without MOX fuel

RT_{T0j} = 3 °C at 3.40E+19 n/cm^2 (E > 1MeV) for loading with MOX fuel after cycle 38.

6.2.5 Discussion of the Results

The predicted RT_{NDTj} and RT_{T0j} values and the differences between them are given in Table 11.

Material	EFPY	Fluence [n/cm^2]	RT_{NDTj}* (R.G. 1.99) [°C]	RT_{T0j}** (TRS 429) [°C]	Difference [K]
BM Ring 03	15	9.71E+18	10	-67	77
	17	1.36E+19	11	-66	77
	55	3.22E+19	13	-64	77
	55	3.40E+19*	13	-64	77
BM Ring 04	15	9.71E+18	7	-60	67
	17	1.36E+19	8	-59	67
	55	3.22E+19	12	-55	67
	55	3.40E+19*	12	-55	67
WM W03	15	9.71E+18	7	-8	15
	17	1.36E+19	10	-5	15
	55	3.22E+19	18	3	15
	55	3.40E+19*	18	3	15

*) determined with specimen orientation L-S
**) determined with specimen orientation T-L

Table 11: RT_{NDTj} and RT_{T0j} values,

The KCB RPV materials have a low irradiation embrittlement and the adjusted reference temperatures have a big margin to the RT_{limit} curve. For the two base metals Ring 03 and Ring 04 the RT_{T0j} values are by about 77 K and 67 K, respectively lower than the RT_{NDTj} values at end of life. For the leading weld metal the difference in the values from the RT_{NDTj} - determination and the RT_{T0}- determination after 55 EFPY is 15 K.

After 55 EFPY the adjusted reference temperatures of the leading weld metal according to the RT_{NDT} as well as the RT_{T0} concept with and without MOX fuel loading are:

RT_{NDTj} = 18 °C

RT_{T0j} = 3 °C.

6.3 Structural Integrity Assessment

The brittle safety assessment of the RPV Borssele in case of loss of coolant accidents has been performed according to state-of-the-art. All relevant locations, not only the core weld have been considered. The allowable reference temperatures and the material reference temperatures for 55 EPFY are compared .

It turns out that for all considered locations a large safety margin between allowable and material reference temperature exists, which will be even larger, if RT_{T0} instead of RT_{NDT} is used.

For the most severe transients and flaw assumptions a safety margin between the adjusted material reference temperature and the allowable RT_{NDT} of 44 K was calculated for the unirradiated and 104 K for the irradiated regions of the RPV for 55 EFPY.

In addition to that the material reference temperatures are far below the limit values in relevant standards like the KTA 3203 limit curve (40 °C limit at no fluence and 65 °C limit at 3.5E+19 n/cm^2), even larger conservatism is obvious, if the Borssele material values are compared with the American standard 10 CFR 50.61 (149 °C limit value for welds, see 6.3.1).

6.3.1 According to U.S. 10 CFR 50

The NPP Borssele is designed and manufactured by Siemens/KWU according to German requirements. A comparison with American NPPs according to American design and manufacturing procedures can be made, but there are differences in the two designs, which complicates a quantitative comparison. The rules in the U.S 10 CFR 50 are based on the American design plants and can therefore be only used for a qualitative statement for the integrity assessment of the KCB RPV. The PTS screening criterion in the 10 CFR 50 is based on probabilistic analyses of US plants and is 149 °C for circumferential welds (reference temperature limit for welds). It says that if the material reference temperature of a considered weld after irradiation is below that value, no plant specific analysis has to be performed, because the general analyses which lead to the screening criterion covers this material. If the Borssele material value of RT_{NDTj} = 18 °C for the weld (which is leading) is compared to the 149 °C screening limit, a huge safety margin can be observed.

6.3.2 End of Life Prediction

The updated state of the art thermal-hydraulic analysis includes

- realistic strip and plume cooling
- fluid and heat transfer coefficient input files for the whole RPV

instead of the axisymmetric load case “Circumferential Thermal shock” in the core weld region of the former analysis.

The simulation of the plant behavior during the most severe loading condition of a loss of coolant accident has been done by state-of-the-art fracture mechanics analysis and variation of the relevant leakage load paths in case of safety injection. Following areas of interest:

- nozzle, hot and cold (new)
- flange (new)
- core weld (updated)

were considered simultaneously and cover all relevant locations with high load and/or irradiation response.

All these RPV areas were considered for each transient. It turned out that in the nozzle area the leading transients are those with small leak sizes < 100 cm². For the irradiated core region the leading transient is a LOCA with 100 cm² leak size, which is comparable to analyses performed for German NPPs. In the unirradiated flange region the leading transient is a LOCA with 40 cm² leak size.

If the conventional RT_{NDT} material concept is applied, the irradiated core weld region is no longer the leading area, but the nozzle regions are more relevant because of the higher loading.

But for all specific RPV areas and for the relevant loss of coolant paths investigated, the exclusion of brittle fracture is demonstrated with large safety margins for all different injection modes.

6.3.3 Allowable Pressure during Service Conditions Level A / B (ASME)

During normal operation the pressure-temperature limit curve cannot be exceeded since this is prevented by the reactor coolant pressure limitation systems. The existing pressure-temperature limit curve for the RPV was predicted for 40 years of service with an adjusted reference nil-ductility temperature RT_{NDTj} of 44 °C and is given in Work report .

In the pressure-temperature limit curve was re-evaluated with the new predicted RT_{NDTj} of 18 °C for 55 EFPY.

For calculation of the allowable reactor coolant pressure in dependence of the coolant inlet temperature a procedure is given in ASME XI, Appendix G [87]. This procedure in Appendix G has changed since the last pressure-temperature limit calculation in 1994. For heat up the 1/4-thickness defect is postulated at the outside surface. For steady-state condition the heat up stress is 0 but the 1/4-thickness defect is postulated at inside surface. The requirement from which the allowable pressure for any assumed rate of temperature change can be determined has changed from K_{IR} (reference fracture toughness) to K_{Ic}:

$$2K_{Im} + K_{It} < K_{Ic}$$

K_{Ic} = fracture toughness

K_{Im} = stress intensity factor corresponding to membrane tension

K_{It} = stress intensity factor corresponding to radial thermal gradient

The allowable internal pressure p given in [87] is as follows:

$$p = \frac{K_{Ic} - K_{It}}{2 \cdot M_m} \cdot \frac{t}{r_i}$$

M_m = correction factor of the geometry depending on flaw size and location

t = wall thickness

r_i = inner radius.

For startup condition the allowable pressure must be calculated for steady-state (no heat up stress) and for consideration of heat up stress.

For cool down condition the allowable pressure will be calculated for an inside surface flaw as the positive fraction of the thermal stress is acting on the inside flaw.

The allowable pressure temperature relationship is the minimum pressure at any temperature determined during startup and cool down condition.

Due to the new prediction of RT_{NDTj} = 18 °C instead of 44 °C and due to the change of the fracture toughness curve from K_{IR} to K_{Ic} the allowable pressure temperature curve is shifted more than 50 K to lower wall temperatures. This means that a correction of the existing reactor coolant pressure limitation system is not necessary in respect to safety reasons.

6.4 Final Assessment for 60 Years LTO

The safe operation of the KCB RPV in terms of structural integrity is guaranteed for all load cases considered with large safety margins.

For the most severe transients and flaw assumptions a safety margin between the adjusted material reference temperature according to the RT_{NDT}-concept and the allowable RT_{NDT} of 44 K was calculated for the unirradiated and 104 K for the irradiated regions of the RPV for 55 EFPY. Using the RT_{T0}-concept instead of the RT_{NDT}-concept the safety margin could be enhanced for the unirradiated region to 121 K and for the irradiated region to 119 K .

After 60 years of operation a low irradiation embrittlement is expected for the core beltline region of the KCB RPV which will be experimentally validated by the ongoing RPV irradiation surveillance program.

Moreover, a correction of the existing reactor coolant pressure limitation system is not necessary in respect to safety reasons.

7. Summary and Conclusion

See first page.

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9. List of Figures

10. List of Tables

Table 1: P-, Ni-, Cu- and Mn- content of cylindrical shell 03, 04 and of the circumferential weld W03

Table 2: RT_{NDT}-temperature of cylindrical shell 03 and 04 and of weld material W03 from SOP 0,

Table 3: Changes in criteria temperature and upper shelf energy due to irradiation,

Table 4: Transition temperatures and upper shelf energy for the unirradiated sets SOP 0 with L-S orientation of the base metal and SOP 0a with T-L orientation of the base metal

Table 5: T_0 values for SOP 0a for the materials of the KCB RPV determined according to ASTM E 1921 [62], and reference temperatures RT_{T0} for the materials of the KCB RPV determined according to ASME Code [48]

Table 6: Comparison of criteria temperatures NDT-T, T_{AV} (68J) – 33K and T_{AV} (0.9mm) – 33K for the determination of RT_{NDT} (n.d. = not determined)

Table 7: Adjusted reference temperatures RT_{NDTj} according to KTA 3203 [36]

Table 8: Results of RT_{NDTj}-determination according to Reg.-Guide 1.99 Rev.2 Position 2 [40]

Table 9: Adjusted reference temperatures RT_{T0j} values

Table 10: Results of RT_{T0j}-determination according to IAEA Guideline TRS No. 429 [50]

Table 11: RT_{NDTj} and RT_{T0j} values,