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Assessment and management of ageing of major nuclear power plant components important to safety:

PWR pressure vessels

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FOREWORD

At present, there are over four hundred operational nuclear power plants (NPPs) in IAEA Member States Operating experience has shown that ineffective control of the ageing degradation of the major NPP components (e g , caused by unanticipated phenomena and by operating, maintenance or manufacturing errors) can jeopardize plant safety and also plant life Ageing in these NPPs must be therefore effectively managed to ensure the availability of design functions throughout the plant service life From the safety perspective, this means controlling within acceptable limits the ageing degradation and wear-out of plant components important to safety so that adequate safety margins remain, i e integrity and functional capability in excess of normal operating requirements

This TECDOC is one in a series of reports on the assessment and management of ageing of the major NPP components important to safety The reports are based on experience and practices of NPP operators, regulators, designers, manufacturers and technical support organizations and a widely accepted Methodology for the Management of Ageing of NPP Components Important to Safety which was issued by the IAEA in 1992

The current practices for the assessment of safety margins (fitness-for-service) and the inspection, monitoring and mitigation of ageing degradation of selected components of Canada deuterium–uranium (CANDU) reactors, boiling water reactors (BWRs), pressurized water reactors (PWRs), including water moderated, water cooled energy reactors (WWERs) are documented in the reports These practices are intended to help all involved directly and indirectly in ensuring the safe operation of NPPs, and also to provide a common technical basis for dialogue between plant operators and regulators when dealing with age related licensing issues Since the reports are written from a safety perspective, they do not address life or life-cycle management of the plant components, which involves the integration of ageing management and economic planning The target audience of the reports consists of technical experts from NPPs and from regulatory, plant design, manufacturing and technical support organizations dealing with specific plant components addressed in the reports

The NPP component addressed in the present publication is the PWR pressure vessel The work of all contributors to the drafting and review of this report is greatly appreciated In particular, the IAEA would like to acknowledge the contributions of T R Mager, M Brumovsky, M Erve, M J Banic, C Faidy and Ph Tipping J Pachner (IAEA Division of Nuclear Installation Safety) and P E MacDonald (INEEL) directed the preparation of the report

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

1.1 BACKGROUND

Managing the safety aspects of nuclear power plant (NPP) ageing requires implementation of effective programmes for the timely detection and mitigation of ageing degradation of plant systems, structures and components (SSCs) important to safety, so as to ensure their integrity and functional capability throughout plant service life. General guidance on NPP activities relevant to the management of ageing (maintenance, testing, examination and inspection of SSCs) is given in the International Atomic Energy Agency (IAEA) Nuclear Safety Standards (NUSS) Code on the Safety of Nuclear Power Plants: Operation [1] and associated Safety Guides on in-service inspection [2], maintenance [3] and surveillance [4].

The Operation Code requires that NPP operating organizations prepare and carry out a programme of periodic maintenance, testing, examination and inspection of plant SSCs important to safety to ensure that their level of reliability and effectiveness remains in accord with the design assumptions and intent and that the safety status of the plant has not been adversely affected since the commencement of operation. This programme is to take into account the operational limits and conditions, any other applicable regulatory requirements, and be re-evaluated in the light of operating experience. The associated Safety Guides provide further guidance on NPP programmes and activities that contribute to timely detection and mitigation of ageing degradation of SSCs important to safety.

The Safety Guide on In-Service Inspection [2] provides recommendations on methods, frequency and administrative measures for the in-service inspection programme for critical systems and components of the primary reactor coolant system aimed at detecting possible deterioration due to the influences of stress, temperature, radiation, etc. and at determining whether they are acceptable for continued safe operation of the plant or whether remedial measures are needed. Organizational and procedural aspects of establishing and implementing an NPP programme of preventive and remedial maintenance to achieve design performance throughout the operational life of the plant are covered in the Maintenance Safety Guide [3]. Guidance and recommendations on surveillance activities for SSCs important to safety (i.e. monitoring plant parameters and systems status, checking and calibrating instrumentation, testing and inspecting SSCs, and evaluating results of these activities) are provided in the Surveillance Safety Guide [4]. The aim of the surveillance activities is to verify that the plant is operated within the prescribed operational limits and conditions, to detect in time any deterioration of SSCs as well as any adverse trend that could lead to an unsafe condition and to supply data to be used for assessing the residual life of SSCs. The above Safety Guides provide general guidance, but do not give detailed technical advice for particular SSCs.

Guidance specific to ageing management is given in the reports entitled Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety [5] and Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing [6]. Guidance provided in these reports served as a basis for the development of component specific technical documents (TECDOCs) on the Assessment and Management of Ageing of Major NPP Components Important to Safety. The present publication on pressurized water reactor (PWR) pressure vessels is one of these TECDOCs.

The first PWR in the West was the Yankee-Rowe plant in Rowe, Massachusetts, USA. The first reactor pressure vessel (RPV), Yankee-Rowe's vessel, weighed 210 000 kg (470 000 pounds) and had an inside diameter of 277 cm (109 in.). Today, depending on the design of the nuclear steam supply system (NSSS), two, three, four or six loops, the RPV can weigh as much as 427 000 kg (941 600 pounds) and have an inside diameter of 440 cm (173 in.). PWRs have been operating in Argentina, Armenia, Belgium, Brazil, Bulgaria, China, the Czech Republic, Finland, France, Germany, Hungary, Italy, Japan, the Republic of Korea, the Netherlands, the Russian Federation, Slovakia, Slovenia, South Africa, Spain, Sweden, Switzerland, Taiwan (China), Ukraine, the United Kingdom and the USA.

The PWR pressure vessel is the most important pressure boundary component of the NSSS because its function is to contain the nuclear core under elevated pressures and temperatures. Additional RPV functions are to provide structural support for the reactor vessel internals and the core. The RPV design attempts to protect against rupture by considering all the postulated transients that the NSSS may undergo. Since each postulated transient constitutes a loading–unloading cycle, a fatigue analysis is performed for each RPV. The load restriction and fatigue life on as-fabricated RPVs are governed by industrial codes and regulatory requirements throughout the world. In addition, RPV design allows for changes in material properties due to fast neutron exposure and other effects (ageing) of the vessel wall surrounding the core. The RPV is designed so that the vessel wall around the core region is free of structural discontinuities or other stress inducers. The radiation and service condition induced material property changes are thus confined to a portion of the reactor vessel with a straight cylindrical wall in which stresses are theoretically simple to analyse.

RPVs are fabricated in accordance with strict quality assurance (QA) programmes. Information about how to produce a RPV is well documented. All phases are covered, beginning with the technical requirements and ending with the monitoring of all work performance activities. During fabrication activities, the RPV undergoes non-destructive examinations (NDE) and concludes fabrication with a shop hydrostatic test at some given value above operating limits. Further, once a NPP is in operation, the RPV is subjected to comprehensive periodic in-service inspection, including material radiation damage assessment via the surveillance programme. (There is no surveillance programme for a number of older water moderated, water cooled energy reactor (WWER) power plants and some western PWRs; however, information from other power reactors is used in conjunction with empirical correlations to predict the radiation damage).

Pressurized light water reactor vessels experience service at 250-320°C and receive significant levels of fast neutron fluence, ranging from about 5×10^{22} to about 3×10^{24} n/m^2, depending on the plant design. There are also differences in materials used for the various designed reactors. Weldments also vary in type and impurity level. Accordingly, the assessment of ageing degradation of major components such as the pressure vessel is a common objective for the safe operation of all PWRs.

1.2. OBJECTIVE

The objective of this report is to document the current practices for the assessment and management of the ageing of NPP RPVs. The report emphasizes safety aspects and also provides information on current inspection, monitoring and maintenance practices for managing ageing of RPVs.

The underlying objective of this reports series is to ensure that the information on the current assessment methods and ageing management techniques is available to all involved, directly and indirectly, in the operation of NPPs in the IAEA Member States.

NPP operators, regulators, technical support organizations, designers and manufacturers are likely to be interested in this report.

The TECDOC does not address life or life-cycle management of PWR RPVs because it is written from the safety perspective and life management includes economic planning.

1.3. SCOPE

This report provides the technical basis for managing the ageing of the PWR and pressurized heavy water RPVs to assure that the required safety and operational margins are maintained throughout the plant service life. The scope of the report includes the following RPV components; vessel shell and flanges, structural weldments, closure studs, nozzles, penetrations and top and bottom closure heads. The scope of this report does not treat RPV internals, the control rod drive mechanisms (CRDMs), or the primary boundary piping used in PWRs. All the various size and types of PWR pressure vessels are covered by this report including the WWER (Vodo-Vodianyi Energeticheskii Reactor) plants built in Russia and elsewhere. The boiling water reactor (BWR) pressure vessels and Canadian deuterium–uranium (CANDU) pressure tubes and calandria are covered in separate companion reports.

1.4. STRUCTURE

The designs, materials of construction and physical features of the various PWR pressure vessels are described in Section 2. The codes, regulations and guides used in a number of countries to design RPVs are summarized in Section 3. Section 4, Ageing Mechanisms, identifies the dominant ageing mechanisms, sites, consequences and operating experience. Section 5, Inspection and Monitoring Requirements and Technologies, addresses the application of various inspection technologies to assess the condition of the RPV. Section 6, Ageing Assessment Methods, gives the current practices and data required in assessing degradation of an RPV. Section 7, Ageing Mitigation Methods, describes operational methods used to manage ageing mechanisms (i.e. to minimize the rate of degradation) and maintenance methods used to manage ageing effects (i.e. to correct unacceptable degradation). Section 8 describes an RPV ageing management programme utilizing a systematic ageing management process.

2. DESCRIPTION OF REACTOR PRESSURE VESSEL

This section provides a description of the PWR pressure vessels and includes design features, applicable material specifications and differences amongst the various RPV components.

Western type PWR pressure vessels were designed by Babcock & Wilcox (B&W) Company, Combustion Engineering, Inc., Framatome, Mitsubishi Heavy Industries, Ltd, Siemens/KWU, and Westinghouse. The RPVs were fabricated by B&W Company, Chicago Bridge and Iron Company, Combustion Engineering, Inc., Creusot-Loire, Klöckner, Rotterdam Dry Dock Company, MAN GHH, Mitsubishi Heavy Industries, Ltd and Udcomb.

The WWER RPVs were designed by OKB Gidropress, the general designer for all NPPs in the former Soviet Union and the Community for Mutual Economical Assistance (CMEA) countries. Some small modifications were made in the Czech designs by ŠKODA Co. The WWER plants have been built in two sizes; the WWER-440s which are 440 MWe plants and the WWER-1000s which are 1000 MWe plants. There are two designs for each size; the WWER-440 Type V-230, the WWER-440 Type V-213, the WWER-1000 Type V-302 and the WWER-1000 Type V-320. The Type V-230s were built first and the V-320s were built last. The WWER-440 RPVs are similar as are the WWER-1000 RPVs; the differences in the two designs for the two plant sizes are mainly in the safety systems. There are only two WWER-1000 Type V-302 pressure vessels, so only WWER-1000 Type V-320 information is presented in this report. The WWER pressure vessels were manufactured at three plants, the Izhora Plant near Saint Petersburg (Russia), the Atommash Plant on the Volga (Russia) and the ŠKODA Nuclear Machinery Plant in the Czech Republic.

2.1. RPV DESIGN FEATURES

2.1.1. Western pressure vessels

A Westinghouse designed RPV is shown in Fig. 1. This vessel is fairly typical of the reactor vessels used in all the so-called western designed RPVs. However, there are significant differences in size, nozzle designs, penetration designs and other details among the various suppliers. The RPV is cylindrical with a hemispherical bottom head and a flanged and gasketed upper head. The bottom head is welded to the cylindrical shell while the top head is bolted to the cylindrical shell via the flanges. The cylindrical shell course may or may not utilize longitudinal weld seams in addition to the girth (circumferential) weld seams. The body of the vessel is of low-alloy carbon steel. To minimize corrosion, the inside surfaces in contact with the coolant are clad with a minimum of some 3 to 10 mm of austenitic stainless steel.

Numerous inlet and outlet nozzles, as well as control rod drive tubes and instrumentation and safety injection nozzles penetrate the cylindrical shell. The number of inlet and outlet nozzles is a function of the number of loops or steam generators. For the majority of operating NPPs, the nozzles are set-in nozzles. However, there are a number of operating NPPs with RPVs with set-on nozzles. A set-in nozzle has the flange set into the vessel wall, a set-on nozzle has the flange placed on the vessel wall surface as shown in Fig. 2.

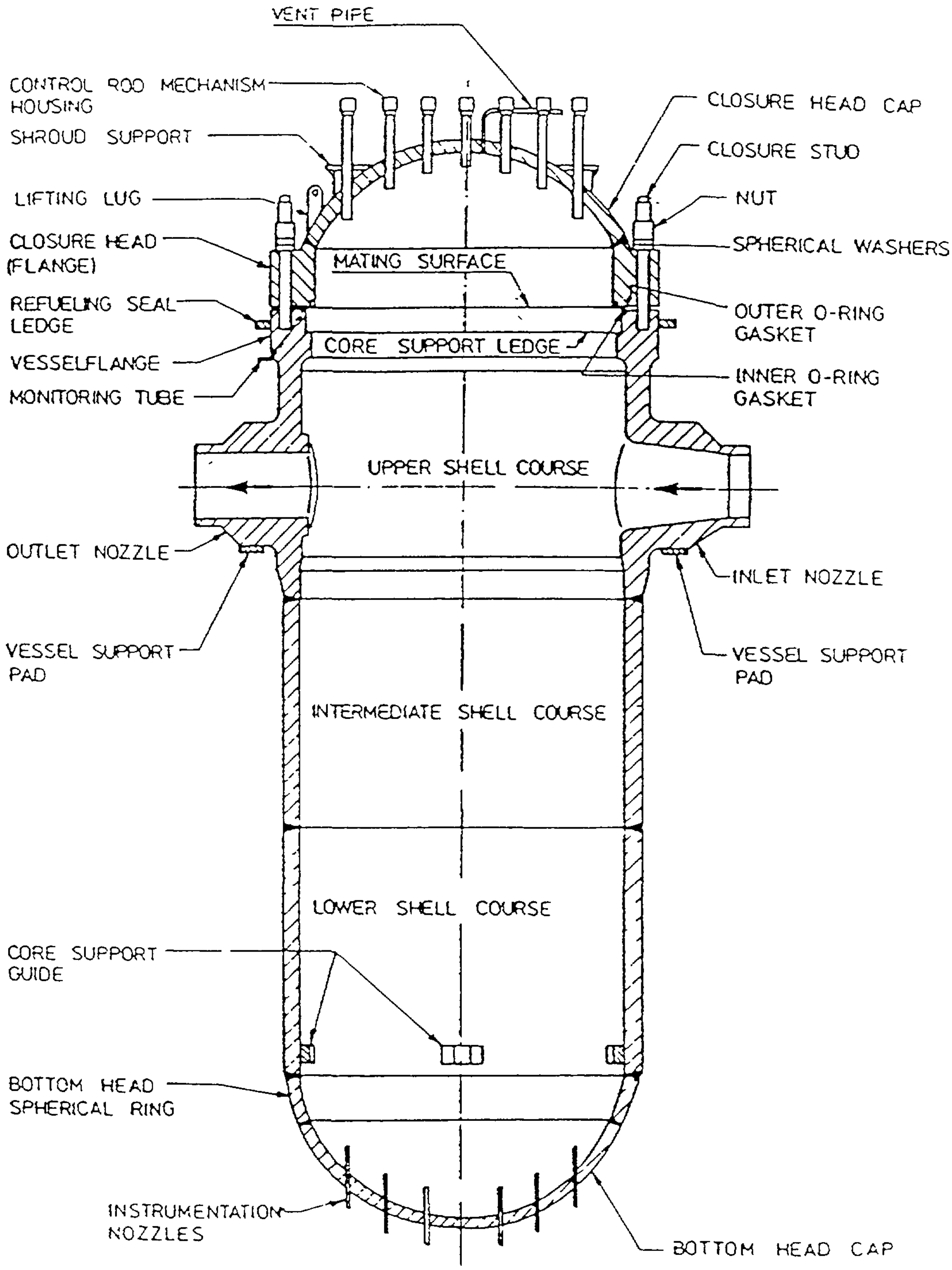


FIG. 1 A typical Westinghouse reactor pressure vessel

The PWR pressure vessel design pressure is 17.24 MPa (2500 psi) and the operating pressure is 15.51 MPa (2250 psi). The usual vessel preservice hydrostatic pressure is 21.55 MPa (1.25 × design pressure). The PWR pressure vessel design temperature is 343°C (650°F) while the operating temperature is typically 280 to 325°C (540 to 620°F).

An ABB-CE (formally Combustion Engineering) designed RPV is shown in Fig. 3. The ABB-CE design is somewhat different from some other western designed RPVs in that there are a relatively large number of penetrations which are made from Alloy 600. As will be discussed in a later section, reactor penetrations fabricated from Alloy 600 can be of concern to ageing management of the RPV.

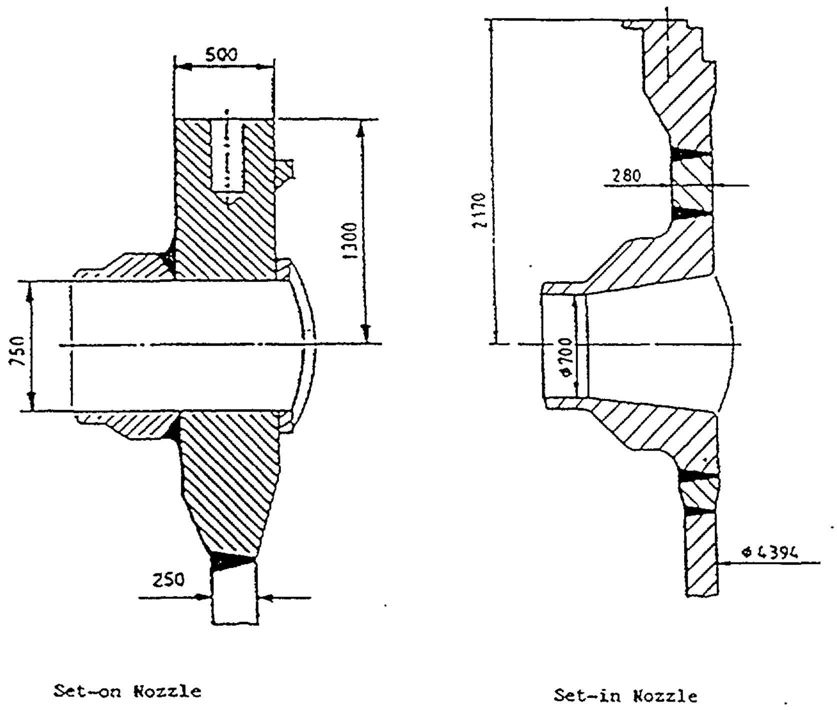


FIG. 2. Sketches of typical set-on and set-in nozzles used in reactor pressure vessels.

A Siemens (KWU) designed RPV is shown in Fig. 4. The features of the Siemens RPV which significantly differ from other western design are as follows:

- set-on inlet and outlet nozzles
- reinforcement of the flange portion
- no nozzles or guide tubes within the lower part of RPV (no risk of breaks and leaks below the loops)
- one piece upper part section
- special screwed design for the control rod drive and instrumentation nozzle penetrations made from co-extruded pipe.

The French RPVs are designed by Framatome and manufactured by Creusot-Loire. Sketches of the French 3-loop (900 MWe) and 4-loop (1450 MWe) RPVs are presented in Fig. 5 and the major characteristics of the RPVs used for the 4-loop N4 plants are listed in Table I. The French RPVs are constructed with ring sections and, therefore, there are no longitudinal (vertical) welds. Generally, the core beltline region consists of two parts, although the Sizewell B vessel has only one ring and some old vessels have three rings in the beltline region. Six or eight set-in nozzles are used along with stainless steel safe ends connected to the nozzles with dissimilar metal welds. The design pressure is 17.2 MPa, the operating pressure is 15.5 MPa, the initial pre-service hydrostatic pressure is 22.4 MPa (1.33 × design pressure) and the design life is 40 years.

2.1.2. WWER pressure vessels

The WWER pressure vessels consist of the vessel itself, vessel head, support ring, thrust ring, closure flange, sealing joint and surveillance specimens (the latter were not in the

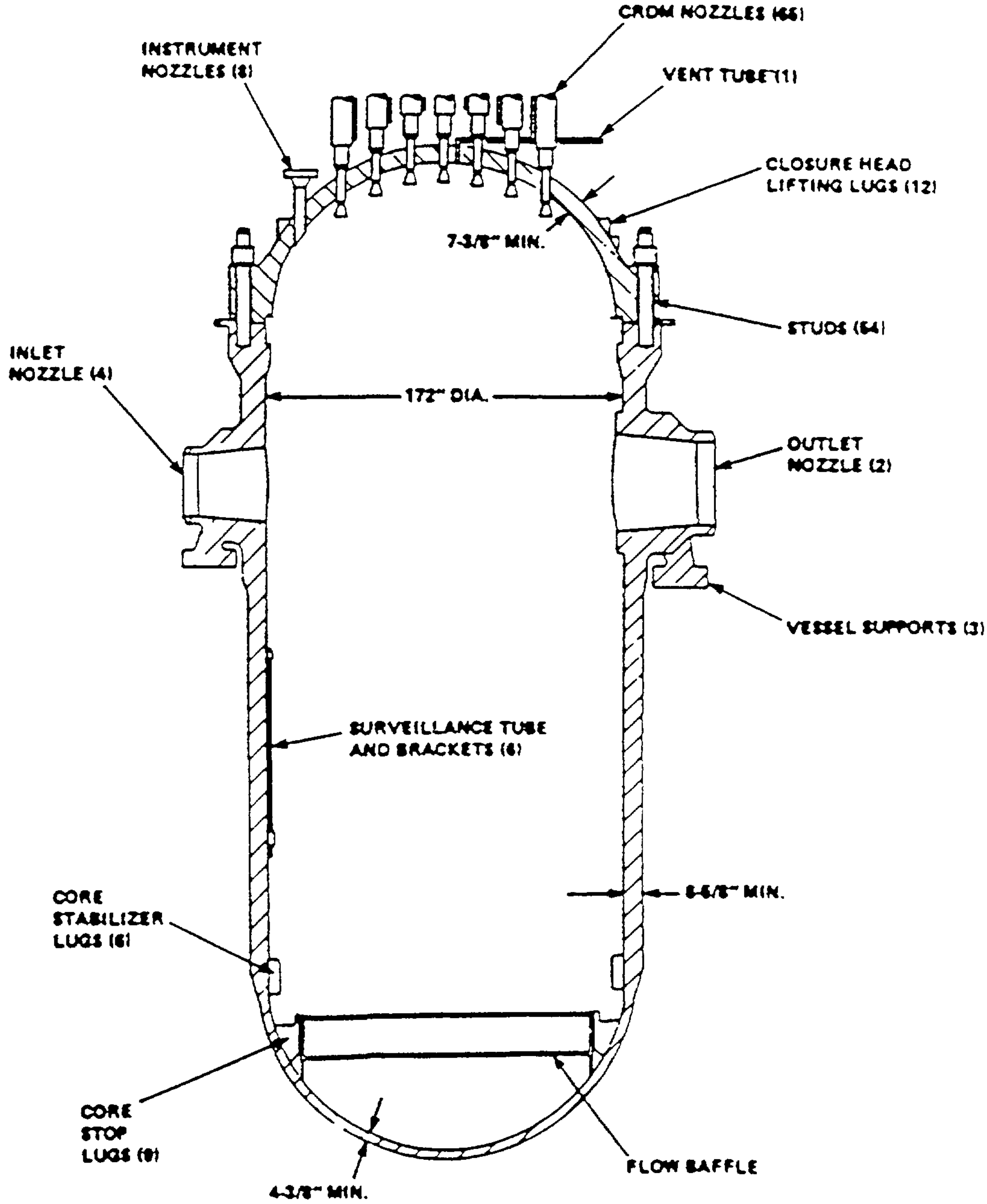


FIG. 3. A typical ABB-CE reactor pressure vessel.

WWER/V-230 type of reactors). The RPVs belong to the "normal operation system", seismic Class I and are designed for:

- safe and reliable operation for over 40 years,
- operation without damage for not less than 24 000 hours (damage in this sense includes leaks in the bolted joints and the threaded control rod drive nozzle joints, thread surface damage, etc.),
- non-destructive testing (NDT) of the base and weld metal and decontamination of the internal surfaces,

- materials properties degradation due to radiation and thermal ageing monitoring (not in the case of WWER/V-230 type of reactors),

- and all operational, thermal and seismic loadings.

The WWER RPVs have some significant features that are different from the western designs. A sketch of a typical WWER pressure vessel is shown in Fig. 6 and the main design parameters and materials are listed in Tables IIa and IIb.

- The WWER RPVs (as well as all other components) must be transportable by land, i.e. by train and/or by road. This requirement has some very important consequences on vessel design, such as a smaller pressure vessel diameter, which results in a smaller water gap thickness and thus a higher neutron flux on the reactor vessel wall surrounding the core and, therefore, requirements for materials with high resistance against radiation embrittlement.

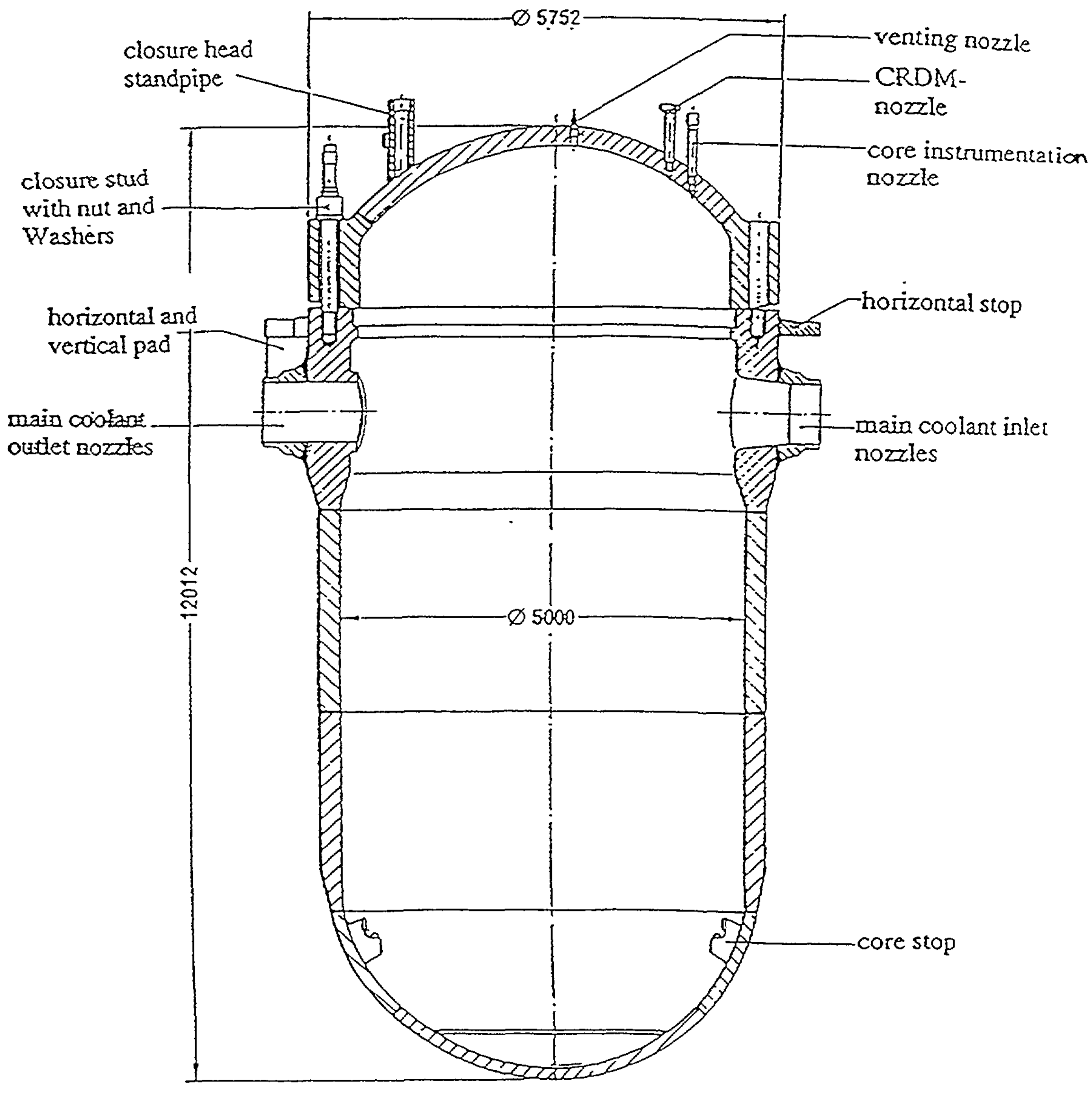


FIG 4. A typical Siemens/KWU reactor pressure vessel for a 1300 MWe plant.

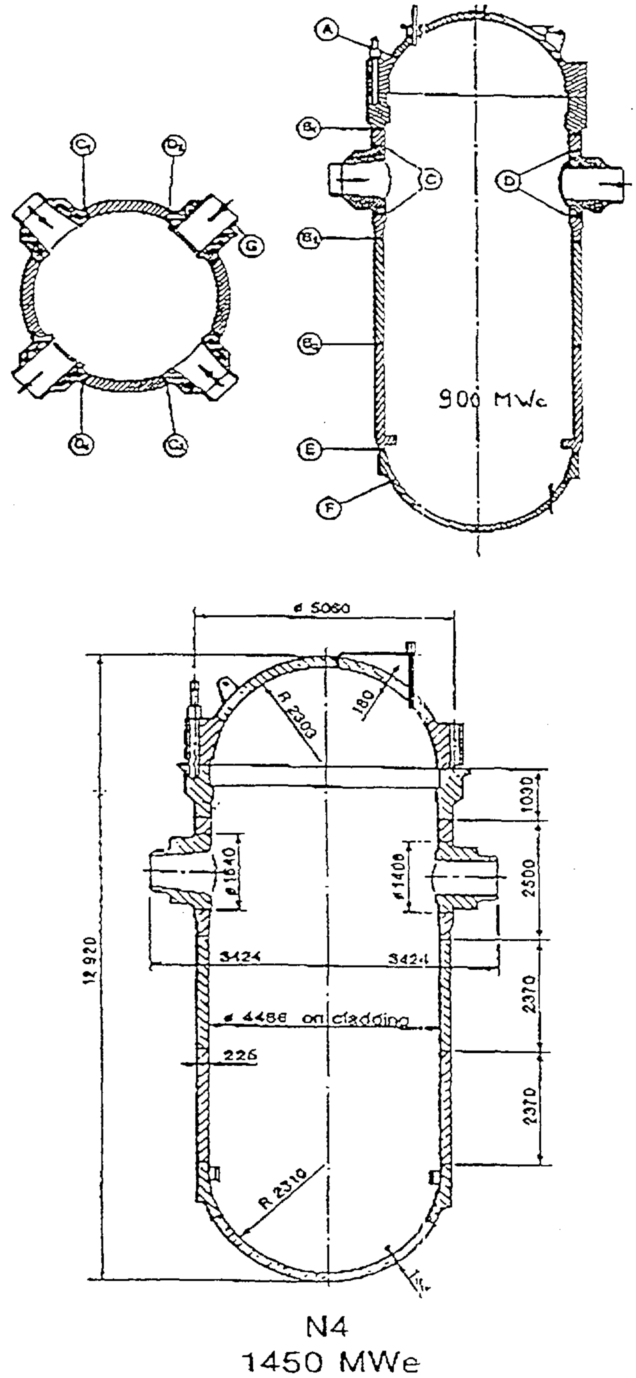


FIG. 5. Sketchs of French 3- and 4-loop RPVs; typical dimensions.

- Transport by land also results in a smaller vessel mass and, therefore, thinner walls which require higher strength materials.

- The upper part of the vessel consists of two nozzle rings, the upper one for the outlet nozzles and the lower one for the inlet nozzles. An austenitic stainless steel ring is welded to the inside surface of the vessel to separate the coolant entering the vessel through the inlet nozzles from the coolant exiting the vessel through the outlet nozzles. This design results in a rather abrupt change in the axial temperature distribution in the vessel, but uniform temperatures around the circumference.

TABLE I. WESTERN REACTOR PRESSURE VESSEL DESIGN PARAMETERS

	French 4-loop N4 type plants	German Konvoi design values	Westinghouse 4-loop plant
Thermal power (MWth)	4270	3765	3411
Electric output (MWe)	1475	> 1300	1125
Number of loops	4	4	4
Type of fuel assembly	17 × 17	18 × 18–24	17 × 17
Active length (mm)	4270	3900	366
Core diameter (mm)	4490	3910	337
Water gap width* (mm)	424	545	51.2
Linear heating rate (W/cm)	179	166.7	183
Number of control rods	73	61	53
Total flow rate (m^3/h)	98 000	67 680	86 800
Vessel outlet temperature (°C)	329.5	326.1	325.5
Outlet/inlet temperature difference (°C)	37.5	34.8	33.0
Specified RT_{NDT} at 30 L		–12°C	
ΔT_{41} at EOL (based on design values)	–	23°C	–

*Distance from the outer fuel element and the RPV inner surface.

- The WWER vessels are made only from forgings, i.e. from cylindrical rings and from plates forged into domes. The spherical parts of the vessels (the bottom and the head) are either stamped from one forged plate, or welded from two plates by electroslag welding, followed by stamping and a full heat treatment. There are no axial welds.

- The WWER inlet and outlet nozzles are not welded to the nozzle ring but they are either machined from a thicker forged ring, for the WWER-440 vessels, or forged in the hot stage from a thick forged ring for the WWER-1000 vessels. A typical WWER-440 forged and machined nozzle is shown in Fig. 7.

2.2. VESSEL MATERIALS AND FABRICATION

2.2.1. Western pressure vessels

Materials

The western PWR pressure vessels use different materials for the different components (shells, nozzles, flanges, studs, etc.). Moreover, the choices in the materials of construction changed as the PWR products evolved. For example, the Westinghouse designers specified American Society for Testing and Materials (ASTM) SA 302 Grade B for

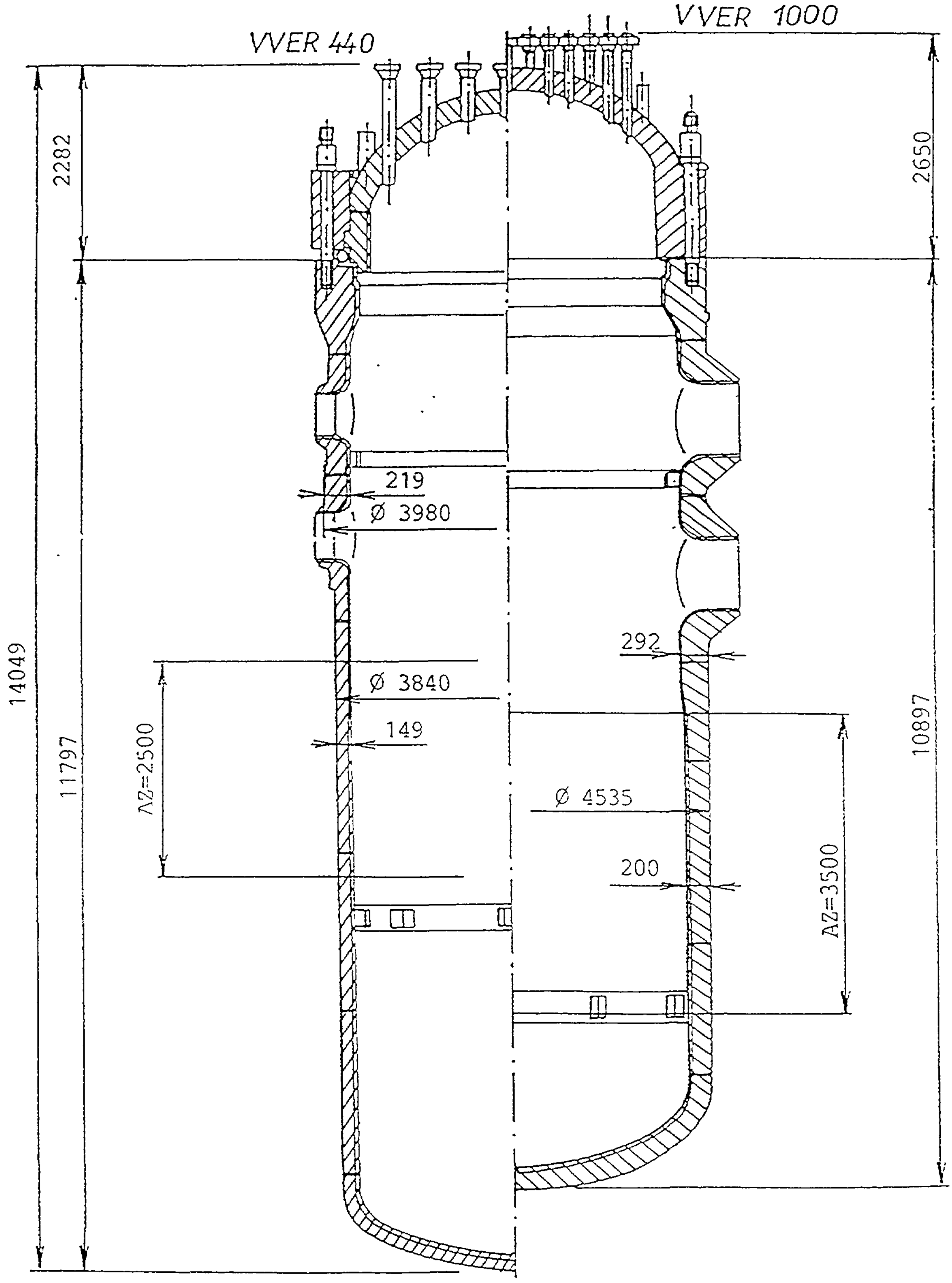


FIG. 6. WWER reactor pressure vessels (split diagram).

TABLE IIa. WWER REACTOR PRESSURE VESSEL DESIGN PARAMETERS

Reactor	WWER-440		WWER-1000
	V-230	V-213	V-320
mass [t]	215		320
length [m]	11.800		11.000
outer diameter [m]:			
- in cylindrical part	3.840		4.535
- in nozzle ring	3.980		4.660
wall thickness (without cladding) [m]:			
- in cylindrical part	0.140		0.193
- in nozzle ring	0.190		0.285
number of nozzles	2×6 [(1)]	2×6 [(1)] $+ 2 \times 3$ [(2)]	2×4 [(1)] $+ 3$ [(2)]
working pressure [MPa]	12.26		17.65
design pressure [MPa]	13.7		19.7
hydrotest pressure [MPa]	17.1	19.2[(3)]	24.6
Operating wall temperature [°C]	265		288
design wall temperature [°C]	325		350
vessel lifetime [year]	30	40	40
neutron fluence during design lifetime [n/m^2] (E_n greater 0.5 MeV)[(4)]:			
- base metal	1.6×10^{24}	2.6×10^{24}	6.3×10^{23}
- weld metal	1.3×10^{24}	1.8×10^{24}	5.7×10^{23}
cover mass [t]	50		90
number of penetrations	37 + 18		61 + 30

[(1)] Primary nozzle.
[(2)] Emergency core cooling system (ECCS) nozzle.
[(3)] Test pressure has been recently decreased to 17.2 MPa in Hungary, the Czech Republic and Slovakia.
[(4)] The fast fluence at energies greater than 0.5 MeV is about 1.67 times the fast fluence at energies greater than 1.0 MeV.

the shell plates of earlier vessels and ASTM SA 533 Grade B Class 1 for later vessels [7, 8]. Other vessel materials in common use include American Society of Mechanical Engineers (ASME) SA 508 Class 2 plate in the USA, 22NiMoCr37 and 20MnMoNi55 in Germany, and 16MnD5 in France. In addition to using plate products, all the NSSS vendors use forgings in the construction of the shell courses. Table III lists the main ferritic materials used for PWR vessel construction over the years and summarizes their chemical composition [9]. Table IV lists the individual vessel components and the various materials used for each component in the US and French N4 RPVs. These materials are discussed in somewhat more detail in the following paragraphs.

TABLE IIb MATERIALS SPECIFIED FOR WWER PRESSURE VESSEL COMPONENTS

Reactor	WWER-440		WWER-1000
	V-230	V-213	V-320
Vessel components			
- cylindrical ring	15Kh2MFA	15Kh2MFA	15Kh2NMFAA
- other parts of vessel	15Kh2MFA	15Kh2MFA	15Kh2NMFA
- cover	18Kh2MFA	18Kh2MFA	15Kh2NMFA
- free flange	25Kh3MFA	25Kh3MFA	-
- stud bolts and nuts	25Kh1MF	38KhN3MFA	38KhN3MFA
Welding process			
- automatic submerged arc	Sv-10KhMFT + AN-42	Sv-10KhMFT + AN-42M	Sv-12Kh2N2MA + FC-16A
- electroslag	Sv-13Kh2MFT + OF-6	Sv-13Kh2MFT + OF-6	Sv-6Kh2NMFTA + OF-6

SA-302, Grade B is a manganese-molybdenum plate steel used for a number of vessels made through the mid-1960s Its German designation is 20MnMo55 As commercial nuclear power evolved, the sizes of the vessels increased For the greater wall thicknesses required, a material with greater hardening properties was necessary The addition of nickel to SA-302, Grade B in amounts between 0 4 and 0 7 weight per cent provided the necessary increased hardening properties to achieve the desired yield strength and high fracture toughness across the entire wall thickness This steel was initially known as SA-302, Grade B Ni Modified

Forging steels have also evolved since the mid-1950s The SA-182 F1 Modified material is a manganese-molybdenum-nickel steel used mostly for flanges and nozzles in the 1950s and 1960s Another forging material used then was a carbon-manganese-molybdenum steel, SA-336 F1 Large forgings of these materials had to undergo a cumbersome, expensive heat treatment to reduce hydrogen blistering Eventually these steels were replaced with a steel, first described as ASTM A366 Code Case 1236 and is now known as SA-508 Class 2, that did not require this heat treatment [10] This steel has been widely used in ring forgings, flanges and nozzles It was introduced into Germany with the designation 22NiMoCr36 or 22NiMoCr37 With slight modifications, this steel became the most important material for German reactors for a long time In addition, SA-508 Class 3 (20MnMoNi55 in Germany and 16 MnD5 and 18MnD5 in France) is used in the fabrication of western RPVs

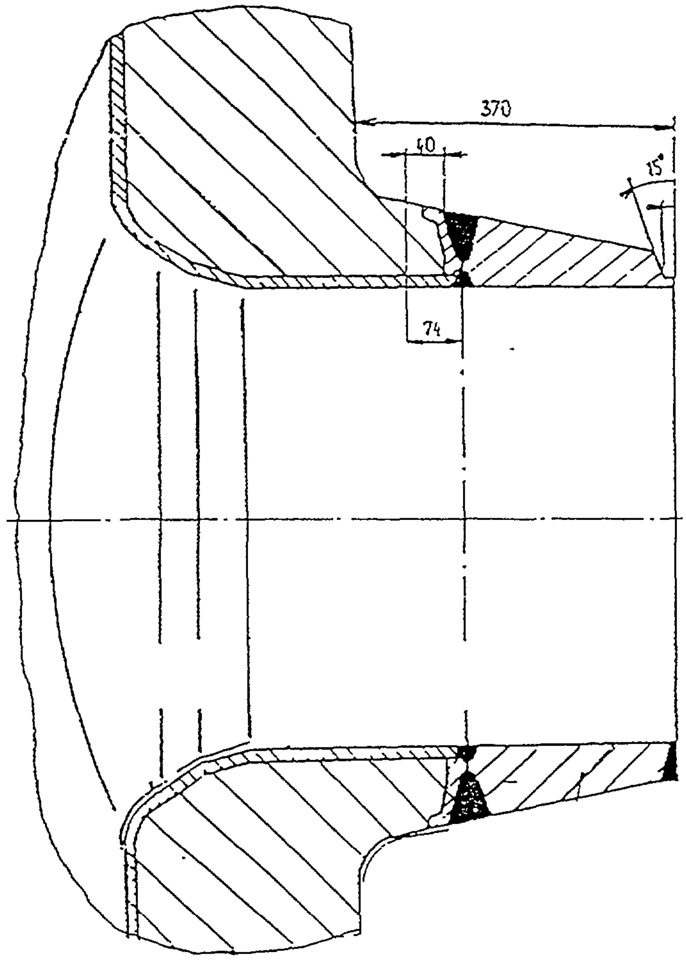


FIG. 7. Sketch of a typical forged and machined WWER-440 pressure vessel nozzle.

Although many materials are acceptable for reactor vessels according to Section III of the ASME Code, the special considerations pertaining to fracture toughness and radiation effects effectively limit the basic materials currently acceptable in the USA for most parts of vessels to SA-533 Grade B Class 1, SA-508 Class 2 and SA-508 Class 3 [11].

The part of the vessel of primary concern with regard to age related degradation is the core beltline — the region of shell material directly surrounding the effective height of the fuel element assemblies plus an additional volume of shell material, both below and above the active core, with an end-of-life fluence of more than 10^{21} n/m^2 (E >1 MeV) [10]. It is typically located in the intermediate and lower shells (Fig. 8). The low alloy steels making up the beltline are subject to irradiation embrittlement that can lead to loss of fracture toughness. When early vessels were designed and constructed, only limited data existed about changes in material properties caused by radiation damage. Now we know that the susceptibility of RPV steel is strongly affected by the presence of copper, nickel and phosphorus. Because operating vessels fabricated before 1972 contain relatively high levels of impurity copper and phosphorous, irradiation damage becomes a major consideration for their continued operation.

TABLE III CHEMICAL REQUIREMENTS (HEAT ANALYSIS) MAIN FERRITIC MATERIALS FOR REACTOR COMPONENTS IN WESTERN COUNTRIES

Designation	Elements (weight %)													
	C	Si	Mn	P_{max}	S_{max}	Cr	Mo	Ni	V_{max}	Cu_{max}	Al	Sn	N2	As
ASTM A 302B		0.15	1.15				0.45							
	0.25	0.30	1.50	0.035	0.040		0.60							
ASTM A 336, Code Case 1236	0.19	0.15	1.10				0.50	0.40						
	0.25	0.35	1.30	0.035	0.035	0.35	0.60	0.50						
ASME A 508 cl 2 (1971)		0.15	0.50			0.25	0.55	0.50						
	0.27	0.35	0.90	0.025	0.025	0.45	0.70	0.90	0.05					
ASME A 533 gr B (1971)		0.15	1.15				0.45	0.40						
	0.25	0.30	1.50	0.035	0.040		0.60	0.70						
ASME A 508 cl 2 (1989)[1]		0.15	0.50			0.25	0.55	0.50						
	0.27	0.40	1.00	0.015	0.015	0.45	0.70	1.00	0.05	0.15				
ASME A 508 cl 3 (1989)[1]		0.15	1.20				0.45	0.40						
	0.25	0.40	1.50	0.015	0.015	0.25	0.60	1.00	0.05					
ASME A 533 gr B (1989)		0.15	1.15				0.45	0.40						
	0.25	0.40	1.50	0.035	0.040		0.60	0.70						
16 MnD5 RCC M 2111[2]		0.10	1.15				0.43	0.50	0.01	0.20	0.040			
	0.22	0.30	1.60	0.02	0.012	0.25	0.57	0.80						
18 MnD5 RCC M 2112 (1988)		0.10	1.15				0.45	0.50						
	0.20	0.30	1.55	0.015	0.012	0.25	0.55	0.80	0.01	0.20	0.040			
20 Mn Mo Ni 5 5 (1983, 1990)[3][4]	0.17	0.15	1.20				0.40	0.50			0.010			
	0.23	0.30	1.50	0.012	0.008	0.20	0.55	0.80	0.02	0.12[5]	0.040	0.011	0.013	0.036
22 Ni Mo Cr 3 7 (1991)[6]	0.17	0.15	0.50			0.25		0.60			0.010			
	0.23	0.35	1.00	0.012	0.008	0.50	0.60	1.20[7]	0.02	0.12[5]	0.050	0.011	0.013	0.036

(1) Supplementary Requirement S 9 1(2) and S 9 2 for A 508 cl 2 and 508 cl 3
(2) Forgings for reactor shells outside core region Restrictions for Core Region (RCC M 2111) S ≤0.008, P ≤0.008, Cu <0.08
(3) VdTUV Material Specification 401, Issue 1983
(4) KTA 3201 1 Appendix A Issue 6/90
(5) Cu content for RPV (Core Region) shall be <0.10%
(6) According *Siemens/KWU* under consideration of SR 10 (MPa Stuttgart)
(7) For flanges and tube sheets the Ni content shall be ≤1.40%

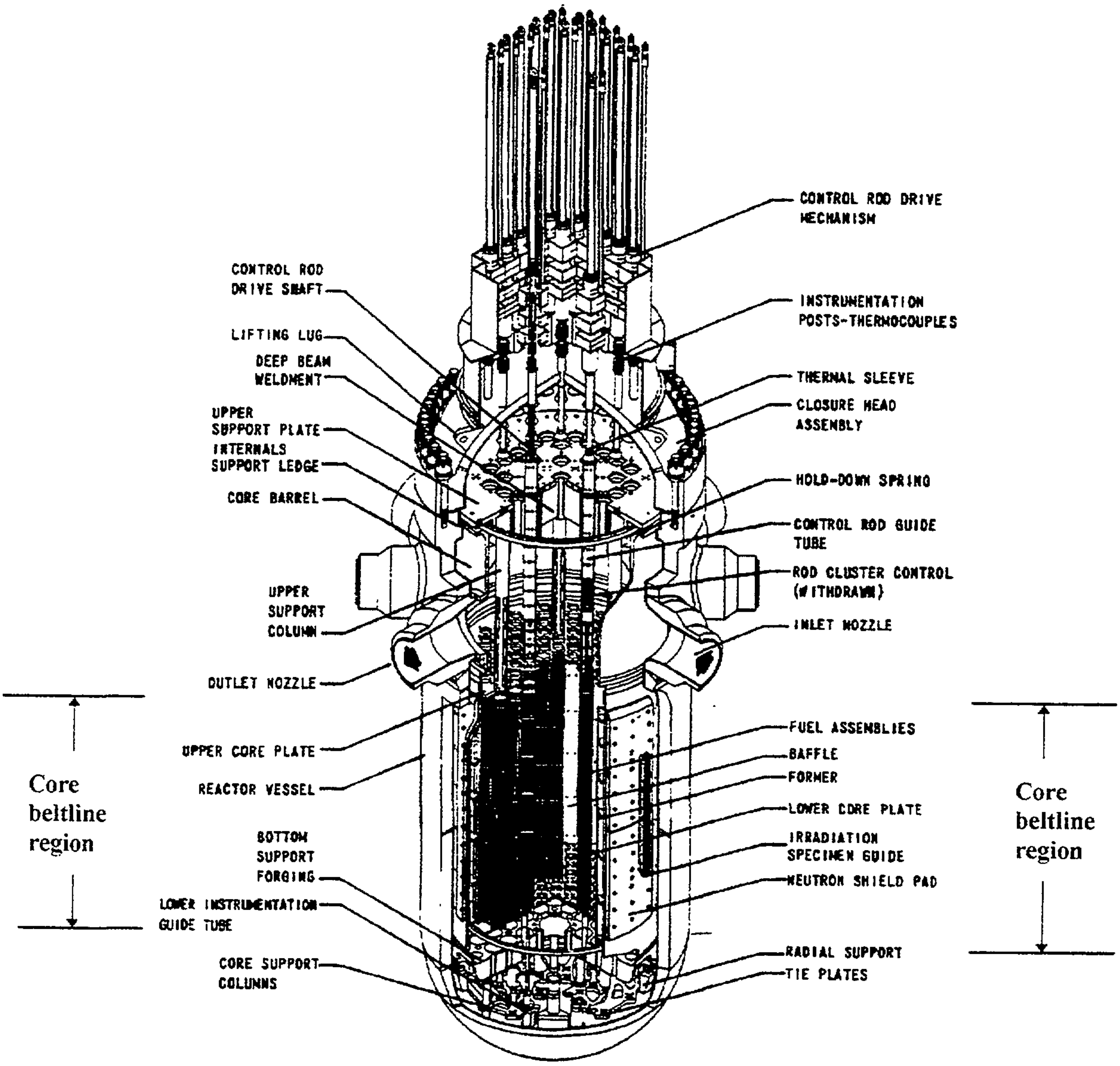


FIG. 8. Typical arrangement of reactor vessel plates and welds.

Other components of current concern with regard to ageing are certain CRDM nozzles. CRDM nozzles are made of stainless steel and Alloy 600 (ASME Code specification SB-166 bar or SB-167 pipe), a nickel base alloy. In Siemens RPVs, CRDM nozzles are made from ferritic steel, cladded with stainless steel (manufactured as co-extruded pipes), except the Obrigheim RPV, which is equipped with CRDM nozzles made from Alloy 600. Nozzles with Alloy 600 are of concern because some have experienced primary water stress corrosion cracking (PWSCC). The composition of this alloy is about 75 weight per cent nickel, 15 weight per cent chromium and 9 weight per cent iron, with trace amounts of carbon, manganese, sulphur, silicon, copper and cobalt.

The French have recently introduced the use of hollow ingots to make the beltline ring sections. The beltline material used in France is 16 MnD5. The chemical requirements for this material are listed in Table III along with the other western materials. The materials used for other N4 RPV components are listed in Table IV. The heat treatments and minimum material properties for 16MnD5 are listed in Table V. The French practices in terms of K_{CV} and hot

TABLE IV. MATERIALS SPECIFIED FOR PWR VESSEL COMPONENTS

Plants in the USA:

Closure head dome	Closure head flange	Lifting lugs	Shroud support ring	Closure head stud assembly	Vessel flange	Shells	Bottom head	Nozzles	CRDM housings	Stainless steel cladding	Leakage monitoring tubes	Core support pads (lugs)	Instrumentation tubes/ penetrations	Refuelling seal ledge
SA302 GR B	SA336	SA302 GR B	SA212 GR B	SA320 L43	SA336	SA302 GR B	SA302 GR B	SA302 GR B	SA182 TYPES 304, 316	TYPE 308L, 309L	SA312 TYPE 316	SB166	SB166	SA212 GR B
SA533 GR B Class 1	SA508 Class 2 SA508 Class 3	SA533 GR B Class 1	SA516 GR 70	SA540 B23, B24	SA508	SA533 GR B Class 1	SA533 GR B Class 1	SA533 GR B Class 1	SB166	TYPE 304	SB166	SB167	SB167	SA516 GR 70
				SA320 L43 Class 3		SA336		SA336			SB167			SA533
						SA508 Class 2 SA508 Class 3		SA508 Class 2 SA508 Class 3						

French 4-loop N4 plants.

Shells, flanges, heads, nozzles	16MnD5
Safe ends, adapter flanges	72CND18-12
Adapter sleeves, instrumentation penetrations	NC15Fe/NC30Fe
Studs, nuts, washers	40NCDV7 03
Internal supports	NC15Fe

TABLE V. HEAT TREATMENTS AND MINIMUM MATERIAL PROPERTIES FOR 16MND5

Austenisation			850–925°C
Tempering			635–668°C
Stress relief			600–630°C
Rp 0.2% at 20°C (yield stress)			>400 MPa
Rm at 20°C (tensile strength)			550–670 MPa
A% at 20°C (total elongation)			>20%
Rp 0.2% at 350°C			>300 MPa
Charpy energy in J (TL or L orientation)	at 0°C	TL:	Ind. >40 J[(1)] Mean >56 J[(2)]
		L:	Ind. >56 J Mean >72 J
	at -20°C	TL:	Ind. >28 J Mean >40 J
		L:	Ind. >40 J Mean >56 J
	at +20°C	TL:	Ind. >72 J
		L:	Ind. >88 J

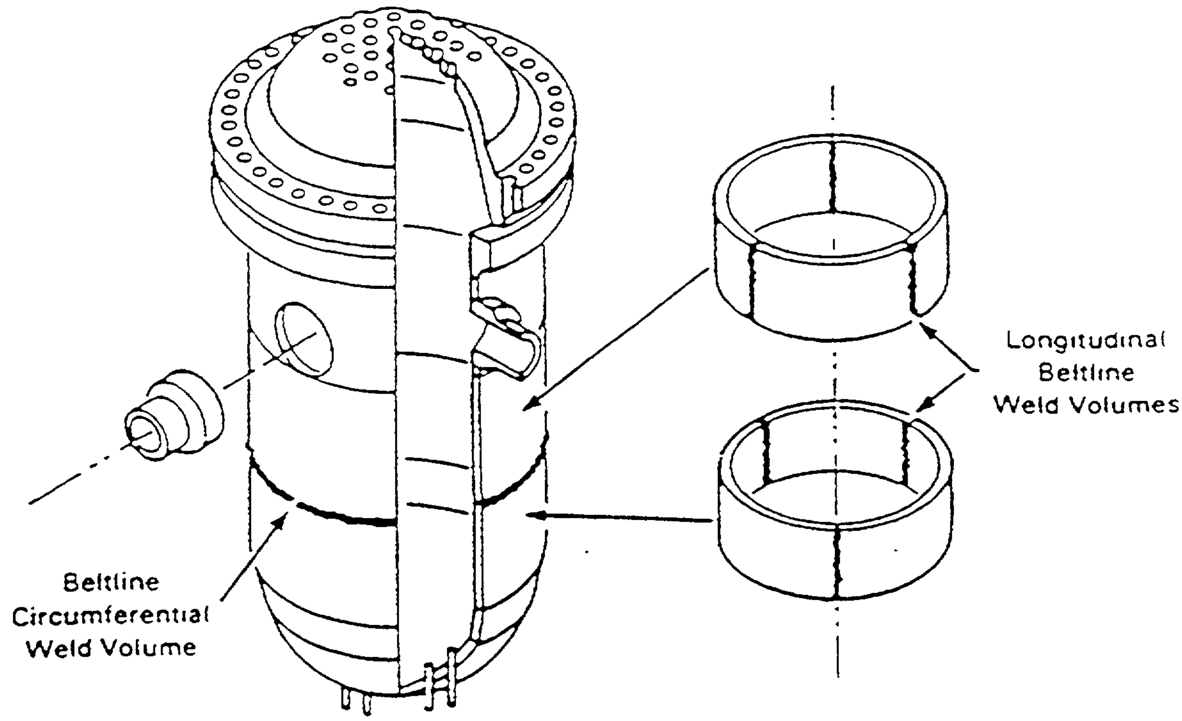
(1) Measurement is from one individual specimen.
(2) Measurements from three specimens which are averaged.

test requirements should be noted. As a general rule, material with a tensile strength at room temperature above 700 MPa cannot be used for pressure boundaries. The other western RPVs are designed with a minimum tensile strength of 350 MPa (50 Ksi).

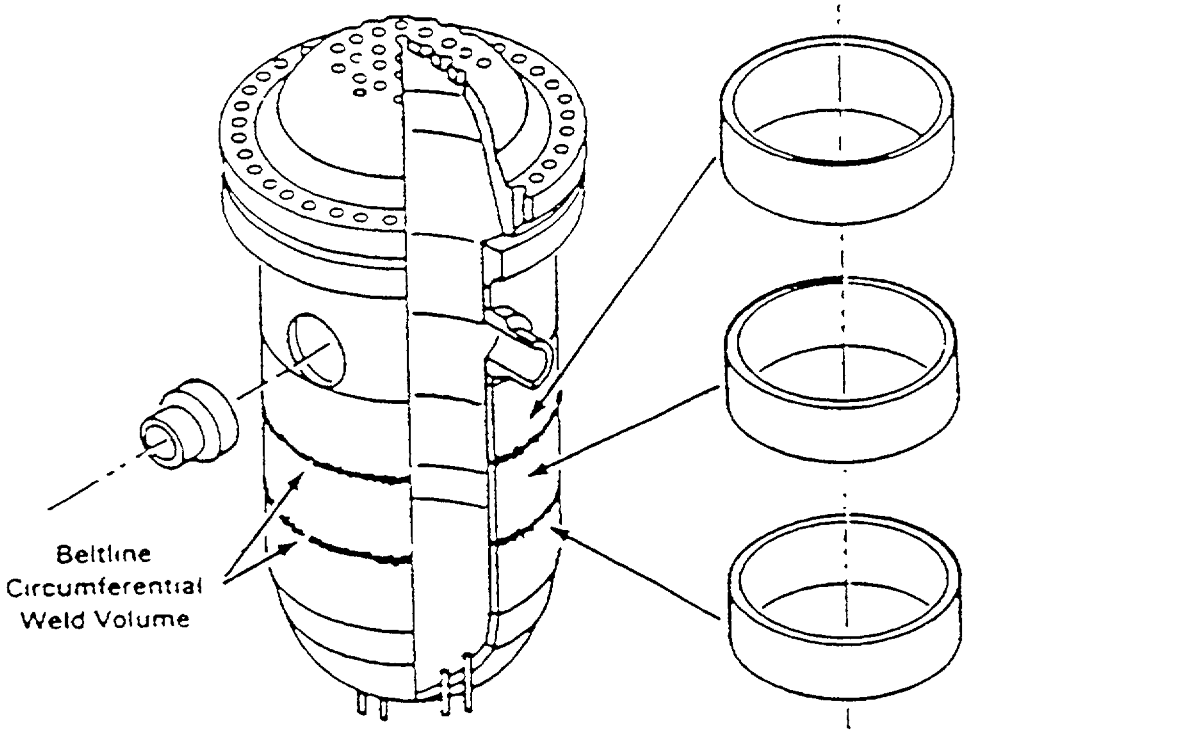
Fabrication practices

Fabrication of RPVs has also been an evolving technology, and later vessels were fabricated using knowledge gained from the surveillance programmes and more modern methods such as the use of large forgings to reduce the number of welds in the beltline [10, 12].

Most RPVs in the USA were fabricated by either Combustion Engineering, Chicago Bridge and Iron, or Babcock and Wilcox. Westinghouse did not fabricate vessels but had them fabricated at another shop. Some vessels were fabricated in Europe by Rotterdam Drydock Company and by Creusot-Loire. In some cases, vessels were constructed by more than one fabricator because of scheduling problems in the shops.



(a) Rolled and Welded Beltline Shell



(b) Welded-Ring-Forging Beltline Shell

FIG. 9. Fabrication configuration of PWR beltline shells

Large vessels are fabricated by two methods. In the first method, rolled and welded plates are used to form separate steel courses. Such a vessel has both longitudinal and circumferential weld seams (Fig. 9a). In some older vessels (before 1972), the longitudinal welds are of particular concern with regard to vessel integrity because they contain high levels of copper and phosphorous. In the second method, large ring forgings are used (Fig. 9b). This method improves component reliability because of the lack of longitudinal welds. Weld seams are located to avoid intersection with nozzle penetration weldments. Weldments within the beltline region were minimized once research showed that weld metal could be more sensitive to neutron radiation than base material. In general, parts of the longitudinal shell course welds are within the beltline region when the RPV is fabricated using plate material. At least one circumferential weld is near, or marginally within, the beltline region when the RPVs are fabricated from either plates or ring forgings. Recently, NSSS vendors are designing the RPV such that the beltline region does not contain any weldments. This is accomplished by utilizing very large ring forgings to fabricate the shell course.

Western RPV heads may be fabricated by welding a central dished plate to multiple toroidal plates, sometimes called "orange peel" sections, forming a hemisphere. The lower head is welded to the lower shell course while the top head is joined to the shell course by a flanged and bolted joint. However, the modern French and German RPVs do not have welds in the heads except for the circumferential weld which connects the head to the flange (top) or shell (bottom).

The interior surfaces of the steel vessel, closure head and flange area are typically clad with stainless steel, usually Type 308 or 309. Cladding was used to prevent general corrosion by borated coolant and to minimize the buildup of corrosion products in the reactor coolant system. The cladding was applied in one or two layers by multiple-wire, single-wire, strip-cladding, or resistance welding processes. Some vessels have areas of Alloy 82 or 182 weld cladding where Alloy 600 components were welded to the vessel.

During the fabrication of some RPVs it was discovered that small cracks were present in the base metal beneath the cladding of the steel. The first incident of underclad cracking was discovered in the early 1970s in Europe and later in the USA. This cracking was defined as "reheat cracking" because the cracks appeared after the final stress relief heat treatment of the RPVs. Reheat cracking was limited to RPVs fabricated from A508 Class 2 forging steel or the equivalent European grades. Reheat cracking only occurred when the cladding was applied utilizing a high heat input welding procedure. During the cladding process, grain coarsing occurred due to the high heat input of the welding procedure, thus weakening the underclad grain boundaries. Then the subsequent post-weld stress relief heat treatment at elevated temperature resulted in decohension of the grain boundaries, e.g., small cracking occurred. Underclad reheat cracks are approximately 2 to 3 mm in depth and can be detected during the preservice NDE by using straight beam transducers. However, it is virtually impossible to size these cracks with NDT. Reheat cracking is, for the most part, confined to the cylindrical portion of the RPV. The beltline region can contain many millions of small reheat cracks.

The second incident of underclad cracking occurred in the late 1970s in Europe followed by discovery of cracks in the USA. The second incident of underclad cracking was identified as "cold cracking". Cold cracking only occurred during the cladding process of the RPV when the second layer of cladding was applied without preheat. Cold cracking was, for the most part, limited to the highly constrained nozzle regions in the RPV. The mechanism for cold cracking was hydrogen diffusion into the base metal during the application of the second layer of cladding The cracking occurred following cooldown of the component at locations where there was hydrogen and a high strain due to the RPV nozzle configuration The size of the cold crack beneath the cladding is of the order of 6 to 8 mm and these cracks are readily discovered during NDE Unlike reheat cracking, the cracks that occurred due to cold cracking were removed by grinding prior to the vessel going into service All RPV steels are susceptible to cold cracking if the cladding is applied without preheat in regions of high constraint It is unlikely that cold cracking will occur at the beltline region of the RPV

The USNRC reviewed the issue of reheat cracking and concluded that it was not a safety issue [13] However, the USNRC also prohibited the use in the USA of high heat input welding procedures for cladding of RPVs To date there has not been any growth of the reheat cracks detected during the inservice inspections (ISIs) Cold cracking is not considered to be a significant issue because, for the most part, the cold cracks were removed prior to plant startup Also, any cold cracks that were inadvertently missed prior to startup would have been readily detected during the ISIs

Whitman et al [14], Griesbach and Server [15], Griesbach [10] describe fabrication methods in detail, the Electric Power Research Institute (EPRI) [16] gives additional references Kanninen and Chell [17] discussed the effect of the cladding on vessel integrity Radiation embrittlement of beltline materials and the computer database containing data on beltline materials used in US reactors are covered in Ref [12]

Welding

The welding processes used were mostly submerged-arc and shielded-metal-arc Before the early 1970s, copper-coated weld wire was used to improve the electrical contact in the welding process and to reduce corrosion during storage of the weld wire hence the generation of hydrogen When it was discovered that copper and phosphorus increased the weld s sensitivity to radiation embrittlement, RPV fabricators imposed strict limits on the per centage of copper and phosphorus in the welds as well as in plates [10, 14. 15] The use of copper coated weld wire was eliminated due to the strict limits on the per centage of copper in the weld The weld wire or stick electrodes were kept in storage in plastic bags and/or low temperature furnaces to eliminate the formation of moisture on the weld wire and electrodes

For the circumferential welds, many beads of weld material and consequently a large volume of weld wire are needed This becomes important when determining the properties of each individual weld in the beltline for sensitivity to neutron irradiation For example, the chemistry of the weld (copper and nickel content) may vary through the thickness and around the circumference because of variations in the weld wire used in fabrication Each weld in the vessel can be traced by the unique weld wire and flux lot combination used [12]

The sensitivity of welds to radiation can be inferred from the chemical composition The degree of embrittlement [shift in transition temperature or decrease in upper shelf energy (USE)] is determined as a function of the chemical composition and the level of neutron exposure Copper, nickel and possible phosphorus content in the weld are the most important elements from the standpoint of radiation damage The embrittlement of high copper and high nickel welds plays a key role in the assessment of the significance of pressurized thermal shock (PTS) [12]

TABLE VI. CHEMICAL COMPOSITION OF WWER REACTOR PRESSURE VESSEL MATERIALS (weight %)

MATERIAL	C	Mn	Si	P	S	Cr	Ni	Mo	V
15Kh2MFA	0.13 0.18	0.30 0.60	0.17 0.37	max 0.025	max 0.025	2.50 3.00	max 0.40	0.60 0.80	0.25 0.35
18Kh2MFA	0.15 0.21	0.30 0.60	0.17 0.37	max 0.025	max 0.025	2.50 3.00	max 0.40	0.60 0.80	0.25 0.35
25Kh3MFA	0.22 0.25	0.30 0.60	0.17 0.37	max 0.025	max 0.025	2.803.30	max 0.40	0.60 0.80	0.25 0.35
15Kh2NMFA	0.13 0.18	0.30 0.60	0.17 0.37	max 0.020	max 0.020	1.80 2.30	1.00 1.50	0.50 0.70	max 0.10
Sv-10KhMFT + AN-42	0.04 0.12	0.60 1.30	0.20 0.60	max. 0.042	max 0.035	1.20 1.80	max 0.30	0.35 0.70	0.10 0.35
Sv-10KhMFT + AN-42M	0.04 0.12	0.60 1.30	0.20 0.60	max 0.012	max 0.015	1.20 1.80	max 0.30	0.35 0.70	0.10 0.35
Sv-12Kh2N2MA + FC-16	0.05 0.12	0.50 1.00	0.15 0.45	max 0.025	max 0.020	1.402.10	1.20 1.90	0.45 0.75	–
Sv-12Kh2N2MA + FC-16A	0.05 0.12	0.50 1.00	0.15 0.45	max 0.012	max 0.015	1.40 2.10	1.20 1.90	0.45 0.75	–
Sv-13Kh2MFT + OF-6	0.11 0.16	0.40 0.70	0.17 0.35	max 0.030	max 0.030	1.40 2.50	–	0.40 0.80	0.17 0.37

TABLE VII. ALLOWABLE IMPURITY CONTENT IN THE WWER BELTLINE MATERIALS (max. weight %)

MATERIAL	P	S	Cu	As	Sb	Sn	P+Sb+Sn	Co
15Kh2MFAA	0.012	0.015	0.08	0.010	0.005	0.005	0.015	0.020
15Kh2NMFAA	0.010	0.012	0.08	0.010	0.005	0.005	0.015	0.020

TABLE VIII. GUARANTEED MECHANICAL PROPERTIES OF WWER VESSEL MATERIALS[1]

MATERIAL	20°C				350°C				T_{k0}
	$R_{p0.2}$ [MPa]	R_m [MPa]	A_5 [%]	Z [%]	$R_{p0.2}$ [MPa]	R_m [MPa]	A_5 [%]	Z [%]	[°C]
15Kh2MFA – base metal	431	519	14	50	392	490	14	50	0
– A/S weld metal	392	539	14	50	373	490	12	45	20
25Kh3MFA	628	736	12	50	590	838	12	45	
15Kh2NMFA – base metal	490	608	15	55	441	539	14	50	–10
15Kh2NMFAA – base metal	490	608	15	55	441	539	14	50	–25
– A/S weld metal	422	539	15	55	392	510	14	50	0

[1] $R_{p0.2}$ is the 0.2 per cent offset yield strength, R_m is the ultimate tensile strength, Z is the per cent reduction in area at failure and T_{k0} is the initial ductile-brittle transition temperature.

2.2.2. WWER pressure vessels

The WWER pressure vessel materials are listed in Table IIb. The chemical compositions of the various WWER materials are listed in Table VI, the allowable impurities in the beltline region are listed in Table VII and the guaranteed mechanical properties are listed in Table VIII. As indicated by the information in these tables, the WWER pressure vessel materials are basically different from the western RPV materials. The Type 15Kh2MFA(A) material used for the WWER-440 pressure vessels contains 0.25 to 0.35 weight per cent vanadium and very little nickel (maximum of 0.40 weight per cent). The Type 15Kh2NMFA(A) material used for the WWER-1000 pressure vessels contains 1.0 to 1.5 weight per cent nickel and almost no vanadium. Material with vanadium alloying was first used in the Soviet naval RPVs because the vanadium carbides make the material relatively resistant to thermal ageing, fine grained (tempered banite) and strong. However, the Type 15Kh2MFA(A) material is harder to weld than nickel steels and requires very high preheating to avoid hot cracking. This became more of a problem for the large WWER-1000 pressure vessels and a material with nickel rather than vanadium alloying was chosen. The influence of vanadium on the susceptibility of those materials to radiation embrittlement was shown to be negligible.

Not all the WWER pressure vessels were covered by austenitic stainless steel cladding on their whole inner surface: only approximately half of the WWER-440/V-230 pressure vessels were clad. However, all of the WWER-440/V-213 and WWER-1000 pressure vessels were covered on the whole inner surface. The cladding was made by automatic strip welding under flux with two layers — the first layer is made of a Type 25 chromium/13 nickel unstabilized austenitic material (Sv 07Kh25N13), and the second layer is at least three beads made of Type 18 chromium/10 nickel stabilized austenitic stainless steel (Sv 08Kh18N10G2B) to achieve a required total thickness of cladding equal to ~8 mm. Therefore, all the austenitic steels which are in contact with water coolant are stabilized. The stabilized austenitic stainless steels contain an alloying element (usually titanium) which forms stable gain boundary carbides. This prevents chromium depletion along the grain boundaries and makes the material immune to stress corrosion cracking. Unstabilized material was used for the first layer because the thermal expansion coefficient of that material is closer to the thermal expansion coefficient of the low-alloy pressure vessel material.

The WWER vessel head contains penetrations with nozzles. The nozzles are welded to the vessel head from inside (buttering) and are protected by stainless steel sleeving (0Kh18N10T).

The WWER quality control and QA procedures are applied during manufacture, assembly and installation of the reactor in accordance with applicable standards. The required quality is assured by:

- design by analysis,
- quality control of base and weld materials used,
- quality control during manufacture,
- acceptance testing prior to installation at the site.

Testing is performed using ultrasonic, radiographic, dye-penetrant and magnetic particle methods and includes hydrotests, if applicable. RPVs made in the Czech Republic were also monitored by acoustic emission during the pressure hydrotests at the manufacturing site (ŠKODA), in Plzeň.

3. DESIGN BASIS: CODES, REGULATIONS AND GUIDES FOR REACTOR PRESSURE VESSELS

The load restrictions on as-fabricated RPVs in various national standards and codes are generally based on Section III of the ASME Boiler and Pressure Vessel Code [18]. The objective of designing and performing a stress analysis under the rules of Section III to the ASME Boiler and Pressure Vessel Code is to afford protection of life and property against ductile and brittle RPV failure. The ASME Section III requirements are discussed in the next section. Some important differences exist in the RPV design requirements of certain other countries (e.g. Germany, France and Russia) and these differences are discussed in Sections 3.3, 3.4 and 3.5.

3.1. (ASME SECTION III) DESIGN BASIS

The reactor vessel has been designated as Safety Class 1, which requires more detailed analyses than Class 2 or 3 components. The rules for Class 1 vessel design are contained in Article NB-3000 [18], which is divided into three sub-articles:

(a) NB-3100, General Design Rules
(b) NB-3200, Design by Analysis
(c) NB-3300, Vessel Design

Sub-article NB-3100 deals with loading conditions specified by the owner (or his agent) in the form of an equipment specification. The specification identifies the design conditions and operating conditions (normal conditions, upset conditions, emergency conditions, faulted conditions and testing conditions).

Sub-article NB-3200 deals with the stresses and stress limits which must be considered for the analysis of the component. The methods of analysis and stress limits depend upon the category of loading conditions, i.e., the requirement for normal conditions are considerably more stringent than those for faulted conditions.

Sub-article NB-3300 gives special requirements that have to be met by Class 1 vessels. This article gives tentative thickness requirements for shells, reinforcement requirements for nozzles and recommendations for welding nozzles, for example.

3.1.1. Transient specification

It is impossible to determine accurately the stresses in a component without a correct description of the loads applied to that component. The loads themselves are divided into two broad categories: static and dynamic, the dynamic loads arising primarily from seismic conditions. The distinction between static and dynamic loads is based primarily on the comparison of the time span of the load variation to the response time of the structure.

The operating conditions themselves are divided into five categories depending on the severity of the transient and the number of occurrences:

(a) Normal conditions
(b) Upset conditions
(c) Emergency conditions
(d) Faulted conditions
(e) Testing conditions

Normal conditions are those which exist during normal running of the plant. Upset conditions are deviations from the normal conditions but are anticipated to occur often enough that provisions for them must be made in the analysis. These transients are those that do not result in forced outage, or if forced outage occurs, the restoration of power does not require mechanical repair. Emergency conditions are deviations from normal which require shutdown and may require repair and must be considered in order to assure no gross loss of structural integrity. Faulted conditions are deviations from normal, are extremely low probability, but may result in loss of integrity and operability of the system. Testing conditions are pressure overload tests, or other tests on the primary system.

For a PWR, the definitions of all operating transients are contained in the equipment specifications and are designed to represent the conditions under which a specific plant would operate. The interrelationship of the many groups within an organization needed to produce such a document is shown in Fig. 10. A listing of the transients, categories and number of occurrences contained in a typical specification is shown in Table IX.

3.1.2. Analysis of normal and upset conditions

Description of stress categories

The rules for design of Class 1 vessels make use of both realistic and accurate analysis techniques and failure criteria and therefore have relaxed overly restrictive safety factors used in the past. The calculated value of stress means little until it is associated with a location and distribution in the structure and with the type of loading which produced it. Different types of stress have different degrees of significance and must, therefore, be assigned different allowable values. For example, the average hoop stress through the thickness of the wall of a vessel due to internal pressure must be held to a lower value than the stress at the root of a notch in the wall. Likewise, a thermal stress can often be allowed to reach a higher value than

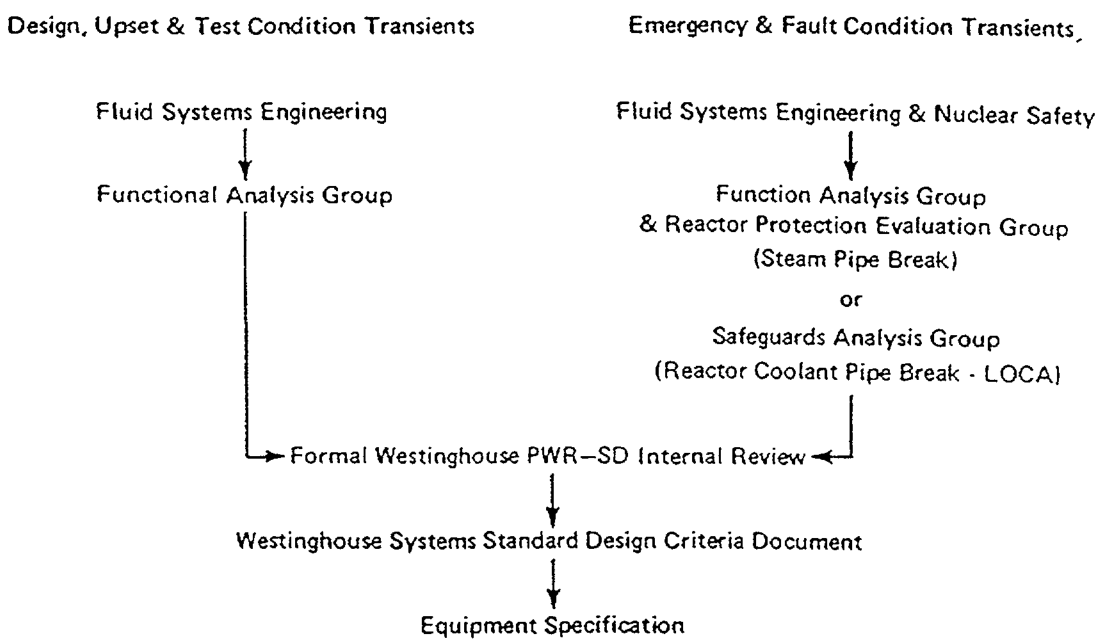


FIG. 10. Development of design transients.

TABLE IX TYPE OF TRANSIENT, NUMBER OF OCCURRENCES, AND TRANSIENT CLASSIFICATION IN A TYPICAL PWR DESIGN SPECIFICATION

Type of transient	Occurrences	Classification
Plant heatup at 55°C (100°F)/h	200	Normal
Plant cooldown at 55°C (100°F)/h	200	Normal
Plant loading at 5% of full power per minute	18 300	Normal
Plant unloading at 5% of full power per minute	18 300	Normal
Step load increase of 10% of full power	2000	Normal
Step load decrease of 10% of full power	2000	Normal
Large step load decrease (with steam dump)	200	Normal
Steady state fluctuations	Infinite	Normal
Loss of load (without immediate turbine or reactor trip)	80	Upset
Loss of power (blackout with natural circulation in reactor coolant system)	40	Upset
Loss of flow (partial loss of flow-one pump only)	80	Upset
Reactor trip from full power	400	Upset
Inadvertent auxiliary spray	10	Upset
Turbine roll test	10	Test
Primary side hydrostatic test before ttartup at 3105 psig (218 3 kg/cm^2)	5	Test
Primary side leak test at 174 7 kg/cm^2 (2485 psig)	50	Test
Steam pipe break	1	Faulted
Reactor coolant pipe break	1	Faulted

one which is produced by dead weight or pressure Therefore, a new set of design criteria were developed which shifted the emphasis away from the use of standard configurations and toward the detailed analyses of stresses The setting of allowable stress values required dividing stresses into categories and assigning different allowable values to different groups of categories The failure theory used here is the maximum shear stress theory which has been found appropriate to reactor vessel applications and has the advantage of simplicity Other criteria like the Mises criteria could be used as well The maximum shear stress calculated from the failure theory defines stress intensities

Different types of stress require different limits, and before establishing these limits, it was necessary to choose the stress categories to which limits should be applied. The categories and sub-categories chosen were as follows:

A. Primary stress

   1. General primary membrane stress
   2. Local primary membrane stress
   3. Primary bending stress.

B. Secondary stress

C. Peak stress.

The chief characteristics of these stresses may be described as follows:

(a) Primary stress is a stress developed by the imposed loading which is necessary to satisfy the laws of equilibrium between external and internal forces and moments. The basic characteristic of a primary stress is that it is not self-limiting. If a primary stress exceeds the yield strength of the material through the entire thickness, the prevention of failure is entirely dependent on the strain-hardening properties of the material.

(b) Secondary stress is a stress developed by the self-constraint of a structure. It must satisfy an internal strain pattern rather than equilibrium with an external load. The basic characteristic of a secondary stress is that it is self-limiting. These stresses are caused by thermal expansion or discontinuity conditions. The main concern with secondary stresses is that they may result in localized yielding or distortion.

(c) Peak stress is the highest stress in the region under consideration. The basic characteristic of a peak stress is that it causes no significant distortion and is objectionable mostly as a possible source of fatigue failure.

Stress intensity limits

The choice of the basic stress intensity limits for the stress categories described above was accomplished by the application of limit design theory tempered by some engineering judgment and some conservative simplifications. The principles of limit design which were used can be described briefly as follows.

The assumption is made of perfect plasticity with no strain-hardening. This means that an idealized stress-strain curve of the type shown in Fig. 11 is assumed. Allowable stresses, based on perfect plasticity and limit design theory, may be considered as a floor below which a vessel made of any sufficiently ductile material will be safe. The actual strain-hardening properties of specific materials will give them larger or smaller margins above this floor.

In a structure as simple as a straight bar in tension, a load producing yield stress, S_y results in "collapse". If the bar is loaded in bending, collapse does not occur until the load has been increased by a factor known as the "shape factor" of the cross section; at that time a "plastic hinge" is formed. The shape factor for a rectangular section in bending is 1.5. When the primary stress in a rectangular section consists of a combination of bending and axial tension, the value of the limit load depends on the ratio between the tensile and bending loads. Figure 12 shows the value of the maximum calculated stress at the outer fiber of a rectangular

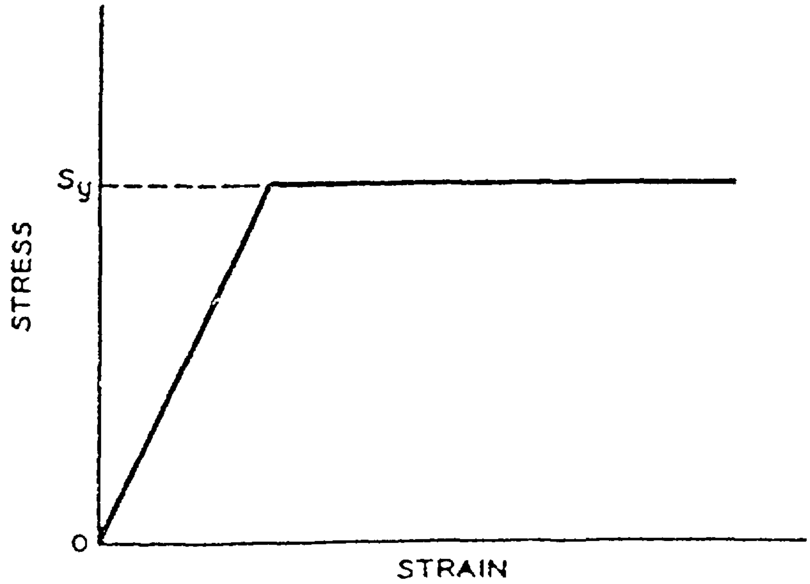


FIG 11 Idealized stress-strain relationship

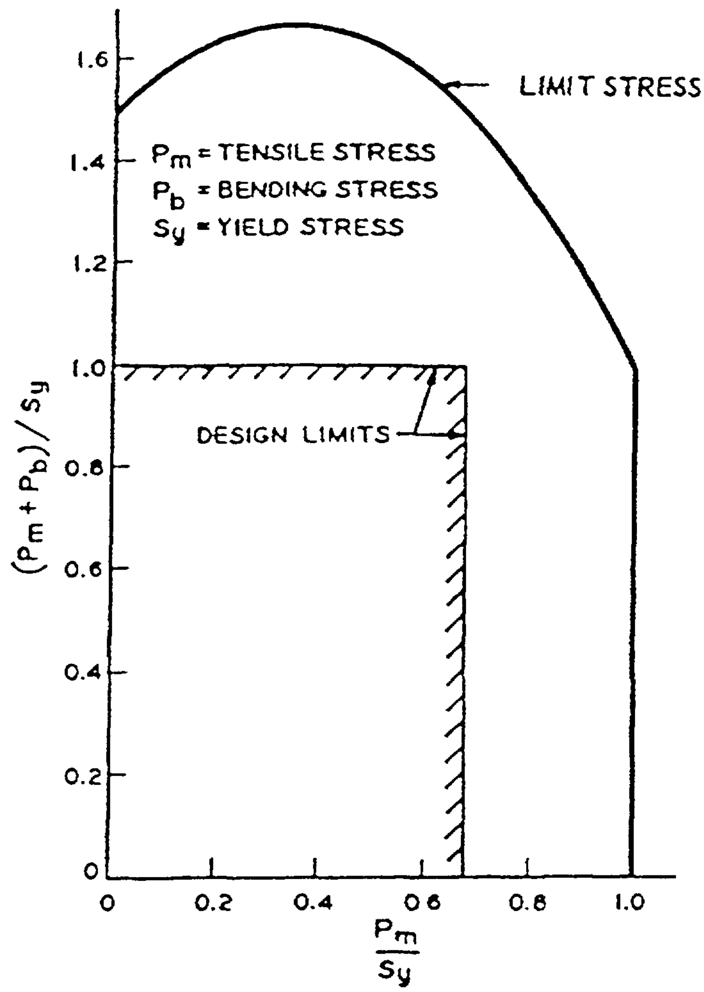


FIG 12 Limit stress for combined tension and bending (rectangular section)

section which would be required to produce a plastic hinge, plotted against the average tensile stress across the section, both values expressed as multiples of the yield stress, S_y. When the average tensile stress, P_m is zero, the failure stress for bending is 1.5 S_y. When the average tensile stress is S_y no additional bending stress, P_b, may be applied.

Figure 12 was used to choose allowable values, in terms of the yield stress, for general primary membrane stress, P_m and primary membrane-plus-bending stress, $P_m + P_b$. It may be seen that limiting P_m to (2/3) S_y and $P_m + P_b$ to S_y provides adequate safety. The safety factor is not constant for all combinations of tension and bending, but a design rule to provide a uniform safety factor would be needlessly complicated.

In the study of allowable secondary stresses, a calculated elastic stress range equal to twice the yield stress has a very special significance. It determines the borderline between loads which, when repetitively applied, allow the structure to "shake down" to elastic action and loads which produce plastic action each time they are applied; 2 S_y is the maximum value of calculated secondary elastic stress which will "shake down" to purely elastic action.

We have now shown how the allowable stresses for the first four stress categories listed in the previous section should be related to the yield strength of the RPV material. The last category, peak stress, is related only to fatigue and will be discussed later. With the exception of some of the special stress limits, the allowables in Codes are not expressed in terms of the yield strength, but rather as multiples of the tabulated value S_m which is the allowable for general primary membrane stress. In assigning allowable stress values to a variety of materials with widely varying ductilities and widely varying strain-hardening properties, the yield strength alone is not a sufficient criterion. In order to prevent unsafe designs in materials with low ductility and in materials with high yield stress-to-tensile strength ratios, the Code has always considered both the yield strength and the ultimate tensile strength in assigning allowable stresses. The stress intensity limits for the various categories given are such that the multiples of yield strength described above are never exceeded.

The allowable stress intensity for austenitic steels and some nonferrous materials, at temperatures above 38°C (100°F), may exceed (2/3) S_y and may reach 0.9 S_y at temperature.

TABLE X. ASME SECTION III STRESS LIMITS AND POTENTIAL FAILURE MODE FOR EACH TYPE OF STRESS CATEGORY

Stress intensity limit		Mode of failure
Primary stress		Burst and gross distortion
General membrane	S_m	
Local membrane + Primary bending	1.5 S_m	
Primary and secondary	3.0 S_m	Progressive distortion
Peak stresses	Design Fatigue Curve	Fatigue failure

TABLE XI BASIS FOR THE ALLOWABLE DESIGN STRESS-INTENSITY VALUES (S_m) IN SECTION III OF THE ASME CODE

- Ferritic steels

  Design stress intensity value (S_m) is lowest of
  - 1/3 of the specified minimum tensile strength at room temperature
  - 1/3 of the tensile strength at temperature
  - 2/3 of the specified minimum yield strength at room temperature
  - 2/3 of the yield strength at temperature

- Austenitic steels, nickel-chromium-iron and Ni-Ch-Fe alloys

  Design stress intensity value (S_m) is lowest of
  - 1/3 of the specified minimum tensile strength at room temperature
  - 1/3 of the tensile strength at temperature
  - 2/3 of the specified minimum yield strength at room temperature
  - 90% of the yield strength at temperature, but not to exceed 2/3 of the specified minimum yield strength at room temperature

- Bolting materials

  Design stress intensity value (S_m) based on lowest of
  - 1/3 of minimum specified yield strength at room temperature
  - 1/3 of the yield strength at temperature up to a temperature of 426 7°C (800°F)

Some explanation of the use of up to 0 9 S_y for these materials as a basis for S_m is needed in view of Fig 12 because this figure would imply that loads in excess of the limit load are permitted The explanation lies in the different nature of these materials' stress strain diagram These non-ferrous materials have no well-defined yield point but have strong strain-hardening capabilities so that their yield strength is effectively raised as they are highly loaded This means that some permanent deformation during the first loading cycle may occur, however, the basic structural integrity is comparable to that obtained with ferritic materials This is equivalent to choosing a somewhat different definition of the "design yield strength" for those materials which have no sharply defined yield point and which have strong strain-hardening characteristics Therefore, the S_m value in the code tables, regardless of material, can be thought of as being no less than 2/3 of the "design yield strength" for the material in evaluating the primary and secondary stresses

The basic stress limits for each type of stress category are/is shown in Table X The basis for the allowable design stress intensity values (S_m) is shown in Table XI for typical reactor vessel materials

Fatigue evaluation

The last stress category to be examined is that of peak stresses This category is only a concern in fatigue. The ASME Code gives specific rules for fatigue strength reduction factors and design curves for each type of material For the component design to be acceptable, the cumulative usage factor at the end of life must be less than unity Under some conditions outlined in the Code, a fatigue analysis is not necessary, however, conditions are then fairly restrictive

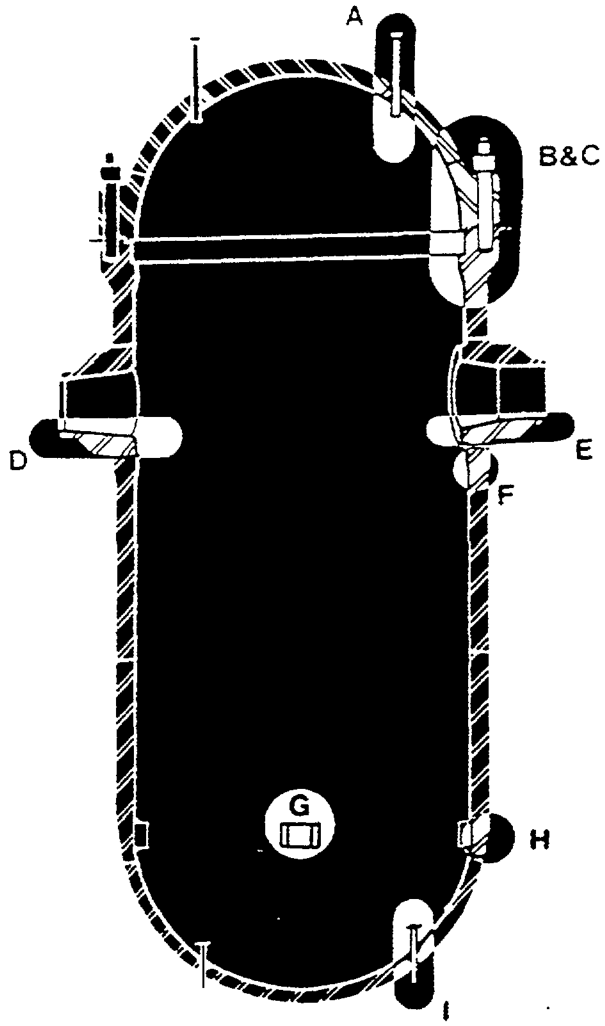


FIG. 13. Regions of a RPV to be analysed in order to determine compliance with the ASME Code.

Areas of the vessel analysed

The regions of the vessel which are examined in order to determine compliance with the ASME Code are shown in Figs 13–17. They are the areas which have potentially the highest stresses.

Stress analysis methods

Depending on the vendor, several different methods are used to determine the stresses in components. Two of the most popular are discontinuity analysis and finite element analysis as shown in Figs 18 and 19, respectively for the reactor vessel inlet nozzle.

Typical results of analysis for normal and upset conditions

For normal and upset conditions, Table XII shows the maximum calculated primary stress intensities for the general membrane category and the local membrane plus bending category. Note, the stresses are all below the allowables for both categories.

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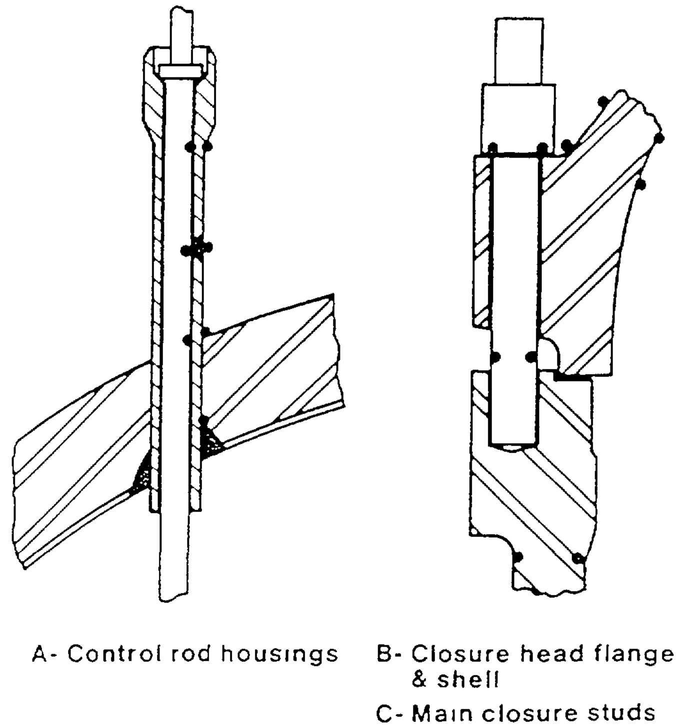


FIG 14 Typical control rod housing and closure head flange, shell and studs locations to be evaluated in an ASME stress analysis

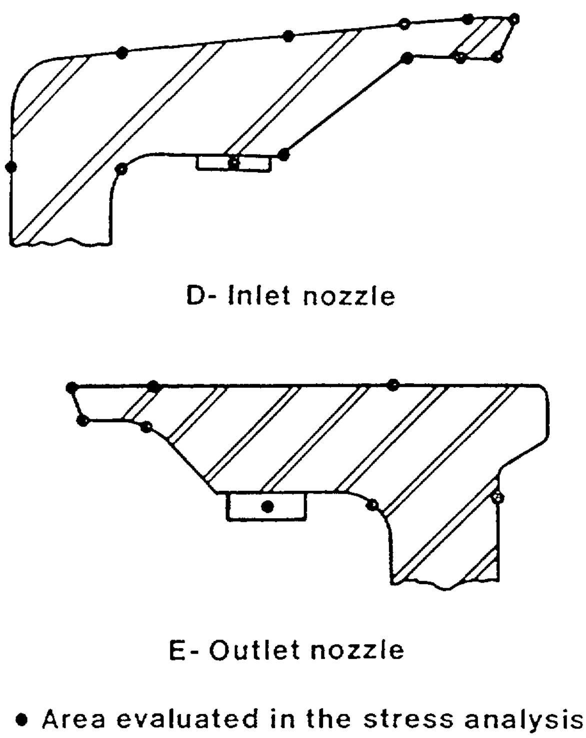


FIG 15 Typical inlet and outlet nozzle locations to be analysed in order to determine compliance with the ASME Code

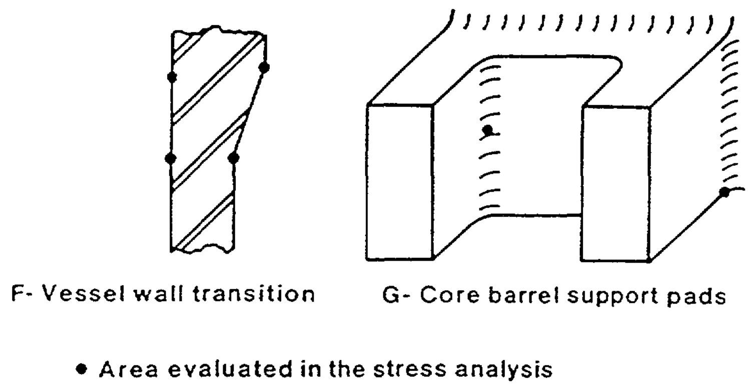


FIG. 16. A typical vessel wall transition and core barrel support pad locations to be analysed in order to determine compliance with the ASME Code.

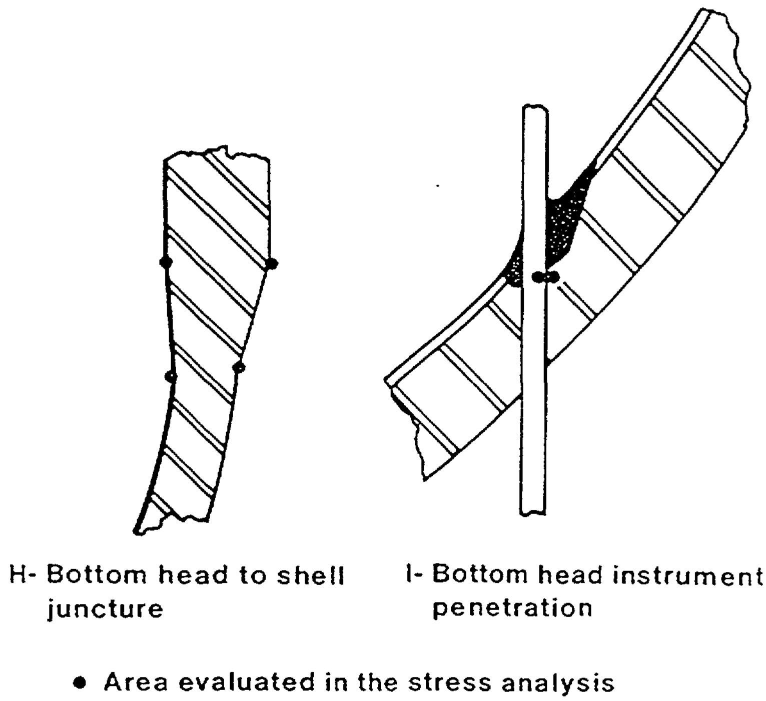


FIG. 17. Typical bottom head to shell juncture and bottom head instrument penetration locations to be analysed in order to determine compliance with the ASME Code.

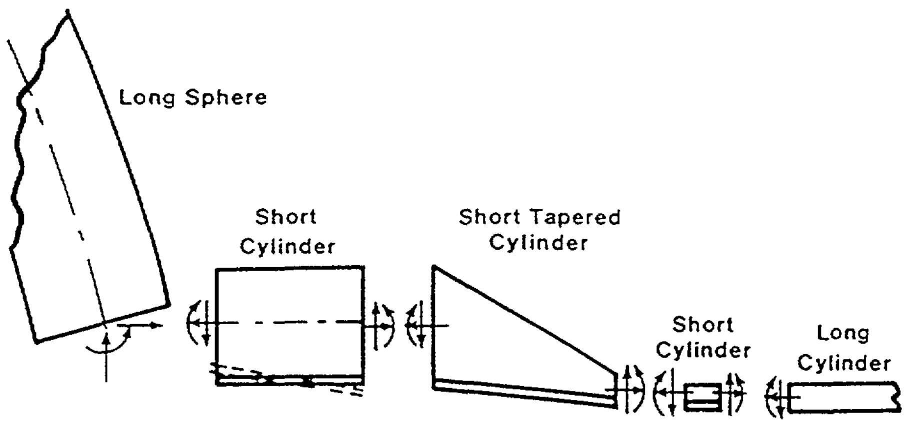


FIG. 18. Discontinuity analysis model of reactor vessel inlet nozzle.

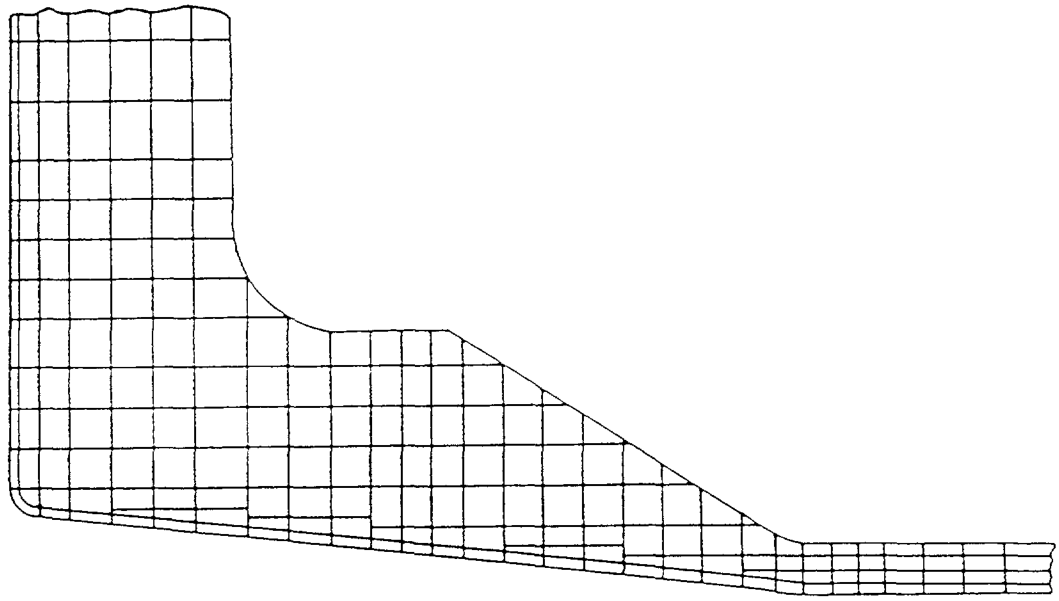


FIG 19 Finite element analysis model of reactor vessel inlet nozzle

TABLE XII MAXIMUM CALCULATED PRIMARY STRESS INTENSITIES VERSUS ASME SECTION III ALLOWABLE LIMITS

NORMAL AND UPSET CONDITIONS				
	General membrane		Local membrane and bending	
Location	Calculated	Allowable (S_m)	Calculated	Allowable (1 5 S_m)
CRDM housings	24.3	26 7	16 9	35 0
Closure head-flange region	14 7	26 7	39 5	40 0
Vessel-flange region	19 9	26 7	29 7	40 0
Closure studs	34.5	34 8	47 6[1]	73 6
Inlet nozzle	15.8	26 7	37 4	40 0
Outlet nozzle	16 4	26 7	38 1	40 0
Vessel wall transition	26.3	26 7	24 3	40 0
Core support pads[2]			28 9	35 0
Bottom head to shell juncture	26 3	26 7	22 5	40 0
Bottom instrumentation tubes region	26.5	26 7	15 2	35 0

[1] Maximum average bolt service stress
[2] Not pressure retaining part

TABLE XIII. MAXIMUM RANGE OF PRIMARY PLUS SECONDARY STRESS INTENSITIES COMPARED AGAINST THE ALLOWABLE LIMITS IN SECTION III OF THE ASME CODE

NORMAL AND UPSET CONDITIONS				
	Primary and secondary stress intensity		Usage factor	
Location	Calculated	Allowable ($3S_m$)	Calculated	Allowable
CRDM housings	54.6	69.9	0.09	1.0
Closure head-flange region	41.1	80.0	0.04	1.0
Vessel-flange region	58.2	80.0	0.02	1.0
Closure studs	91.5	110.4	0.57	1.0
Inlet nozzle	44.4	80.0	0.038	1.0
Outlet nozzle	48.0	80.0	0.06	1.0
Vessel wall transition	26.3	80.0	<0.01	1.0
Core support pads	46.2	69.9	0.37	1.0
Bottom head to shell juncture	27.1	80.0	0.01	1.0
Bottom instrumentation tubes region	53.2	69.9	0.13	1.0

TABLE XIV. ALLOWABLE STRESS INTENSITY LIMITS IN SECTION III OF THE ASME CODE FOR EMERGENCY CONDITIONS

Primary stresses		Allowable limits
General membrane	(P_m)	Greater of 1.2 S_m or S_y for elastic analysis
Local membrane + Primary bending	($P_L + P_b$)	Greater of 1.8 S_m or 1.5 S_y for elastic analysis
		0.8 C_L for limit analysis (C_L denotes collapse load)

No evaluation of secondary stresses (including thermal stresses) is required since they are self-relieving.

These conditions need not be considered in the component fatigue evaluation since limited to a total of 25 occurrences.

TABLE XV ALLOWABLE PRIMARY STRESS LIMITS FOR FAULTED CONDITIONS IN SECTION III OF THE ASME CODE

System or (subsystem) analysis	Components analysis	Stress limits for components		Components supports		
		P_m	P_l or $[P_m(\text{or}P_l)+P_b]$	P_m	P_l or $[P_m(\text{or}P_l)+P_b]$	Test
Elastic	Elastic	Smaller of $2.4S_m$&$0.70S_u$	Smaller of $3.6S_m$&$1.05S_u$ [4]	Larger of $1.5S_m$ or $1.2S_y$ [1]	Larger of $2.25S_m$ or $1.8S_y$ [4,2]	
Elastic	Plastic	Larger of $0.70\ S_u$ or $S_y + 1/3\ (S_u\ S_y)$ [5]	Larger of $0.70\ S_u$ or $S_y + 1/3\ (S_{ut}\ S_y)$ [5]	S_y [1] L_2 [3,5]	$1.5\ S_y$ [4,2]	$0.8\ L_1$
	Limit analysis	$0.9\ L_1$	[3]	$0.9\ L_1$ [3]		L_2 [5]

S_u = Ultimate stress from engineering stress-strain curve at temperature
S_{ut} = Ultimate stress from true stress-strain curve at temperature
S_m = Stress intensity from ASME Section III at temperature
L_1 = Test load

[1] But not to exceed $0.70\ S_u$
[2] But not to exceed $1.05\ S_u$
[3] L_1 and L_2 = Lower bound limit load with an assumed yield point equal to 2.3. S_m and S_y (but not to exceed $0.70\ S_u$), respectively
[4] These limits are based on a bending shape factor of 1.5. For simple bending cases with different shape factors, the limits will be changed proportionally
[5] When elastic system analysis is performed, the effect of component deformation on the dynamic system response should be checked.

TABLE XVI. GOVERNING MECHANICAL LOAD STRESS VERSUS ALLOWABLE FAULT CONDITION LIMITS

		Inlet nozzle	Outlet nozzle	Allowable limits	
Loading	Stress category	S.I. (KSI)	S.I. (KSI)	Value	Limit
Reactor vessel nozzle safe ends					
Normal + DBE	P_m	17.65	16.14	40.08	2.4 S_m
	P_L+P_b	26.06	26.49	60.12	3.6 S_m
Normal + DBA	P_m	24.88	16.33	40.08	2.4 S_m
	P_L+P_b	34.28	26.07	60.12	3.6 S_m
Nor+DBE+DBA	P_m	30.55	17.69	40.08	2.4 S_m
	P_L+P_b	46.45	34.23	60.12	3.6 S_m
Reactor vessel nozzle to shell juncture					
Normal + DBE	P_L+P_b	35.82	36.46	74.92	1.8 S_y
Normal + DBA	P_L+P_b	41.58	49.32	74.52	1.8 S_y
Nor+DBE+DBA	P_L+P_b	45.64	53.36	74.52	1.8 S_y
Reactor vessel support pads					
Normal + DBE	Horiz.	47.71	29.37	56.68	1.2 S_y
	Vert.	7.83	9.29	56.68	1.2 S_y
Normal + DBA	Horiz.	59.94	76.43	109.14	0.8 test
	Vert.	15.45	14.71	56.58	1.2 S_y
Nor+DBE+DBA	Horiz.	107.66	105.80	108.14	0.8 test
	Vert.	21.12	21.85	56.58	1.2 S_y

Table XIII shows the maximum range of primary plus secondary stress intensities compared against the allowable limits; also the table shows the calculated usage factors in fatigue. Note, the Code requirements are satisfied in all cases.

3.1.3. Analysis of emergency and faulted conditions

Description of stress categories and analysis methods

For these types of operating conditions, the rate of occurrence is significantly less than normal and upset conditions and the primary concern is to prevent burst and gross distortion.

For this reason limits are only placed upon the general membrane category and the local membrane plus primary bending category Also, because inelastic analysis is often required, the stress limits are considerably more detailed The system analysis used to determine the loads which act on the components is generally a dynamic analysis because of the nature of the events postulated (earthquakes/air crashes) This system analysis is generally elastic and the system design is modified by adding supports and stiffness to control structural resonance conditions If significant inelastic response occurs within the component the original elastic system analysis requires modification The stress intensity limits for emergency conditions are shown in Table XIV Depending upon the analysis method the applicable primary stress limits for faulted conditions are given in Table XV

Typical results for faulted conditions

For the purposes of illustration, only the critical locations around the nozzles thought to be critical will be considered The results of the analysis are shown in Table XVI In this table DBE is defined to be the Design Basis Earthquake and DBA is the Design Basis Accident

3.1.4. Analysis of test conditions

The major interest for this transient is to prevent burst or permanent distortion In the general primary membrane stress category, the stress intensity is limited to 0 9 of the tensile yield strength (σ_y) in the primary membrane plus primary bending stress category, the stress intensity is limited to 1 35 σ_y For the cold hydrotest transient the results of the analysis are shown in Table XVII

TABLE XVII MAXIMUM CALCULATED STRESS INTENSITIES DURING A COLD HYDRO TEST TRANSIENT COMPARED WITH THE ALLOWABLE LIMITS IN SECTION III OF THE ASME CODE

The reactor vessel hydro test initial pressure was 21 5 MPa (3125 psi)

Location	Calculated	Allowable (1 35σ_Y)
CRDM housing	23 8	47 25
Closure head - flange region	49 3	67 5
Vessel - flange region	37 2	67 5
Closure studs	85 0	130 0[(1)]
Inlet nozzle	45 6	67 5
Outlet nozzle	48 0	67 5
Vessel wall transition	31 9	67 5
Core support pads	38 8	47 25
Bottom head to shell juncture	34 1	67 5
Bottom instrumentation tubes	23 4	47 25

[(1)] Minimum bolt yield stress

3.1.5. Design and analysis against non-ductile failure (heatup and cooldown limit curves for normal operation)

At the recommendation of the Pressure Vessel Research Committee, the ASME Boiler and Pressure Vessel Code introduced criteria into Section III — Nuclear Power Plant Components — to provide assurance against brittle failure. The criteria required the component materials to satisfy certain fracture toughness requirements (NB-2330 of the Code). The criteria also introduced non-mandatory Appendix G, "Protection Against Non-Ductile Failure", into the ASME Code [19]. Appendix G of Section III presents a procedure for obtaining the allowable loading for ferritic pressure-retaining materials in Class 1 components. The procedure is based on the principles of linear elastic fracture mechanics (LEFM). Appendix G provides a reference critical stress intensity factor (K_I) curve as a function of temperature, a postulated flaw and a K_I expression.

The basic premise of LEFM is that unstable propagation of an existing flaw will occur when the value of K_I attains a critical value for the material designated as K_{IC}. K_{IC} is called the linear elastic fracture toughness of the material. In the case of ferritic materials, it has been found that the fracture toughness properties are dependent on temperature and on the loading rates imposed. Dynamic initiation fracture toughness obtained under fast or rapidly applied loading rates is designated K_{Id}. Further, in structural steels, a crack arrest fracture toughness is obtained under conditions where a propagating flaw is arrested within a test specimen. The crack arrest toughness is designated K_{Ia}. Appendix G to Section III presents a reference stress intensity factor [K_{IR}] as a function of temperature based on the lower bound of static K_{IC}, dynamic K_{Id} and crack arrest K_{Ia} fracture toughness values. The K_{IR} vs. temperature curve is shown in Fig. 20. No available data points for western-type ferritic RPV material yet tested for static, dynamic or arrest tests fall below the curve given. The value of K_{IR} represents a very conservative assumption as to the critical stress intensity vs. temperature properties of materials similar to those tested, as related to the measured nil-ductility temperature. The Code (NB-2331a) identifies a reference nil-ductility transition temperature (RT_{NDT}) to index the K_{IR} curve to the temperature scale. The reference temperature RT_{NDT} is defined (NB-2331) as the greater of the drop weight nil-ductility transition temperature or a temperature 33.3°C (60°F) less than the 68 J (50 ft-lb) [and 0.9 mm (35 mils) lateral expansion temperature] as determined from Charpy specimens oriented normal (NB-2322.2) to the rolling direction of the material (the T-L orientation). The requirements of Charpy tests at 33.3°C (60°F) above the nil-ductility temperature serve to sort out nontypical materials and provide assurance of adequate fracture toughness at "upper shelf" temperatures. In addition, the requirement of lateral expansion values provides some protection from variation in yield strength. Measurement of lateral expansion can also serve as an index of ductility.

G-2120 of Appendix G gives a postulated defect to be used in determining the allowable loading. As shown in Fig. 21, it consists of a sharp surface flaw, perpendicular to the direction of maximum stress, having a depth of 1/4 of the section thickness over most of the thickness range of interest. The assumed shape of the postulated flaw is semi-elliptic, with length six times its depth. In sizing the postulated flaw, it was assumed that (with the combination of examinations required by Section III and the volumetric examination required by ASME Section XI) there is a very low probability that defects larger than four times the allowable size as defined in Section III will escape detection.

G-2200 outlines the recommended procedure for protection against nonductile failure for normal and upset operating conditions. Included in G-2200 is G-2214 which defines methods to calculate linear elastic stress intensity factors, K_I. G-2215 provides the bases for determining allowable pressure at any temperature at the depth of the postulated defect during normal, upset and operating conditions The requirements to be satisfied and from which the allowable pressure for any assumed rate of temperature change can be determined are

$$2K_{IM} + K_{IT} < K_{IR} \tag{1}$$

where

K_{IM} is the stress intensity factor for primary stresses, and
K_{IT} is the stress intensity factor for secondary stress

This must be maintained throughout the life of the component at each temperature with K_{IM} from G-2214 1, K_{IT} from G-2214 2 and K_{IR} from G-2212 The recommended safety factor of 2 on K_{IM} adds to the conservatism of the assumptions Due to its secondary and self-relieving nature, no safety factor is given for K_{IT} G-2410 relaxes the conservatism by reducing the safety factor for K_{IM} to 1.5 during system hydrostatic testing

Heatup and cooldown limit curves (P–T limit curves) are calculated using Appendix G and the most limiting value of the reference nil-ductility transition temperature (RT_{NDT}) for a given RPV The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material fracture toughness properties and estimating the radiation-induced change in the reference nil-ductility transition temperature (ΔRT_{NDT}) RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 0 9 mm (35-mil) lateral expansion (normal to the major working direction) minus 33°C (60°F)

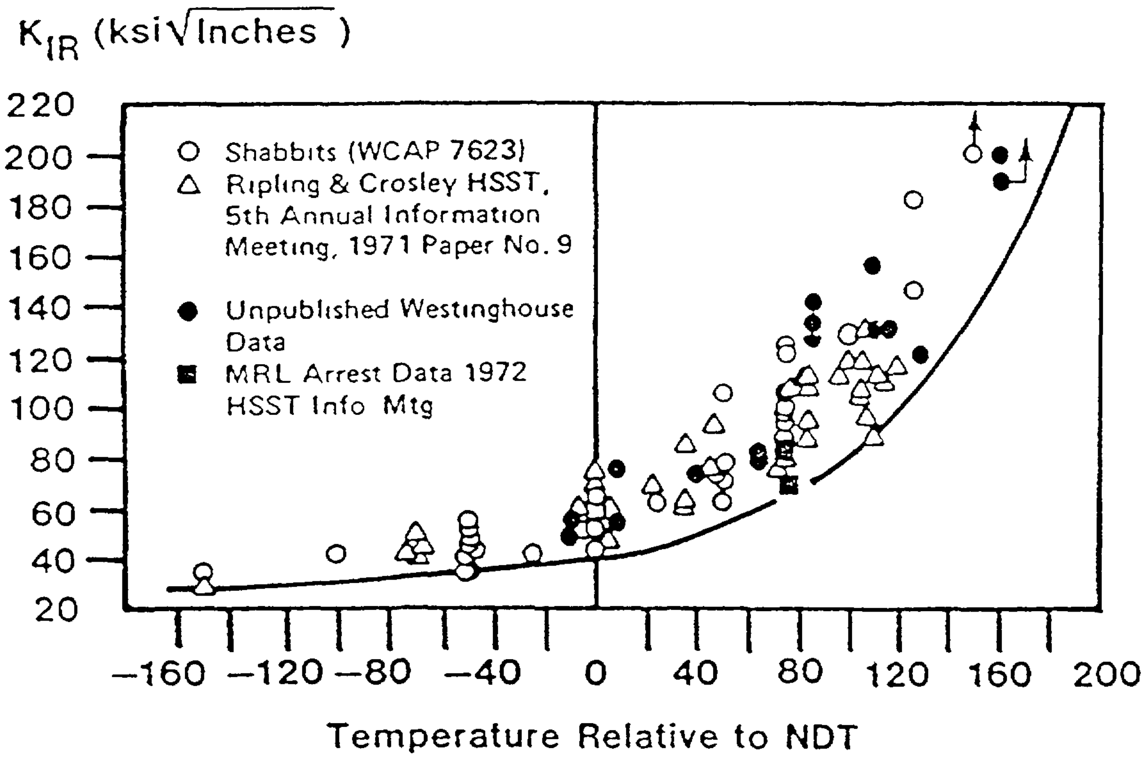


FIG 20 Derivation of curve of reference stress intensity factor (K_{IR})

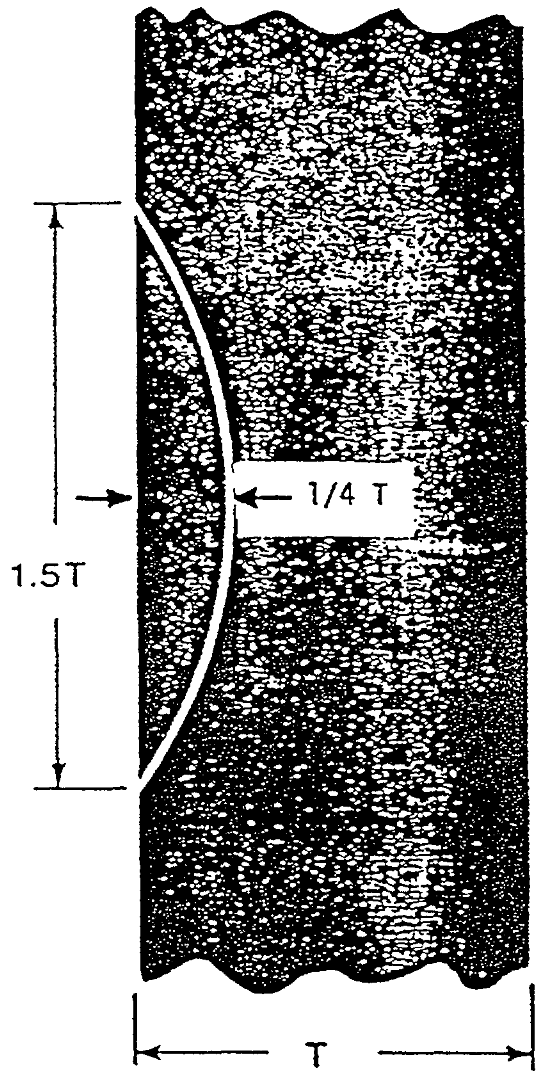


FIG 21. ASME Section III, Appendix G Reference Flaw.

The fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review Plan. Appendix G to the ASME Codes specifies that for calculating the allowable limit curves for various heatup and cooldown rates, the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the following equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145 (T-RT_{NDT} + 160)] \quad (2)$$

where

K_{IR} = reference stress intensity factor in British units (ksi · in$^{0.5}$) as a function of the metal temperature T (°F) and the metal reference nil-ductility temperature RT_{NDT}.

Therefore, the governing equation for the heatup-cooldown analysis is defined in Appendix G of Section III of the ASME Code [19] as follows

$$C\,K_{IM} + K_{IT} \leq K_{IR} \tag{3}$$

where

K_{IM} = stress intensity factor caused by membrane (pressure) stress

K_{IT} = stress intensity factor caused by the thermal gradients

K_{IR} = function of temperature relative to the RT_{NDT} of the material

C = 2 0 for Level A and Level B service limits

C = 1 5 for hydrostatic and leak test conditions during which the reactor core is not critical

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT} and the reference fracture toughness curve The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed From Equation (3) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside which increase with increasing cooldown rates Allowable P-T relations are generated for both steady-state and finite cooldown rate situations From these relations, composite limit curves are constructed for each cooldown rate of interest

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw

During cooldown, the 1/4 wall thickness location is at a higher temperature than the fluid adjacent to the vessel inside diameter This condition, of course, is not true for the steady-state situation It follows that at any given reactor coolant temperature, the temperature change developed during cooldown results in a higher value of K_{IR} at the 1/4 wall thickness location for finite cooldown rates than for steady-state operation Furthermore if conditions exist so that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value

The above procedures are needed because there is no direct control on temperature at the 1/4 wall thickness location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period

Also, the 1993 Amendment to 10 CFR 50 has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 67°C (120°F) during normal operation when the pressure exceeds 20% of the preservice hydrostatic test pressure.

Vendors, owners and regulatory bodies can perform or require an ASME Section III Appendix G analysis for normal, upset and test conditions for all RPVs. Stresses are obtained from the pertinent stress report and the methods of ASME Appendix G are applied to four locations in the reactor vessel: closure head to flange region, nozzle to shell course region, beltline region and the bottom closure head to shell course region. Neutron radiation effects are factored into the analysis, where applicable. The analysis demonstrates the existence of adequate margins for continued operation over the life time of the plant in the presence of a flaw one quarter the vessel wall thickness in depth.

3.2. REGULATORY REQUIREMENTS FOR RPV DESIGN IN THE USA

Part 50 of the US Code of Federal Regulations, Title 10 (10 CFR 50) [20] regulates the construction of NPPs. Section 10 CFR 50.55(a) defines the reactor vessel to be part of the reactor coolant boundary and requires that the vessel meet the requirements for Class 1 vessels contained in the ASME Boiler and Pressure Vessel Code Sections III [18] and XI [21].

The pressure vessels in the USA were designed and fabricated in accordance with the version of Section III of the ASME Boiler and Pressure Vessel Code applicable at the time of fabrication, except for RPVs built before Section III existed (prior to 1963). Earlier plants, such as Yankee Rowe, Connecticut Yankee and a few others, were constructed to predecessors of Section III, such as Section I (power boilers) and Section VIII, Division 1 (unfired pressure vessels) [22, 23]. The allowable stress levels for pressure boundary materials were about 25% lower for Section VIII than those permitted by Section III for similar materials, which resulted in thicker walls and larger nozzle corner radii for Section VIII vessels. However, Section III requires a more limiting NDE of the welds, so the probability of having manufacturing defects in a Section III vessel is smaller than for a Section VIII vessel.

The US Code of Federal Regulations, Title 10, contains other regulations which are applicable to the vessel, such as 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation", 10 CFR 50.61, "Fracture toughness requirements for protection against PTS events", and Appendices A [24], G [25] and H [26] of 10 CFR 50. The quality, fracture prevention and inspection of the reactor coolant pressure boundary are addressed in General Design Criteria 30, 31 and 32 of Appendix A. Appendix G specifies fracture toughness requirements for ferritic RPV materials based on ASME Code, Section III. Requirements for the reactor vessel material surveillance programme are based on the ASTM requirements and are specified in Appendix H of Federal Regulation 10 CFR 50.

The following is a summary of the requirements of 10 CFR 50.60, 10 CFR 50.61, 10 CFR 50.66 and Appendices G and H to 10 CFR 50.

Under 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation", all nuclear power reactors must meet the fracture toughness and material surveillance programme requirements for the reactor coolant pressure boundary set forth in Appendices G and H to 10 CFR Part 50. The fracture toughness of the reactor coolant pressure boundary required by 10 CFR 50.60 is necessary to provide adequate margins of safety during any condition of normal plant operation. The required material surveillance programme monitors changes in the fracture toughness properties of ferritic materials in the beltline resulting from exposure to neutron irradiation and the thermal environment. Under the programme, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the vessel.

Under CFR 50.61, "Fracture toughness requirements for protection against PTS events", the plant operators are required to assess the projected values of reference temperature. If the projected reference temperature exceeds the screening criteria in 10 CFR 50.61, the plant operator must submit an analysis and schedule for a flux reduction programme that is reasonably practicable and avoids exceeding the screening criteria. If no such flux reduction programme will avoid exceeding the screening criteria. the plant operator must submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel if continued operation beyond the screening criteria is allowed. 10 CFR 50.61 has recently been modified to explicitly cite thermal annealing as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}. PTS is discussed in more detail in Section 3.2.1 below.

Under 10 CFR 50.66, "Requirements for thermal annealing of the RPV", the nuclear plant operators in the USA are provided a consistent set of requirements for the use of thermal annealing to mitigate the effects of neutron irradiation. The thermal annealing rule impacts both 10 CFR 50.61 [pressure thermal shock (PTS) rule] and Appendix G of 10 CFR 50.

Appendix G to 10 CFR Part 50, "Fracture toughness requirements", requires that the beltline materials have Charpy upper shelf energies of no less than 68 J (50 ft-lb) throughout the life of the vessel. Otherwise, licensees must show equivalent margins of safety in accordance with Paragraph IV.A.1 or perform actions in accordance with Paragraph V.C of the Appendix.

Paragraph V.A of Appendix G to 10 CFR Part 50 requires a prediction of the effects of neutron irradiation on the reactor vessel materials. The extent of the neutron embrittlement depends on the material properties, thermal environment and results of the material surveillance programme. In Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations", the USNRC stated that it will use the guidance in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," in estimating the embrittlement of the materials in the vessel beltline. All the nuclear plant operators in the USA have responded to Generic Letter 88-11 and committed to use the methodology in Regulatory Guide 1.99, Revision 2 for predicting the effects of neutron irradiation. This methodology is also the basis in 10 CFR 50.61 for projecting the reference temperature.

Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Programme Requirements", requires the surveillance programme to meet the ASTM Standard E 185. "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels", and specifies the applicable edition of ASTM E 185. Nuclear plant operators in the USA, especially those with reactor vessels purchased before ASTM issued the 1973 edition of ASTM E 185, may have surveillance programmes that do not meet the requirements of Appendix H to 10 CFR Part 50. They can use these alternative surveillance programmes if they have been granted an exemption. The plant operators must monitor the test results from the material surveillance programmes. According to Paragraph III.C of Appendix H to 10 CFR Part 50, the results may indicate that a plant Technical Specifications change is required, either in the pressure–temperature (P–T) limits or in the operating procedures required to meet the limits.

3.2.1. Pressurized thermal shock

Pressurized thermal shock (PTS) transients can be described as those transients, such as a small loss-of-coolant accident, that result in a strong decrease in the coolant temperature due to the activation of the safety injection system, followed by repressurization of the system. In the USA, the issue of PTS is covered by 10 CFR 50, part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events". The USNRC has also published a PTS screening criteria which addresses longitudinal and circumferential flaws in RPVs. For longitudinal flaws, the RT_{NDT} cannot exceed 121°C (250°F) and for circumferential flaws, the RT_{NDT} cannot exceed 149°C (300°F). If the PTS screening criteria is exceeded, a safety evaluation in accordance with Regulatory Guide 1.154, "Format and Content of Plant Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors", is required. Regulatory Guide 1.154 requires the following analyses:

(1) A probabilistic risk assessment to determine the probability of occurrence for the most severe PTS-type transient that can be postulated;

(2) Thermal hydraulic analyses to establish the pressure-temperature-time histories for the various transients;

(3) Probabilistic fracture mechanics analyses to determine the conditional probability of vessel failure for each postulated PTS transient; and

(4) A summation of the event frequencies multiplied by the conditional probabilities of vessel failure. This must be less than a throughwall crack penetration mean frequency of 5×10^{-6} per reactor year.

A Regulatory Guide 1.154 PTS analysis is very costly (~US $2 million) and most likely will result in system changes to meet or satisfy regulatory requirements. However, as mentioned above, 10 CFR 50.61 has recently been modified to explicitly cite thermal annealing as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}. More recently, EPRI, Westinghouse and Sartrex have initiated a programme to develop an alternative approach that would simplify the probabilistic fracture mechanics analysis procedure and be economically efficient to implement, without requiring the resolution of a substantial number of technical issues. The fundamental concept of the approach is that there is a direct relationship between the probability of crack initiation and a critical crack size (A_C) computed using deterministic fracture mechanics methods.

3.3. DESIGN BASIS IN GERMANY

The reactor vessel designs in Germany follow the German KTA standards for light water reactors, published by the NUSS Commission. The KTA requirements are very similar to those in the ASME Code, regarding the definition of stress intensities and allowable stresses. However, considerable differences exist in the design requirements for USE and mid-thickness tensile and Charpy values, as well as for in-service inspections. Also, the German KTA has a limit on the allowable fluence whereas the ASME Code and the Codes in a number of other countries do not.

3.3.1. Non-ductile failure

To provide assurance against brittle failure, the KTA Standards require:

- an analysis of the brittle fracture transition temperature according to the Pellini/Porse methods and,
- a LEFM analysis (which is in accordance with Appendix G of Section III of the ASME Code).

1. The brittle fracture transition temperature must be determined and shown to be well below the operating temperature range. However, the brittle fracture transition temperature concept is applied only to the core region, since that is where the maximum fast neutron fluence and the maximum primary stress occur.

2. The allowance for detected flaw indications during ISIs is based on the principles of LEFM which are in accordance with Appendix A of Section XI of the ASME Code.

   The acceptability of the observed flaws are met for all Service Limits if a safety factor of, at least, K_{IC}/K_I equal to 1.5 is shown. For locations other than the beltline region, a safety factor of 2 for the calculated membrane stress intensity factor K_I is, in contrast to ASME, not necessary for the level A and B Service Limits, also a surface flaw with a depth of 1/4 of the section thickness is not required if it can be justified.

   For level C and level D Service Limits, assurance against brittle failure must be provided for the beltline region. KTA specifies that the critical flaw size which is still allowable must be twice as large as the flaw size which can reliably be detected by NDE. Crack instability is allowable if crack arrest can be proven within 3/4 of the section thickness.

3.3.2. Ductile failure and plastic collapse

This part of the design of the German RPVs follows the requirements of KTA 3201.2. In the main subjects, this part of KTA corresponds to the ASME Code, Section III, NB 3000. Load cases are given in a plant specification. The relation of the load cases to the service stress limits is done in the "design sheets" for the RPV for its whole or for parts of it. In addition, external loads, acting on nozzles or brackets, are also provided in the design sheets. The design stress intensity for low alloy ferritic RPV material is the smallest value of:

$$Sm = \left\{ \frac{R_{mRT}}{3}, \frac{R_{mT}}{2.7}, \frac{R_{p0.2T}}{1.5} \right\} \tag{4}$$

TABLE XVIII. GERMAN KTA STRESS LIMITS FOR THE VARIOUS SERVICE LEVELS

Service levels				Design limits	Service limits				
Stress category				(Level 0)	Level A	Level B	Level p [2]	Level C [4]	Level D
Primary plus secondary plus peak stresses	Primary plus secondary stresses	Primary stresses	P_m	S_m	-----	$1.1 \cdot S_m$	$0.9 \cdot R_{p0.2PT}$	$R_{p0.2T}$ [3]	$0.7 \cdot R_{mT}$
			P_l	$1.5\ S_m$	-----	$1.65 \cdot S_m$	$1.35 \cdot R_{p0.2PT}$	$1.5 \cdot R_{p0.2T}$ [3]	R_{mT}
			$P_m + P_b$ or $P_l - P_b$	$1.5\ S_m$	-----	$1.65 \cdot S_m$	$1.35 \cdot R_{p0.2PT}$	$1.5 \cdot R_{p0.2T}$ [3]	R_{mT}
		P_e		-----	$3 \cdot S_m$ [1]	$3 \cdot S_m$ [1, 5]	-----	-----	-----
		$P_m + P_b + P_e + Q$ or $P_l + P_b + P_e + Q$		-----	$3 \cdot S_m$ [1]	$3 \cdot S_m$ [1, 5]	-----	-----	-----
	$P_m + P_b + P_e + Q + F$ or $P_l + P_b + P_e + Q + F$			-----	$D \leq 1.0$; $2 \cdot S_a$	$D \leq 1$; [6] $2 \cdot S_a$	-----	-----	

(1) When the $3 \cdot S_m$ stress intensity limit is exceeded, an elastic-plastic analysis shall be performed taking the stress cycles into account. Provided the applicable requisites are fulfilled, this may take the form of simplified elastic-plastic analysis.

(2) If the total of stress cycles is greater than 10, the number of stress cycles in excess of 10 shall be included in the fatigue analysis as for levels A and B Service Limits.

(3) But not more than 90% of the value for level D Service Limits.

(4) If the total number of stress cycles is greater than 25, the number of stress cycles in excess of 25 shall be included in the fatigue analysis as for levels A and B Service Limits.

(5) These verifications are not mandatory in those cases in which stresses and strains of emergency and faulted service conditions are assigned to these Service Limits for reasons of operability or for any other reasons.

(6) Fatigue analysis is not mandatory in those cases in which stresses and strains of the emergency and faulted service conditions are assigned to these Service Limits for reasons of operability or for any other reasons and in which these service conditions are part of the group of 25 stress cycles for level C Service Limits for which fatigue analysis is not required.

where

R_{mRT} is the minimum specified tensile strength at room temperature
R_{mT} is the minimum specified tensile strength at the design temperature
$R_{p0\,2T}$ is the 0.2 per cent offset minimum specified yield strength at the design temperature.

In addition to the limitations on the loadings, the major RPV ferritic materials must initially have an USE of at least 100 J, measured with transverse Charpy V-notch specimens and the end-of-life USE must be at least 68J.

The stress limits of all service levels are given in Table XVIII. According to this table and the stress classifications, the number of calculations is fixed and corresponds to the requirements in the ASME Code.

Methods used to perform stress analyses are also given in KTA, especially:

- method of finite elements
- method of discontinuities.

Modelling of the RPV, or parts of it, allows the stress calculation to be performed everywhere in the component; but in general stresses are shown in sections or single points, covering the neighbourhood.

3.3.3. Heatup and cooldown limit curves for normal operation

In general, the same procedure as specified in the ASME Code and described in Section 3.1.5 above is used in Germany and defined as the "fracture mechanics approach" in KTA 3201.2. Alternatively, the KTA accepts the use of a modified Porse-diagram as the so-called "RT_{NDT} approach", according to which the stress limits are calculated as a function of the minimum RPV-wall temperature according to the Pellini/Porse method.

3.4. DESIGN BASIS IN FRANCE

3.4.1. Code rules

The oldest 3-loop plants in France were designed under ASME Section III, Appendix G. The newer 4-loop plants are being designed under RCC-M B 3200, Appendix ZG [27]. The RCC-M B 3200 rules are similar to the rules in ASME Section III (however, the fabrication, welding, examination and QA rules are different) [28, 29]. The allowable stress, Sm, is equal to the minimum of:

Rm/3, Su/3, 2Re/3, or 2Sy/3 for ferritic steels
Rm/3, Su/3, 2Re/3, or 0.9Sy for austenitic steels.

where Rm is the specified tensile strength at room temperature, Su is the minimum tensile strength at temperature, Re is the specified elastic limit at room temperature and Sy is the minimum yield limit at temperature. A value of 1.8 Sm is used for the Level C criteria rather than 2.25 Sm. Also, specific fatigue analysis requirements and specific methods for brittle and ductile fracture protection are included.

3.4.2. Brittle and ductile fracture assessments

Two methods for assessing the fracture toughness of the RPV steel are presented in RCC-M:

Method 1: similar to ASME Section III, Appendix G

- 1/4 thickness defect
- Level A: $2K_{Im} + K_{Ith} < K_{IR}$
- Level C & D $1.5\ K_{Im} + K_{Ith} < K_{IR}$

The K_{IR} curve is the same function of T-RT_{NDT} as in the ASME Code.

Method 2: An initial 15 mm crack is postulated, the end of life size is then evaluated using the Level A transient fatigue crack growth, the end of life K_J (based on J estimation scheme) is evaluated and the various criteria presented in Table XIX are used.

TABLE XIX. RCC-M APPENDIX ZG CRITERIA

Level A criteria:	$T < RT_{NDT} + 50°C$	$K_{cp} < \min(0.4\ K_{JC};\ 0.5\ K_{Ia})$
	$T > RT_{NDT} + 50°C$	$K_{cp} < \min(0.7\ K_{Ia};\ 0.7\ K_{JC})$
Level C criteria:	$T < RT_{NDT} + 50°C$	$K_{cp} < \min(0.5\ K_{IC};\ 0.85\ K_{Ia})$
	$T > RT_{NDT} + 50°C$	$K_{cp} < \min(0.85\ K_{Ia};\ 0.85\ K_{JC})$
Level D criteria:	$T < RT_{NDT} + 100°C$	$K_{cp} < \min(0.8\ K_{IC};\ 0.9\ K_{JC})$ and crack arrest before 75% of the thickness
	$T > RT_{NDT} + 100°C$	$K_{cp} < 0.9\ K_{JC}$ and limited stable crack growth

where RT_{NDT} is the material nil ductility transition temperature,

K_{IC}, K_{Ia}, K_{JC} are the material static, arrest and ductile initiation toughness, and

K_{cp} is the elastic stress intensity factor with a plastic zone correction.

3.4.3. Heatup and cooldown limit curves for normal operation

The governing equation for this analysis is defined in RCGH Appendices ZG Method 1 (which is similar to ASME III Appendix G) and is based on a 20 mm depth crack:

$$2K_{IM} + K_{It} \leq K_{IR} \tag{5}$$

The RSE-M gives in B2140, a figure for two pressure limits versus (T-RT_{NDT}), one for hydro testing at 1.2 times the design pressure and one for a maximum rate of cooldown of 20°C/hour [30].

3.5. WWER DESIGN BASIS

All the WWER RPVs were designed according to the Soviet (Russian) Codes in effect at the time of their design and manufacturing. Requirements for assuring general safety and design life were summarized in *Rules for Design and Safe Operation of Components of NPPs, Test and Research Reactors and Stations* [31] issued in 1973; these rules were updated in 1990 as *Rules for Design and Safe Operation of Components and Pipings of NPPs* [32]. The design itself (including the necessary stress analysis and the design lifetime calculations) was carried out mostly according to the *Code for Strength Calculations of Components of Reactors, Steam-Generators and Pipings of NPPs, Test and Research Reactors and Stations* [33] issued in 1973, which was updated in 1989 as the *Code for Strength Calculations of Components and Piping of Nuclear Power Plants, Moscow, 1989* [34]. The former Code was used for the design and analysis in the Pre-operational Safety Reports and the Supplementary Manufacturing Reports, the newer one is now also used for calculations within the Operational Safety Reports and other assessments. All these Soviet Codes were accepted also by all the national regulatory bodies of the countries operating these reactors.

3.5.1. Code requirements in Russia

The RPVs and primary system piping at all the major nuclear facilities, i.e. the PWRs, nuclear heating centres, as well as research and test reactors with operating temperatures over 600°C (i.e. with gas or liquid metal coolants) are safety related components and must be evaluated according to the Codes and Rules [31–34]. With respect to the WWER RPVs, special analysis requirements are also provided for radiation embrittlement.

The Code [33] is divided into 5 parts:

(1) *General Statements* deal with the area of Code application and basic principles used in the Code.

(2) *Definitions* gives full description of the most important operational parameters as well as parameters of calculations.

(3) *Allowable stresses, strength and stability conditions.*

(4) *Calculation of basic dimensions* deals with the procedure for choosing the component wall thickness, provides strength decrease coefficients and hole reinforcement values. Further, formulas for analysis of flange and bolting joints are also given.

(5) *Validating calculations* are the most important part of the Code. These detailed calculations contain rules for the classification of stresses as well as steps for stress determination.

Further, detailed calculations for different possible failure mechanisms are required and their procedures and criteria are given:

- calculation of static strength,
- calculation of stability,
- calculation of cyclic strength (fatigue),
- calculation of long-term cyclic strength (creep–fatigue) [not applicable for WWER RPV],
- calculation of resistance against brittle fracture,

TABLE XX. TYPICAL LIST OF WWER TRANSIENTS USED FOR DESIGN OF THE RPVs

CLASSIFICATION	TITLE	OCCURRENCE
NORMAL	Primary side pressure test	100
	Primary side leak test	30
	Plant heatup (20°C)	130
	Plant loading at 1% of full power per minute	5600
	Plant unloading at 1% of full power per minute	5000
	Change in 30-100% of full power	10 000
	Lost of load (without immediate or reactor trip)	150
	Step load decrease of 20% of full power	150
	Step load increase of 20% of full power	150
	Steady state fluctuations (+/- 5%)	Not limited
	Lost of power (blackout with natural circulation in reactor coolant system)	120
	Fault reactor trip	150
	Plant cooldown (max. 30°C)	70
UPSET	Loss of flow (partial loss of flow-one pump only)	30
	Inadvertent auxiliary spray into steam generator	10
	Tube failure of steam generator	30
	Fast plant cooldown (60°C)	30
EMERGENCY	Small break loss-of-coolant accident (inside diameter less than 100 mm)	15
	Loss-of-coolant accident (inside diameter more than 100 mm)	1
	Non-closure of safety valve in pressurizer	1
	Non-closure of safety valve in steam generator	1
	Steam pipe break	1

- calculation of long-term static strength (creep) [not applicable for WWER RPV],
- calculation of progressive form change [not applicable for WWER RPV],
- calculation of seismic effects,
- calculation of vibration strength (ultra-high frequency fatigue).

A mandatory part of this Code is also a list of the materials (and their guaranteed properties) to be used for manufacturing the components of the NSSS, including the RPVs. These appendices also contain methods for the determination of the mechanical properties of these materials and some formulas for designing certain structural features (e.g. nozzles, closures, etc.) of the vessel, as well as typical equipment units strength calculations.

3.5.2. Transient specification

In accordance with the NPP elements and systems classification as described in the *General Provisions on NPP Safety Assurance* [35], the WWER pressure vessel belongs to the 1st class of safety. Therefore, appropriately more rigid requirements are placed on the quality of the design, as well as the fabrication and operation of the RPV.

In accordance with Ref. [32] there are operation modes for equipment and pipings (including RPVs) which are defined as follows:

Normal mode of operation
- working conditions in normal operation

Violation of normal mode of operation
- any deviation from the normal mode of operation (as to pressure, temperature, loads, etc.), requiring a shutdown of the reactor to eliminate these deviations but without actuating the ECCS.

Emergency situation
- any deviation from the normal mode of operation which could result in poor core cooling and actuation of the ECCS.

Additionally, the normal mode of operation is subdivided into the following categories: steady mode, startup, CPS work, reactor power change, shutdown, as well as pressure hydrotests for strength and tightness testing.

A list of the expected operational modes is prepared when the RPV lifetime is calculated. Faulted conditions, like earthquakes, are analysed in a special part of the validating calculations. Definitions of these conditions are similar to those in the ASME Code or other Rules. For a given type of reactor, these conditions are specified in the design specification as well as in the Pre-operational Safety Report and are plant specific mostly only in the definitions of seismic events and conditions. A typical list of transients for the WWER-1000 reactor type with their categorization and design number of occurrences is given in Table XX.

3.5.3. Stress analysis

The validating calculations require a detailed stress analysis to determine the different types of stresses and classify of them so as to be able to apply prescribed stress limits and safety coefficients. Detailed analysis of various failure mechanisms are also required.

Categories of stresses

In principle, the stresses are divided into the following categories:

σ_m general membrane stresses

σ_{Ml} local membrane stresses

σ_b general bending stresses

σ_{Bl} local bending stresses

σ_T general temperature stresses

σ_{TL} local temperature stresses

σ_k compensation stresses

σ_{mw} mean tensile stresses in bolted sections, created by mechanical loading.

Checking calculations are carried out, applying to all existing loadings (including temperature effects) and all operating regimes.

Stress intensities, which are compared with allowable ones, are determined using the theory of maximum shear stresses with the exception of calculations of resistance against brittle failure, in which the theory of maximum normal stresses is applied.

Linear-elastic analysis techniques are used to calculate stresses in locations without stress concentrations. For fatigue calculations in the elastic-plastic region of loading, so-called pseudo-elastic stresses are used. These stresses are obtained by multiplication of the elastic-plastic strains in a given location by the Young's modulus.

Stress intensities are divided into four groups, according to their type:

$(\sigma)_1$ stress intensities calculated from the general membrane stress components

$(\sigma)_2$ stress intensities calculated from the sum of the general or local membrane and bending stress components

$(\sigma)_{3w}$ stress intensities calculated from the sum of the mean tensile stresses in a bolted section, including the tightening loads and the effects of temperature

$(\sigma)_{4w}$ stress intensities caused by mechanical and temperature effects, including tensioned bolt loadings and calculated from stress components of tension, bending and twisting in bolts

while the stress intensity ranges for RPVs are defined as:

$(\sigma)_{RV}$ the maximum stress intensity range calculated from the sum of the general and local stress components, the general and local bending stresses, the general temperature stresses and the compensation stresses

Stress intensity limits

The WWER Codes [33, 34] do not contain allowable stress intensity values (i.e. stress intensity limits), therefore these values must be calculated using:

(a) guaranteed mechanical properties, given in the Code,
(b) safety coefficients, also given in the Code.

Nominal allowable stresses [ó] caused by internal pressure. are defined as a minimum value:

$$[\sigma] = \min \{ R_{mT}/n_m; R_{p0\,2T}/n_{0\,2} \}, \quad (6)$$

where safety factors for vessels loaded by internal pressure are defined as:

n_m = 2.6 with respect to ultimate tensile strength, R_m
$n_{0\,2}$ = 1.5 with respect to yield strength. $R_{p0\,2}$

The nominal allowable stresses in bolting materials, as a result of pressure and bolt tightening, are given as:

$$[\sigma] = R_{p0\,2T}/n_{0\,2} \quad (7)$$

where the safety factor is given by:

$n_{0\,2} = 2.$

The allowable stresses in the WWER pressure vessel components are governed by the values calculated from the ultimate tensile strength of the material. These allowable stresses reach a value slightly less than 0.5 $R_{p0\,2T}$, similar as for bolted joints, i.e. even somewhat lower than that according to the ASME Code.

The validating calculation for static strength serves to control the strength requirements taking into account pressure, weight, additional loading, reaction loading and temperature effects in all operational regimes. All stresses obtained during these calculations must not exceed the values given in Table XXI. Mean bearing stresses must not exceed 1.5 $R_{p0\,2T}$. At the same time, mean shear stresses, as a result of mechanical loadings, must not be larger than 0.5 $[\sigma]$ (and, in bolt threads, no more than 0.25 $R_{p0\,2T}$). Mean shear stresses, as a result of mechanical loadings and temperature effects, must not be larger than 0.65 $[\sigma]$ (and, in bolt threads, no more than 0.32 $R_{p0\,2T}$). The general membrane stresses during hydraulic (or pneumatic) pressure tests must not be larger than 1.35 $[\sigma]_{Th}$ and the total stresses, determined as a sum of general and local membrane and general bending stresses must not be larger than 1.7 $[\sigma]_{Th}$, where $[\sigma]_{Th}$ is the allowable stress at the temperature of the pressure test. The maximum allowable stresses in the bolts during the pressure tests must not be larger than 0.7 $R_{p0\,2Th}$. In calculations of static strength using stress range $(\sigma)_R$, the maximum or minimum absolute values of stresses, put into calculations of $(\sigma)_R$, must not be larger than R_{mT}. Supplementary requirements for these stresses are also given in Table XXI.

TABLE XXI. ALLOWABLE STRESS INTENSITY LIMITS FOR WWER RPVs

COMPONENT	REGIMES	$(\sigma)_1$	$(\sigma)_2$	$(\sigma)_{3w}$	$(\sigma)_{4w}$	$(\sigma)_{RV}$
REACTOR PRESSURE VESSELS	NORMAL OPERATING CONDITIONS	$[\sigma]$	$1.3\ [\sigma]$	-	-	$(2.5 - R_{p0.2T}/R_{mT})\ R_{p0.2T}$
	UPSET CONDITIONS	$1.2\ [\sigma]$	$1.6\ [\sigma]$	-	-	-
	EMERGENCY CONDITIONS	$1.4\ [\sigma]$	$1.8\ [\sigma]$	-	-	-
BOLTING JOINTS	NORMAL OPERATING CONDITIONS	$[\sigma]_w$	-	$1.3\ [\sigma]_w$	$1.7\ [\sigma]_w$	-
	UPSET CONDITIONS	$1.2\ [\sigma]_w$	-	$1.6\ [\sigma]_w$	$2.0\ [\sigma]_w$	-
	EMERGENCY CONDITIONS	$1.4\ [\sigma]_w$	-	$1.8\ [\sigma]_w$	$2.4\ [\sigma]_w$	-

TABLE XXII. ALLOWABLE STRESSES FOR WWER PRESSURE VESSELS SUBJECTED TO SEISMIC EVENTS

LOADINGS	GROUP OF STRESSES	ALLOWABLE STRESSES	GROUP OF STRESSES	ALLOWABLE STRESSES
ALLOWABLE STRESSES				
NORMAL + DBE	$(\sigma_s)_1$ $(\sigma_s)_2$	1.4 $[\sigma]$ 1.8 $[\sigma]$	$(\sigma_s)_{mw}$ $(\sigma_s)_{4w}$	1.4 $[\sigma]_w$ 2.2 $[\sigma]_w$
NORMAL + DBA	$(\sigma_s)_1$ $(\sigma_s)_2$	1.2 $[\sigma]$ 1.6 $[\sigma]$	$(\sigma_s)_{mw}$ $(\sigma_s)_{4w}$	1.2 $[\sigma]_w$ 2.0 $[\sigma]_w$
ALLOWABLE BEARING STRESSES				
NORMAL + DBE	$(\sigma_s)_s$	2.7 $[\sigma]$	–	–
NORMAL + DBA		2.5 $[\sigma]$	–	–
ALLOWABLE SHEAR STRESSES				
NORMAL + DBE	$(\tau_s)_s$	0.7 $[\sigma]$	$(\tau_s)_s$	0.7 $[\sigma]_w$
		0.6 $[\sigma]$		0.6 $[\sigma]_w$

Calculation for earthquake effects must be performed for all sites characterized by MSK-64 grade 5 and more. In this case, new categories of stresses are defined:

$(\sigma_s)_s$ bearing stress including seismic loading
$(\tau_s)_s$ shear stress including seismic loading.

Requirements for these stresses are given in Table XXII, separately for the pressure vessel and the bolting joints. (In this table DBE is defined to be the design basis earthquake [with frequency 1/10 000 year] and DBA is the design basis accident [with frequency 1/100 year]).

Areas of the vessel analysed and examined

Stress analysis of the vessel is carried out for the whole RPV volume; however emphasis is placed on those regions with stress concentrations. Therefore the in-service inspections concentrate on regions with:

- the highest stress levels,
- potential sources of defects (welding joints, cladding, etc.).

Stress analysis methods

The Code provides unified methods for calculated and experimental determination of stresses, strains, displacements and loads. These methods are taken as recommended, other more precise methods can be also used. In this case, the organization performing this calculation is fully responsible for the results. Only computing programmes which have been approved by the regulatory body can be used for WWER stress analysis.

3.5.4. Design and analysis against brittle failure

All necessary requirements and analysis procedures as well as material data are given in the new version of the Code [34] (only the temperature approach was given in the previous version of the Code [33]). The whole procedure is summarized in the Chapter "*Calculation of Resistance Against Brittle Fracture*". The Code can also be used for components manufactured before the Code was issued, which are now in operation, or under completion, if the procedure has been approved by the regulatory body. The procedures in the Code are based on the principles of LEFM with the use of static plain strain fracture toughness, K_{IC}, only. The Code provides allowable stress intensity factor curves (defined also by formulas) as a function of reference temperature, a postulated flaw and a K_I expression for normal operating conditions, pressure tests and upset conditions and emergency conditions. In principle, the procedure is very similar to the one from the ASME Code, some differences result from the different materials and reactor designs used.

Allowable stress intensity factors

The Code gives as the main condition for fulfillment of component resistance against brittle failure the following formula:

$$K_I \leq [K_{IC}]i \tag{8}$$

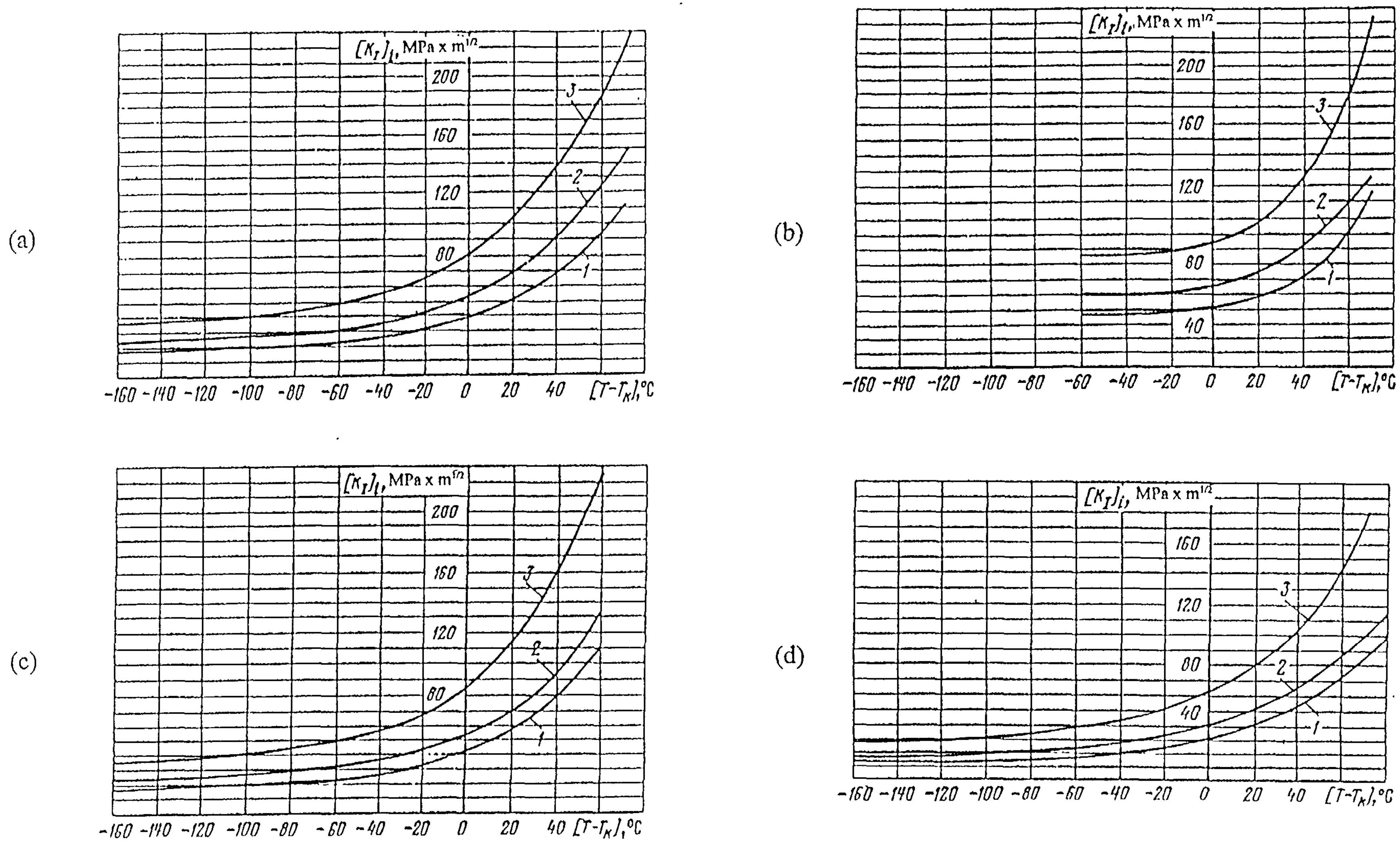
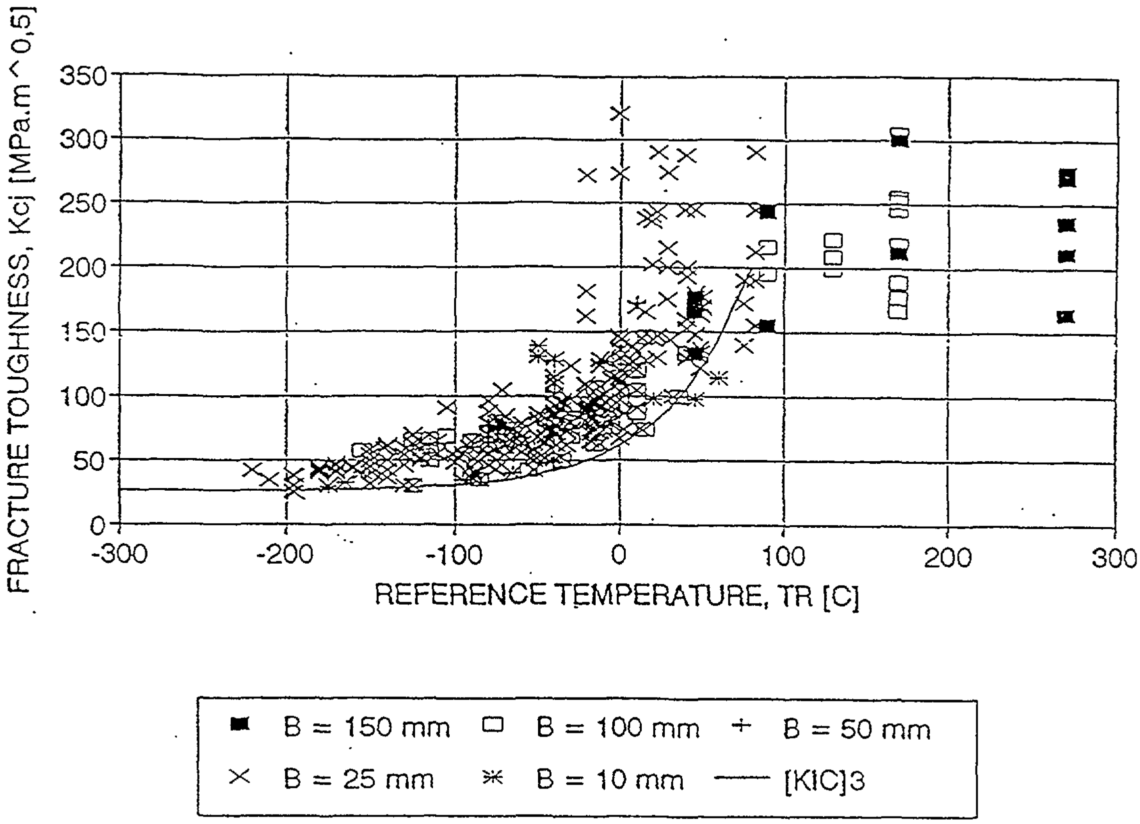


FIG. 22. The allowable fracture toughness curves in the Russian Code for three material types and for other low-Alloy steels. The curves labelled 1 are used for normal operating conditions, the curves labelled 2 are for pressure testing and upset conditions, and the curves labelled 3 are for emergency conditions. (a) Types 12Kh2MFA, 15Kh2MFA and 15Kh2MFAA steel. (b) Types 15Kh2NMFA and 15Kh2NMFAA steel (c) 15Kh2MFA, 15Kh2MFAA, 15Kh2NMFA and 15Kh2NMFAA weld metal. (d) Other low-alloy steels.

(a)



(b)

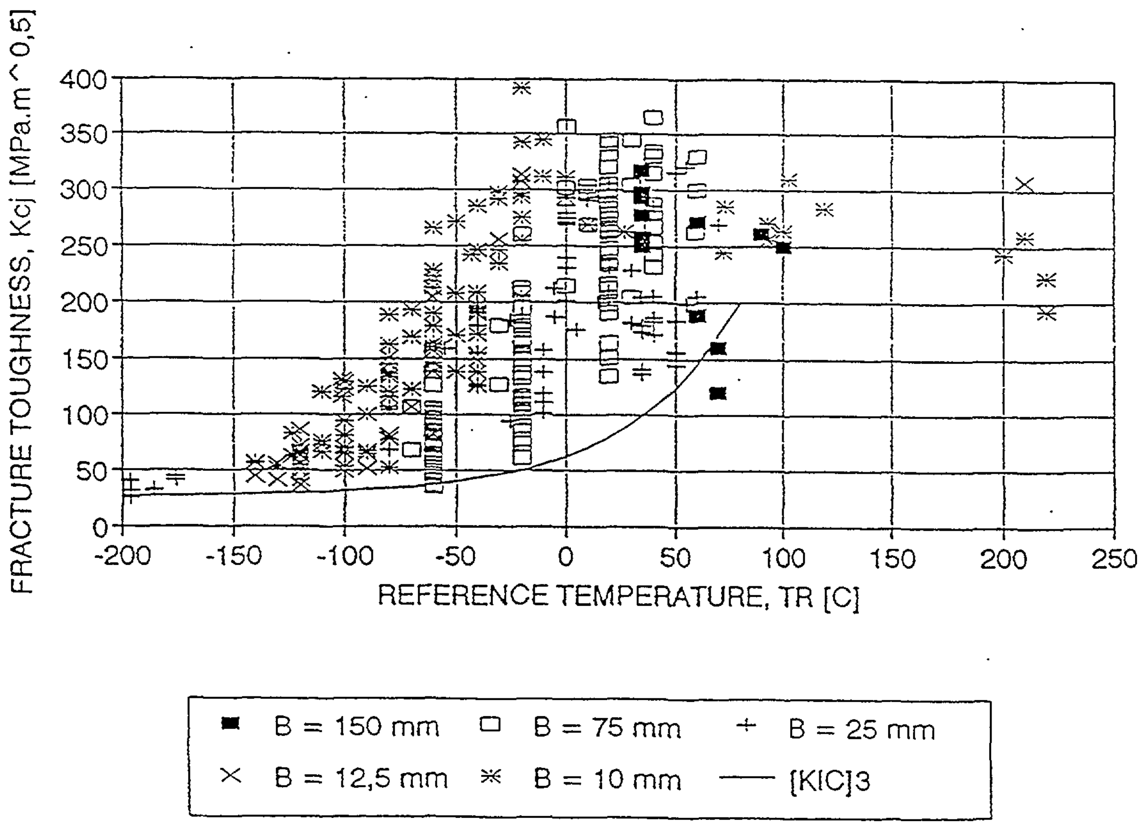


FIG. 23. Fracture toughness (K_{IC}) data plotted versus reference temperature (a) Type 15Kh2MFA and 15Kh2MFAA steels (b) Type 15Kh2NMFA and 15Kh2NMFAA steels. B is the specimen thickness.

where

$[K_{IC}]i$ is the allowable stress intensity factor for regime i

i = 1 normal operating conditions
i = 2 pressure hydrotest (pneumatic pressure test) or upset conditions
i = 3 emergency conditions.

The allowable stress intensity factor curves for three material groups and a set of general allowable stress intensity factor curves are plotted in Fig. 22. The curve labelled 1 on each plot is the K_{IC} curve for normal operating conditions, the curve labelled 2 is the K_{IC} curve for pressure testing and upset conditions and the curve labelled 3 is the K_{IC} curve for emergency conditions. The curves in Fig. 22a are for Types 12Kh2MFA, 15Kh2MFA and 15Kh2MFAA steels. The curves in Fig. 22b are for Types 15Kh2NMFA and 15Kh2NMFAA steels. The curves in Fig. 22c are for Type 15Kh2MFA, 15Kh2MFAA, 15Kh2NMFA and 15Kh2NMFAA weld metal. In addition, the curves in Fig. 22d are general formulas for use with other low alloy steels. These curves were constructed from lower bound curves of all the relevant experimental data for each material type; almost no available data fall under the curves. The fracture toughness data (K_{IC} versus reference temperature) for the Type 15Kh2MFA and 15Kh2MFAA steel are plotted in Fig. 23a. The fracture toughness data for the Type 15Kh2NMFA and 15Kh2NMFAA steels are plotted in Fig. 23b. Then two curves of allowable stress intensity factors were constructed for each of the three operating conditions using two types of safety factors:

n_K: a safety factor applied to the stress intensity
n_T: a safety factor applied to the temperature.

One curve was obtained from the initial lower bound data curve by dividing K_{IC} by the safety factor n_K. The other curve was obtained by shifting the temperature scale by n_T. The values of n_K and n_T used for the three operating conditions were:

Operating condition	n_K	n_T
Normal operating conditions	2.0	+30°C
Pressure tests and upset conditions	1.5	+30°C
Emergency conditions	1.0	0°C

The final allowable stress intensity factor curves shown in Fig. 22 were then constructed by fitting a lower bound curve to the curves adjusted by n_K and n_T for each operating condition. The result was allowable stress intensity factor curves ($[K_{IC}]i$ curves) as a function of reference temperature, defined as $[T-T_K]$, where T_K is the ductile to brittle transition temperature. As mentioned, allowable stress intensity factor curves were developed for three specific material types, as well as general ones. Figure 24 shows a comparison of the general curves for the three operating conditions with the ASME K_{IC} and K_{IR} curves. The equations which describe the curves shown in Fig. 22 are listed in Table XXIII.

Calculated (postulated) defect

A postulated defect was chosen to be much larger than any defect which could be missed during the pre-service or in-service non-destructive inspections. The postulated defect is defined as a semi-elliptical fatigue type crack with a depth (a) equal to 25% of the

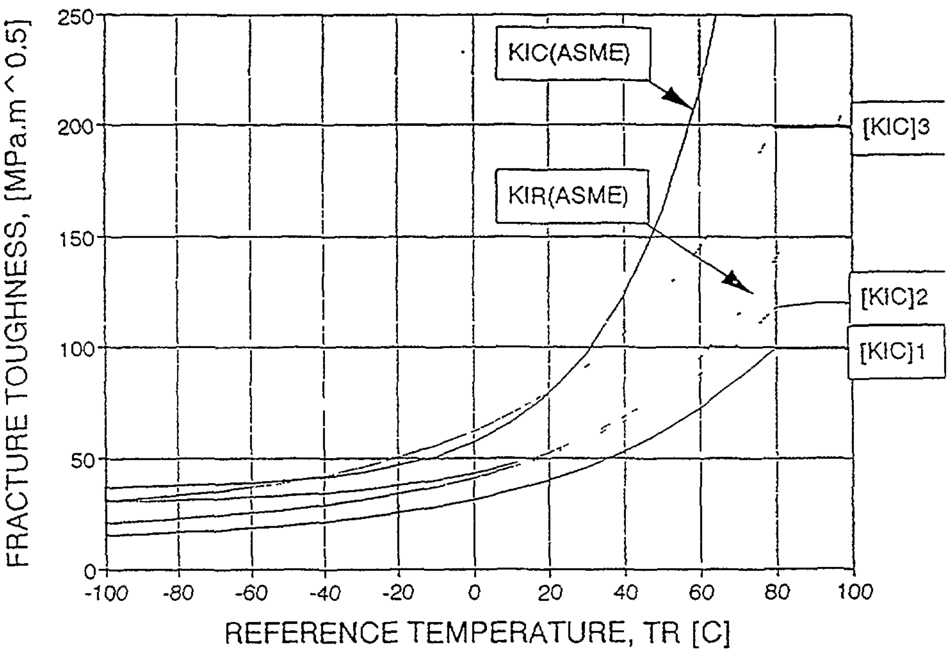


FIG 24 Comparison of the allowable WWER fracture toughness curves for low-alloy steels with the K_{IC} and K_{IR} reference fracture toughness curves in the ASME Code $[K_{IC}]1$ is the allowable fracture toughness for normal operating conditions, $[K_{IC}]2$ is the allowable fracture toughness for hydraulic testing, and $[K_{IC}]$ 3 is the allowable fracture toughness for emergency conditions

component thickness (S) without cladding and a crack shape equal to a/c = 2/3 where c is the crack length. These dimensions are independent of the vessel thickness and are applicable if the vessel thickness S fulfils the requirement

$$S \geq 8\ ([K_I]/\ R_{p0\ 2T})^2 \tag{9}$$

This postulated defect is put into the calculations for the normal and upset conditions. For emergency conditions, defects which range in size from a = 0 to a = 0.25 S must be taken into account.

Stress intensity factors

The Code allows the analyst to determine the stress intensity factors using analytical, numerical, or experimental methods, but all must be approved by the Regulatory Body. The Codes also give formulas for cylindrical, spherical, conical, elliptical as well as flat elements, loaded by inner pressure and temperature effects. In these formulas, stresses are divided into membrane and bending components using an integral type of mean stress determination. For elements with concentrators (due to thickness changes, holes, or nozzles) special correcting coefficients are provided. All these formulas are supposed to be as conservative as possible.

Transition temperatures

Fracture toughness is a temperature dependent mechanical property of a material (fracture toughness depends also on load rate, but in the Code, only static fracture toughness, i.e. failure initiation, is taken into account). Therefore, reference fracture toughness curves are

TABLE XXIII. METHODS FOR CALCULATING THE ALLOWABLE FRACTURE TOUGHNESS OF VARIOUS WWER MATERIALS

MATERIAL	$[K_{IC}]_1$	$[K_{IC}]_2$	$[K_{IC}]_3$
STEELS 12Kh2MFA, 15Kh2MFA, 15Kh2MFA-A	$17.5 + 22.5 \exp(0.02\ T_R)$	$23.5 + 30.0 \exp(0.02\ T_R)$	$35.0 + 45.0 \exp(0.02\ T_R)$
STEELS 15Kh2NMFA, 15Kh2NMFA-A	$37.0 + 5.5 \exp(0.0385\ T_R)$	$50.0 + 5.1 \exp(0.0041\ T_R)$	$74.0 + 11.0 \exp(0.0385\ T_R)$
WELDING METALS FOR STEELS 15Kh2MFA, 5Kh2MFAA, 15Kh2NMFA, 15Kh2NMFA-A	$17.5 + 26.5 \exp(0.0217\ T_R)$	$25.0 + 27.0 \exp(0.0235\ T_R)$	$35.0 + 53.0 \exp(0.0217\ T_R)$
GENERAL — ALL MATERIALS	$13.0 + 18.0 \exp(0.02\ T_R)$	$17.0 + 24.0 \exp(0.018\ T_R)$	$26.0 + 36.0 \exp(0.02\ T_R)$

constructed using so-called reference temperatures. In the Russian Codes, the so-called *critical temperature of brittleness* is a basis for an assessment of resistance against brittle failure. This critical temperature of brittleness, T_K, is determined using notch toughness testing of Charpy-V type specimens, only. In principle, this temperature is defined as a temperature, at which the mean value from 3 notch toughness tests is equal to a criterial value $(KCV)_c$ which is dependent on the yield strength ($R_{p0.2}$) of the material:

$R_{p0.2}$ [MPa]	$(KCV)_c$ [J.cm^{-2}]	$(KV)_c$ [J]
less than 300	30	24
300–400	40	32
400–550	50	40
550–700	60	48

At the same time, at a temperature equal to T_k + 30°C the following supplementary requirements must be fulfilled:

$$KCV \geq 1.5\ (KCV)_c \tag{10}$$
$$(KCV)_{min} \geq 0.7 \times 1.5\ (KCV)_c = 1.05\ (KCV)_c$$
$$(\text{fracture appearance})_{min} \geq 50\ \%\ (\text{fibrous fracture, }\%)$$

Differences between these critical temperatures, as determined experimentally for Types 15Kh2MFA and 15Kh2NMFA steel and ASTM A 533-B steel are:

$$\delta T = RT_{NDT} - T_k = \pm\ 10°C \tag{11}$$

Evaluation of the brittle resistance of the RPV at the design state is performed in accordance with the former Soviet “Code for Strength Calculation...” [34]. The evaluation is performed using linear elastic fracture mechanics techniques. Temperature and stress fields in the vessel during a PTS sequence are calculated for the whole vessel wall thickness, i.e., the austenitic cladding is taken into account, if it exists. A finite element method is usually used for the calculation of the temperature distribution in the wall, as well as the stress calculation.

The stress intensity factors, K_I, are determined only for the deepest part of each postulated defect and are calculated for the entire loading path and for a whole set of postulated defects with depths ranging from 0 to 25% of the wall thickness. These calculated stress intensity factors are then compared with the allowable stress intensity factors for emergency conditions, $[K_I]_3$, taking into account the temperature dependence of these factors. From those comparisons, the maximum allowable critical brittle fracture temperatures, $T_k^a(j)$, for the analysed PTS sequences are obtained. In other words, the K_I values are plotted versus the temperature at the deepest point in the crack during the whole PTS sequence. Then the $[K_{IC}]_3$ curve is shifted to a higher temperature, up to the point where it contacts the K_I curve. The value of the shift determines the maximum allowable critical temperature, $T_k^a(j)$ for the j-event which fulfils the requirement that K_I is lower than $[K_{IC}]_3$. The lowest of these temperatures for the whole set of analysed PTS sequences is taken as the maximum allowable critical brittle fracture temperature, T_k^a. This temperature is material independent and depends only on the RPV and reactor design, especially on the PTS sequences. This temperature is then compared with the critical brittle fracture temperature T_k of the analysed vessel. Decisions on further operation can be made based on this comparison.

The brittle to ductile transition temperature (critical temperature of brittleness) of the WWER pressure vessel materials is time or use dependent, since many damaging mechanisms can affect it, and can be expressed in the form:

$$T_k = T_{k0} + \Delta T_F + \Delta T_T + \Delta T_N \qquad (12)$$

where

T_K is the instant critical temperature of brittleness
T_{k0} is the initial critical temperature of brittleness
ΔT_F is the shift of critical temperature due to radiation embrittlement
ΔT_T is the shift of critical temperature due to thermal ageing
ΔT_N is the shift of critical temperature due to cyclic damage

Individual component shifts are also defined in the Code where formulas for their determination are also given.

The transition temperature shift due to radiation embrittlement (ΔT_F) can be expressed as

$$\Delta T_F = A_F \,.\, (F \times 10^{-22})^{1/3} \qquad (13)$$

where

A_F is the radiation embrittlement coefficient
F is the neutron fluence with energies greater than 0.5 MeV.

Fluences with energies higher than 0.5 MeV are used in the Soviet Code, as it is suggested that this criterion better describes the damaging part of the fluence. The ratio between fluences with energies higher than 0.5 and energies higher than 1.0 MeV depends on the place in the reactor where it is determined — for PWR types, it mainly depends on the reflector thickness and the surveillance position, inner or outer vessel wall. For the inner surface of a WWER pressure vessel this ratio is approximately

$$F(E_n \geq 1.0\ \text{MeV}) \,/\, F(E_n \geq 0.5\ \text{MeV}) \propto 0.6 \qquad (14)$$

The coefficient A_F depends not only on the radiation temperature but also on material composition, mainly on the phosphorus, copper and nickel contents (for 15Kh2NMFA). The Code provides specific values or formulas for the A_F coefficients which are necessary to put into the calculations. These values have been obtained as upper bound values from experimental data. All the necessary data are summarized in Table XXIV.

Thermal ageing should also be taken into account, and for the WWER-440 and WWER-1000 RPV materials, this shift is given as

$$\Delta T_T = 0^{\circ}\text{C for Type 15Kh2MFAA steel} \qquad (15)$$

$$= +5^{\circ}\text{C for Type 15Kh2NMFAA of steel}$$

TABLE XXIV. WWER RADIATION EMBRITTLEMENT COEFFICIENTS

MATERIAL		IRRADIATION TEMPERATURE T_{irr} [°C]	IRRADIATION EMBRITTLEMENT COEFFICIENT A_F [°C]
15Kh2MFA	BASE METAL	250	22
		270	18
		290	14
	A/S WELD METAL	250	800(P+0.07 Cu)+8
		270	800(P+0.07 Cu)
15Kh2MFA-A	BASE METAL	270	12
		290	9
	A/S WELD METAL	270	15
		290	12
15Kh2NMFA	BASE METAL	290	29
15Kh2NMFA-A	BASE METAL	290	23
	A/S WELD METAL	290	20

The shift ΔT_N represents the changes in the material properties caused by low-cycle fatigue damage. All transients are considered, including heatup and cooldown, pressure testing, scram, etc. For WWER pressure vessel materials, the Code provides the following formula to be used in the calculations:

$$\Delta T_N = 20 \,.\, A \ \ [^\circ C] \qquad (16)$$

where A is the usage factor from the fatigue calculations, which means that the maximum shift due to cyclic damage is equal to + 20°C. This shift is, of course, only taken into account in locations with high stress concentrators, where a high usage factor is obtained - i.e. mostly for nozzles.

However, it must be mentioned that both of the WWER pressure vessel materials are cyclically softened and thus this formula gives very conservative values. In fact, some negative shift of the transition temperature is usually found during the early part of the fatigue life.

The Code strictly requires a material surveillance programme for all reactor vessels. Requirements for the type of specimens and the time schedule for their withdrawal are also presented.

3.5.5. WWER heatup and cooldown limit curves for normal operation

Heatup and cooldown limit curves (P–T limit curves) are calculated using a linear elastic fracture mechanics approach and a reference critical (brittle) temperature, T_k, defined on the basis of Charpy V-notch impact tests, only, but taking into account the potential effects of degrading mechanisms such as radiation embrittlement, thermal ageing and fatigue damage.

The allowable stress intensity factors values are shown in Table XXIII. They were constructed from the lower bound fracture toughness values for the listed materials and certain prescribed safety factors. Then, allowable P-T limit curves are obtained when:

$$K_I(T) \leq [K_{IC}]_i \tag{17}$$

where

i = 1 for normal operating conditions, and

i = 2 for hydrostatic testing.

The stress intensity factors, $K_I(T)$, are calculated for the "postulated defect" discussed in Section 3.5.4 above, which is assumed to be at a surface without cladding and semielliptical in shape with a depth equal to 25% of the wall thickness and an aspect ratio, a/c. equal to 2/3. The defect is assumed to be perpendicular to the principal stresses. Only the deepest point of the defect is considered when calculating the stress intensity factors. The following formula is recommended:

$$K_I = \eta\ (M_m\sigma_m + M_b\sigma_b)(\pi a)^{1/2}Q^{-1} \tag{18}$$

where

η is a correction to the stress concentration (= 1 for a cylindrical part)

σ_m is the membrane stress

σ_b is the bending stress

M_m is a membrane correction factor

M_b is a bending correction factor

a is a crack depth (m)

Q is a shape factor.

The mechanical, as well as the thermal stress components, are added together, and the membrane and bending stress components are then derived using summary stress integration through the vessel wall. The following type of equation is obtained when the required dimensions and aspect ratio of the postulated defect are put into Equation (18):

$$K_I = \eta(0.7\sigma_m + 0.4\sigma_b)(s)^{1/2} \tag{19}$$

where s is the RPV wall thickness (m).

This formula is then used for calculation of the P-T limit curves. It must be also mentioned that the maximum allowable heatup and cooldown rates are 30 K/h, only.

4. AGEING MECHANISMS

This section describes the age related degradation mechanisms that could affect PWR RPV components and evaluates the potential significance of the effects of these mechanisms on the continued safety function performance of these components throughout the plant service life.

The set of age related degradation mechanisms evaluated in this section is derived from a review and evaluation of relevant operating experience and research. This set consists of the following mechanisms:

1. Radiation embrittlement
2. Thermal ageing
3. Temper embrittlement
4. Fatigue
5. Corrosion
   - Intergranular attack and PWSCC of Alloy 600 components, Alloy 82/182 welds, radial keys, etc.
   - General corrosion and pitting
   - Boric acid corrosion
6. Wear.

The technical evaluation of a particular age related degradation mechanism and its effects on the continued safety or functional performance of a particular PWR RPV component leads to one of two conclusions: (1) the degradation mechanism effects are potentially significant to that component and further evaluation is required relative to the capability of programmes to effectively manage these effects; or (2) the age related degradation effects are not significant to the ability of that component to perform its intended safety function throughout the remainder of plant life. For the latter case, specific criteria and corresponding justification are provided in this section. These criteria can be used as the basis for generic resolution of age related degradation mechanism/component issues.

The most important ageing degradation mechanism is the radiation embrittlement of the cylindrical part of the RPV surrounding the core.

4.1. RADIATION EMBRITTLEMENT

4.1.1. Radiation embrittlement of western pressure vessels

The degree of embrittlement and hardening induced in ferritic steels after exposure to fast neutron radiation is an issue of the utmost importance in the design and operation of NPPs. The area of the RPV surrounding the core (called the beltline region) is the most critical region of the primary pressure boundary system because it is subjected to significant fast neutron bombardment. The overall effect of fast neutron exposure is that ferritic steels experience an increase in hardness and tensile properties and a decrease in ductility and toughness, under certain conditions of radiation.

For example:

1. *The effect of neutron fluence* on radiation hardening and embrittlement has been reported to be significant at fluences above 10^{22} n/m^2 (E >1 MeV). Unless a steady state or saturation condition is reached, an increase in neutron fluence results in an increase in RT_{NDT}, yield strength and hardness, and a decrease in the Charpy toughness, also in the upper shelf temperature region. There are significant variations in the fluence and radiation damage around the circumference and in the longitudinal direction of RPVs.

2. *Alloy composition* (especially when consideration is given to impurity copper and phosphorus and alloying element nickel) is known to have a strong effect on radiation sensitivity. Data have been generated on both commercial and model alloys to show the effects of alloy composition.

3. *Radiation temperature* has long been recognized to have an effect on the extent of the radiation damage. Data from the early 1960s demonstrated that the maximum embrittlement occurred during radiation at temperatures below 120°C (250°F). Recent studies have reported a decrease in radiation embrittlement at higher temperatures (>310°C), which is attributed to the dynamic in-situ "annealing" of the damage.

4. *Microstructural characteristics*, such as grain size and metallurgical phases (lower or upper bainite, ferrite), can influence the severity of radiation damage associated with a given fluence.

5. *The neutron flux energy spectrum* contributions to the embrittlement behaviour of ferritic steels are secondary effects. However, recent reactor experience has suggested that, under certain conditions, the flux spectrum may influence the degree of radiation embrittlement caused in ferritic steels.

The most important parameters listed above are fluence and alloy and impurity content. The deleterious effect of copper (Cu) as an impurity element on radiation embrittlement and hardening of pressure vessel steels and welds was recognized nearly 20 years ago. The dramatic increase in ductile-to-brittle transition temperature and reduction in USE observed in a variety of pressure vessel welds after neutron radiation at ~288°C was broadly correlated with the nominal impurity Cu content in the steels. The increased sensitivity to embrittlement was more pronounced for welds because Cu-coated welding rods had frequently been used in the fabrication of the reactor vessels leading to Cu levels of ~0.3 wt%. For an equivalent copper level, the cast structure of weld metals is more sensitive to neutron radiation damage than the base metal.

Early methods used to quantify the effect of impurity elements on radiation sensitivity in western RPV materials indicated that both copper and phosphorus played a role [36, 37]. Later on, USNRC Regulatory Guide 1.99, Revision 2 omitted the effect of phosphorus but included nickel as a factor [38]. As a result, it is often assumed that copper and nickel play the dominant role in creating sensitivity to neutron radiation in low phosphorus steels. However, there are variations in alloying content and impurity element ranges in the various countries in which the RPV materials were produced and it is still necessary to consider the contribution of phosphorus, particularly when low levels of copper and nickel are present.

The radiation embrittling mechanism attributed to copper impurity level is well understood in terms of small copper-rich clusters or precipitates formed under the creation of minute matrix damage caused by fast neutron bombardment. Such precipitates can act as blocks to dislocation movement and cause hardening and embrittlement. Hawthorne and co-workers [39] examined the action of Cu and P in a variety of A533B and A302B steels. Phosphorus contents greater than 0.014 weight% exerted a strong effect on the sensitivity of A302B steels to radiation embrittlement.

Unlike Cu and P, the role of nickel (Ni) in radiation hardening/embrittlement has been unclear. The contradictory reports concerning the influence of Ni in the embrittlement behaviour of RPV steels indicated that its effect was a subtle one. The effect of Ni can be demonstrated qualitatively by studying the HY and A350LF steels (~3 wt. % Ni). Although studies by Lucas et al. [40] and Igata et al. [41] showed no effect of Ni on radiation embrittlement of RPV steels, several other studies show a significant effect. In 1981, Guionnet et al. [42] concluded that Ni was deleterious to the behaviour of A508 irradiated to a fluence of 5×10^{23}n/m^2 at 290°C. A pronounced Ni effect in increasing the radiation sensitivity of high Ni (0.7 wt. %) welds was reported by Hawthorne [43]. Similarly, Fisher and Buswell [44] noted that high Ni steels (i.e., those containing >1% Ni) were much more sensitive to neutron radiation than steels containing <0.85% Ni. Soviet experience with chromium (Cr) and Ni bearing RPV steels also indicated that Ni exerted a pronounced effect on embrittlement behaviour [45].

Recently, Odette and Lucas [46] examined the effect of Ni (0 to 1.7 wt%) on the hardening behaviour of A 533-B type steels as a function of neutron fluence, flux, temperature and manganese and copper content. Irradiations at fluxes of 5×10^{15} and 5×10^{16}n/m^2/s gave final fluences of 9×10^{22} to 1.5×10^{23}n/m^2 (E>1 MeV). Low fluence irradiations were done at 306°C and 326°C; the higher fluences were accumulated at 271°C to 288°C. The results indicated that Ni increased the sensitivity to radiation embrittlement in these materials, with increasing fluence, lower flux levels, lower irradiation temperature and increased manganese (Mn) levels causing more damage. The synergisms and complex nature of the response of the alloys examined makes a complete interpretation of the mechanisms difficult.

Although the roles of Cu, P and Ni as promoters of radiation hardening and embrittlement are well-recognized, the contribution of other elements such as manganese (Mn), molybdenum (Mo), Cr, arsenic (As) and tin (Sn), to the radiation induced behaviour of RPV steels has not been unambiguously identified.

4.1.2. Significance for western pressure vessels

A fluence value of 1×10^{22} n/m^2 (E >1 MeV) is approximately the threshold for neutron induced embrittlement of the ferritic steels used in western PWRs. Therefore, the beltline region is the region most likely to undergo significant changes in mechanical properties due to neutron radiation. Components made of materials such as Alloy 600 or Alloy 182 are less susceptible to neutron embrittlement. The following components are subjected to lifetime fluences less than 1×10^{22} n/m^2 (E >1 MeV) or are made of materials not susceptible to neutron embrittlement:

- Core supports
- Nozzles
- Head penetrations
- Bottom head
- Top head
- Vessel flange
- Closure studs
- Safe ends.

Therefore, neutron embrittlement is potentially significant only for that part of the RPV shell beltline region which is located in a high flux region.

4.1.3. Radiation embrittlement of WWER pressure vessels

Three different types of steels, 15Kh2MFA, 15Kh2MFAA and 15Kh2NMFAA were used to fabricate the WWER pressure vessels. These steels are affected by different embrittlement mechanisms and behave differently from each other as discussed below.

The 15Kh2MFA type steel is used for the WWER-440 V-230 pressure vessels and the Loviisa pressure vessels. It has almost no nickel and so its behaviour is controlled by its phosphorus and copper impurity content. Contrary to western practice, in which specifications strictly limit the phosphorus content in both the base and weld metal, the original specifications used in the CMEA countries imposed only very mild requirements on the phosphorus content, allowing as much as 0.040 weight %. The phosphorus content was originally not even measured in the weld metal; it was measured only in the welding wires. The resulting phosphorus contents are listed in Table XXV and are mostly close to or even higher than the 0.040 weight % limit in the weld metal. The copper content in the 15Kh2MFA type steel typically ranges from 0.15 and 0.20 weight % and so its effect on embrittlement is small. Thus *phosphorus is practically the only controlling impurity* in the steel used for the WWER V-230 type of RPVs.

Phosphorus causes embrittlement because of thermal and radiation induced diffusion to and segregation at the grain boundaries. However, intercrystalline (intergranular) fracture of Charpy surveillance specimens is very rare, even after high neutron fluences. Most of the Charpy failures are transgranular failures. Therefore, the effects of the high phosphorus segregation are not fully understood.

The 15Kh2MFA vessels become, of course, very embrittled during radiation and most of them have been annealed in the last several years. Radiation embrittlement remains the main concern for these types of vessels.

The beltline regions of the WWER-440/V-213 pressure vessels were also manufactured from 15Kh2MFA steel, but many of the V-213 pressure vessels have low phosphorous and copper contents and are similar in impurity content to the WWER-1000 pressure vessels made of 15Kh2MFAA steel with strict requirements on the residual element (Cu, P, As, Sn and Sb) content. Thus, radiation embrittlement does not seem to be a limiting factor for a 40 year vessel lifetime. Moreover, the degree of radiation embrittlement of the WWER-440/V-213 pressure vessels is lower than that of the western PWR vessels made of

TABLE XXV. COPPER AND PHOSPHORUS CONCENTRATIONS (WEIGHT %) IN THE BASE AND WELD METAL OF VARIOUS WWER-440 PRESSURE VESSELS

Plant	Cladding	Weld metal No. 4			Base metal		
		Cu	P	METHOD	Cu	P	METHOD
KOLA 1	N	0.13 0.146	0.032 0.033	calc. scrape inside	–	0.012	certif.
KOLA 2	N	0.154	0.036 0.0375	calc. scrape inside	–	0.012	certif.
ARMENIA 1	N	0.16	0.030	scrape inside	–	0.013	certif.
ARMENIA 2	Y	–	–	–	–	–	–
NOVOVORONEZH 3	N	0.15	0.031	template	0.16	0.012	template
NOVOVORONEZH 4	N	0.17	0.030	template	–	0.011	certif.
KOZLODUY 1	N	0.12	0.0515 0.036	scrape inside calc.	0.15	0.010	certif.
KOZLODUY 2	N	0.18	0.036 0.0375	template	0.17	0.017	template
KOZLODUY 3	Y	0.20	0.036	certif. based on test coupon	0.17	0.016	certif.
KOZLODUY 4	Y	0.04	0.021	certif. based on test coupon	0.10	0.012	certif.
BOHUNICE 1	Y	0.15 0.103	0.035 0.043	certif. scrape outside	0.13 0.091	0.012 0.014	certif. scrape outside
BOHUNICE 2	Y	0.20 0.109	0.036 0.026	certif. scrape outside	0.08 0.082	0.010 0.010	certif. scrape outside
GREIFSWALD 1	N	0.104 0.10	0.034 0.043	scrape inside template	0.17 0.18	0.010 0.015	certif. template
GREIFSWALD 2	N	0.157 0.15	0.037 0.032 0.036	certif. scrape inside	–	0.012	certif.
GREIFSWALD 3	Y	0.12	0.035	certif. based on test coupon	–	0.012	certif.
GREIFSWALD 4	Y	0.16	0.035	certif. based on test coupon	0.12	0.016	certif.

ASME A 533-B material even though the V-213s are irradiated at a relatively low temperature, about 265°C. This is probably due to the higher structural stability of the 15Kh2MFAA type steel, relative to the A 533-B steel, caused by the presence of vanadium carbides, which are very stable, together with the steel microstructure and the absence of nickel.

The beltline regions of the WWER-1000 pressure vessels are fabricated from Type 15Kh2NMFAA steel. This steel has almost no vanadium and much more nickel than the Type

15Kh2MFAA material used for the WWER-440 vessels. As mentioned in Section 2, a nickel rather than vanadium alloy steel was chosen for the WWER-1000 vessels so that it would be easier to weld the relatively large WWER-1000 forgings, while still retaining the desired strength characteristics. The limits on the residual element content for this steel are very strict (similar to 15Kh2MFAA type).

The nickel in the base metal was controlled to values between 1.00 and 1.50 weight %, however, the nickel in the weld metal of many of the WWER-1000 RPVs is as high as 1.90 weight %. Thus, the nickel content in the weld metal is the controlling element for radiation embrittlement. The inlet water temperature of the WWER-1000 plants is higher than in the WWER-440 plants by about 20°C (i.e. 288°C) and is similar to western PWR inlet water temperatures. Since the operating temperatures and nickel contents are similar, the radiation embrittlement of the beltline of the WWER-1000 vessels is somewhat comparable to the embrittlement of the beltline regions of the western vessels fabricated with A 533-B and A 508.

The radiation coefficient, A_F, given in Ref. [34] and discussed in Section 3.4.4 of this report, was developed from weld metal data with a nickel content lower than 1.5 weight %. Therefore, use of the standard values of these coefficients for determining the allowable fracture toughness of weld metal with a high nickel content (using the K_{IC} curves for weld metal from the Code) may not be conservative. Also, the data from the WWER-1000 surveillance specimens (except the data from the specimens from the three ŠKODA made vessels) may not be representative of the radiation embrittlement of the beltline materials because the surveillance specimens are located above the reactor core and the core barrel in a steep flux gradient where their temperatures are up to 10 to 20°C higher than the temperature of the RPV beltline region. This subject is discussed in greater detail in Section 6.

To summarize, the 15Kh2MFA weld metal in the WWER-440 V-230 pressure vessels is very susceptible to radiation damage because of its low operating temperature and high phosphorus content and the segregation of the phosphorus on the grain boundaries. The Type 15Kh2MFAA material in the WWER-440 V-213 pressure vessels is relatively resistant to radiation damage because of its good chemistry (lack of impurities) and vanadium carbides. However, this material is also exposed to relatively low operating temperatures where there will be more radiation damage than at higher temperatures. The Type 15Kh2NMFAA material used for the WWER-1000 pressure vessels sometimes contains relatively high levels of nickel in the weld metal, but relatively low levels of copper and other impurities. Its radiation damage may be somewhat comparable to some of the materials used in western PWR pressure vessels.

4.1.4. Significance for WWER pressure vessels

Radiation damage becomes significant at neutron fluences greater than 1×10^{22} n/m^2 (E >0.5 MeV). The design end-of-life neutron fluence for the beltline region of the WWER-440/V-230 pressure vessels has been calculated to be to approximately 1.5×10^{24} n/m^2 while for the V-213 type it is somewhat higher — up to 2.5×10^{24} n/m^2. The actual RPV life depends very strongly on the operation history and mitigation activities. Most of the V-230 plants use dummy elements in the periphery of the active core to decrease the neutron flux on the RPV wall; in other WWER reactors, a low leakage core (LLC) strategy has been implemented to reduce the flux hence fluence.

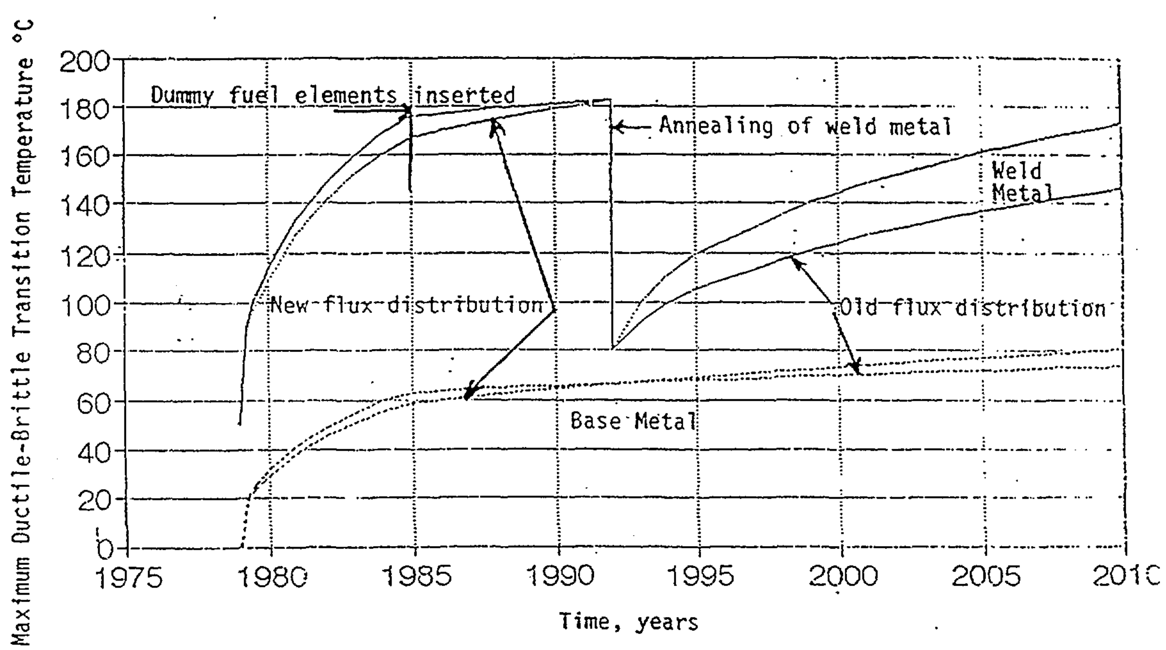


FIG. 25. Transition temperature values as a function of operation time for Bohunice Unit 2. The upper lines are the weld material and the lower lines are the base metal material.

Although the most irradiated part of the RPV is the base metal situated around the axial centre region of the reactor core, the most degraded *material* in the WWER V-230 vessels is the circumferential weld metal located in the lower part of the core, which is labelled the 0.1.4 or 5/6 weld. Its neutron fluence reaches only about 70% of that of the maximum fluence in the beltline region, but its embrittlement is much higher because of its high phosphorus content. This weld metal controls the vessel lifetime even after vessel annealing, as shown in Fig. 25. The circumferential weld at the top of the centre shell ring is subjected to much lower fluences, its neutron fluence being equal to about 3% of the maximum fluence in the beltline region. Therefore, it is necessary to anneal only a small region around the most embrittled weld to improve the state of the whole vessel and to extend its lifetime.

In contrast to the embrittlement behaviour of the weld and base metals used in the V-230s, there is no substantial difference between the embrittlement of the base and weld metals in the WWER-440/V-213 pressure vessels. The only difference is that the initial transition temperature of the weld metal is much higher than in the base metal. Thus, the weld metals located at 0.1.4 again control the vessel lifetime.

There is only a small difference between the fluences in the circumferential weld metal situated in the lower part of the active core and the base metal exposed to the maximum fluences in the beltline region of the WWER-1000 pressure vessels. In this case, for weld metals with nickel content lower than 1.5 weight %, there is no substantial difference in the embrittlement of the base and weld metal. However, weld metals with high nickel content (up to 1.9 weight %) experience substantially greater embrittlement than the base metal. Thus, in most cases, the weld metals remain the controlling materials for the RPV embrittlement.

4.2. THERMAL AGEING

4.2.1. Description of mechanism

Thermal ageing is a temperature, material state (microstructure) and time dependent degradation mechanism. The material may lose ductility and become brittle because of very small microstructural changes in the form of precipitates coming out of solid solution. In the case of RPV steel with impurity copper, the important precipitates are copper-rich (however, there could be other precipitates). The precipitates block dislocation movement thereby causing hardening and embrittlement. The impurity copper in RPV steel is initially trapped in solution in a super-saturated state. With time at normal PWR operating temperatures (~290°C), it may be ejected to form stable precipitates as the alloy strives toward a more thermodynamically stable state, even if there is no radiation damage. As discussed in Section 4.1, neutron-induced structural damage promotes the copper precipitation process.

The effects of long-term aging at temperatures up to 350°C on the ductile-to-brittle transition temperature of RPV steels have recently been summarized in a paper by Corwin et al. [47]. The work was sponsored by the USNRC and performed at the Oak Ridge National Laboratory and the University of California, Santa Barbara. Corwin et al. concluded that "most of the data from the literature suggest that there is no embrittlement in typical RPV steels at these temperatures for times as great as 100 000 h...."

Some of the more important data is discussed next. The reader is referred to the Corwin et al. paper for additional references and discussions. Data are available on the behaviour of A 302 Grade B, A 533 Grade B and A 508 Class 2 and Class 3, and equivalent non-US steels. Limited thermal ageing studies by Potapovs and Hawthorne [48] for P-bearing A 302 Grade B steels at 290°C revealed no significant shift in the ductile to brittle transition temperature (decrease of 5 to 14°C). DeVan et al. [49] have reported that A 533 Grade B Class 1 plate materials from the Arkansas 1 reactor shifted –4 to 10°C after thermal ageing at 280°C for 93 000 hours. A 508 Class 2 forging materials encapsulated outside the beltline region of the Oconee Unit 3 reactor showed an increase of about 1°C after exposure to a temperature of 282°C for 103 000 hours [49]. Also, weld metal specimens from the Arkansas 1 and Oconee Unit 3 reactors showed changes in the ductile-to-brittle transition temperature of –8 to 0°C after exposure up to 103 000 hours at about 280°C [49]. The weld metal specimens were made with Linde 80 MnMoNi weld wire typical of that used for submerged arc welds.

Fukakura et al. [50] studied the effect of thermal ageing on A 508 Class 3 steel and concluded that after thermal ageing for 10 000 hours at temperatures of 350°C, 400°C and 450°C, the increase in the nil-ductility transition temperature was small. The J_{IC} and J-R resistance curves also decreased somewhat as a result of thermal ageing. It appears that grain size may be an important variable in assessing thermal ageing embrittlement. The effect of grain size on thermal ageing embrittlement may be due to grain boundary embrittlement by impurity segregation (e.g. P) at the grain boundaries.

Three studies have been conducted in Germany of the effects of thermal ageing of low alloy pressure vessel steels. All the tests were conducted at temperatures typical of operating temperatures and for durations of 10 000 to 100 000 hours. All found negligible detrimental effects. In the first study, base, butt-weld heat-affected zone, and weld-simulated heat-affected zone 20 Mn Mo Ni 55 material was clamped to the main coolant lines of three NPPs for about 60 000 hours (7 years). The ageing temperature was approximately 290°C (555°F). The toughness transition temperature curves for these materials were unchanged at the end of the exposure period. In the second study, which was part of the German Component Safety Programme, base and heat-affected zone 20 Mn Mo Ni 55 and 22 Ni Mo Cr 37 material was placed in a laboratory furnace at 320°C (608°F) for up to 10 000 hours. Again, there were no significant changes in the toughness transition temperature curves. In the third study, parts of the Obrigheim main coolant line were removed after approximately 100 000 hours of operation at about 285°C (545°F) and then destructively examined (tensile and Charpy testing). The Obrigheim main coolant line was fabricated from 22 Ni Mo Cr 37 material. The mechanical testing indicated that there had been no thermal embrittlement of this material during the 100 000 hours of operation.

In contrast, Hasegawa et al. [51] observed some small shifts in the transition temperature for Cu and P-bearing A 533-B steels after thermal ageing at temperatures near 300°C and a maximum shift at about 500°C, well beyond the operating temperature of PWRs. Similar behaviour was reported for coarse-grain simulated and thermally aged heat-affected zones of A 533-B steel by Druce et al. [52]. However, this ageing was associated with temperatures higher than 400°C and with P segregation to the grain boundaries.

The 15Kh2MFA and 15Kh2MFAA steels used to fabricate the WWER-444 pressure vessels also do not appear to be susceptible to thermal ageing, even when they contain relatively high phosphorous impurity levels. The results from the thermal ageing surveillance specimens located in the upper plenums of the WWER-440 V-213 pressure vessels and removed and tested after 10 years at about 300°C indicate that the shift in the Charpy ductile to brittle transition temperature is small. These results are supported by Charpy ductile to brittle transition temperature measurements from RPV trepans removed from closed plants (Novovoronezh 1 and 2), as well as boat samples taken from operating plants. Laboratory tests carried out at 350°C for 10 000 hours also showed that the transition temperatures remain stable within the normal data scatter.

The type 15Kh2NMFA steel used to fabricate the WWER-1000 pressure vessels is slightly susceptible (a shift in the ductile-brittle transition temperature of 10 to 20°C) to thermal ageing at operating temperatures, due to the high nickel and low vanadium content of this material. Even though the Standard [33] recommends that thermal ageing should not be taken into account for this type of steel, the most recent results show some non-negligible shift that should be considered and incorporated into the Standard [53].

4.2.2. Significance

Thermal ageing does not appear to be generic but depends on the heat treatment, chemical composition and service time at temperature of the material. Microstructural aspects such as grain size and the phases present may also be involved in the thermal ageing of low-alloy steels. The experimental results discussed above show that the thermal ageing mechanism should be classified as an insignificant degradation mechanism for PWR pressure vessels. In addition, it can be argued that thermal ageing degradation is at least partly taken into account in the RT_{NDT} shift prediction methodologies since all the PWR surveillance capsule specimens are irradiated at slightly higher temperatures than the RPV walls.

In addition, thermal ageing does not appear to be significant for WWER type reactors, even for materials with high phosphorous content. The results from the surveillance specimens after 10 years of operation at about 300°C as well as the results from testing of the trepans and boat samples from components aged more than 15 years have not shown any substantial transition temperature increases.

4.3. TEMPER EMBRITTLEMENT

4.3.1. Mechanism

The term "temper embrittlement" has been traditionally used to describe the embrittlement of structural steels, mostly by impurity phosphorus concentrating at the grain boundaries. Temper embrittlement is found in quenched and tempered ferritic materials, especially when a tempering temperature around 450-500°C is used. The role of phosphorus in the overall embrittlement of western-type RPV materials has been a subject of much discussion over the years. The problems have been compounded by the lack of qualified data and the variation of alloy compositions and irradiation conditions. However, the effect of phosphorus in weld metals and the heat affected zones is of concern, particularly when a thermal annealing may be applied to restore toughness. The propensity of phosphorus to migrate to grain boundaries in the RPV materials and thereby cause embrittlement under certain thermal conditions should be accounted for. The generation of a non-hardening embrittled condition is theoretically possible (called temper embrittlement) if phosphorus levels are high enough and the diffusion paths and thermal activation are available.

4.3.2. Significance for western pressure vessels

RPV steels with phosphorus content well above about 0.02 wt% may be susceptible to temper embrittlement during fabrication. However, the western RPV materials normally contained less than 0.020 wt% phosphorus. Therefore, it is unlikely that any western RPVs will exhibit temper embrittlement. If a 450°C thermal anneal of an irradiated RPV is required for recovery of the fracture toughness, the possibility of temper embrittlement should be evaluated.

4.4. FATIGUE

4.4.1. Description of mechanism

Fatigue is the initiation and propagation of cracks under the influence of fluctuating or cyclic applied stresses. The chief source of cyclic stresses are vibration and temperature fluctuations. As discussed previously, the PWR RPV is designed so that no subcomponent of the RPV is stressed above the allowable limits described in the ASME Boiler and Pressure Vessel Code, Section III (or equivalent national codes) during transient conditions and the allowable cyclic fluctuations do not violate Miner's fatigue rule. Once a crack is detected, it's behaviour under cyclic loading is analysed according to Section XI of the ASME Code or similar codes.

The RPV should be designed in such a way that no subcomponent is stressed above the allowable limit, which is a usage factor of 1. Even if the usage factors go slightly above 1, fatigue cracks are not expected because the safety factors discussed in Section 3 are used in the design.

4.4.2. Significance for western pressure vessels

The RPV closure studs have the highest usage factor of any of the subcomponents. However, the usage factor for the RPV closure studs are of the order of 0.66 for the 40-year design life. The head penetrations for the control rod drives and the vent tubes have very low fatigue usage factors. The RPV inlet and outlet nozzles also have relatively low fatigue usage factors. Unless there is some condition that results in extreme vibration to any of the RPV subcomponents, fatigue damage is considered an insignificant degradation mechanism in the assessment and management of the PWR pressure vessels.

4.4.3. Significance for WWER pressure vessels

From a fatigue point of view, the most important subcomponents of the WWER pressure vessels are the closure studs. The lifetime of the WWER-440 closure studs is limited to some 15 years of operation, when the expected usage factor will reach one. However, there is little chance of failure because these studs are tested every 4 years by ultrasonic and eddy current methods. Moreover, their exchange is a standard maintenance procedure, which is planned in advance.

The second most important WWER pressure vessel components are the primary nozzles, especially the cladding on their inner radius. However, the calculated usage factors for these locations are less than one for the whole design lifetime. And again, these parts are included in the in-service inspection performed every 4 years when ultrasonic, eddy-current and dye-penetrant methods are applied.

4.5. CORROSION

Corrosion is the reaction of a substance with its environment that causes a detectable change which can lead to deterioration in the function of the component or structure. In the present context, the material is steel and the reaction is usually an electrochemical (wet) reaction. The appearance of corrosion is governed by the so-called corrosion system consisting of the metal and the corrosive medium (the environment) with all the participating elements that can influence the electrochemical behaviour and the corrosion parameters. The variety of possible chemical and physical variables leads to a large number of types of corrosion, which can be subdivided into:

- corrosion without mechanical loading (uniform corrosion and local corrosion attack, selective corrosion attack as e.g. intergranular corrosion)

- corrosion with mechanical loading (stress corrosion cracking, corrosion fatigue)

- flow assisted corrosion attack (erosion-corrosion, flow induced corrosion, cavitation).

During the electrochemical processes, the metal ions dissolve in liquid electrolyte (anodic dissolution) and hydrogen is produced. This is the process of material loss and creation of corrosion products. When mechanical stresses or strains are also present, the anodic dissolution of the metal can be stimulated, protection layers (oxide layers) can rupture or hydrogen interaction with the metal (absorption) can be promoted which can produce secondary damage. The combined action of a corrosive environment and mechanical loading can cause cracking even when no material degradation would occur under either the chemical or the mechanical conditions alone.

TABLE XXVI TYPICAL PRIMARY COOLANT SYSTEM WATER CHEMISTRY PARAMETERS

Parameter[1]	Siemens KWU (FRG)	EPRI (USA)	Westinghouse (USA)	VGB (FRG)	J PWER (Japan)	EdF (France)	WWER 440/1000 (SU)	WWER 440 (Finland)
Lithium hydroxide	0 2–2[2]	0 2–2 2[2]	0 7–2 2[2]	0 2–2 2[2]	0 2–2 2[2]	0 6 2 2[2] 0 45–2 2[3]		
Potassium hydroxide	–	–	–	–			2–16 5[1]	2–22[4]
Ammonia	–	–	–	–			>5	>5
Hydrogen	2–4	2 2–4 5	2 2–4 4	1–4	2 2–3 15	2 2–4 4	2 7–4 5	2 2–4 5
Oxygen	<0 005	<0 01	<0 005	<0 005	<0 005	<0 01	<0 01	<0 01
Chloride	<0 2	<0 15	<0 15	<0 2	<0 05	<0 15	<0 1	<0 1
Fluoride	–	<0 15	<0 15	–	<0 1	<0 15	<0 05	<0 1
Conductivity (25°C)	<30	[2]	[2]	–		1–40[2]	4–80[2]	–
pH (25°C)	5–~8 5	[2]	4 2–10 5[2]	[2]	4 2–10 5	5 4–10 5	>6	>6
Dissolved iron	(<0 05)[5]	–	–	–	–	–	–	–
Total iron		–	–	(<0 01)[5]	–	–	<0 2	
Sulphate		0 1	–	–	–	–	–	–
Silica	(<0 5)[b]	–	<0 2	–	–	<0 2	–	–
Suspended solids	(<0 1)[b]	0 35	<1	–	<0 5	<1	–	
Aluminum	–	–	<0 05	–	–	<0 1	–	–
Calcium	–	–	<0 05	–	–	<0 1	–	–
Magnesium	–	–	<0 05	–	–	<0 1	–	

(1) Concentrations in mg/kg (ppm), conductivities in µS/cm (µmhos/cm)
(2) According to Li and B concentration
(3) According to Li and B concentration, new treatment
(4) Calculated taking into account ∑K + Na + Li
(5) Normal operating value
– Not applicable/not specified

Water chemistry control during operation, as well as during shutdown, is very important with respect to avoiding corrosion problems. Thus the content of all additives has to be carefully monitored and the ingress of impurities has to be strictly avoided, e.g. during stand still periods and maintenance work. The water chemistry regimes which are used in the primary coolant circuits of the various types of reactors and which have proven effective are presented in Table XXVI.

4.5.1. Primary water stress corrosion cracking (PWSCC) of the PWR CRDM penetrations

Most western PWRs have CRDM (control road drive mechanism) penetrations in the pressure vessel head made of stainless steel and Alloy 600. The lower portion of each penetration is made of Alloy 600, a high nickel content material, and is attached to the inside surface of the vessel head with a partial-penetration weld. The weld and the nozzle wall above the weld are part of the reactor coolant pressure boundary. The upper portion of each penetration is made of stainless steel which is joined to the Alloy 600 material with a dissimilar metal weld and joined to the CRDM housing with a screw fitting and seal weld. A typical Westinghouse-type CRDM penetration is shown in Fig. 26. The CRDM penetration design and materials are essentially the same at all PWRs except for the PWRs in Germany; this includes plants in Belgium, Brazil, China, France, Japan, the Republic of Korea, Taiwan (China), South Africa, Spain, Sweden, Switzerland and the USA. The only German PWRs with Alloy 600 penetrations are Obrigheim and Mülheim-Kärlich.

There are typically 40 to 90 penetrations, depending on plant size, distributed over each pressure vessel head as shown in Fig. 27. These may include some spare penetrations which are not fitted with CRDMs. In some plants, these penetrations also include in core instrumentation. The vessel head is spherical and the CRDM penetrations are vertical, so the penetrations are not perpendicular to the vessel surface as shown in Fig. 26, except at the centre. The uphill side (toward the centre of the head) is generally designated as the 180-degree location and the downhill side (toward the outer periphery of the head) the 0-degree location. The uphill side is also sometimes referred to as the centre side or upside, and the downhill side is sometimes referred to as the hillside or downside. The slope of the vessel head at a given penetration is called the setup or hillside angle, which is zero for the central penetration and maximum for the outer-most penetrations. Typical values for the setup angle for the peripheral CRDM penetrations vary from 38 to 50 degrees. A majority of the CRDM penetrations have thermal sleeves to guide the control rod drive shaft.

Mechanism description

PWSCC requires the simultaneous presence of high tensile stresses, a corrosive environment (in this case, high temperature water) and a susceptible microstructure [54]. The PWSCC damage rate increases as a function of stress to the power of 4 to 7, i.e.,

$$\text{damage rate} \propto \sigma^4 \qquad (20)$$

where σ is the maximum principal tensile stress, which includes both applied and residual stresses. This correlation suggests that a 50% reduction in the effective stress will result in a 16-fold decrease in the damage rate and a corresponding increase in PWSCC initiation time.

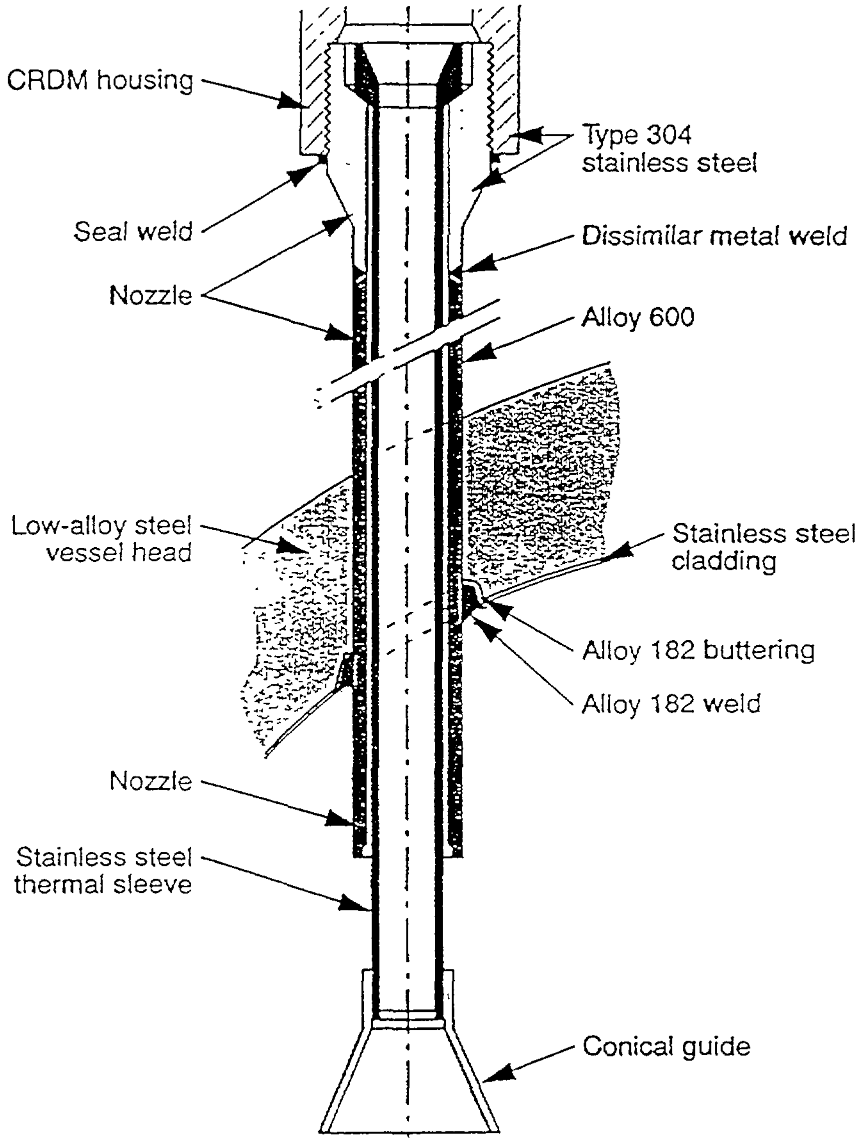


FIG 26 Typical control rod drive mechanism penetration in a Westinghouse-type PWR (Buisine et al. 1994) Copyright the Minerals, Metals & Materials Society; reprinted with permission

PWSCC is also a thermally activated process that can be described by an Arrhenius relationship of the form

$$\text{damage rate} \propto e^{-Q/RT} \tag{21}$$

where Q is activation energy, R is universal gas constant and T is temperature. Various estimates for the activation energy, Q, of Alloy 600 tube materials have been derived from laboratory studies and field experience. The estimates range from 163 to 227 kJ/mole (39 to 65 kcal/mole), with a best-estimate value of 209kJ/mole (50 kcal/mole) (Ref. [54] and references therein). Estimates for the activation energy for Alloy 600 components fabricated from bar material may be different than those fabricated from tube materials. Both the initiation and growth of PWSCC are very sensitive to temperature. For example, a PWSCC initiation time would typically be reduced by a factor of 2 for a 10°C (18°F) increase from an operating temperature of 315°C (600°F).

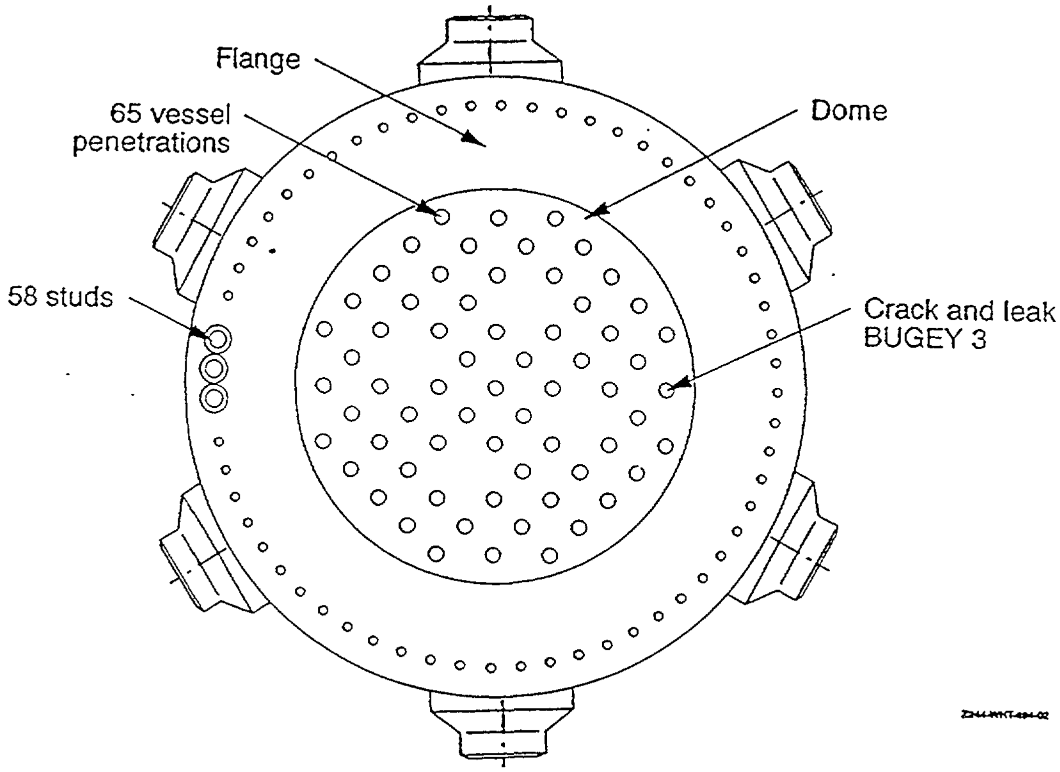


FIG. 27. Plan view of the PWR pressure vessel head (Buisine et al. 1994). Copyright the Minerals, Metals & Materials Society; reprinted with permission.

Field experience and research results show that the PWSCC resistance of Alloy 600 is highest when the grain boundaries are covered with continuous or semicontinuous carbides. For example, the PWSCC initiation time increases by a factor of five as the grain boundary carbide coverage increases from 0 to 100% [55]. The per centage of the grain boundaries covered with intergranular carbides depends on the material heat treatment temperature and time, carbon content and grain size. High temperature heat treatments which put the carbides back in solution result in good carbide coverage of the grain boundaries and a microstructure that is more resistant to PWSCC. Also, larger grain material has less grain boundary area than small grain material, so it is easier to get complete coverage with larger grains.

Primary coolant chemistry has a secondary effect on the time of PWSCC initiation. However, increasing the hydrogen concentration in the primary coolant decreases the PWSCC initiation time. Therefore, EPRI's PWR Primary Water Chemistry Guidelines [56] recommend that plant operators maintain hydrogen concentrations in the range of 25 to 35 cm^3/kg, which is near the lower end of the typically used range of 25 to 50 cm^3/kg.

Penetration fabrication and installation

The penetration material and the installation process can determine whether the penetrations are susceptible to PWSCC or not. The penetrations in many Westinghouse plants, all B&W plants, and several Combustion Engineering plants were fabricated from Alloy 600 pipes, as were the penetrations in the Swedish plants. The penetrations in all French, Swiss and Belgian plants and the remaining Westinghouse and Combustion Engineering plants were fabricated from Alloy 600 bars. One difference in the bar materials is that the French, Swiss and Belgian plants used forged bars whereas the Westinghouse and Combustion Engineering plants generally used rolled bars. A machining process was used to fabricate the penetrations. Machining introduces a thin layer of cold-worked material on the machined surfaces. The yield strength of the cold-worked material is higher than that of the base metal. In addition, machining also introduces compressive residual stresses. The penetration material was usually heat treated in the temperature range of 870 to 980°C (1600 to 1800°F) for 90 minutes or longer. One exception is the French PWRs, where the material was heat treated in the temperature range of 710 to 860°C (1310 to 1580°F) if the yield strength exceeded 343 MPa (49.7 ksi). A higher heat treatment temperature results in lower yield strength and lower residual stresses. Also, a higher heat treatment temperature (above the solution temperature) is one of the parameters that can result in a more PWSCC resistant microstructure. The Alloy-600 penetrations at Obrigheim were stress relieved during fabrication, and this is probably the reason that cracks have not been detected in this material.

The penetrations are shrunk fit into the vessel head openings by dipping them into liquid nitrogen and quickly inserting them into the openings. When the penetration returns to ambient temperature, a tight fit results. Then, the penetrations are attached to the bottom of the head with a partial penetration weld, shown in Fig. 26. These attachment welds are made with multiple passes (up to an estimated 50 passes) of Alloy 182 weld metal. Due to the geometry of the vessel head, the attachment welds are not axisymmetric, except the one for the central penetration, and therefore, the amount of weld metal deposited around the penetration is not uniform. The 180-degree location has a wider weld bead than the 0-degree location. Also, the average volume of weld metal deposited around the penetration varies from plant to plant. The standard minimum partial penetration weld size is given in Figure NB-4244(d)-1(d) of the ASME Code. Some fabricators may have used larger weld sizes to ensure that the minimum size was met. As their experience increased, the fabricators may have been able to use smaller weld sizes on later heads. In addition, the size requirements in the ASME Code were changed between the 1968 and 1971 editions to permit an alternate configuration for partial penetration weld connections. Use of the alternate configuration will reduce the depth of the weld groove by about 40% and the length of the weld leg (length of the weld in contact with the nozzle) by about 17%.

The weld metal shrinks as it cools and pulls the lower end of the penetration radially outward (because of the difference in the axial location of the 0 and 180° weld metal), bending the penetration and ovalizing its cross-section at the weld region [57]. The ovality (the difference in major and minor diameters, as a per centage of the original diameter) can be as high as 2% in the penetration cross-section at the downhill location of the weld. A penetration with a larger setup angle or larger welds has more welding-induced deformation, that is, bending of the penetration and ovalization of its cross-section. The ovalized cross-section has its major diameter along the circumferential direction of the vessel head. The deformation also causes the penetration above the weld to lose its interference fit with the head opening.

The welding-induced residual stresses on the inside surface of the penetration have been measured using a mockup of a typical CRDM penetration [58, 59]. As expected, the measurements show that the highest welding-induced stresses on the inside surface are in the peripheral penetrations, for which the setup angle is the largest. The stresses are highest at the 0-degree location, toward the periphery of the vessel head. The circumferential stresses exceed the axial stresses by about a factor of 1.6. Stresses at the 180-degree location on peripheral penetrations are lower, but the circumferential stress is still higher than the axial stress. As the setup angle decreases, the magnitude of the measured circumferential and axial stresses and the difference between them also reduce.

Axial tensile stresses are present on some portion of the nozzle outside surface and are of higher magnitudes than the corresponding hoop stresses. These stresses could initiate a circumferential stress corrosion crack on the outside surface, provided the surface is exposed to primary coolant that has leaked from a through-wall crack. Such a circumferential crack, if propagated through the penetration wall and around its circumference, would lead to a rupture of the penetration. However, this is an unlikely event because the axial stresses across the penetration wall thickness vary from tensile to compressive and, therefore, cannot support the growth of a circumferential crack. One conservative analysis shows that the crack propagation would take a very long operational time, well beyond the current license period.

Penetration environment

The penetration temperature is determined by the temperature of the coolant in the upper head. Estimated head temperatures vary from 289 to 327°C (552 to 621°F). The penetration temperatures could be affected by the bypass of the vessel inlet flow into the upper head, which varies from an estimated 5% in some Westinghouse-designed plants to 0.5% in Combustion Engineering-designed plants. Framatome is modifying the vessel internals to increase the bypass flow in some of its plants. Inasmuch as lower penetration temperatures are considered to be beneficial in mitigating PWSCC, there is still a debate over how much this small leakage flow cools the upper head. Measurement of the temperature on the outside surface of the vessel head could help in better estimating the penetration wall temperatures.

Operating experience

A CRDM penetration began leaking in September 1991 at Bugey 3, a French PWR. The leak occurred during a hydrotest (after 10 years of operation) conducted at a pressure of 20.7 MPa (3000 psi) and about 80°C (175°F) and was detected with acoustic emission monitoring equipment [57]. The leak rate was about 0.7 L/h (0.003 gpm).

Subsequent inspection revealed that the leaking crack was axially oriented and located on the downhill side at an elevation corresponding to the lowest portion of the partial penetration weld attaching the penetration to the RPV head. Several other approximately axial (within about 15 degrees of being axial) cracks were also found on the inside surface of the penetration at both the downhill and uphill locations. A sketch showing the crack locations is shown in Fig. 28. In addition to the leaking crack, there was another through-wall axial crack located at the counterbore and below the weld at the uphill location, shown in Fig. 28. This crack is in the portion of the penetration wall that does not constitute a part of the primary pressure boundary.

Destructive examination of the damaged penetration material revealed that a through-wall PWSC crack was initiated on the inside surface (downhill location) at the upper corner of the counterbore; it was 25 mm (1.0 in.) long on the inside surface and 2 mm (0.08 in.) long on the outside surface. The crack length was greater underneath the inside surface; its maximum value was 52 mm (2.0 in.) [60]. The crack also penetrated the Alloy 182 weld metal over a length of about 15 mm (0.6 in.) and to a maximum depth of 2.7 mm (0.1 in.) The examination also revealed oxidation at the crack tip, which implies that the through-wall

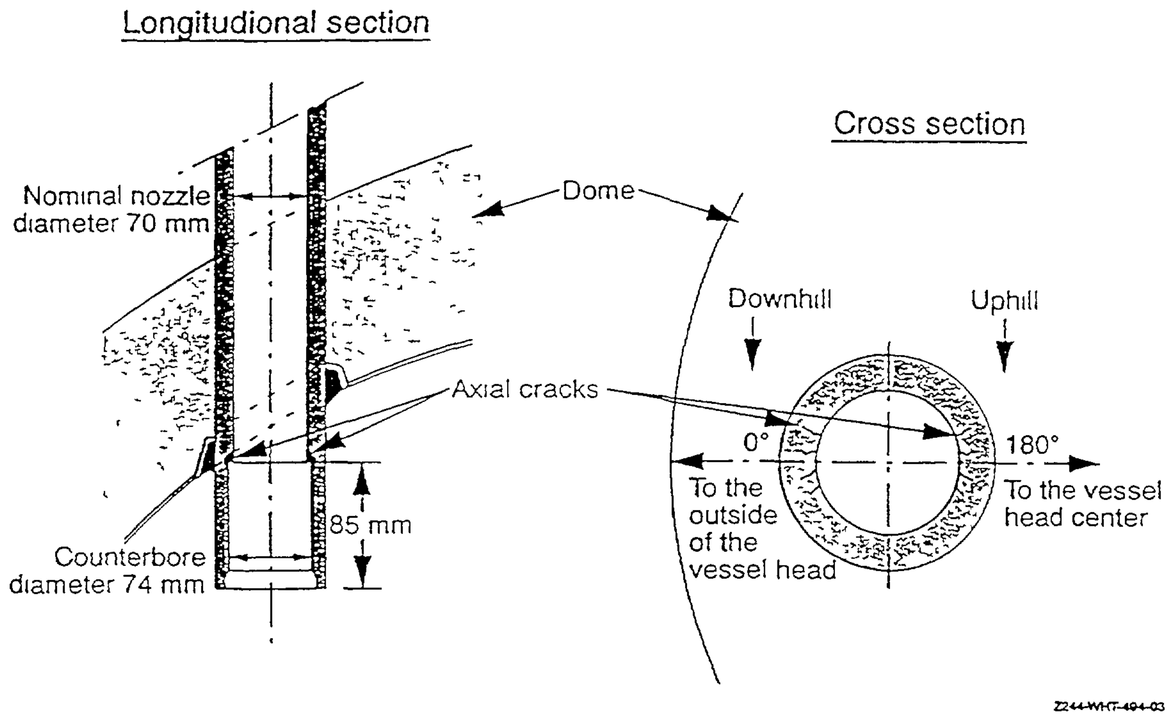


FIG 28 Location of cracks on the Bugey 3 CRDM nozzle (Buisine 1994) Copyright the Minerals Metals & Material Society reprinted with permission

TABLE XXVII WORLDWIDE VESSEL HEAD PENETRATION PWSCC INSPECTION RESULTS [62][(1)]

Country	Plants inspected	Total penetrations	Penetrations inspected	Penetrations with indications
Belgium	7	435	435	0
Brazil	1	40	40	0
France	47	3225	3213	105
Japan	17	960	834	0
South Africa	1	63	63	0
Spain	5	325	102	0
Sweden	3	195	190	7
Switzerland	2	72	72	2
USA	4	249	197	2
TOTALS	87	5565	5146	116

(1) Available inspection results as of January 1996

crack was present prior to the hydrotest [58]. In addition, there were no boric acid deposits on the surface of the head opening, indicating that significant leakage did not occur during operation [57].

The destructive examination also revealed the presence of two circumferential cracks on the outside surface of the penetration: one in the weld, which was found to be a *hot crack* resulting from the original welding process; and one in the base metal, which was connected to the axial through-wall crack. The crack in the base metal was on the downhill side just above the weld, making an angle of 30 degrees with the horizontal plane. The crack was about 3 mm long and 2.25 mm deep. Further examination verified that the crack was caused by PWSCC [61]. It appears that the primary coolant may have leaked through the axial crack into the annular region between the nozzle wall and the vessel head and caused PWSCC on the outside surface, or the circumferential crack in the base metal was part of the through-wall axial crack.

After the leakage was detected at Bugey 3, the CRDM penetrations at plants in Belgium, Brazil, France, Japan, South Africa, Spain, Sweden, Switzerland and the USA were inspected. The results of the those inspections are summarized in Table XXVII [62]. By the beginning of 1996, approximately 5150 CRDM penetrations had been inspected and indications of cracks were found in approximately 2% of the inspected penetrations. Most of the indications were found in the French PWRs [62, 63], in part, because more French penetrations have been inspected; however, indications were also found in Sweden, Switzerland and the USA. If the inspection results from France are excluded, the per centage of penetrations with indications in the remainder of the world is approximately 0.5%. The somewhat higher defect rate in France may be due to differences in the material fabrication methods.

Most of the penetration cracks were short (less than 25 mm) and axial, making a small angle with the vertical, initiated on the inside surface and located at either the uphill or downhill side of the peripheral penetrations and near the partial penetration weld. The maximum angle of inclination was about 30 degrees. Circumferential indications were found at three plants. These indications were on the outside surface of penetrations at Bugey 3, in the attachment weld at Ringhals 2 and at Zorita [64]. The indication at Ringhals 2 was in the weld and consisted of multiple unconnected small cracks that did not form a large continuous crack. Most likely, they were manufacturing defects and they are now repaired for limited operation until the head is replaced. None of the detected cracks located at or above the attachment weld was through-wall except the one at Bugey 3 which leaked during the hydro test. However, a few cracks in the French penetrations were through-wall (less than 5) but located below the welds in the region of the penetrations which is not part of the reactor coolant pressure boundary [63, 65].

CRDM penetration cracking that was probably caused by chemical attack due to intrusion of demineralizer resins containing sulphur into the reactor coolant system has occurred in the 160-MW, one-loop Spanish PWR, Zorita. A total of 171 crack indications were found in 34 of the 37 Zorita penetrations [64, 66]. Most of the indications were axial and located in the free span region of the penetrations rather than near the attachment welds. These indications were not included in Table XXVII because the cracking was not PWSCC and the Zorita water chemistry excursion is not typical of what might occur in most PWRs.

Safety significance

The inspection results have shown that most cracks are initiated on the inside surface of the penetration and have an axial orientation, but in at least one case the crack was on the outside surface and had a circumferential orientation. (As discussed above, the outside circumferential crack was caused by the presence of an axial throughwall crack.) Crack growth analyses show that these cracks are not likely to lead to a catastrophic failure of the penetration because Alloy 600 is ductile and, therefore, the critical flaw length is long [67–70]. One analysis shows that a through-wall axial crack has to grow some 216 mm (8.5 in.) above the vessel head before it can cause penetration rupture [71]. The analyses also show that the stresses in the penetration are not high enough to support such crack growth away from the partial penetration weld. In addition, a leak would be detected long before the crack reaches a critical flaw size.

The safety significance of circumferential cracks initiated on the outside surface may also be limited because the crack growth rates are likely to be low. The circumferential crack has to grow through the thickness and around the circumference before the penetration can rupture. The results of one analysis show that such crack propagation would take much more than the current license period [72]. Other analyses have shown that short circumferential cracks on the outside surface are possible; however, these cracks are not expected to become through-wall and cause rupture because of the comprehensive axial stresses present in front of the cracks [73].

Limited field experience suggests that, during normal operation, leakage of the primary coolant from a through-wall axial crack is unlikely because of the tight fit between the penetration and the reactor vessel head will prevent opening of the crack and will restrict the leakage. (Note that no boric acid deposits were detected at the Bugey 3 plant, where leakage was detected only during a hydrotest; this means that little or no leakage took place during operation.) However, if a leak occurs it will be at least 9 years, according to one analysis, before the boric acid corrosion of the vessel head could challenge the structural integrity of the head [71]. It is very unlikely that such leakage could remain undetected.

There has been little or no experience feedback on certain other Alloy-600 components such as the radial keys, vent nozzles, or bottom head penetrations. Some investigations of the bottom penetrations are still in progress, mainly in France. The bottom penetrations operate at lower temperatures than the CRDM penetrations and are generally stress relieved, but some have been installed without stress relief and were distorted during the fabrication process. These penetrations may be susceptible to stress corrosion cracking.

To date no cracks have been found in the Alloy 82 or 182 weld material.

Stress corrosion cracking of WWER pressure vessel components

Nickel based alloys have not been used in the WWER pressure vessels. The cladding, penetrations and welded joints between the head materials and austenitic tubings (for control rods instrumentation, etc.) are made of austenitic materials of the same type as the cladding itself, i.e. the first layer/bead on the ferritic material is Type 25/13 material and the upper layers are stabilized Type 18/10 austenitic stainless steel.

4.5.2. General corrosion and pitting on the inside surfaces

Corrosion can commonly lead to uniform material loss, shallow pit formation, pitting or selective attack at the surface. Often the metal is relatively uniformly removed. However, when there are inhomogeneities at the metal surface and/or local differences in the electrochemical reactivity of the environment, the creation of local cells is possible which commonly results in local corrosion attack, causing shallow pit formation or severe pitting. Pitting of chromium or chromium nickel alloyed steels is mainly caused by the action of chloride ions. Pitting is often combined with transgranular stress corrosion cracking of austenitic stainless steel material. Such incidents occurred for example with the sealing surfaces of the nozzle flanges (close to the O-ring) in some WWER pressure vessels. In the case of selective corrosion, the attack is concentrated on distinct material phases or regions

TABLE XXVIII. WWER-440/V-230 RPVs

PLANT	BOL	CLADDED	ANNEALED	DUMMIES
KOLA 1	1973	N	1989	1985
KOLA 2	1974	N	1989	1985
ARMENIA 1[(1)]	1976	N	1988	N
ARMENIA 2	1979	Y	N	N
NOVOVORONEZH 3	1971	N	1987, 1991	N, LLCAA
NOVOVORONEZH 4	1972	N	1991	N, LLCAA
KOZLODUY 1	1974	N	1989	1987
KOZLODUY 2	1975	N	1992	1988
KOZLODUY 3	1980	Y	1989	1987
KOZLODUY 4	1982	Y	N	N, LLC 1986
BOHUNICE 1	1978	Y	1993	1992, LLC 1983
BOHUNICE 2	1980	Y	1993	1985, LLC 1984
GREIFSWALD 1[(2)]	1973	N	1988	1986
GREIFSWALD 2[(2)]	1974	N	1990	N
GREIFSWALD 3[(2)]	1977	Y	1990	1986
GREIFSWALD 4[(2)]	1977	Y	1990	1986

(1) shut down 1989
(2) shut down 1990
LLCAA low leakage core after annealing
LLC low leakage core
Y yes
N no
BOL beginning of operating life

along the grain boundaries of the metal A well known type of selective corrosion is the intergranular corrosion of sensitized austenitic stainless steel, which in principle can be neglected in PWR pressure vessel environments due to the reducing atmosphere

The interior of the western RPVs are clad with austenitic stainless steel that provides good general corrosion resistance in the PWR environment (the metal loss rate caused by uniform corrosion attack is smaller than 5 μm per year) Even in the one known case where one region of the RPV (the beltline region) was unclad, due to a poor cladding process, the general corrosion rate was so low that we have concluded that general corrosion is not a significant factor in western RPV service life

Most of the WWER 440/V-213 and V-230 RPVs are protected against corrosion by a relatively thick (8 mm) austenitic stainless steel cladding, with stabilized austenitic material used for the outer layers However, there are some WWER 440/V-230 RPVs which do not have their inside surfaces clad with stainless steel (see Table XXVIII) These low alloy steel vessels are therefore exposed directly to the primary coolant All unalloyed or low alloyed ferritic steels are subject to the formation of a magnetite protection layer as a consequence of the reaction between the water and the steel (iron) at operating temperature Nevertheless, large scale surface corrosion and pitting has been observed in most of these vessels (in the core region and in the nozzle to safe-end zone) This corrosion was caused by the oxygen pick-up during shutdown periods which remained in the primary system for a period after startup

Significance

As long as the water chemistry regime is controlled within its specified limits, general corrosion, pitting and selective corrosion on the inside surface is not a severe matter of concern for the ageing management of the RPV Care has to be taken to protect unclad surfaces during shutdown periods and to use maintenance auxiliaries to avoid the ingress of impurities in unacceptable concentrations

4.5.3. Boric acid corrosion of outer surfaces

Aerated solutions of boric acid can attack carbon and low alloy steels If a leak exists somewhere in the vicinity of the RPV head, boric acid corrosion of the low alloy steel plates or forgings is possible Boric acid leakage can result in very high and localized corrosion rates, i e , mm per year Sporadic leakage has been observed from flanges (O-ring seals), closure studs and instrumentation tubes of western PWRs Sporadic leakage has also been observed from some of the WWER control rod drive nozzle flanges and from around the threads for the vessel head closure studs To exclude more leakage of this type, the nickel O-rings used in the WWER control rod drive nozzle flanges were exchanged for graphite O-rings While boric acid leakage is not considered a safety issue, it can become an economic problem Therefore, boric acid leakage should be considered a significant degradation mechanism in the assessment and management of the PWR pressure vessels

Mechanism

Boric acid is a relatively weak acid, however, when it leaks onto a hot surface, the water evaporates and leaves behind a concentrated solution of boric acid and, finally boric acid crystals. A saturated solution of boric acid at 95°C (200°F) has a pH of less than 3 and is very corrosive, causing localized areas of carbon or low-alloy steel components to dissolve and corrode. At some point, the corrosion reaction stops or slows down, but the steel has already been damaged [74].

In most cases, boric acid crystals build up near a leak. Significant corrosion of the area beneath the crystals has been observed. The crystallized dry form of boric acid is not considered benign. A constant state of wetting and drying renders the crystals an active part of the corrosion process. Corrosion rates of 0.2-0.5 mm (10-20 mils) per year have been reported for this condition.

Another mechanism for boric acid corrosion is galvanic interaction between dissimilar metals, as are often used in dissimilar metal welds. If such a weld becomes wetted with a boric acid solution, the carbon steel is likely to corrode quickly.

As for crevice corrosion, tests have shown that any small amount of boric acid concentrated in a crevice is likely to boil out or become consumed by the reaction and therefore be short lived. Preferential attack of these regions, therefore, is not expected to occur. The carbon steel in the area will still dissolve, however.

Operating experience

The EPRI has surveyed the problem of boric acid corrosion, searching various databases and contacting plants [74]. The survey included licensee event reports, Institute of Nuclear Power Operations Nuclear Network data, NRC publications and technical literature describing the results of test programmes. Another EPRI report concentrated on the degradation and failure of bolting in NPPs and recommended corrective actions [75]. The study concluded that materials resistant to boric acid corrosion were not always satisfactory for fasteners because of inadequate strength or other problems. Coatings and platings also have not always proven satisfactory in preventing boric acid corrosion. The study made the key point that preventing reactor coolant leaks was the best protection against the threat of boric acid corrosion. Additional information can be found in Refs [76–79].

Some of the worst cases of reported boric acid corrosion of RPB components are summarized below.

Turkey Point Unit 4

In 1987, the plant operating staff found over 230 kg (500 pounds) of boric acid crystals on the RPV head. They also found crystals in the exhaust cooling ducts for the CRDMs. After removing the boric acid and steam cleaning the head, the plant staff noted severe corrosion in several areas [74].

The cause of the boric acid buildup was a leak from a lower instrument tube seal (conoseal) onto one of the in core instrument tubes. The "small leak" was noted during an outage in August 1986 because of the buildup of some boric acid crystals. In October 1986, during an unrelated shutdown, the staff found about 0.03 m^3 (1 cubic foot) of boric acid crystals on the head; they subsequently removed the crystals. In both cases, the staff deemed the leak rate acceptable for continued operation. The borated reactor coolant leaking from the

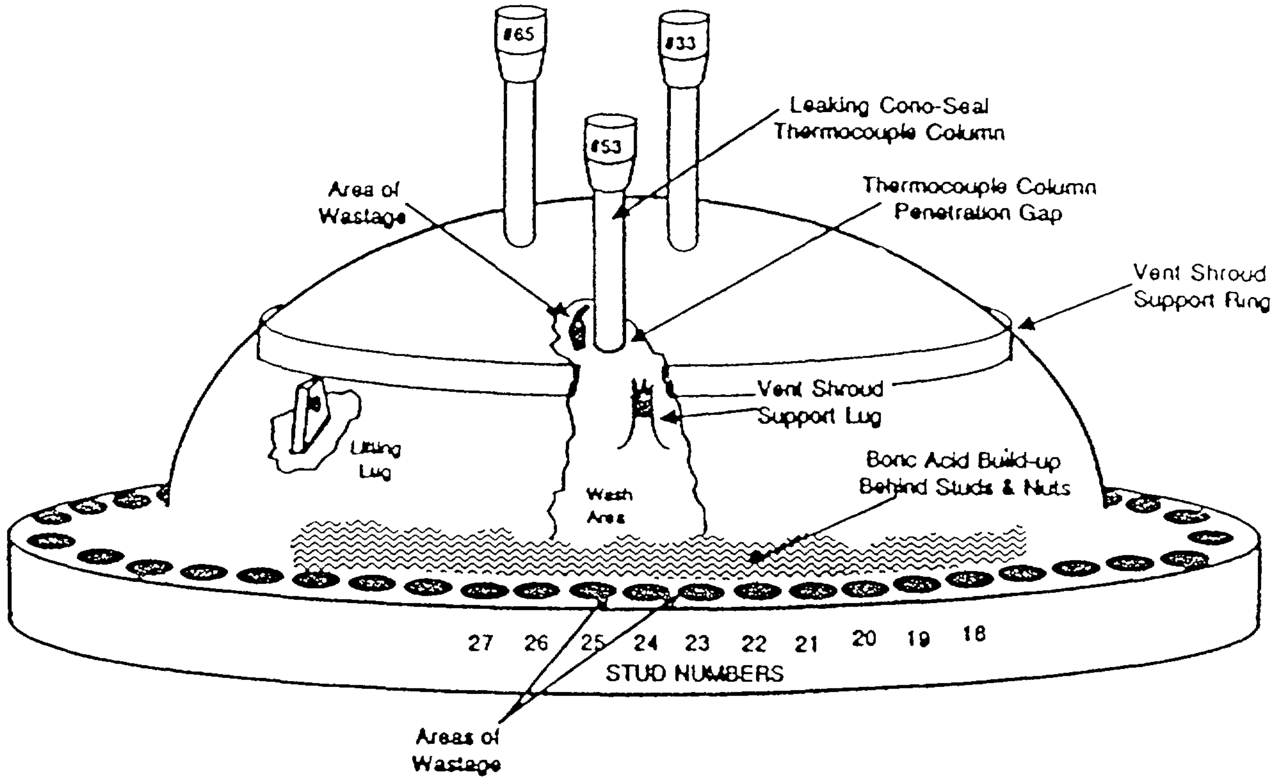


FIG 29 A sketch of the Turkey Point Unit 4 vessel head showing the areas affected by boric acid corrosion

conoseal flowed down the head insulation and beneath the insulation to the exposed head This caused damage to the head, the conoseal clamps and some of the head bolts Three of the 58 head studs were so corroded that the bolts and nuts had to be replaced

Additionally, vapors containing boric acid had been borne into the CRDM cooling coils and ducts and condensed there, forming crystals The control rod drive cooling shroud support was also so severely corroded it had to be replaced

During extensive inspections of the entire head area, the plant staff found either heavy deposits and/or general corrosion in many areas. Several components had wasted away [74] Figure 29 shows the corroded areas on the Turkey Point head.

Salem Unit 2

During an unplanned cold shutdown in 1987, the plant staff found boric acid crystals on a seam in the ventilation cowling around the head An inspection team removed some of the cowling and insulation and discovered a pile of boric acid residue near the head The size of the pile was about 0 9 × 1.5 × 0 3 m Pitting was found beneath the deposit. Nine pits ranged from 25 to 76 mm in diameter by 9 to 10 mm deep The minimum thickness of the material in these areas still exceeded the minimum required design thickness

The boric acid buildup was attributed to a leak in a seal weld at the base of the threaded connection for the thermocouple instrumentation Borated water had leaked onto the head from ventilation supports.

Ringhals Unit 2

The plant staff noted a somewhat higher leakage rate than usual from the primary system during the summer of 1985. The leakage rate was about 2–3 times higher than expected but well within the limit of the technical specifications. Despite extensive searching, the staff could not find any leaking valve and thus assumed that maintenance of the steam generator had been less effective than usual and attributed the higher rate to steam generator leakage.

During startup after a shutdown in December 1985 for preventive steam generator maintenance, inspection showed that the reactor flange was leaking. The head was lifted and the reactor flange and head flange were cleaned and inspected. The reactor flange had some minor defects in the groove for the O-rings. Four to six head studs had been affected by the leakage. The studs were cleaned and inspected.

4.6. WEAR

Wear is defined as the motion between two surfaces that results in the removal of material surface layers. Wear occurs in parts that experience intermittent relative motion. Wear may occur due to either flow induced vibration or displacement of adjacent parts. Incore instrumentation tubes (flux thimble tubes) in some Westinghouse plants have exhibited wear due to flow induced vibration. However, wear is not considered to be a significant ageing mechanism because wear between two surfaces subjected to relative motion is readily detectable by visual inspection long before the effects of wear begin to compromise the structure integrity or function of the component.

5. INSPECTION AND MONITORING REQUIREMENTS AND TECHNOLOGIES

5.1. NDE REQUIREMENTS

5.1.1. Requirements in the USA

RPVs in the USA are inspected in accordance with Section XI of the ASME Code [21]. There are three types of examinations used during in-service inspection: visual, surface and volumetric. The three types of in-service inspections are a carry-over from the pre-service inspection (PSI) that is required for the RPVs. Inspection plans are prepared for the PSI (if required), the first in-service inspection interval and subsequent in-service inspection intervals.

Each NPP follows a pre-service and in-service inspection programme based on selected intervals throughout the design life of the plant. The RPV inspection category is described in Table IWB 2500-1 of Section XI of the ASME Code, which details the inspection requirements. The in-service inspection intervals are determined in accordance with the schedule of Inspection Programme A of IWA-2410, or optionally Inspection Programme B of IWA-2420. Programme A is modeled on the traditional bi-modal distribution which is based on the expectation that most problems will be encountered either in the first few years of operation or late in plant service life. Programme B is modeled on the expectation that plant problems will be uniformly distributed with respect to time. For Programme B, 16 per cent of the required inspections are to be completed in the third year, another 34 per cent of the required inspection by the seventh year and the remainder by the tenth year of operation.

All shell, head, shell-to-flange, head-to-flange and nozzle-to-vessel welds and repair welds (repair depth greater than 10% of wall thickness) in the beltline region must be subjected to a 100% volumetric examination during the first inspection interval (over 3 to 10 years). Successive inspection intervals also require 100% volumetric examination of all of these welds. The nozzle inside radius sections must all be subjected to a volumetric examination during each of the four inspection intervals. The external surfaces of 25% of the partial-penetration nozzle welds (CRDM and instrumentation) must have a visual examination during each inspection interval (leading to total coverage of all nozzles). All of the nozzle-to-safe end butt welds with dissimilar metals (i.e., the ferritic steel nozzle to stainless steel or Alloy 600 safe end weld) must be subjected to volumetric and surface examinations at each interval. All studs and threaded stud holes in the closure head need surface and volumetric examinations at each inspection interval. Any integrally welded attachments must have surface (or volumetric) inspections of their welds at each inspection interval.

Thus, the inspection plan for the RPV results in close monitoring of potential fatigue-crack formation and growth in all the relevant welds. Any additional monitoring and recording of transients is usually done in accordance with the plant technical specifications.

Many vessels in older plants were fabricated prior to Section XI of the ASME Code, hence these older plants did not undergo a PSI in accordance with the current rules. However, all of the RPVs in the USA have undergone in-service inspections in accordance with Section

TABLE XXIX. INSPECTION TECHNIQUES IN GERMANY — SCOPE AND INTERVALS FOR RPVs

Area to be inspected	Inspection method/technique	Orientation of flaws	Scope of inspections	Inspection intervals
Longitudinal and circumferential welds	UT single-crystal angle probe, UT tandem, UT special technique	longitudinal transverse	All welds, entire length, 100% including surfaces and near-surface areas	4 years
Nozzle-to-pipe and nozzle-to-shell welds ^{3}250 mm diameter	UT single-crystal angle probe, UT tandem or single-crystal special technique	longitudinal transverse	All welds, entire length, 100% including surfaces and near-surface areas	4 years
Nozzle inner radius ^{3}250 mm diameter	UT special technique	radial	All inner radius surfaces and near-surface areas of all nozzles	4 years
Ligaments in nozzle arrays	UT single-crystal angle probe, UT special technique	radial	All ligaments, 100% including surfaces and near-surface areas	4 years
Threaded studs	UT special technique or magnetic testing or eddy-current testing	perpendicular to stud axis	Surfaces and near surface areas of all studs; entire pre-tensioned length including thread region	At least 25% of threaded studs and associated blind tapped holes, nuts and washers during a 4-year period; 100% inspection to be completed within 3 successive 4-year inspection intervals
Blind tapped holes	UT single-crystal special technique or eddy-current testing	perpendicular to thread axis	Surfaces and near-surface areas of all blind holes, entire thread	At least 25% of threaded studs and associated blind tapped holes, nuts and washers during a 4-year period; 100% inspection to be completed within 3 successive 4-year inspection intervals
Nuts	Visual inspection	–	Threaded area and contact area (bearing surface) of all nuts	At least 25% of threaded studs and associated blind tapped holes, nuts and washers during a 4-year period; 100% inspection to be completed within 3 successive 4-year inspection intervals

TABLE XXX. KINDS, METHODS, AND TECHNIQUES OF INSPECTION IN GERMANY

Kind of inspection	Method	Technique
1. Inspection for surface breaking cracks and surface-near cracks	Magnetic testing Penetrant testing Electromagnetic testing	Magnetic particle inspection, magnetographic inspection, or eddy-current
	Ultrasonic testing	Special techniques, e.g., surface waves, mode conversion, corner effect, or TRL-wave probes
2. Inspection of volume	Ultrasonic testing	Impulse-echo-technique with zero degree or angle, tandem technique, or special techniques
	Radiographic testing	X rays or isotope rays
	Electromagnetic testing (for thin walls)	Single frequency or multi-frequency
3. Visual inspection	Visual testing	–
4. Pressure test	–	–
5. Function test	–	–

XI of the ASME Code. In addition, the majority if not all the RPVs in the western world have undergone in-service inspection in accordance with Section XI of the ASME Code or local rules similar to Section XI. Vessels fabricated before Section XI was issued have been reconciled to the requirements of Section XI using the results of a subsequent in-service inspection. The resolution of indications or flaws detected during a scheduled inspection is discussed below. Periodically, indications or flaws have been detected during an inspection. The indications or flaws have been evaluated in accordance with the acceptance standards of IWB-3500, Section XI, ASME Code. To date, all indications have been found acceptable by either the standards given IWB-3500 or by analysis in accordance with Appendix A to Section XI of the ASME Code. There has never yet been an occasion when a PWR RPV had to have a flaw removed during service and undergo a weld repair.

5.1.2. Requirements in Germany

ISI in Germany dates back to the late 1960s, when a large research and development programme funded by the Federal Ministry for Research and Technology was launched. In 1972, a draft version for the Inservice Inspection Guidelines [80] of the Reactor Safety Commission was published and this document remained almost unchanged in the subsequent issues. This became the basis for the formulation of the German KTA 3201.4 Code [81], which today specifies the NDE requirements for ISI.

The inspection scope and the NDE-methods to be applied to a RPV are listed in Tables XXIX and XXX. The ISI includes all welds, the nozzle radii, the control rod ligaments in the top head, the studs, nuts and threaded stud boreholes. The inspection intervals for the RPV are 4 years (for conventional vessels it is 5 years); however, the scope of an inspection may be subdivided and each part carried out separately during the 4-year period, e.g., each year at the refuelling outage for BWRs.

The inspection technique usually used is UT. The tandem technique is required for wall thicknesses larger than 100 mm. The inspection method and techniques have to be chosen to detect all safety relevant flaws in the planes perpendicular to the main stresses, the planes parallel to the fusion lines of the welds and the planes perpendicular to the welds.

UT inspection sensitivity — detection level

The detection level is defined as the limit of the echo height which can be detected over the noise observed when moving the probe (see Fig. 30). Therefore, it is mainly a function of the inspection technique and the data acquisition equipment should have little, if any, negative influence.

UT inspection sensitivity — recording level for subsurface defects

The size of defects to be detected is not defined explicitly in the code; but implicitly the authors had in mind that embedded flaws of 10 mm diameter should be detected in walls greater than 40 mm thick. Therefore, they defined the 10 mm diameter flat bottomed borehole as the recording level for specular reflection, e.g., by the tandem technique, and a 3 mm diameter flat bottom borehole as the recording level for detection using flaw tip scattering and diffraction techniques. A comparison of these reflectors with the ASME Section XI recording level sensitivity in terms of echo height is given in Fig. 31.

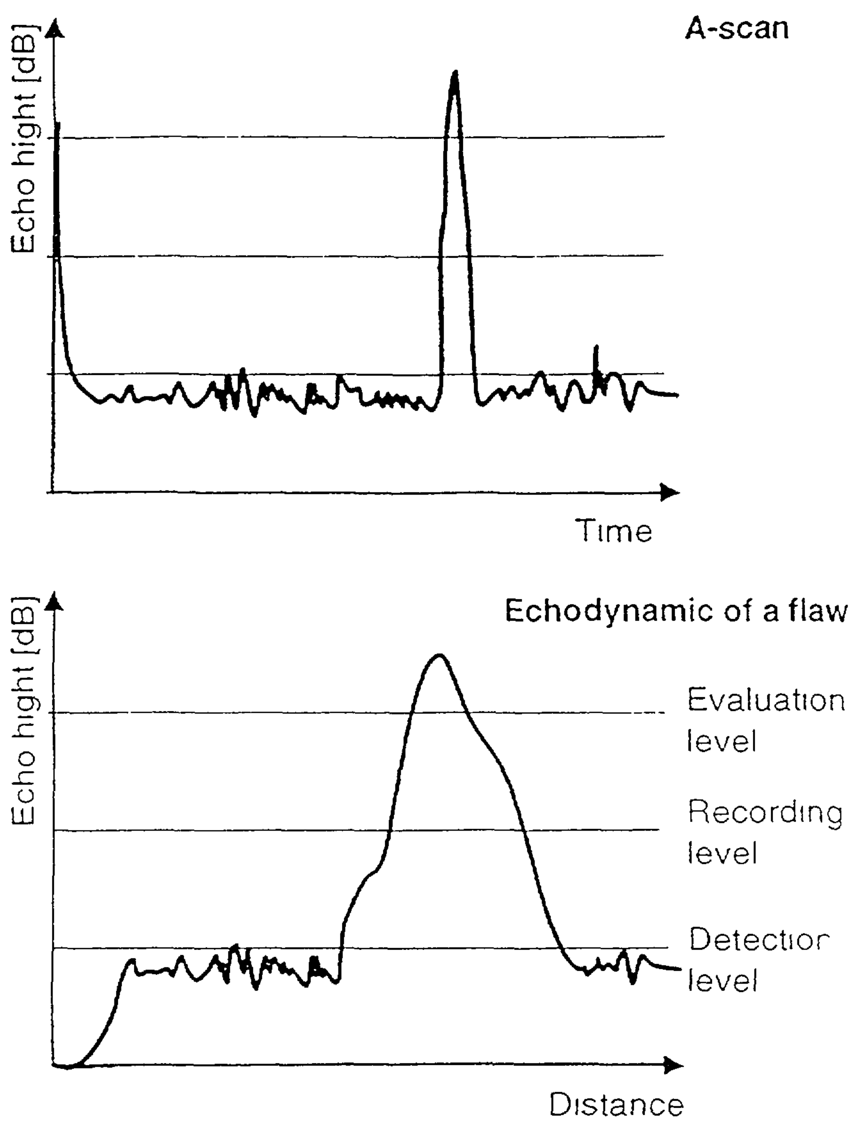


FIG 30 Schematic diagram of the different amplitude levels of indicators The upper sketch displays the signal amplitude as function of time, the so called A-scan, the usual screen picture from an UT - instrument In most cases, only the high signal requires further evaluation The lower sketch displays the flaw signal as function of the probe movement and illustrates the detection level as defined by the surrounding noise signals This is the relevant level for the sensitivity of any automated inspection technique

UT inspection sensitivity — recording level for surface defects

The calibration and sensitivity settings for surface inspection have to be done with notches. The depth of these calibration notches depends on the wall thickness and is 3 mm for wall thicknesses larger than 40 mm. The notches must have reflecting planes perpendicular to the surface: i.e., rectangular or triangular notches or spark eroded slots.

All indications with an echo height of the above mentioned notch echo, minus 6 dB or more have to be recorded The influence of the cladding or the structure on the signals has to be evaluated using the component and has to be considered during data acquisition and data evaluation. For the commonly used 70° transmitter-receiver-longitudinal (TRL) wave-probe-technique, the echo height is comparable to a 6 mm diameter flat bottom borehole or more,

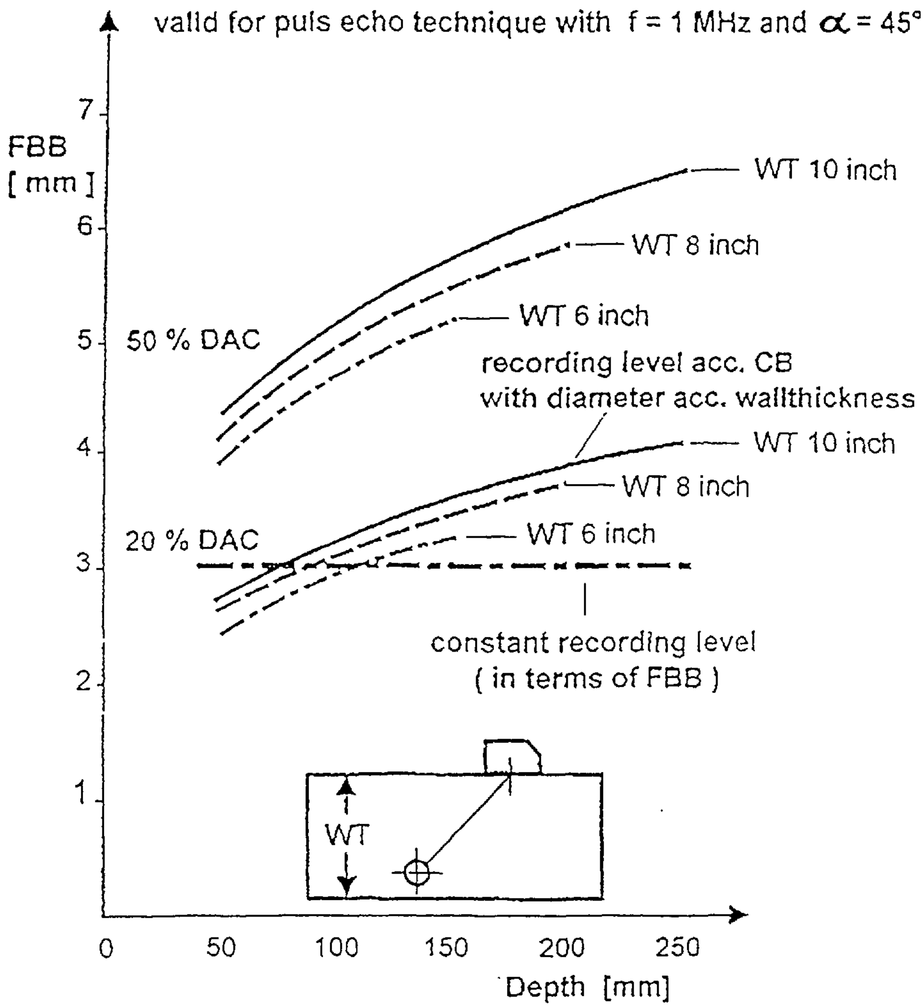


FIG. 31. Comparison of recording levels required by the ASME and KTA 3101.4 Codes for the case of a pulse echo technique with a 45 degree angle and a 1 MHZ frequency. The KTA-required recording level of a 3 mm flat bottom borehole (FBB) is constant over the wall thickness. The echos from the cylindrical boreholes follow a less strong distance dependence than do the echos from a FBB and therefore the recording level rises (the sensitivity decreases) with an increasing sound path.

and is near the echo height of a backwall. Since 1991, the recording level has been lowered 6 dB below the echo height of the notch, to give more distance to the backwall echo, but in some inspections from the outside, this level is also near to the noise from the cladding.

UT inspection sensitivity — evaluation level

It is the philosophy of KTA 3201.4 that important indications should be evaluated and compared with earlier data to check on any possible growth of the flaws. Surface and near surface indications must be evaluated if the indication has an echo height 6 dB over the recording level or more, or if the indication has an echo height over the recording level and the indication length is more than half the wall thickness, or more than 50 mm. The method to evaluate the indication length has to be agreed with the TÜV. Subsurface indications must be evaluated if the same criteria are met, but the length of the reflector is measured by the length of the indication at −12 dB under the recording level.

TABLE XXXI INSPECTION TECHNIQUES IN FRANCE — SCOPE AND INTERVALS FOR RPVs

Area to be inspected	Inspection method/technique	Orientation of flaws	Scope of inspections	Inspection intervals
Circumferential welds core shells/ shell-flange/ nozzle-shell/shell-bottom head	Underwater UT	longitudinal/ transverse	100% welds and wear surface areas	Each CV[1]
Flange-head circumferential weld	UT	Longitudinal/ transverse	–	Between first and second CV
Underclad inner surface	Underwater UT	Longitudinal/ transverse	Irradiated area first 30 mm from the inner surface	After 20 years
Cladding surfaces	Visual/televisual	–	All the inner surface	After 20 years
Threaded stud boreholes	Televisual	–	–	If stud deterioration, every 2 years
Ligament in thread-hole area	Underwater UT	Longitudinal/ transverse	–	If stud deterioration, every 2 years
Radial key welds	Televisual	–	–	Each CV
CRDM and vent penetrations	Eddy-current, UT	–	Inner surface	Every year
Studs	Eddy-current	–	–	Three times in 10 years
Nuts	Eddy-current	–	–	Three times in 10 years
CRDM penetration welds, BMI penetration welds	Televisual + acustic emission	–	–	

[1] The first complete visit (CV) is after the first hydro test, the second CV is before 30 months of operation, the third is before 10 years of operation, and the rest are every 10 years

Procedure for indications above evaluation level

The Code requires the analysis of indications above the evaluation level when they are found for the first time or if it is suspected that they are growing. Indications above the evaluation level must be compared with the results of the last inspection. If there is a change to a higher amplitude or a longer length beyond the usual tolerances, the results of all earlier inspections are compared to see if there has been a change in the course of the time. If there is evidence of a new or growing indication, one has to analyse the data for evidence of the kind, position and size of the flaw. New measurements with specialized techniques may be necessary. If it is thereby confirmed that the defect is new or has grown, then it is necessary to find its root cause and prepare a safety analysis using, for example, the operation records. The safety analysis may include: fracture mechanics analysis, experimental investigations and evaluations of the experience at other plants. The fracture mechanics analysis method (analysis of brittle fracture) applied for the RPV is dealt with in KTA 3201.2. (The ASME, Section XI, procedure could also be used.) The safety factors and the crack growth velocity are usually taken from ASME Section XI.

The results of the safety analysis should determine whether the flaw can be accepted in the component or not; there is no general acceptance level independent of the specific circumstances.

5.1.3. Requirements in France

The requirements for in-service inspection programmes in France are published in RSE-M[30]. The Code requires periodic hydrotests with acoustic emission monitoring during the hydrotests, NDE during the outages, a material surveillance programme, loose-parts (noise) monitoring during operation, leak detection during operation and fatigue monitoring. The Code specifies a complete programme including both the utility and regulatory agency inspections. Areas of the RPV that must be inspected are listed in Table XXXI and include the beltline region of the shell, all the welds, the top and bottom heads, the nozzles, the penetrations, the control rod drive housings, the studs, the threaded holes and the supports.

One of the major differences between the in-service inspection programme in France and programmes in other countries is hydrotesting. A hydrotest at 1.33 times the design pressure (22.4 MPa) is required after the RPV fabrication is completed. A hydrotest at 1.2 times the design pressure 20.4 MPa) is then performed after every 10 years of operation. The 10-year internal tests must be performed at a temperature of RT_{NDT} + 30°C.

The NDE techniques which are used in France are also listed in Table XXXI and include focused under water UT, radiography, visual examinations, tele-visual examinations under water, die-penetrant tests, acoustic emission monitoring and eddy-current testing (ECT). Ultrasonic testing of the welds generally covers the weld area plus about 50 mm of base metal on both sides of the weld. Base metal regions of the RPV shell subjected to fluences above 10^{22} n/m^2 are also inspected with ultrasonics, generally at a depth of 7 to 25 mm from the inside surface.

5.1.4. Requirements for WWERs

The WWER RPV in-service inspection is carried out at least every 4 years (30 000 hours) of operation and includes NDE (visual, dye-penetrant, magnetic particle, ultrasonic

TABLE XXXII. WWER REQUIREMENTS FOR NDE

REACTOR TYPE REACTOR PART	WWER-440/V-230	WWER-440/V-213	WWER-1000/V-320
CYLINDRICAL PART CORE BELTLINE	INSIDE	OUTSIDE/INSIDE	OUTSIDE/INSIDE
BOTTOM HEAD	INSIDE/OUTSIDE	INSIDE/OUTSIDE	INSIDE/OUTSIDE
NOZZLE AREA	INSIDE/OUTSIDE	OUTSIDE/INSIDE	OUTSIDE/INSIDE
SAFE-ENDS	INSIDE(1)/OUTSIDE	INSIDE/OUTSIDE	INSIDE/OUTSIDE
CLADDING	INSIDE	INSIDE	INSIDE

(1)Only in some NPPs.

and eddy-current), surveillance specimen evaluation and hydraulic testing. Parts and sections of the reactor to be inspected, locations, volume and periodicity are specified in the procedure. A change to an 8-year inspection interval for examination of the RPV inner surface is now under consideration by the regulatory bodies in the Czech Republic, Hungary and Slovakia.

Examination of the RPV base and weld metal in the zones with stress concentrations or high neutron flux, the cladding/base metal interface, the nozzle transition areas, sealing surfaces, outer and inner surfaces of the vessel bottom and top heads, bolts, nuts and threaded holes is obligatory. The inspections are carried out according to the requirements listed in Table XXXII. A special shielded cabin is used at some NPPs for visual and dye-penetrant inspections from the inside of the RPV, as well as for the repair of any defects. The in-service inspection of the vessel head includes only a visual inspection (and sometimes also a dye-penetrant inspection) of sealing surfaces, welds and cladding, performed at the locations which are accessible. Ultrasonic inspection of the circumferential weld is also performed.

The examination results are evaluated using the former Soviet procedure PK 1514-72 [82] which was originally developed as a manufacturing defect rejection criteria. These standards and procedures have been approved by the Russian regulatory body. Although they are not officially accepted by all the safety authorities responsible for WWER in-service inspections, they are used in general at most of the WWER plants since no other procedure or standard for defect acceptance/ rejection is available, except in Hungary, the Czech Republic and Slovakia where newly developed national procedures are applied.

The ultrasonic examination equipment is calibrated using a flat bottomed hole according to PK 1514-72. However, the most recent inspections in some plants have been performed using calibration methods similar to those used in the West.

5.2. NDE TECHNIQUES

5.2.1. Advanced ultrasonic examination methods

Smooth, sharp-edged flaws oriented in a plane normal to the vessel surface and located in the beltline region near the cladding/base-metal interface are the most critical type of flaws because the material in that area of the pressure vessel exhibits the highest degree of neutron embrittlement and corresponding high RT_{NDT} and high tensile stresses (thermal) occur
near the vessel inner surface during a PTS accident or a cooldown violating the P-T limits. Such flaws are difficult to detect and size with an ultrasonic technique based on signal-amplitude alone, which was the technique originally developed for ISIs in the USA and elsewhere.

In the amplitude-based technique, the sensitivity setting of the ultrasonic equipment is referenced to a distance-amplitude correction curve, which can be obtained from an ASME reference block with one 3-mm (0.125-in.) side-drilled hole [83]. The ASME Section XI code (1986 Edition) specifies an amplitude cut-off level of 20% of the distance-amplitude correction; only defect indications that exceed that level are recorded. ASME Section XI Code also specifies use of an additional scan angle of 70-degrees longitudinal wave to inspect clad-base metal interface regions [84, 85].

The amplitude-based technique uses the decibel-drop method to determine *flaw sizes* much larger than the width of the sound field [83, 86]. In the decibel-drop method, the transducer is positioned to obtain a maximum height for an echo from the defect, and then it is traversed until the height of the echo drops to a specified threshold (50% of the maximum height for the 6-decibel-drop method). This position of the transducer is assumed to be over the edge of the flaw. Similarly, the transducer is moved in other directions from the maximum height position, and finally the flaw size is determined. A flaw size much smaller than the width of the sound field can be determined by the 20-decibel-drop method (beam edge method) or by comparing the amplitude of the reflection from the flaw with a range of reflection amplitudes from various flat-bottomed holes in test blocks. The accuracy of flaw sizing by the amplitude-based technique depends not only on the transducer sound field size, acoustic impedance differences between the flaw and the surrounding material (that is, the ultrasonic reflectivity of the flaw) and the flaw size, but also on the orientation of the flaw, the surface condition and the ultrasonic scattering properties of the flaw. This technique is effective in sizing a smooth, flat flaw that is at a right angle to the ultrasonic beam and away from the clad-metal interface, but it under sizes near-surface and other flaws. Cladding surface roughness also affects sizing of the flaws; it causes scattering of the ultrasound, which may result in under sizing of near-surface flaws [87].

Tip-diffraction techniques developed in the United Kingdom more accurately size underclad and embedded flaws. With one of the tip-diffraction techniques, the time-of-flight diffraction technique, the difference in the travel times of ultrasonic waves diffracted from each of the flaw tips is measured to estimate the flaw size [88]. Examples of time-of-flight diffraction are depicted in Fig. 32 [89]. The technique consists of a separate transmitter and

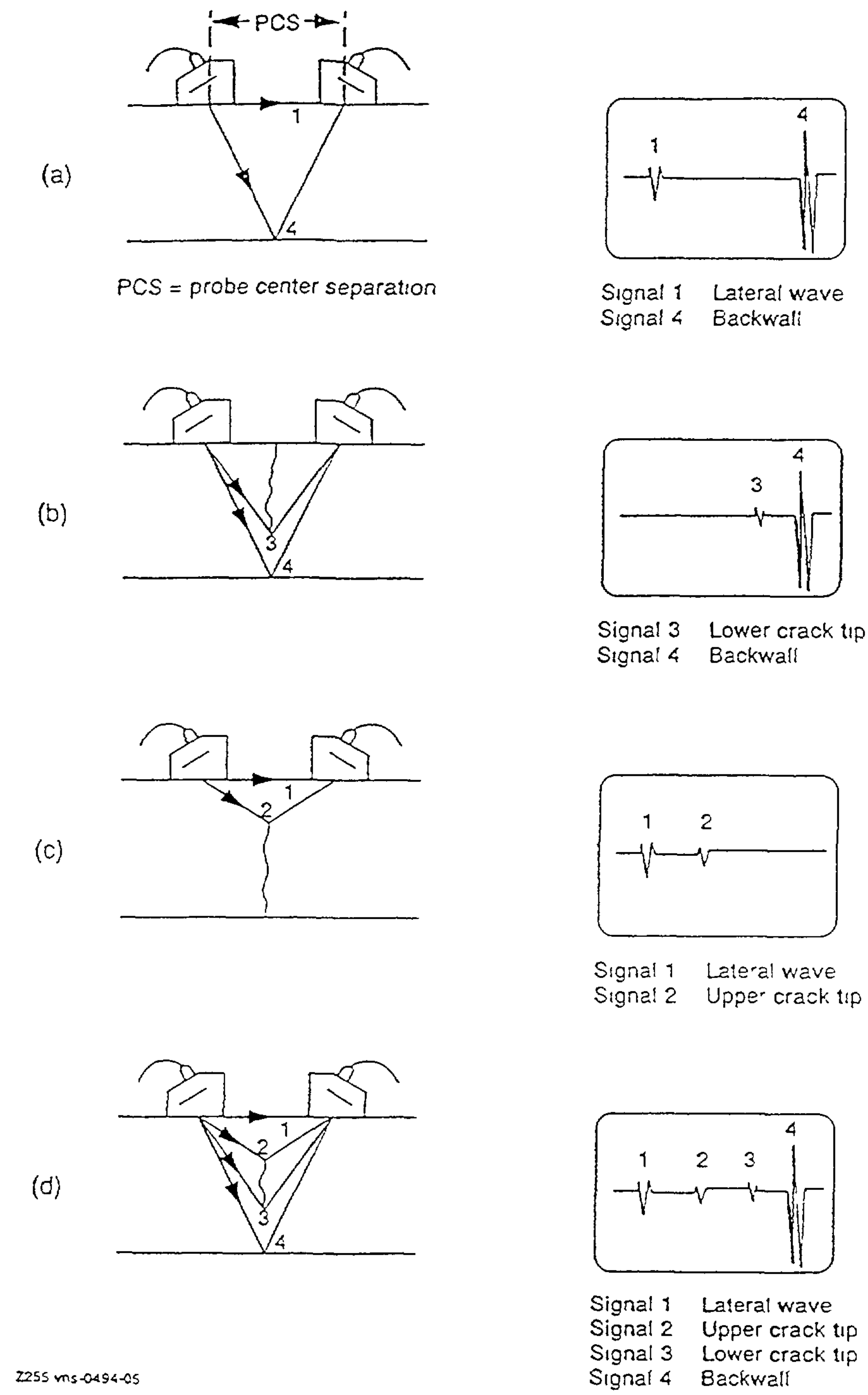


FIG 32 Examples of time-of-flight diffraction (TOFD) signals (Pers-Anderson 1993) Copyright TRC, reprinted with permission

receiver oriented in opposite directions, as shown in Fig. 32(a). Two signals are present in the absence of a crack, a direct lateral wave signal and a backwall reflection signal from the opposite surface. Diffraction occurs when the incoming sound beams impinges upon a finite planar reflector such as a crack. The diffracted sound energy from the end or "tip" of the crack acts as a point source and radiates a sound wave to the receiving transducer. The time of arrival of this signal can then be used to pinpoint the tip of the crack and determine crack depth. Figure 32(b) illustrates such a diffracted signal produced by the tip of a surface crack: note the presence of a backwall reflection signal and the absence of a lateral wave signal.

Although cracking on the inside surface is a primary concern, cracking on the outside (back wall) surface could also occur. As illustrated in Fig. 32(c), the presence of an outside surface crack will cause the loss of a backwall reflection signal, but the lateral wave and the diffracted signal from the crack tip are present. In Fig. 32(d), two diffracted signals from the ends of an embedded crack are evident, and both a lateral wave and a backwall reflection signal are present.

Flaw orientation and roughness, which interfere with flaw sizing using amplitude-based techniques, have very little effect on flaw sizing with tip-diffraction techniques. Laboratory test results, including the Programme for the Inspection of Steel Components II test results, show that the tip-diffraction techniques are the most accurate for sizing underclad and embedded flaws [86, 90]. One disadvantage of the time-of-flight diffraction method is that the diffracted crack tip signals are often small in amplitude and can easily be confused with grain noise or other small amplitude reflectors. In addition, crack branches may interfere with the interrogating sound beam or cause additional diffracted signals. These additional signals may cause cracks to be undersized.

Flaws located in the nozzle-to-shell welds are also of considerable interest in assessing RPV integrity. The nozzle-to-shell welds can be ultrasonically inspected from the nozzle bore; however, sizing of the flaws is difficult when conventional (unfocused) transducers are used [86]. The main reason for this difficulty is the large distance between the nozzle bore and nozzle weld. At these distances, the ultrasonic beam of conventional transducers provides poor resolution of flaws in the welds. A large-diameter, focused ultrasonic transducer produces a small diameter beam at the flaw location and can be used for accurate mapping of flaw edges. Laboratory results show that the large-diameter focused transducers are substantially more accurate than unfocused transducers in sizing flaws in the nozzle-to-shell welds [91].

Focused transducers are used commonly in France and Belgium, but infrequently in the United States. Examples of the applications of focused transducers are inspection of RPV welds and heat-affected zones in Westinghouse 350-MWe and Framatome 900-MWe reactors in Belgium and in a 660-MWe reactor at Krsko in Slovenia [92]. Also, a large-diameter focused transducer was used to inspect the nozzle-to-vessel welds of the Ginna reactor vessel during its second ISI interval [93]. This inspection with the focused transducer characterized the earlier detected ultrasonic indications as closely spaced slag inclusions; a conventional transducer was unable to resolve these closely spaced indications. Earlier, a focused transducer was used to characterize the flaws in the cladding under the head of the Yankee Rowe RPV [94].

Ultrasonic examination methods based on a phased array technique have also been developed for ISI of components which have complex geometries and have limited access and clearance. One such technique developed by Siemens has been used for inspection of the BWR feedwater nozzle inner radius regions, nozzle bore and nozzle-to-vessel welds; the BWR bottom head ligaments; and the PWR closure head ligaments [95]. This technique has also been used for inspection of PWR feedwater nozzles inner radius regions.

A phased array transducer consists of multiple elements that can be controlled individually to create a variety of beam patterns. The use of multiple elements with a computer controlled pulsing sequence results in the ability to steer and/or focus the sound beam. With an appropriate phase-shifting of the transducer elements, the focal length of the transducer can be changed and the specimen can be scanned in depth. The transducer design can be tailored to the needs of the specific examination. For example, the examination of a nozzle inner radius region employs a fixed incident angle with a variable skew angle whereas the vessel shell welds require a fixed skew angle with a variable incident angle. Echoes received in many cross-sectional directions are stored during inspection and echo tomography utilizes the spatial relationships of the signals in order to enhance the signal to noise ratio. The combination of these modes allows a rapid and accurate analysis of the reflectors. Flaw sizing is typically done with a tip diffraction method [96].

Recently, the ASME Section XI Code has developed more stringent requirements for demonstrating the performance of ultrasonic inspection procedures, equipment and personnel used to detect and size flaws at the susceptible sites in pressure vessels. The susceptible sites include the clad-base metal interface, nozzle inside radius section, reactor vessel structural welds, nozzle-to-vessel welds and bolts and studs. These requirements are needed to ensure that inspectors apply the appropriate ultrasonic inspection techniques in the field to correctly characterize the flaws at the susceptible sites in the vessel. These requirements are presented in two appendices of ASME Section XI: Appendix VII, Qualification of Nondestructive Examination Personnel for Ultrasonic Examination; and Appendix VIII, Performance Demonstration for Ultrasonic Examination Systems. Implementation of Appendices VII and VIII will take several years. The enhanced inspection programme will provide more reliable ISI data on US RPVs, which then may be used for the development of a plant-specific vessel flaw distribution or a generic flaw distribution more representative of operating PWR vessels than currently used distributions such as the Marshall distribution [97].

5.2.2. Acoustic emission monitoring

Acoustic emission methods may be used to monitor potential flaw growth in the beltline region welds and base metal if the outside surface of the vessel is accessible. Some PWR vessels are supported by neutron shield tanks, which will prevent access to the vessel outside surface.

An acoustic emission method for crack growth detection was tested at Watts Bar Unit 1 during hot functional testing. A preloaded, precracked fracture specimen was placed in the primary system to test the capability of the acoustic emission method to detect a signal during reactor operation. The specimen was designed such that the system operating temperature would impose thermal loads and cause crack growth. The test results showed that the coolant flow noise could be filtered out and that crack growth acoustic emission signals can be detected under operating conditions [98]. Acoustic emission was also used to monitor possible crack growth during the 1987 hydro test of the High Flux Isotope Reactor located at the Oak Ridge National Laboratory; no evidence of crack growth was detected [99].

Several significant steps have been taken to validate continuous, on-line acoustic emission monitoring in the field. Work on the application of the acoustic emission method at Watts Bar Unit 1 has shown that it can be effectively used for inservice monitoring of crack growth in thick wall, geometrically complicated components such as RPV nozzles [100]. Continuous acoustic emission monitoring has also been used by the Pacific Northwest National Laboratory to monitor a flaw indication in an inlet nozzle safe end weld at the Limerick Unit 1 reactor [101]. In addition, ASME Code Case N-471 has been developed and approved, which provides for continuous on-line acoustic emission monitoring for growth of known flaws. The Code Case applies to components in which flaws exceeding the acceptance criteria (ASME Section XI, IWB-3410.1) have been identified, and for which the analytical evaluation of the flaws found the components acceptable for continued service according to ASME Section XI, IWB-3132.4.

All of the WWER-440/V-213C and WWER-1000/V-320 RPVs manufactured at ŠKODA Plzeň were subjected to a hydraulic test in the shop with acoustic emission monitoring. The same acoustic monitoring techniques are also applied during the hydraulic tests at the plants in the Czech Republic, Hungary and Slovakia.

5.2.3. Inspection of PWR CRDM penetrations

PWSCC in Alloy 600 CRDM penetrations in European PWRs has stimulated the development of special NDE techniques to detect and size the cracks in these penetrations. The primary challenge associated with the inspection is assessing the examination area. With the head removed and on its stand, the penetrations are physically accessible from the underside of the head, but high radiation fields dictate the use of extensive shielding or remotely operated inspection systems. In addition, direct access to the inside surface of most CRDM penetrations is impeded by the stainless steel thermal sleeve. An air gap of only approximately 3 to 4 mm (0.12 to 0.16 in.) between the sleeve and the nozzle inside surface is available for the access. Removal of the sleeve is time and dose intensive. Therefore, examination of the penetration with the sleeve in place is highly desirable. For penetrations without sleeves, examination is possible, and conventional techniques (visual, penetrant testing, ECT and UT) can be applied.

The current industry practice for examining CRDM penetrations for PWSCC on the inside diameter surface is to use ECT for detection; and UT for sizing the detected indications. Small-diameter ECT probes have been developed by several inspection vendors to inspect penetrations with thermal sleeves. The probes are attached near the tip of the long thin (1.5 mm) blade and are typically spring-loaded to maintain continuous contact with the penetration inside surface. With these "gap scanners," cracks as shallow as 1 mm (0.04 in.) can be detected. In addition, information on the crack length can be obtained more accurately, and small, closely spaced cracks can be resolved.

The primary physical limitation to this approach is that the gap can vary by as much as 30% because the penetrations might have deformed and the sleeves may not be centered [102]. The deformation includes ovalized nozzle cross sections and bending of the penetrations, as discussed in Section 4.5.1. This variation in the gap can prevent full inspection for the peripheral nozzles with thermal sleeves. In addition, boric acid deposits in the gap can obstruct access; however, this obstruction can be removed by rotating the thermal sleeve, which is freely hanging inside the penetration.

Once a crack is detected, accurate crack sizing is important to determine if repair of the penetration is necessary. UT is the primary inspection method for sizing cracks. The most widely accepted UT method for sizing a crack in the CRDM penetrations is the crack-tip diffraction or time-of-flight diffraction method discussed in Section 5.2.1 above. Low-profile UT probes which will fit in the gap between the penetration and thermal shield have been developed for this purpose. Inspection of a penetration without a thermal sleeve is performed using rotating ultrasonic time-of-flight probes. Rotating transducers may contain several sets of dual-element probes to optimize the sizing of different type of cracks, that is, isolated or cluster cracks and deep or shallow cracks. UT is also used to search for cracks on the penetration outside surface. Because of the penetration wall thicknesses, the losses are too large for eddy-current techniques to be an effective means of detecting outside surface cracks.

5.3. RPV MATERIAL SURVEILLANCE PROGRAMMES

5.3.1. Requirements in the USA

Every PWR pressure vessel operating in the western world has an ongoing RPV material radiation surveillance programme. To date, close to 300 surveillance capsules have been removed from their host RPV and tested. The results from these surveillance capsules have been used to develop heatup and cooldown curves and to analyse all potential or postulated accident or transient conditions.

Fracture toughness requirements[1]

On 17 July 1973 the USNRC published Appendix G of 10 CFR Part 50, which delineates requirements for prevention of fracture of the ferritic materials in the primary coolant pressure boundaries of the US NPPs, with emphasis on the RPV [25]. The significant points in Appendix G to 10 CFR Part 50 are:

(a) To demonstrate compliance with the minimum fracture toughness requirements of Appendix G, the ferritic materials must be tested in accordance with the ASME Code, Section III NB-2300. Drop weight tests (NB-2321.1) and Charpy V-notch tests (NB-2321.2) are used to define the reference nil-ductility transition temperature RT_{NDT} (NB-2331a). Further, NB-2300 requires that the Charpy V-notch specimens be oriented normal to the main rolling or working direction of the material (NB-2322.2).

(b) The reactor vessel beltline materials must have a minimum initial USE, as determined by Charpy V-notch tests on unirradiated specimens in accordance with NB-2322.2 of the ASME Code of 102 J (75 ft-lbs.) unless it can be demonstrated by data and analysis that lower values of upper shelf fracture energy are adequate.

    10 CFR Part 50 Appendix G also limits the reactor vessel operation to only that service period during which the Charpy impact energy, as measured in the weakest direction, is above 68 J (50 ft-lb) or 0.9 mm (35 mils) lateral expansion. In the event that the RT_{NDT} cannot be defined (Charpy impact energy drops below 68 J), the reactor vessel may continue to be operated provided the requirements listed below are satisfied:

    - An essentially complete volumetric examination of the beltline region of the reactor vessel including 100 per cent of any weldments shall be made in accordance with the requirements of Section XI of the ASME Code.

[1]The industrial technical standards, JEAC 4201-1991 which includes the Japanese embrittlement predictive equation and JEAC 4206-1991 which includes the PTS evaluation method, were published in 1992 by the Japan Electric Association for use in Japan [103, 104]. JEAC 4201-1991 incorporates NRC 10 CFR Part 50 *Appendix G* (1988) and Appendix H (1988), and ASTM E185-82. JEAC 4206-1991 incorporates NRC 10 CFR Part 50 *Appendix G* (1988) and Appendix H (1988) and the material in the ASME Boiler and Pressure Vessel Code Section III Nuclear Power Plant Components (1989).

- Additional evidence of the changes in the fracture toughness of the beltline materials resulting from neutron radiation shall be obtained from results of supplemental tests, such as measurements of dynamic fracture toughness of the beltline materials.

- A fracture analysis shall be performed that conservatively demonstrates the existence of adequate margins for continued operation.

Paragraph IV.A.1 of Appendix G to 10 CFR 50 states, "Reactor vessel beltline materials must have a Charpy upper-shelf energy of no less than 102 J (75 ft-lb) initially and must maintain an upper-shelf energy throughout the life of the vessel of no less than 68 J (50 ft-lb) unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code." This allows licensees to submit an USE equivalent margins analyses instead of performing the three tasks cited here.

If the results of the above tasks do not indicate the existence of an adequate safety margin, thermal annealing of the reactor vessel beltline region is required to recover the reactor vessel beltline material fracture toughness properties or the plant must be shutdown.

(c) The calculated stress intensity factor (K_I) shall be lower than the reference stress intensity factors (K_{IR}) by the margins specified in Appendix G to the ASME Code. However, if there is no fuel in the reactor during the initial pre-operational hydrostatic pressure tests, the safety factor on K_{IM} can be reduced from 1.5 to 1.0.

10 CFR Part 50 Appendix H[2]*, reactor vessel material surveillance programme*

With the publication of Appendix G, "Fracture Toughness Requirements", the USNRC also published Appendix H, a set of rules for the reactor vessel material surveillance rogrammes [26]. The significant points given in Appendix H to 10 CFR Part 50 are:

(a) That part of the surveillance programme conducted with the first capsule withdrawal must meet the requirements of ASTM E185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased.

(b) Surveillance specimen capsules must be located near the inside vessel wall in the beltline region so that the specimen radiation history duplicates to the extent practicable within the physical constraints of the system, the neutron spectrum, temperature history and maximum neutron fluence experienced by the reactor vessel inner wall.

(c) A surveillance capsule withdrawal schedule must be submitted to and be approved by the NRC prior to implementation.

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[2]The industrial technical standards, JEAC 4201-1991, which includes the Japanese embrittlement predictive equation and JEAC 4206-1991 which includes the PTS evaluation method, were published in 1992 by the Japan Electric Association [103, 104]. JEAC 4201-1991 incorporates NRC 10 CFR Part 50 Appendix G (1988) and *Appendix H* (1988), and ASTM E185-82. JEAC 4206-1991 incorporates NRC 10 CFR Part 50 Appendix G (1988) and *Appendix H* (1988), and the ASME Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components (1989).

(d) Each surveillance capsule withdrawal and the test results must be the subject of a summary report submitted to the NRC

Regulatory Guide 1 99

Appendix G, "Fracture Toughness Requirements" and Appendix H, "Reactor Vessel Material Surveillance Programme Requirements", necessitate the calculation of changes throughout the service life in fracture toughness of reactor vessel materials caused by neutron radiation USNRC Regulatory Guide 1.99 [36–38] describes general procedures acceptable to the USNRC staff for calculating the effects of neutron radiation of the low-alloy steels currently used for light-water-cooled reactor vessels in the western world As discussed in more detail in Section 6, the pertinent rules or guidelines are

(a) The ART for each material in the beltline is given by the following expression

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \tag{22}$$

(b) ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by radiation and is calculated as follows

$$\Delta RT_{NDT} = (CF)\, f^{(0\,28 \quad 0\,10 \log f)} \tag{23}$$

where CF is a chemical factor which is a function of the copper and nickel content, f is the fluence in 10^{23} n/m^2 and ΔRT_{NDT} has units of Fahrenheit degrees Regulatory Guide 1 99 Revision 2 presents the CF in tabular form for welds and base metal (plates and forgings) If more than two credible surveillance capsule data are available, the CF should be calculated by curve fitting The neutron fluence f, is the fluence at any depth in the vessel wall The fluence factor, $f^{0\,28 \quad 0\,10 \log f}$ is determined by calculation or from a figure presented in the regulatory guide

Regulatory Guide 1 99 Revision 0 and 1 [36, 37] considered the detrimental effect of copper and phosphorus R G 1 99 Revision 2 introduced the CF and replaced the element phosphorus with nickel

Other regulatory guides

Regulatory Guide 1 43 [105] provides guidance to assure that stainless steel protection cladding complies with ASME Section III and XI requirements to prevent underclad cracking The presence of intergranular cracking in the base metal under the cladding is a possibility in a RPV

Regulatory Guide 1 65 [106] provides guidance on vessel closure bolting materials and inspections PWR plants have closure bolts in compliance with ASME Section III and are inspected according to ASME Section XI All studs are volumetrically examined and receive a surface examination during each 10 year inspection interval

Regulatory Guide 1 150 [107] provides guidance on ultrasonic test procedures which supplement those provided in ASME Section XI PWR procedures for inspection of vessels comply with this guidance

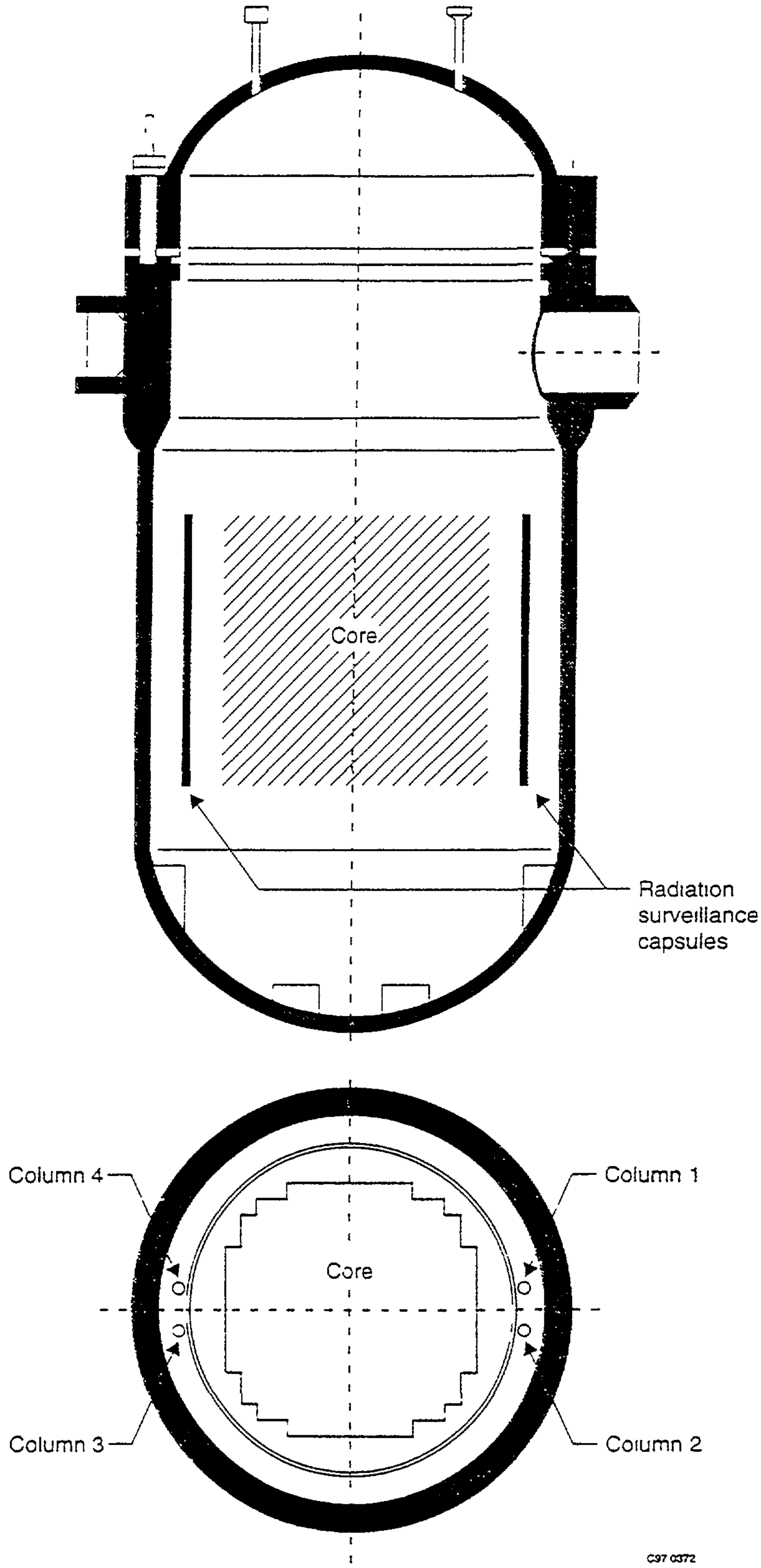
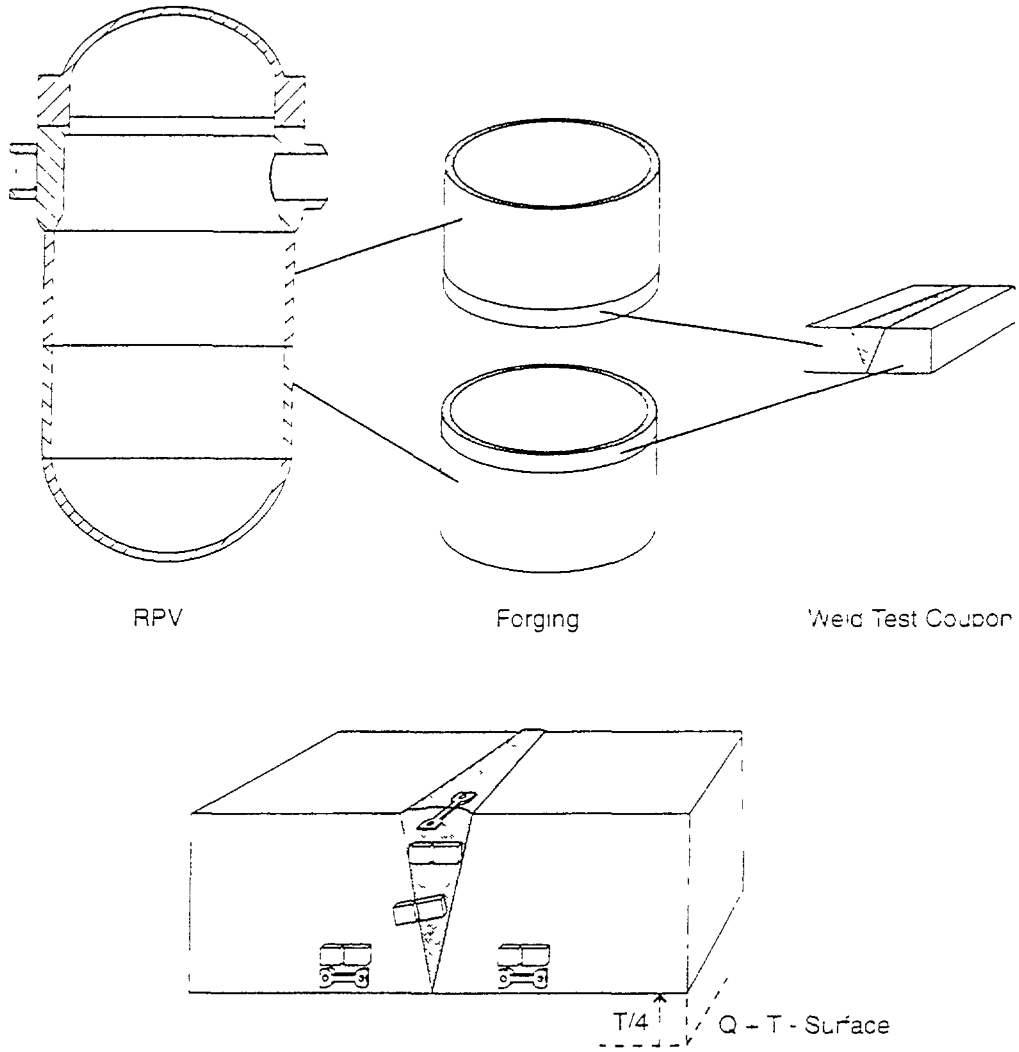


FIG 33 Locations of RPV surveillance specimens in Germany.



Quantities of Specimens

Specimen Set No.	Charpy V-Notch Impact Specimens				Tensile Specimens			Withdrawal Schedule
	Base I	Base II	Weld	HAZ	Base I	Base II	Weld	
1	12	12	12	12	3	3	3	Unirradiated
2	12	12	12	12	3	3	3	50% EOL
3	12	12	12	12	3	3	3	100% EOL

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FIG 34. Configuration, types, and quantities of specimens used in the PRV surveillance programme in Germany

5.3.2. Requirements in Germany

According to the stipulations in the Code, the radiation embrittlement can be neglected when the neutron fluences are lower than 10^{21} n/m^2 (E >1MeV). Since the maximum allowed RPV fast neutron fluence in Germany is limited to 1.1×10^{23}n/m^2 (E > 1 MeV) and KTA 3203 is valid for this fluence or lower values only, the number of radiation sets and the withdrawal schedule (relative to the RPV fluence) are fixed (two sets covering 50% and 100%, respectively, of the RPV design life fluence). KTA 3203 allows higher lead factors (>3) on the radiation capsules. This ensures that the results for the first set of irradiated specimens withdrawn at approximately 50% of the fluence predetermined for the

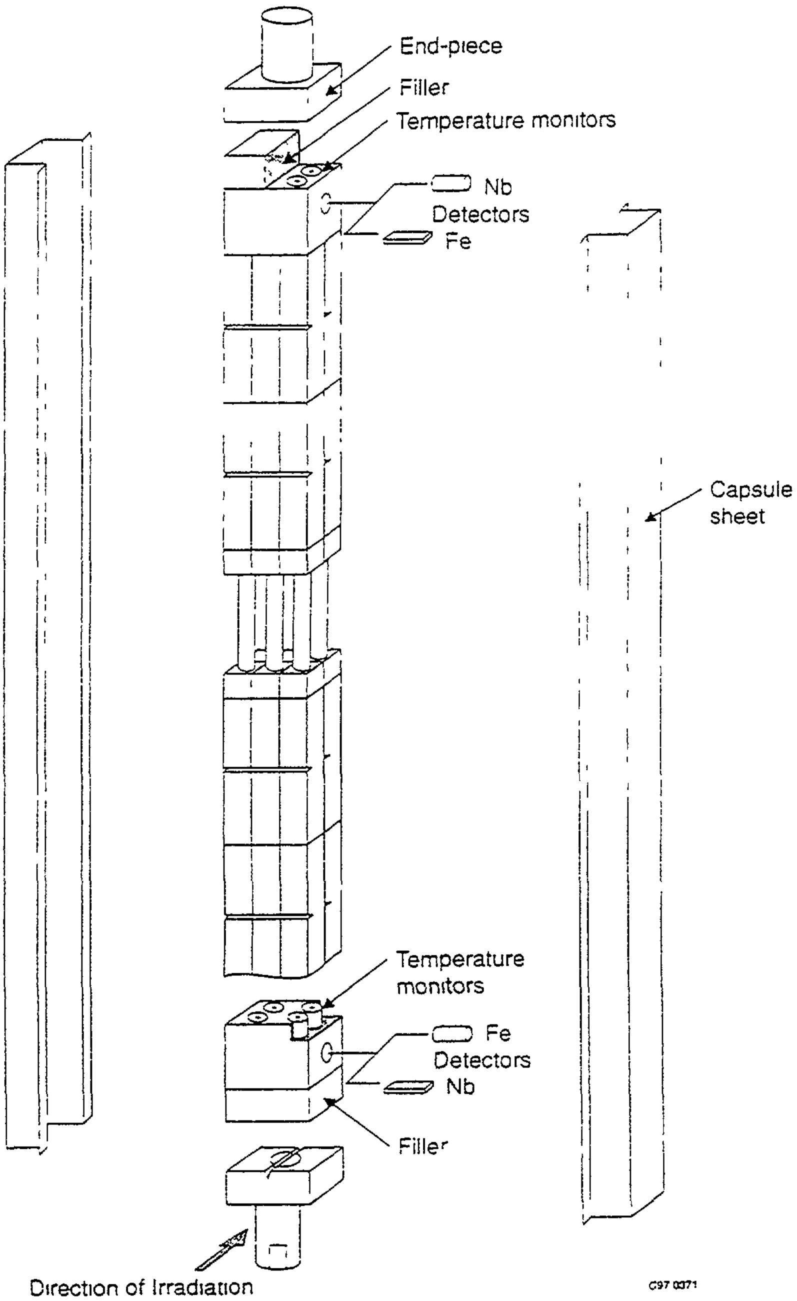

FIG. 35. Irradiation capusles used in the RPV surveillance programme in Germany.

vessel at end-of-life are available prior to the first in-service pressure test of the RPV. The surveillance specimens are located between the core barrel and RPV along the entire core length as shown in Fig. 33. Each set has to contain 12 Charpy V-notch specimens and 3 tensile specimens from both base metals (the upper and lower ring forgings), and the weld metal and 12 Charpy V-notch specimens from one heat-affected zone. However, for end-of-life-fluences between 1.1×10^{21}n/m^2 and 1.1×10^{22}n/m^2 (E>1 MeV), it is sufficient to implement 12 Charpy V - notch specimens from one base metal and the weld metal set each. The specimen configuration and the quantities of specimens are shown in Fig. 34. A sketch of the surveillance programme capsules is shown in Fig. 35.

The differences between the surveillance programmes required by ASTM and KTA, such as the number of the specimen sets and the removal schedule, reserve material (an approximately 1.5 m long section of the fabricated test coupon) instead of optional specimens in the standard capsules, and the magnitude of the lead factor, are justified by the fact that the predicted transition temperature shift does not exceed 40K, and a pre-irradiation nil-ductility transition temperature of <–12°C is required for the steels used in the beltline region of the RPV.

5.3.3. Requirements in France

The material surveillance programme specified in RSE-M [30] is similar to the US programme discussed above. Capsules are regularly removed from the plants and the specimens subjected to Charpy testing. The measured shifts in the Charpy nil-ductility transition temperatures are compared with the predicted values (Equation (27)). As discussed in Section 6.1, the anticipation factor is less than 3. However, all the end-of-life (40-year) ΔRT_{NDT} values for most of the French plants are expected to be available around the year 2000. (In some older plants, a periodic re-arrangement of the capsules is needed to obtain the end-of-life ΔRT_{NDT} values on that schedule.) Presently, only the Charpy specimens are being tested, the fracture mechanics specimens and also the archive material (the same for all the French plants) are being stored for future use, if necessary.

Some changes in the French surveillance programmes (re-arrangement of the capsules, new material in the capsules, laboratory tests, etc.) are being studied to support a possible life extension from 40 to 50 years. The objective of these changes is to provide information for decision making after about 20 years of operation.

5.3.4. WWER material surveillance programme requirements

The requirements for the WWER material surveillance programmes are given in Ref. [31] and updated in Ref. [32], but they were applied only to the WWER-440/V-213 and WWER-1000 NPPs. The oldest design type, the WWER-440/V-230, was not supplied with a material surveillance programme, even though, as was shown later, the materials used for these RPVs are more susceptible to radiation embrittlement than the materials used for the WWER-440/V-213 and WWER-1000 pressure vessels.

WWER surveillance specimens must be removed from the RPV and tested at least six times during the pressure vessel design life. The first batch of specimens must be removed and tested after 1 year of reactor operation. The next three batches of specimens must be removed and tested every 3 years within the first 10 years of operation. This schedule is based on the assumption that the neutron fluence on the RPV wall will be greater than 10^{22}n/m^2

(E ≥0.5 MeV) but less than 10^{23}n/m^2 during the first year of operation. A surveillance programme is not required when the end-of-life RPV fluence is less than 10^{22}n/m^2 (E ≥0.5 MeV) and the RPV operating temperature is greater than 250°C.

The following material properties are measured after each removal:

- tensile properties (yield and ultimate strength and elongation)
- ductile-to-brittle transition temperature
- fracture toughness (or crack opening displacement).

The maximum allowable ductile-brittle transition temperature and the allowable fracture toughness curves (K_{IC} versus reference temperature) for various materials and operating conditions were discussed in Section 3.4.4. There is no lower limit on the USE specified in the WWER codes because experiments have shown that the USE will remain sufficiently high during the expected RPV lifetimes.

The WWER-440/V-213 pressure vessel radiation damage surveillance programmes are characterized by the following features:

- Individual specimens were manufactured from either base metal, weld metal, or heat affected zone material. The base metal specimens were removed from the core beltline ring as it was cut to size during fabrication of the vessel. The weld metal and heat affected zone material were removed from welding coupons for welding joint No. 0.1.4 (the circumferential weld in the lower part of the beltline region).

- Tensile, Charpy V-notch and pre-cracked Charpy (for static fracture toughness testing) type specimens were made from each of the three materials. A complete set of specimens includes 18 tensile specimens (6 of each of the 3 materials), 36 Charpy V-notch specimens (12 of each of the three materials) and 36 pre-cracked Charpy specimens. There are a total of 90 surveillance specimens per set.

- The specimens are put into stainless steel containers, six tensile specimens or two Charpy-type in one container as shown in Fig. 36.

- The containers are connected into chains, each chain consisting of 20 or 19 containers which are then placed adjacent to the active core region.

- Two chains hold one complete set of 90 specimens and contain all the aforementioned specimen types and materials; the two chains are located symmetrically very close to a corner of the hexagon shaped active core and are removed at the same time.

- Six sets of specimens are located in each reactor, one set (two chains) at each corner of the hexagon core.

- The planned withdrawal interval of the individual sets is usually: 1, 3, 5 and 10 years (or 1, 2, 3, 5 and 10 years).

- Some of the containers are supplied with neutron fluence monitors and some are supplied with diamond powder temperature monitors, but the diamond temperature monitors have been found to be unreliable. X ray diffraction techniques are used to decode the diamond powder information.

- The containers are located on the outer wall of the active core barrel, where a high lead factor, between 6 and 18, is obtained. (The lead factor is the ratio of the neutron flux with energies ≥0.5 MeV in the test specimens to the maximum neutron flux on the vessel wall).

- The length of the chain of 20 containers located in the active core region is about 2.4 m. a distance corresponding to a factor of about 10 between the maximum and minimum neutron flux. However, the containers with the Charpy V-notch specimens are all located in the centre (maximum flux) region.

In addition to the surveillance specimens for monitoring the radiation damage in the beltline materials, the WWER surveillance programmes include specimens for monitoring the thermal ageing damage in the pressure vessel materials. Two complete sets of 90 surveillance specimens (39 containers) are located well above the active core (virtually no damage from neutron bombardment) in front of the upper (outlet) nozzle ring. These sets are usually removed and tested after 5 and 10 (or 20) years of operation.

Since the WWER-440/V-213 Charpy specimens for monitoring radiation damage are located on the outer wall of the core barrel in the axial centre region of the core with lead factors ranging from 12 to 18, the original radiation damage surveillance programmes are now essentially finished in all the WWER-440/V-213 plants. (Irradiation for times longer than 5 years has no practical meaning as it represents more than 60 to 90 years of operation, a time much longer than the RPV design life.) At least four complete specimen sets have been removed and tested from each of the 18 WWER-440/V-213 reactors (six in Russia, two in Finland, four in Hungary, two in Slovakia and four in the Czech Republic). However, the radiation damage measured in these specimens may not accurately predict the damage expected in the WWER-440 pressure vessels because the neutron flux in the specimens was so much higher than the highest flux in the vessel walls (12 to 18 times). These problems have lead some of the WWER regulatory bodies to require supplementary surveillance programmes designed to monitor the RPV material behaviour and the neutron flux and fluence throughout the plant life. In the Paks plant (Hungary), quasi-archive and reference material specimens are located in the usual locations, but removed and replaced every 3 years. In the Bohunice (Slovakia) and Dukovany plants (Czech Republic), archive material specimens are located in relatively low flux positions on the outside of the core barrel near the top and bottom of the core (with lead factors below five) and will be withdrawn and tested at periodic intervals during the remaining plant life.

The number and type of surveillance specimens placed in the standard WWER-1000 plants is similar to the number and type of specimens used in the WWER-440/V-213 plants. Both neutron embrittlement and thermal ageing specimens are placed in the WWER-1000 pressure vessels, and the same type of containers (shown in Fig. 36) are used. However, the locations in which the WWER-1000 radiation embrittlement surveillance specimens are placed are quite different than the locations discussed above for the WWER-440 neutron radiation specimens. The axial and radial locations of the WWER-1000 surveillance assemblies are shown in Fig. 37. The locations labelled 1M, 2M, etc. are where the thermal ageing sets are placed, the locations labelled 1L, 2L, etc. are where the neutron radiation embrittlement specimens are placed. The surveillance assemblies hold five containers stacked either one or two high (radiation embrittlement specimens) or five high (thermal ageing specimens) as shown in Fig. 38.

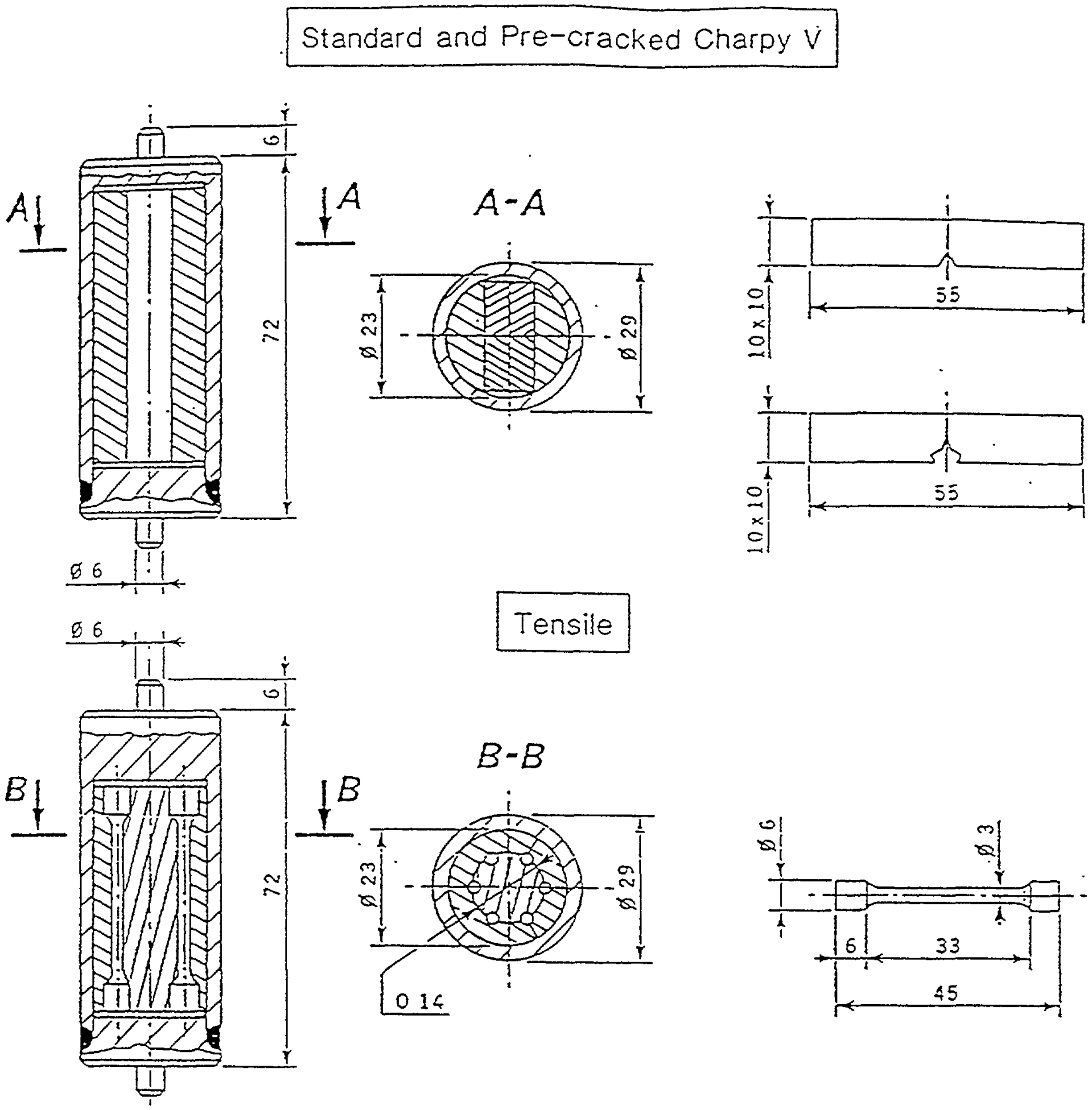

FIG. 36. Charpy and tensile material surveillance specimen containers used in the WWER-440/V-213 and WWER-1000 pressure vessels.

Since the neutron radiation embrittlement surveillance specimens are located above the core, they are in a relatively steep flux gradient as shown in Fig. 39. Also, the mean flux level at their position is approximately the same or lower than that on the RPV wall (a lead factor of less than 1.0) but the energy spectrum is different. In addition, the containers are located in outlet water and therefore the specimens are irradiated at 10 to 20°C above the vessel wall temperature. (The specimen temperature is also monitored with diamond powder, but, as mentioned above, this method has been found unsuitable.) Due to the high temperatures and atypical flux, the use of these surveillance results for vessel radiation embrittlement assessment is not reliable and may yield non-conservative results. The WWER-1000 surveillance programme has not been modified in the current operating plants.

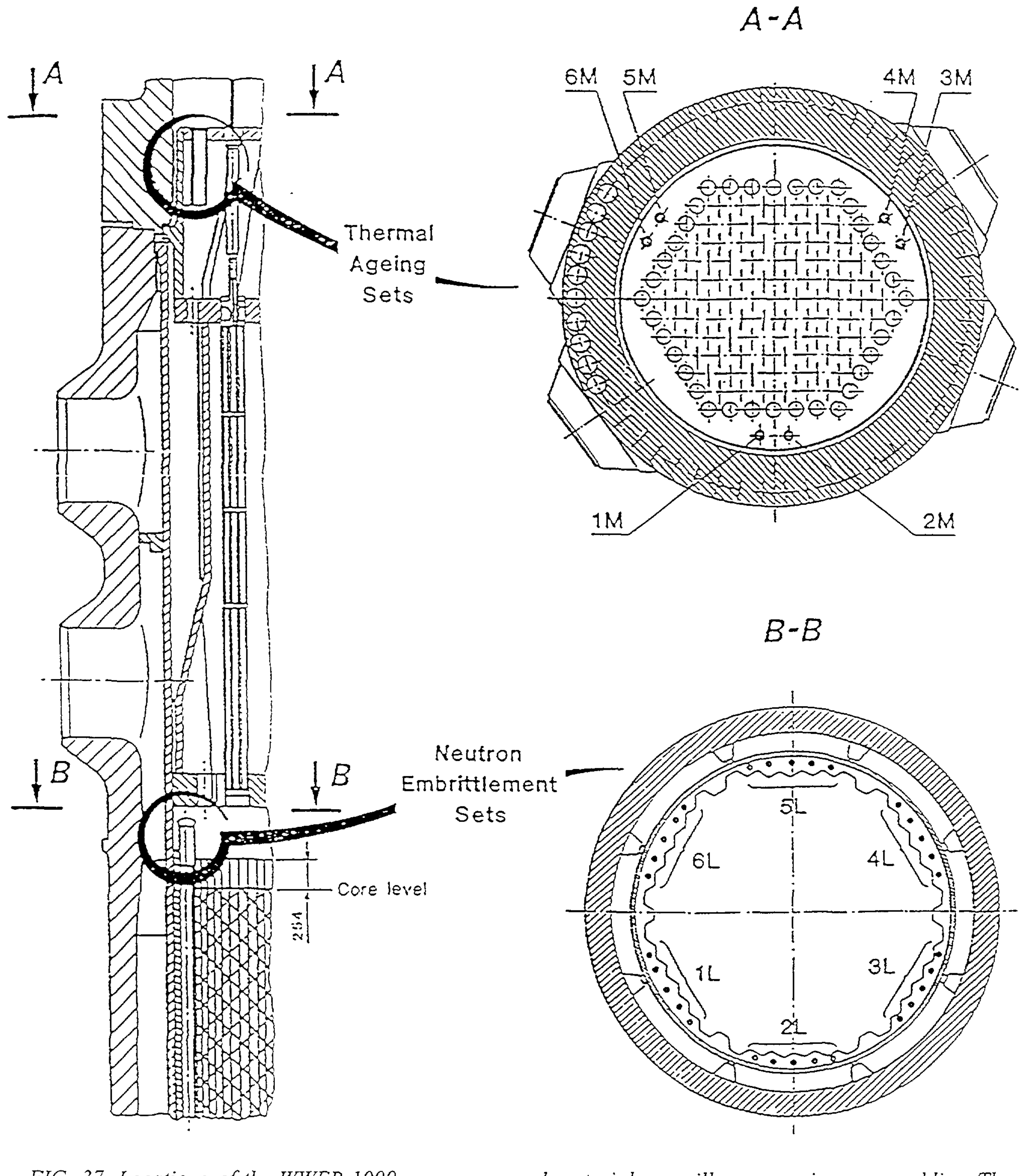

FIG. 37. Locations of the WWER-1000 pressure vessel material surveillance specimen assemblies. The thermal ageing specimens are at 1M, 2M, etc. and the neutron radiation embrittlement specimens are at 1L, 2L, etc.

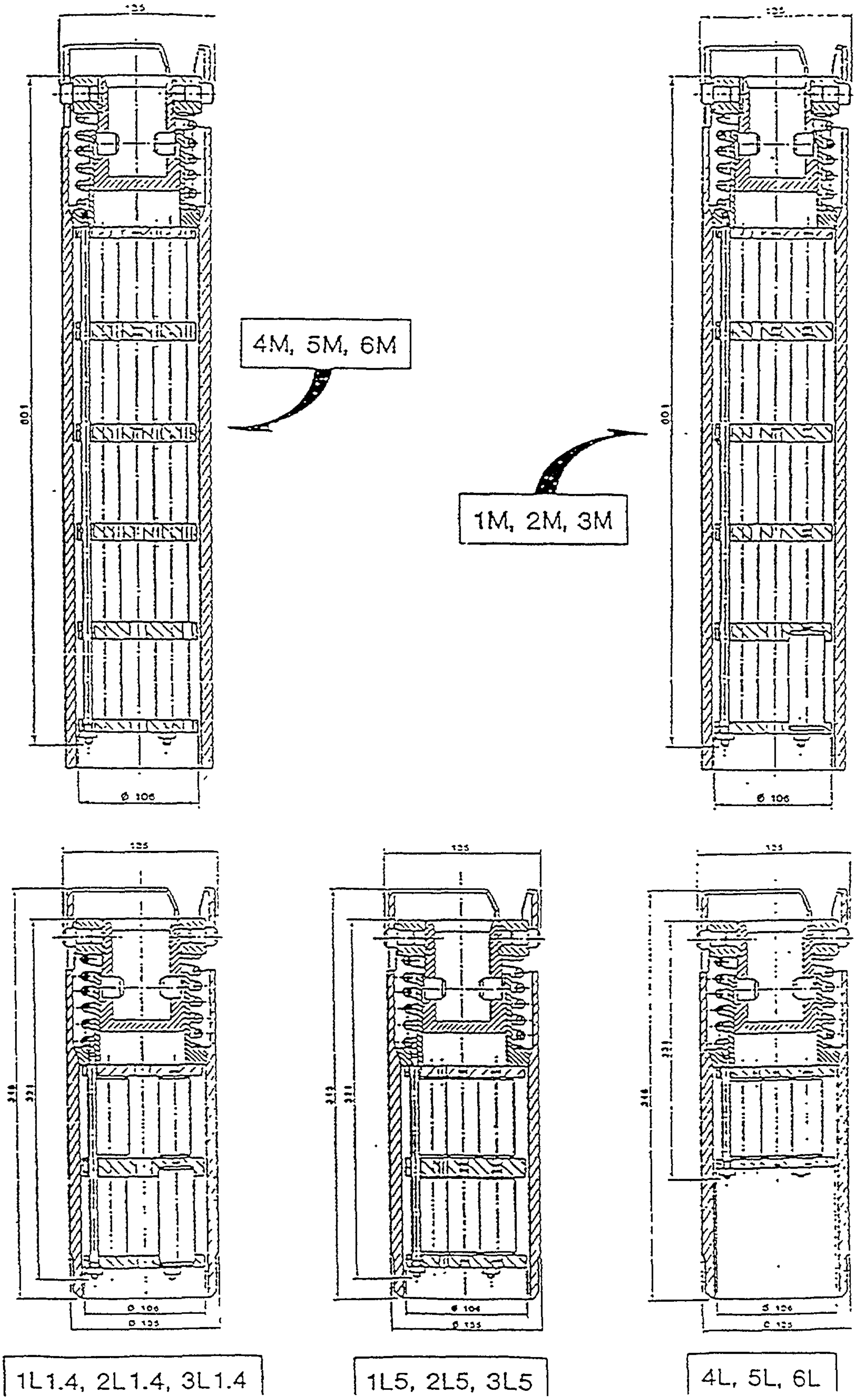

FIG. 38. WWER-1000 pressure vessel material surveillance specimen assemblies The assemblies hold five of the containers shown in Fig. 36 at each axial elevation.

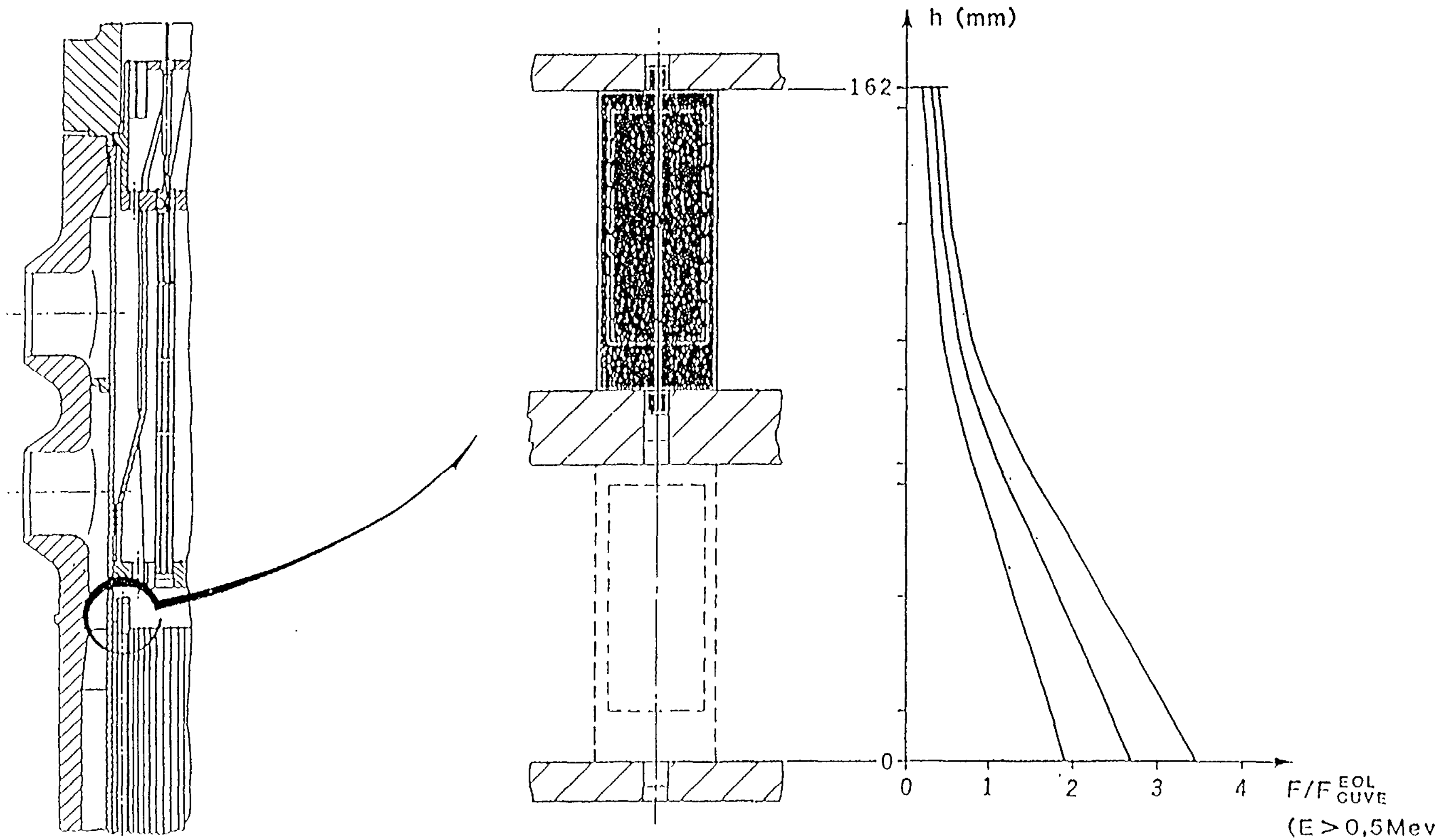

FIG. 39. A typical neutron flux distribution at the locations where the radiation damage surveillance specimens are placed in the WWER-1000 pressure vessels.

The material surveillance programmes for the three WWER-1000 pressure vessels built in the Czech Republic were modified as follows. The specimen specification was expanded to include static fracture specimens of the CT-0.5 type (as defined in ASTM E-399) as well as Charpy pre-cracked specimens for dynamic fracture toughness testing. The specimens were grouped in flat boxes rather than the round containers shown in Fig. 36 (maximum two layers of Charpy-type specimens) and located symmetrically in the active core region near the inner reactor vessel wall, at a small distance below the beltline centre line. This resulted in a lead factor between 1.5 and 2. The planned withdrawal time is 2, 5, 10 and 20 years, while 2 other specimen sets are used to determine possible annealing efficiencies as well as to evaluate further re-embrittlement after an annealing. Neutron fluence monitoring is assured by activation as well as by fission monitors, while temperature is monitored by the use of wire melt monitors.

Destructive examinations

For some WWER-440 type 230 reactors, the initial ductile to brittle transition temperature T_{k0}, as well as the exact chemical composition (phosphorus and copper content) in the weld metal was not measured. Thus, removing material (boat samples) from the pressure vessel is a potential way to obtain these data. However, some problems are connected with such a procedure, mainly:

- it cannot be used for sampling a stainless steel clad vessel from the inside,
- the weld metal is usually covered by a protective surface layer of low-carbon steel electrode material with a thickness up to 5 mm; the surface part of the sample, therefore, does not represent the real weld metal,
- sample dimensions are limited by the lowest allowable wall thickness; therefore, specimens for mechanical testing must be of a subsize type, thus necessitating the development of correlations between standard and subsize Charpy type specimens.
- sampling of the outer surface of the RPV is difficult because of the approximately 40 mm gap between the vessel and the water tank that provides biological shielding.

The phosphorus contamination in the weld metal as well as the mechanical properties of the weld metal vary across the RPV wall thickness. (The phosphorus concentration is somewhat higher near the surface due to geometry and solidification effects.) Therefore, it is difficult to decide whether the surface samples provide a conservative or non-conservative estimate of the material properties across the vessel wall thickness. Nevertheless, boat samples have been removed from unclad vessel inner surfaces, namely in Kozloduy Units 1 and 2, Novovoronezh Units 3 and 4, and Greifswald Units 1 and 2. The chemistry of the templates taken from the inside surfaces of the Novovoronezh Unit 3 and Kozloduy Unit 2 RPVs did not show any noticeable variation of chemical composition in the depth direction of the weld. Also, scrape samples were taken from the Kola Units 1 and 2 and Kozloduy Unit 1 RPV inside surfaces for chemical analysis. The measurements on these samples have been reported to be reliable.

Scrapes for chemical analysis were also taken from the outside of the clad vessels at Bohunice Units 1 and 2. However, due to a protective layer of undefined thickness, even a second and third sampling in the same position might contain a small amount of the low carbon steel cover layer. Recently, small boat samples were also removed from the Bohunice outer surfaces. No samples were taken from other clad vessels of this type.

Sampling from the inside of unclad vessels is a very promising method for measuring the residual transition temperature shift (ΔRT_{NDT}) after annealing. Even though subsize specimens must be used, the effectiveness of the annealing can be evaluated with a high degree of reliability using correlations between subsized and normal specimens. This method has been used in several RPVs after annealing — e.g. Novovoronezh Units 3 and 4, Kozloduy Unit 2, and Greifswald Units 1 and 2. The method has also been used for assessment of the re-embrittlement rate in service, for example, at Kozloduy Unit 1.

5.3.5. IAEA RPV surveillance database

Recently, the IAEA International Working Group on Lifetime Management of NPPs proposed the creation of a worldwide database which would store the results from the RPV surveillance programmes. The primary purpose would be to collect all accessible data from these programmes and specimens and then perform a more general analysis of these results than can be performed using national (or utility) databases, only. The RPV fabrication techniques are slightly different in some countries, even though the manufacturing is performed according to the same standards and general requirements. Thus, vessels from each of the manufacturers represent a family, which can be slightly different from the others. Consequently, results from one surveillance database may not be applicable to RPVs of other manufacturers. Creation of this database started in 1996 under the co-ordination of the aforementioned International Working Group.

5.4. TRANSIENT AND FATIGUE CYCLE MONITORING

5.4.1. Requirements in the USA

As discussed in Section 4.4, the only RPV components likely to experience significant fatigue damage are the RPV studs. However, fatigue can become a significant degradation mechanism if indications or flaws are detected during the RPV in-service inspection or if consideration is given to extending the operating life of the plant. In the former case, fatigue crack growth becomes important in the assessment and management of the ageing of PWR RPVs. In the latter case, fatigue cycles and loading to address Miner's Rule becomes important. In either case, transient and fatigue cycle monitoring is required.

5.4.2. Requirements in Germany

All German PWRs in operation are equipped with a fatigue monitoring system. On the basis of a plant specific weak point analysis of the NSSS, parameters to be monitored are defined and reported in a fatigue manual. Special emphasis is given to thermal loads such as thermal shocks, thermal stratification, and turbulent mixing phenomena which may occur very locally. These transients have been measured by means of special purpose instrumentation. (Thermocouples were installed on selected cross sections of interest.) In addition, global parameters such as internal pressure, fluid temperature, mass flow, water level, etc., have been measured via existing instrumentation and the data combined with the local parameters.

KTA 3201.4 contains requirements for recurring inspections. Parameters which affect the fatigue life must be monitored and the resulting fatigue compared to the design margins. Sophisticated software packages are available to recognize fatigue relevant loadings and to perform automatic fatigue evaluations. Thus the software tools not only satisfy the Code requirements but establish a data base for a reliable evaluation of the fatigue status, end of life predictions, or even life extension evaluations. Also, the German Reactor Safety Commission recommends that the fatigue status of every plant be updated after every 10 years of plant operation. The fatigue status and forecast have to be reported within the safety status report to be presented by the utility.

With respect to the RPV this means that the parameters to be monitored include: internal pressure, inlet and outlet loop temperature, and pressure vessel head temperatures at various locations on the outside surface. The reactor power is also monitored. In order to define the actual service condition several other parameters are made available. Following this way, the RPV nozzles, the flange and bolt connections and the RPV head are also monitored.

5.4.3. Requirements and practices in France

Electricité de France (EdF) implemented a procedure called "transient bookkeeping" when they began operation of their first PWRs and now have a database covering more than 540 reactor-years [108]. This procedure meets a regulatory requirement in the decree of February 26, 1974 and has allowed EdF to confirm that their operating transients are less severe than their design transients. EdF is now developing an automatic system called SYSFAC ("Système de Surveillance en Fatigue de la Chaudière") to survey the fatigue damage of the primary systems and the normal and auxiliary feedwater nozzles [109–112]. This computer system will compare the operating transients and the design transients, will count each type of transient, will record all the transient descriptions and will directly compute the usage factors for specific nozzles and pipes.

Transient bookkeeping relies on the information collected by the units operating sensors: primary loop temperatures, primary and secondary pressures, auxiliary line temperatures, and in addition some logic signals (valve positions, etc.). Approximately 40 parameters are measured and recorded at a 4-loop PWR. Instrumentation has not been installed to measure local phenomena such as flow stratifications; however, transfer functions have been developed to estimate the fatigue associated with such phenomena. The threshold values for calculating fatigue usage are a change in:

– primary loop temperature	5°C in 3 hours
– primary pressure	1 MPa
– secondary pressure	0.5 MPa
– auxiliary circuit temperature	20° at 40°C/hour

These thresholds have been estimated very conservatively. The calculations now being carried out within the framework of the SYSFAC process development show that for transients equivalent to the detection threshold, the calculated stress variations are far below the endurance limit for the most heavily loaded areas of the main primary system [112].

When a transient is detected, the design Transient File is inspected to find "an envelop transient," i.e., a transient at least as severe in terms of its contribution to fatigue. It may or may not be a transient of the same functional nature. In general, the operator looks for a transient of the same functional nature and then he checks the "envelope" character by comparing the amplitudes and rate of change of the various parameters [111]. Should this approach fail, the operator will have to develop a new category for the transient file. Each transient event is added to the number of previous events and compared with the design limits. Twice a year a balance sheet of all the transient consumptions is prepared to assure that the overall fatigue usage is acceptable for all Class 1 components.

5.4.4. WWER practices

The number and magnitude of the WWER plant transients are recorded daily by the main diagnostics systems. This system controls and measures all necessary transient parameters, mainly pressures and temperatures in different places in the reactor system. In some WWERs, a special system for automatic analysis of operational regimes is in service. Such systems are capable not only of recording all necessary data, but also of calculating the fatigue damage in chosen components using either preliminary fatigue usage factors or on-line calculated ones based on measured data.

6. AGEING ASSESSMENT METHODS

This section identifies and outlines methods for assessing age related degradation of an RPV. The assessment of the present and future RPV condition forms the basis for possible RPV maintenance to correct unacceptable degradation and for possible changes in (a) reactor operation to control ageing mechanisms of concern and (b) in-service inspection to monitor these ageing mechanisms and their effects. An age related degradation mechanism is considered significant if, when allowed to continue without any prevention or mitigation measures, it cannot be shown that the component would maintain a capability to perform its safety functions during future operation or that it may result in forced outages or other economic consequences.

6.1. RADIATION EMBRITTLEMENT ASSESSMENT METHODS

Requirements for assessing radiation embrittlement are specified in regulatory rules or regulatory guides issued by the responsible regulatory body in a given country. In the USA and in several other countries, requirements for assuring RPV integrity against brittle fracture are specified in Appendices G and H of 10 CFR Part 50 (discussed in Section 5) [25, 26]. Appendix G addresses operating requirements and Appendix H addresses RPV material surveillance requirements. Surveillance data is collected according to the guidelines of ASTM E185 which is referenced in Appendix H to 10 CFR Part 50. Surveillance data is factored into the assessment and management of ageing in PWR pressure vessels in accordance with guidelines set forth in Regulatory Guide 1.99 Revision 2 [38]. Appendices G and H to 10 CFR Part 50, Regulatory Guide 1.99 Revision 2 and the surveillance capsule data are used to assess and manage (a) the radiation embrittlement as it influences the P-T limitation curves (heatup and cooldown pressure and temperature limit curves), (b) the decrease in fracture toughness with radiation embrittlement, and (c) PTS.

The assessment methodology for radiation embrittlement is very straightforward. Namely, perform post-radiation mechanical testing of RPV material from surveillance capsules in accordance with a prescribed surveillance capsule withdrawal schedule. Once the surveillance capsule data are available, pressure/temperature curves can be generated using the methodology given in Appendix G to ASME Section III of the Boiler and Pressure Vessel Code [19] or other national Codes and discussed in Section 6.1.1 below. Appendix G to the ASME Code is discussed in Section 3 of this report. PTS calculations can be made using the methodology given in R.G. 1.99 Revision 2.

There are no specific requirements or procedures for determining the radiation embrittlement of the WWER pressure vessel materials. However, the following approaches are commonly applied.

For RPVs without surveillance specimens (i.e. the WWER-440 V-230 plants), the radiation embrittlement of the pressure vessel steel is estimated using the known chemical composition of the steel and the radiation embrittlement coefficients, A_F, from the Code for Strength Calculations [34]. The chemical composition used in these evaluations is checked by sampling the RPV, when possible. Also, the calculated radiation embrittlement values are compared with measurements from samples (trepans) taken from the RPV whenever possible.

For RPVs with surveillance specimens, the radiation embrittlement of the pressure vessel steel is again calculated using the chemical composition of the RPV material and the radiation embrittlement coefficients from the Code [34]. These results are compared with measurements from the surveillance specimens and the most conservative values from both are used to estimate the residual life of the RPV.

6.1.1. Radiation embrittlement assessment methods

The RT_{NDT} increases as the material is exposed to fast-neutron radiation. To find the most limiting RT_{NDT} at any time period in the reactor's life, the ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. Regulatory Guide 1.99, Revision 2 is used for the calculation of RT_{NDT} values at the 1/4 and 3/4 wall thickness locations in the beltline region as follows:

$$RT_{NDT} = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (24)$$

The initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class. The margin term accounts for the uncertainty in the initial RT_{NDT} and scatter in the data used to estimate ΔRT_{NDT}.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = [CF] f^{(0.28-0.10 \log f)} \qquad (25)$$

where CF is a chemistry factor which is a function of copper and nickel content. f is the fluence in $10^{23} n/m^2$, and ΔRT_{NDT} has units of Fahrenheit degrees [38].

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4 or 3/4 of the wall thickness), the following formula must first be used to attenuate the fluence at the specific depth:

$$f_{(\text{depth } X)} = f_{\text{surface}}(e^{-0.24x}) \qquad (26)$$

where x (in inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then put into Equation (25) along with the chemistry factor to calculate ΔRT_{NDT} at the specific depth.

6.1.2. Radiation embrittlement assessment methods in Germany

The German rules corresponding to the requirements in 10 CFR 50, Appendix G are in KTA 3201.1 and the KTA 3203 which require that the USE must remain above 68J (50 ft-lb) *during operation. If an upper shelf value of greater than or equal to 68J cannot be proven* by the surveillance programme, further measures have to be undertaken to confirm the safety of the vessel (such measures shall be defined in accordance with the authorized expert).

The German rules corresponding to 10 CFR 50, Appendix H as well as to ASTM E 185 are also in KTA 3203 (issued 03/84). The procedure to calculate the fracture toughness by use of the adjusted reference temperature (ART), in principle, is the same in KTA 3203 as in the US rules. Curves to predict the adjustment (the 41 J/30 ft-lb transition temperature shift) are given graphically in KTA 3203 and are dependent on fast neutron fluence (E >1 MeV), copper content and phosphorus content. These curves, which are quite similar to those given in NRC Regulatory Guide 1.99, are shown in Fig. 40 and were developed as an upper bound using data obtained for the 20 MnMoNi 5 5 and 22 NiMoCr 3 7 steels, and for welding materials such as NiCrMo 1 and S3 NiMo 1. This data base largely overlaps with that of the NRC Regulatory Guide but does not contain the older non-nickel-alloyed steels such as A 302 B and A 212 B. In the light of more recent surveillance data, the curves seem to be overly conservative at high fluences (see Fig. 40) but slightly underestimate the 41 J transition temperature shift at low fluences ($<10^{22}$ n/m^2). The latter is due to the lack of data in this fluence region at the time the curves were developed and is of no safety relevance due to the small absolute values of the shift.

6.1.3. Radiation embrittlement assessment methods in France

The French requirements are specified in the 1974 Order and the corresponding rules are presented in RCC-M "Design and Construction Rules for the Mechanical Components of PWR Nuclear Islands" [27–29] and RSE-M "In-service Inspection Rules for the Mechanical Components of PWR Nuclear Islands" [30].

The following formula must be used to calculate the change in the reference nil-ductility transition temperature, ΔRT_{NDT}:

$$\Delta RT_{NDT}(°C) = (8+(24+1537(P-0.008)+238(Cu-0.08)+191.Ni^2(Cu))f^{0.35} \qquad (27)$$

where P is the weight % phosphorus, Cu is the weight % copper, Ni is the weight % nickel, and f is fluence in 10^{23} n/m^2. This equation was specifically developed for the French RPV material and is based on a large number of measurements. Note that this formula is different than the equation specified in USNRC Regulatory Guide 1.99, Rev. 2, and shown above as Equation (25). Changes in both base and weld metal reference nil-ductility transition temperatures calculated with Equation (27) are compared with measurements in Fig. 41. Only four measured values of ΔRT_{NDT} out of 150 measurements exceed the predicted values, which suggest that the correlation is conservative. The fracture surfaces on the four specimens where the measured ΔRT_{NDT} exceed the calculated ΔRT_{NDT}, suggest an intergranular failure (the final interpretation of these results is still in progress).

The quality of a RPV surveillance programme is strongly related to:

- the spectrum, flux, and temperature of the capsule locations,
- whether the material used for the specimen is representative: base and weld metal, clad if necessary,
- the quality of the neutronic and thermal instrumentation of the capsules, and

- the quality of the neutronic and thermal instrumentation of the capsules, and
- the quality of the neutronic and thermal evaluations of the RPV hot point (location of highest radiation damage).

The French regulations require an anticipation factor less than 3. However, a value above 2 is used and the end-of-life material properties are known before the second 10-year inspection. The surveillance programme is just used to check the design shift in order to confirm the pressure/temperature diagram and the design flaw evaluation.

For PTS assessments, a specific study has been developed as complementary rules.

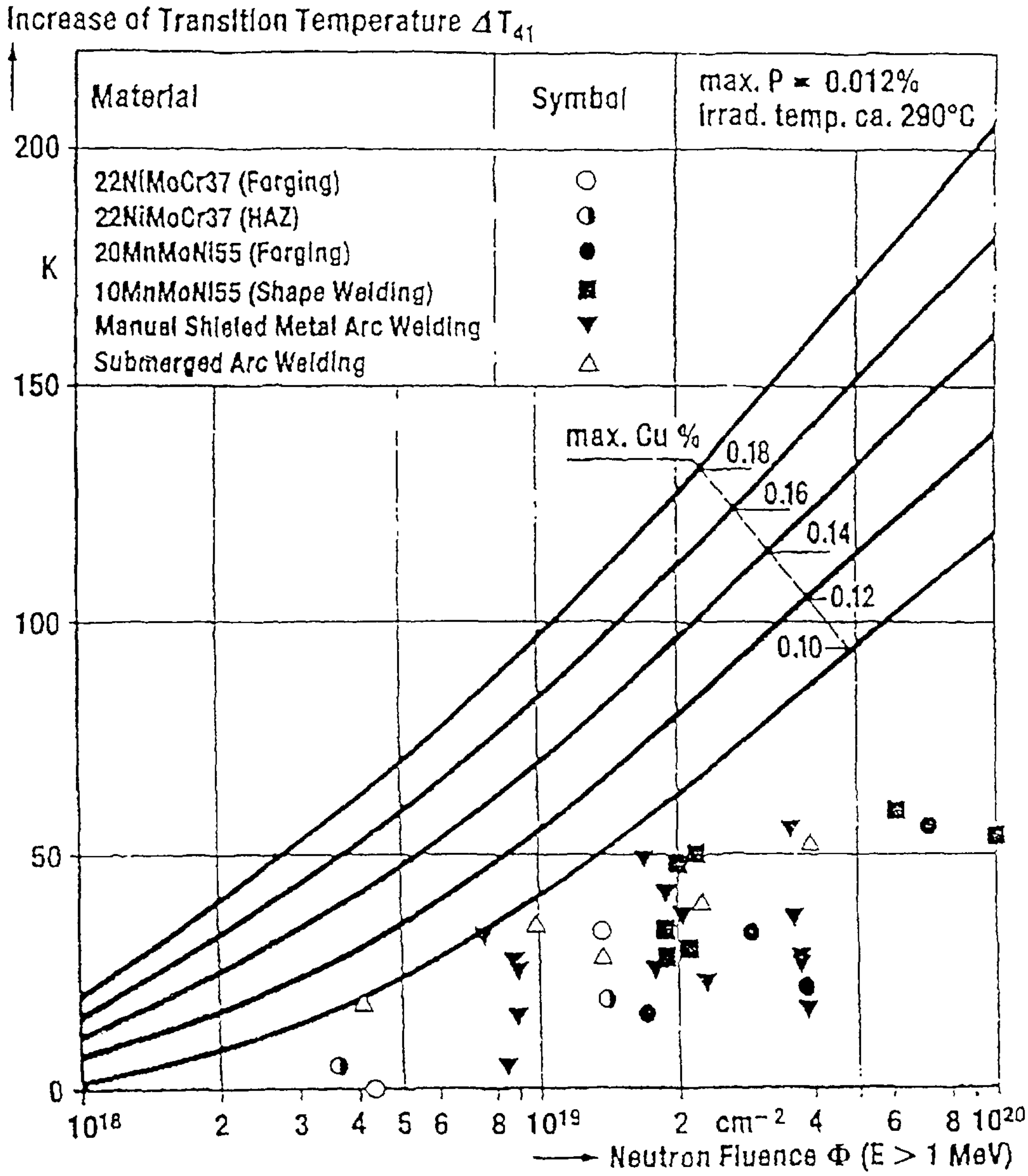

FIG. 40. Increase in the 41J transition temperature (ΔT_{41}) as a function of neutron fluence for German RPV steels with a copper impurity content less than 0.10% and phosphorus impurity content less than 0.012%. Both data and the KTA 3203 design curves are plotted.

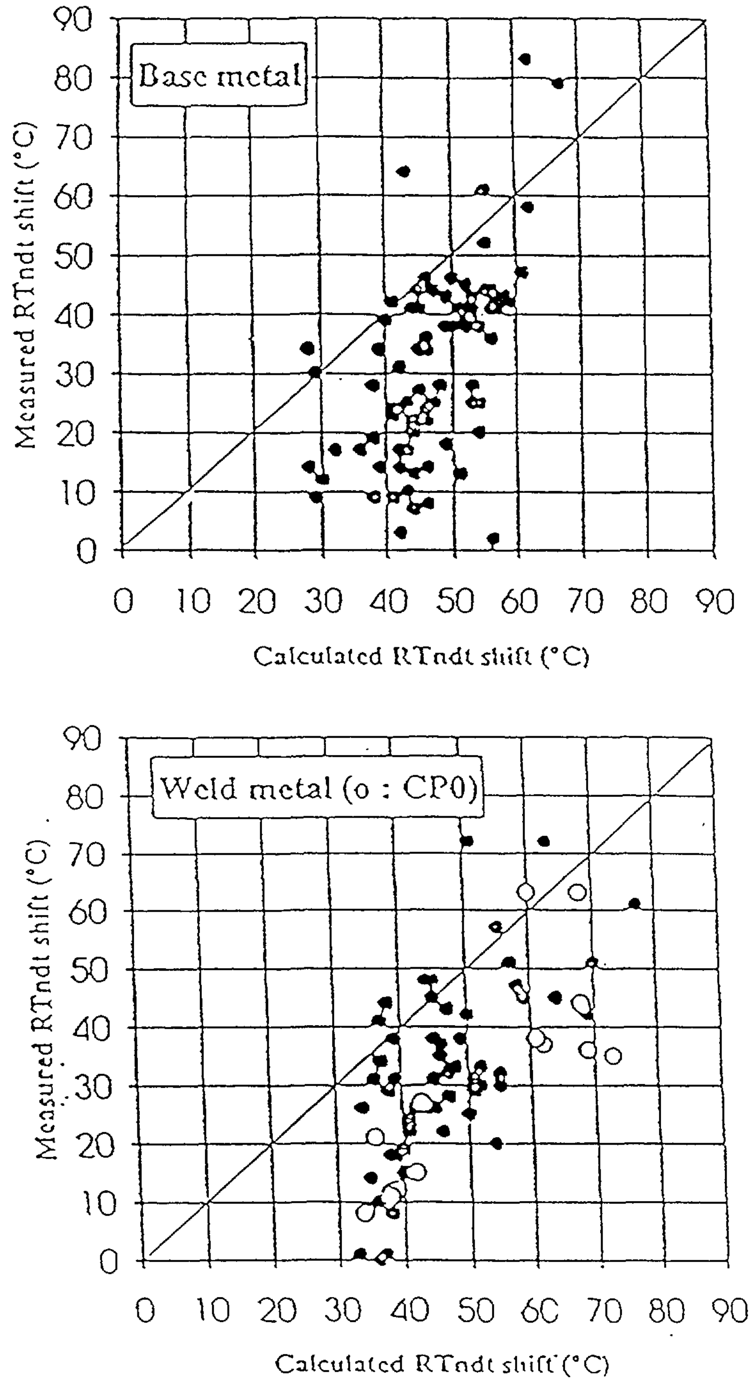

FIG. 41. Comparison of measured and calculated RT_{NDT} shifts in French RPV steel.

6.1.4. WWER radiation embrittlement assessment methods

Even though most of the WWER reactors, including the WWER-1000 and the WWER-440/V 213 types, have RPV surveillance specimen programmes, the Soviet Code [34] is based on calculations. The primary reason for this is because the Code was prepared for RPV design, rather than plant operation. The WWER design basis is discussed in Section 3.5 of this report. Section 3.5.3 discusses the stress analysis procedures including stress categories, stress intensity limits, areas of the RPV to be analysed, and stress analysis methods. Section 3.5.4 discusses design and analysis against brittle fracture including the allowable fracture toughness, postulated defects, stress intensity factors, and transition temperatures and temperature shifts.

The following approach is used in the Russian Codes currently being prepared:

- The change in the transition temperature is measured using both Charpy V-notch impact and static fracture toughness tests. The shift in the transition temperature is then compared to the value calculated using the A_F coefficient in the code and the adjustments for thermal ageing and fatigue (Equations (12) and (13)). The highest value is then used for the assessment.

- However, a new radiation embrittlement trend curve for a given RPV material can be constructed if values from at least three different neutron fluences are obtained and statistically evaluated and temperature safety factors of 10°C for the base metal and 16°C for the weld metal are added.

This overall approach is consistent with the definition of the A_F coefficient as the upper bound value of the experimental data.

6.2. THERMAL AGEING ASSESSMENT METHODS

As discussed in Section 4.2, there is no evidence of a significant change in the ΔRT_{NDT} due to thermal embrittlement. Therefore, we do not recommend the use of any thermal ageing assessment methods.

6.3. FATIGUE ASSESSMENT METHODS

6.3.1. Fatigue assessments in the USA

Evaluation of fatigue damage can be based either on the crack initiation stage or the crack propagation stage. The former is typified by the fatigue design procedures of the ASME Code Section III [18], Subsection NB-3200, while the later is exemplified by the flaw evaluation/acceptance procedures of the ASME Code Section XI [21], Subsection IWB.

Crack initiation

Crack initiation is estimated by determining the fatigue usage at a specific location which results from either actual or design-basis cyclic loads. The time-to-initiation can be predicted only if the applied load sequences and recurrence frequencies are known. If the cycling loading is random, estimates of time to initiation are uncertain.

For a fatigue life evaluation, the data needed are the amplitude and number of stress cycles experienced during a given operating period and the amplitude and number of cycles that lead to crack initiation in laboratory specimens. The sum of the ratios of these quantities gives the cumulative fatigue usage factor. The best source of information for the relatively newer US plants is the certified stress report and the design specification. The certified stress report gives the design-basis cumulative usage factors for vessel components and the Code allowable number of cycles for prescribed events.

The fatigue usage factor is defined according to ASME Code requirements. This value must not exceed 1.0 during the design life of the component. With the conservatism inherent to this calculation, it is presumed that fatigue crack initiation can be prevented by ensuring that the fatigue usage factors remain below the limit of 1.0. The ASME Code fatigue design curves are based on data from smooth-bars tested at room temperature in air. The ASME Code applies a factor of 2 on strain range and a factor of 20 on the number of cycles to the smooth-bar data. The factor of 20 on cycles accounts for data scatter, size effect, surface finish and moderate environmental effects.

Cyclic crack growth

Once a crack has initiated, either by fatigue or some other mechanism such as SCC, continued application of cyclic stresses can produce subcritical crack growth. The Paris crack growth relationship is used to calculate crack growth:

$$da/dN = C(\Delta K)^n \tag{28}$$

where

da/dN	=	fatigue crack growth rate (distance/cycle);
ΔK	=	stress intensity factor range = $(K_{max} - K_{min})$;
C,n	=	constants, related to material and environment; and
K_{max}, K_{min}	=	maximum and minimum stress intensity factors during the loading cycle

Crack growth rates, such as those in the ASME Code, are not constant for all ranges of ΔK. There are three regimes. These are: crack growth at low, medium and high ΔK values. At very low ΔK values, the growth rate diminishes rapidly to vanishingly low levels. A threshold stress intensity factor range (ΔK_{th}) is defined as that below which fatigue damage is highly unlikely.

At the high end of the ΔK range, crack growth increases at a faster rate. This acceleration is partially a result of the increasing size of the plastic zone at the crack tip, which has the effect of increasing the effective stress intensity factor range (ΔK_{eff}). In addition, as the maximum applied stress K_{max} approaches the critical applied stress intensity (K_c), local crack instabilities occur with increasing frequency. Increasing the R ratio (K_{min}/K_{max}) causes an increase in cyclic crack growth rate.

Knowing the history of stress cycle events in conjunction with the appropriate crack growth correlations allows the prediction of crack growth rate in components. Furthermore, information in Section XI of the ASME Code on crack initiation and crack arrest fracture toughness of low alloy steel can be used to calculate the critical crack size of the component, and thus time to failure, or residual life.

The preceding discussions are strictly valid only for metallurgically large cracks (in the literature, the minimum size of a metallurgically large crack ranges from approximately 0.0025 to 0.13 mm (0.0001 to 0.005 in.)). For short cracks (i.e., crack sizes comparable to the size of the high stress field at the tip of the stress raiser at the crack initiation site), the applicability of analyses based on LEFM tends to break down in some instances. Various attempts have been made to address the growth rates of short cracks, but a universally applicable treatment has yet to be established. However, the inspections conducted in accordance with the ASME Code are sufficient to detect crack growth before the acceptance criteria are reached.

It should be noted that for indications found and sized during ISIs such that crack growth evaluation is required, LEFM-based crack growth procedures are adequate and sufficient. Very short cracks for which LEFM-based procedures are not applicable are within acceptance criteria limits for size.

The crack size at the end of a prescribed period of operation can be determined if the cyclic loading sequence is known and a crack growth curve (da/dN versus ΔK), such as that in ASME Section XI, Article A4300, is available.

Fatigue assessment for extended NPP operation

For component locations and parts with no history of fatigue damage, the current ASME Code Section III, Subsection NB-3000 fatigue design basis can be shown to remain valid throughout the design life; in this case, the original design-basis transients. The total usage factor must be shown to be valid, in terms of the numbers and severity of the loads, for any extended operation and the calculated fatigue usage factor, including any modifications to the design-basis transients to account for actual plant operating transients not enveloped by the original design-basis transients, must be shown to be less than unity. The Section III, Subsection NB fatigue evaluation procedures remain valid for these calculations. If the projected fatigue usage factor for the extended operation exceeds unity, detailed fatigue reanalysis considering actual plant operating transients, including partial cycle counting, in lieu of the original assumed design-basis transients, may be used. The fatigue usage factor limit for this reanalysis remains unity. The Section III, Subsection NB evaluation procedures remain valid for these calculations.

For component locations and parts with a history of fatigue damage, or as an alternative to the analytical verification of the adequacy of the original fatigue design basis throughout the design life, an effective in-service examination programme for managing the effects of potentially significant fatigue damage is needed. Formal inservice examination requirements are provided for each plant in its plant ISI and Inservice Testing programmes and are referenced to an applicable edition of the ASME Code Section XI Rules for ISI of Nuclear Power Plant Components. The plant ISI programme, including any commitments to enhanced or augmented inspections as the result of plant operating experience or regulatory enforcement and any special reassessments of loading and material conditions, provides an acceptable basis for continued operation of a component. The intervals for these examinations and the requirements for expansion of the number of locations examined if flaws are detected, assure that significant undetected fatigue degradation of components will not occur.

If the confirmation of the current fatigue design basis for an extended operation is to be demonstrated, the procedure to be followed is similar to that used during the initial plant design. During the design of plant components, in accordance with NB-3000, a set of design-basis transients was defined. These design-basis transients, as described by temperature, pressure, flow rate and number of occurrences, were intended to conservatively represent all transients expected during the design life of the plant. The plant technical specifications require that major cycles be tracked during service, relative to actual operating transients, to assure satisfaction of fatigue design requirements. However, since details of the technical specification transient tracking requirements vary widely from plant to plant, the demonstration that the design-basis transients remain valid for any extended operation, such that the numbers and severity of actual operating transients remain enveloped, is a plant-specific consideration. A variety of methods are available for this demonstration. These include regrouping of design-basis transients, taking credit for partial (versus full) cycle transients, use of actual plant transients rather than design-basis transients, or using a more sophisticated cycle monitoring programme.

The second step in the fatigue design basis confirmation process is demonstrating that the fatigue usage factor calculated for the most critical component location or part remains below unity, as determined by the use of the confirmed design-basis transients extended through the operation. The fatigue analysis procedures of NB-3000 remain valid for these calculations. The ASME Section III rules require that fatigue usage factors calculated for this extended period remain below unity. If this criterion is satisfied, the component is presumed safe (i.e., no fatigue cracks have been initiated).

For components with a reasonably high degree of design margin of safety with regard to fatigue limits, acceptable results for extended life can be demonstrated by conservative evaluation. For more limiting components, a conservative approach may project cumulative fatigue usage factors which approach or exceed a value of 1.0. Unless the excessive conservatism can be removed, more frequent ISIs may be required or, in the worst case, replacement or refurbishment may be recommended far too prematurely.

One way to remove conservatism is to refine the fatigue analysis. The methodology can be enhanced from simple elastic calculations to elastic-plastic or even fully plastic approaches. The definition of loading cycles can also be refined, including regrouping of design basis transients. Credit can be taken for partial versus full design basis transients. Actual plant loading cycles can be used instead of originally assumed design loading cycles. These alternative techniques can be implemented in a manner that is consistent with the ASME Code to show that fatigue damage accumulation will remain within established limits for any extended operation.

Finally, if a refined fatigue analysis is unable to show that the component will remain within the established limits, the component can be examined for detectable fatigue damage and repaired, refurbished or replaced as appropriate.

6.3.2. Fatigue assessments in Germany

The procedure as described in the ASME Code for the assessment of crack initiation and cyclic crack growth is basis for the relevant stipulations in the German KTA 3201.2.

6.3.3. Fatigue assessments in France

The RCC-M general rules [27] and (S,N) fatigue curves are similar to the ASME Section III B3000 rules. However, some specific rules have been developed and incorporated into RCC-M to analyse crack-like defects (RCC-M Appendix ZD), studs (use of experimental results), and plastified areas by Ke optimization, which are not in the ASME Code.

To analyse in-service crack growth, a specific and complete set of rules and material properties are presented in RSE-M A5000 and the corresponding appendices.

All the fatigue re-analysis use the current design procedures except, when it is possible, a different cycle combination procedure ("rain flow").

One major difference with the ASME Code is the effects of the cladding:

- for crack initiation, the stress versus number of cycles curve used is the air environment curve for ferritic steels, and

- for crack growth analysis, the (da/dN, ΔK) curve used is the air environment curve for ferritic steels, without any free surface proximity criteria for underclad defects.

6.3.4. WWER fatigue assessments

Fatigue evaluations

The peak stresses are the main concern in the WWER fatigue evaluations. The Code gives specific rules for fatigue calculations and design curves for different materials as well as fatigue strength reduction factors for welded joints and for some operational factors such as radiation and corrosion.

Two methods are allowed in the Code for determining the fatigue:

(a) design curves for a rough estimate,

(b) design formulas for more detailed calculations or when the design curves cannot be satisfied.

Generally, the following safety factors are used:

- pressure vessel materials:
  $n_\sigma = 2$,
  $n_N = 10$

- bolting materials:
  $n_\sigma = 1.5$,
  $N_n = 3$.

where the factor n_o is applied to the stress and the factor n_N is applied to the number of cycles in the same manner as in the ASME Code. The stress and number of cycle factors listed above are lower for the bolting materials than for the pressure vessel materials because the bolting is changed out periodically. Also, bolting failure should result in leakage rather than rupture. Although the above values are not large, it should be noted that the design curves, as well as the design formulas, are a lower bound of all the experimental data. Moreover, the coefficients $\varphi_w \leq 1$ are incorporated into the calculational formulas. These coefficients conservatively adjust the formulas for the effects of the welded joints on the fatigue life.

Cumulative usage factors are calculated using linear Miner's law; the maximum allowable value is equal to one.

6.4. ASSESSMENT METHODS FOR PWSCC OF ALLOY 600 COMPONENTS

Based on the stress and temperature dependencies as described by Equations (20) and (21) in Section 4.5.1, the damage rate for PWSCC of Alloy 600 can be described by an equation of the form:

$$\text{damage rate} \approx \sigma^4 \exp(-Q/RT) \qquad (29)$$

The time to crack initiation t_i, is given by

$$t_i \approx (\text{damage rate})^{-1} = A\sigma^{-4} \exp\,(Q/RT) \qquad (30)$$

where the constant A is a scaling factor determined by using some standard stress level and reference temperature. The value of A will change whenever there is a systematic change in the material characteristics, the average stress level at the location of interest, or other conditions that may depend on the type of component and differ from plant to plant.

Primary water stress corrosion crack growth can be calculated with the following empirical equation developed by Scott [113]:

$$da/dt = 2.56e^{-(33\,000/RT)}\,(K_I - 9)^{1.16} \qquad (31)$$

where da/dt is the PWSCC growth rate, K_I is a crack tip stress intensity factor in MPa · $m^{1/2}$, T is temperature in degrees kelvin, and R is the universal gas constant. This model was developed by Scott using data obtained by Smialowski et al. of Ohio State University. The specimens used by Smialowski et al. were machined from flattened halves of short lengths of Alloy 600 steam generator tubing and exposed to 300°C water with various chemistries. Only the data from the tests run with standard PWR water chemistry (2 ppm Li, 1200 ppm B, and pH of 7.3) were used by Scott to develop his correlation. In addition, he adjusted the Smialowski et al. data by a factor of 10 to account for the effects of cold work. The specimens used by Smialowski et al. contained significant cold work, whereas, cold work is present in Alloy 600 CRDM nozzles only in a thin layer on the inside surface. Alloy 600 stress corrosion crack growth rate tests performed in 400°C hydrogenated steam and in 360°C PWR primary coolant water environments have shown that 5% cold work leads to crack growth rates between 5 and 10 times faster than those observed in Alloy 600 materials without cold work [114].

The Smialowski et al. data were obtained with a coolant temperature of 300°C. Higher temperatures will cause higher crack growth rates. Scott added the effects of temperature with an Arrhenius-type equation and an activation energy of 33 Kcal/mole. This activation energy is based on the available Alloy 600 PWSCC rates estimated from laboratory tests and plant data for steam generator tubes.

The Scott model indicates that the crack growth rate is proportional to the increase in the stress intensity factor above a threshold value of 9 MPa · $m^{1/2}$ raised to the 1.16 power. This appears reasonable because the data of Rebak et al. [115] also indicate that there is a threshold value of the crack tip stress intensity factor of 5 to 10 MPa · $m^{1/2}$.

6.5. ASSESSMENT METHODS FOR RPV CLOSURE HEAD STUD STRESS CORROSION CRACKING

Once SCC is suspected, detection and sizing of any cracks are required for determining the effects on the RPV closure head studs. In-service inspection by volumetric means, such as ultrasonic testing (UT) is the only way to size SCC indications. Visual examination or dye-penetrant methods may detect SCC flaws but these techniques can only measure the length of the flaw on the surface.

Once flaws are detected and sized in RPV components such as closure head studs, analytical evaluation utilizing fracture mechanics is required to predict life remaining after the initiation of the detected flaw. As with the age related degradation mechanism fatigue, the sub-critical crack growth must be determined to assess and manage SCC in RPV components. As discussed in Section 3, the ASME Boiler and Pressure Vessel Code, Section XI, Appendix A provides an analytical technique for assessing crack growth during the application of cyclic stresses. However, SCC being corrosion driven does not require cyclic loading for the SCC initiation flaw to grow. Therefore, information is required in terms of delta "a" versus delta "t" (da/dt, change in crack length with time).

In summary, volumetric ISI in conjunction with an analytical evaluation is a requirement for the assessment and management of stress corrosion cracking in the PWR RPV.

6.6. ASSESSMENT METHODS FOR BORIC ACID CORROSION

Uniform corrosion or pitting by an aggressive reactant on a metal surface is the most common form of corrosion. The RPV internal surface is generally clad with austenitic stainless steel which provides excellent resistance to corrosion. The outside of the RPV, the vessel flange, stud holes and RPV studs are of concern with respect to corrosion due to the exposure of these ferritic components to boric acid. While boric acid corrosion is considered preventable, boric acid leakage has occurred during the operation of PWR RPVs.

Boric acid corrosion due to leaking reactor coolant has resulted in wastage of the low alloy steels of the RPV flanges, top closure heads and RPV studs at a rate of approximately 25 mm/year. Once a boric acid leak is detected, the wastage level of the given ferritic steel component must be determined. An assessment must be made to determine if the minimum design thicknesses for the given component have been violated. If the wasted component design thickness is violated, refurbishment by welding may be required. If the component's design thickness is marginal following detection of boric acid attack, an analytical evaluation is required to assess the component's "fit for service" status.

6.7. FLAW ASSESSMENT METHODS

6.7.1. Flaw assessment methods in the USA

Article IWA-3000, "Standards for Examination Evaluation", requires evaluation of flaws detected during the inservice examination. The acceptance standards for flaws detected during the ISI are given in IWB-3500, "Acceptance Standards". Flaws that exceed the allowable indication standards of IWB-3500 can be analysed in accordance with Appendix A "Analysis of Flaws" [116] to determine their acceptability. Appendix A to Section XI uses a procedure based upon the principles of LEFM for analysis of flaw indications detected during

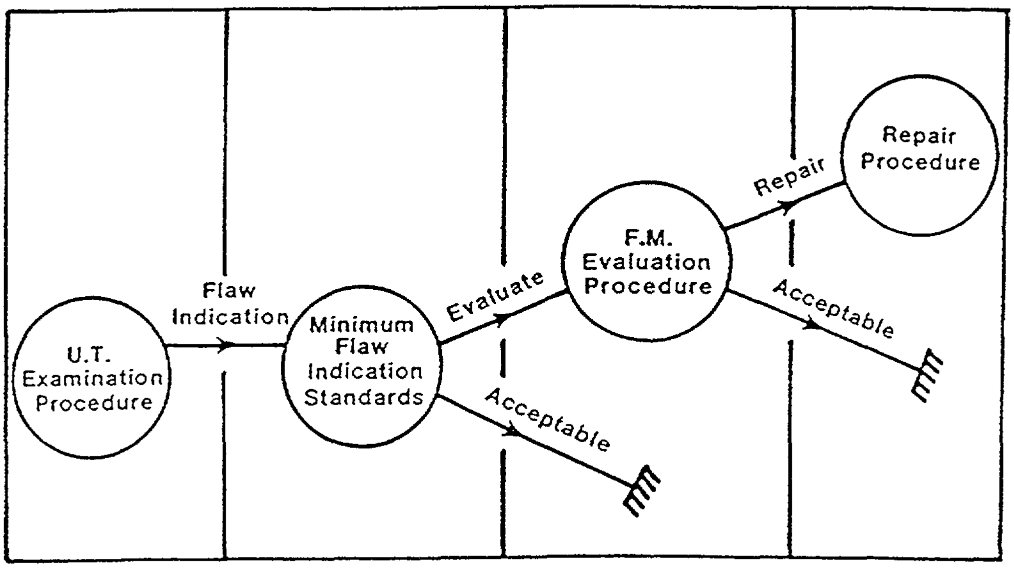

FIG. 42. Functional organization of ASME section XI in-service inspection documents.

ISI. While Section III is a construction code, Section XI provides rules for the integrity of the structure during its service life. The concepts introduced in Appendix G to Section III are carried over to Appendix A to Section XI. Figure 42 shows the functional organization of ASME Section XI. The evaluation procedure can be summarized as follows: set up a simplified model of the observed flaw, calculate stress intensity factors, determine appropriate material properties, determine critical flaw parameters and apply acceptability criteria to the critical flaw parameters.

Models for flaw analysis are given in A-2000 of Appendix A. Definitions are given covering flaw shape, proximity to closest flaw, orientation and flaw location to permit their application into an analytical model for LEFM.

Methods for K_I determination are given in A-3000 of Appendix A. Article A-3000 defines how the applied stresses at the flaw location can be resolved into membrane and bending stresses with respect to the wall thickness and presents a stress intensity factor expression for the flaw model.

Article A-4000 defines the material properties in terms of the fracture toughness of the given material K_{IC} and K_{Ia} (it should be noted that K_{Ia} is equivalent to K_{IR} of Section III) and in terms of the fatigue crack growth rate. As in Appendix G to Section III, the K_{IC} and K_{Ia} versus temperature curves are indexed using RT_{NDT}. For materials that are subjected to radiation, the degradation of the material fracture toughness due to the radiation must be accounted for. This is done through increasing RT_{NDT} by the appropriate indications from standard Charpy impact toughness tests on surveillance programme specimens.

An upper bound curve for fatigue crack growth data was measured on A 533 Grade B Class 1 and A508 steels and included the effects of temperature, frequency of load application and the pressurized water environment.

Finally, Article A-5000 gives the guidelines for determining the critical flaw parameters. These parameters are used in judging the acceptability of the observed flaw:

a_f	=	the maximum size of the observed flaw due to fatigue crack growth
a_{crit}	=	the minimum critical size of the observed flaw under normal operating conditions
a_{init}	=	the minimum critical size for initiation of non-arresting growth of the observed flaw under postulated accident conditions.

After these parameters are determined, they are compared to the acceptance criteria:

$$a_f < 0.1\, a_{crit} \quad \text{or} \quad a_f < 0.5\, a_{init} \tag{32}$$

If these criteria are met, the observed flaw need not be repaired.

Evaluation of flaws in reactor pressure vessels with charpy upper-shelf energy less than 68 J (50 ft-lb)

As discussed in Section 3.2, Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements" [25], requires, in part, that reactor vessel beltline materials must maintain an upper-shelf energy of no less than 68 J (50 ft-lbs), unless it is demonstrated in a manner approved by the USNRC that the lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G to Section III of the ASME Code. In September 1993, the USNRC published draft Regulatory Guide DG-1023, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 ft-lbs" [117]. This Regulatory Guide provides criteria which are acceptable to the USNRC for demonstrating that the margins of safety against ductile fracture are equivalent to those in Appendix G to Section of the ASME Code. The acceptance criteria are to be satisfied for each category of the transients; namely, Levels A and B (normal and upset), Level C (emergency) and Level D (faulted) conditions.

Two criteria must be satisfied for Level A and B conditions, as described below for a postulated semi-elliptical surface flaw with a flaw depth to wall thickness ratio (a/t) equal to 0.25, an aspect ratio or surface length to flaw depth of 6 to 1 and oriented along the material of concern. If the base metal is governing, the postulated flaw must be axially oriented. Smaller flaw sizes may be used on an individual case basis if a smaller size of the above postulated flaw can be justified. The expected accumulation pressure is the maximum pressure which satisfies the requirement of ASME Section III, NB-7311(b). The two criteria are:

(1) The crack driving force must be shown to be less than the material toughness as given below:

$$J_{applied} < J_{0.1} \tag{33}$$

where $J_{applied}$ is the J-integral value calculated for the postulated flaw under pressure and thermal loading where the assumed pressure is 1.15 times expected accumulation pressure, and with thermal loading using the plant specific heatup and cooldown conditions. The parameter $J_{0.1}$ is the J-integral characteristic of the material resistance to ductile tearing ($J_{material}$), as usually denoted by a J-R curve, at a crack extension of 2.54 mm (0.1 inch).

(2) The flaw must be stable under ductile crack growth as given below:

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da} \tag{34}$$

(or with the load held constant, $J_{applied}$ must equal $J_{material}$) where $J_{applied}$ is calculated for the postulated flaw under pressure and thermal loading for all service Level A and B conditions and the assumed pressure is 1.25 times expected accumulation pressure, with a thermal loading as defined above.

The J-integral resistance versus crack growth curve used should reflect a conservative bound representative of the vessel material under evaluation.

For Level C conditions when the upper shelf Charpy energy of any material is less than 68 J (50 ft-lb), postulate interior semi-elliptic surface flaws with their major axis oriented along the material of concern and the flaw plane oriented in the radial direction. Postulate both interior axial and circumferential flaws and use the toughness properties for the corresponding orientation. Consider surface flaws with depths up to one tenth the base metal wall thickness, plus the clad, but with total depth not to exceed 25.4 mm (1.0 inch) and with aspect ratios of 6 to 1 surface length to flaw depth. Similar flaw sizes may be used on an individual case basis if a smaller size can be justified. For these evaluations, two criteria must be satisfied, as described below:

(1) The crack driving force must be shown to be less than the material toughness as given below:

$$J_{applied} < J_{0.1} \tag{35}$$

where $J_{applied}$ is the J-integral value calculated for the postulated flaw in the beltline region of the reactor vessel under the governing level C condition. $J_{0.1}$ is the J-integral characteristic of the material resistance to ductile tearing ($J_{material}$), as usually denoted by a J-R curve test, at a crack extension of 2.4 mm (0.1 inch).

(2) The flaw must also be stable under ductile crack growth as given below:

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da} \tag{36}$$

(or with the load held constant, $J_{applied}$ must equal $J_{material}$) where $J_{applied}$ is calculated for the postulated flaw under the governing level C condition. The J-integral resistance versus crack growth curve shall be a conservative representation of the vessel material under evaluation.

For Level D conditions when the upper shelf Charpy energy of any material is less than 68 J (50 ft-lb), postulate interior semi-elliptic surface flaws with their major axis oriented along the weld of concern and the flaw plane oriented in the radial direction with aspect ratio of 6 to 1. Postulate both interior axial and circumferential flaws and use the toughness properties for the corresponding orientation. Consider postulated surface flaws with depths up to one tenth the base metal wall thickness, plus the clad, but with total depth not to exceed 25.4 mm (1.0 inch) and with aspect ratios of 6 to 1 surface length to depth. Smaller flaw sizes may be used on an individual case basis if a smaller size can be justified. For these evaluations, the following criterion must be met.

The postulated flaw must be stable under ductile crack growth as given below:

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da} \tag{37}$$

(or with the load held constant, $J_{applied}$ must equal $J_{material}$) where $J_{applied}$ is calculated for the postulated flaw under the governing level D condition. The material property to be *used for this assessment is the best estimate J-R curve.*

6.7.2. Flaw assessment methods in Germany

Indications found during ISI have to be considered as being cracks and have to be evaluated on basis of linier elastic fracture mechanics evaluations. Conservatively, the crack has to be treated as a surface crack with an aspect ratio of:

$$a/2c = 1/6 \tag{38}$$

The maximum allowable defect size is defined by the criteria:

$$K_{Imax} = K_{Ic}/1.5 \tag{39}$$

Elasto-plastic fracture mechanics approaches and other advanced methods are only applied and accepted in individual cases. General stipulations for their implementation into the KTA Code are under preparation. Specific requirements for vessels with a Charpy USE less than 68 J (50 ft-lb) are not presented, as there are no RPVs operating in Germany to which this criteria would apply within their design life.

6.7.3. Flaw assessment methods in France

A complete set of rules has been developed and published in RSEM [30, 118] including flaw geometry standards, fatigue crack growth and rupture analysis guidelines, fracture mechanics parameter evaluation guidelines, material properties, etc. All the acceptance criteria are based on elastoplastic fracture mechanic methods with specific safety factors for brittle and ductile behaviour that are completely finalized. As an example, the proposed criteria for the end-of-life flaw are:

for Level A: $T < RT_{NDT} + 50°C$ $\quad K_{cp}(1.2C_A, 1.3a_f) \leq K_{IC}/1.5$ (40)
$T > RT_{NDT} + 50°C$ $\quad J(1.2C_A, 1.3a_f + \Delta a) \leq J_{\Delta a}/1.5$

for Level C: $T < RT_{NDT} + 50°C$ $\quad K_{cp}(1.1C_C, a_f) \leq K_{IC}/1.4$ (41)
$T > RT_{NDT} + 50°C$ $\quad J(1.1C_C, a_f + \Delta a) \leq J_{\Delta a}/1.5$

for Level D: $T < RT_{NDT} + 50°C$ $\quad K_{cp}(C_D, a_f) \leq K_{IC}/1.2$ (42)
$T > RT_{NDT} + 50°C$ $\quad J(C_D, a_f + \Delta a) \leq {}_{J\Delta a}/1.2$

and a limited tearing crack growth or a crack arrest through the thickness of the vessel

where C_A, C_C and C_D are the Level A, C and D loads; a_f is the end-of-life depth of the defect; Δa is the stable tearing crack growth rate, K_{cp} is the elastic stress intensity factor plus plastic zone correction factor; and $J_{\Delta a}$ is the toughness from the J resistance curve of the material.

6.7.4. WWER flaw assessment methods

There is no official international WWER standard for the assessment of flaws found during inservice inspections. Two approaches are used for this procedure. The approach used by the Russian organizations for assessment of any RPV flaws in Russia, Armenia and Bulgaria is based on the procedure "Method for evaluation of allowability of defects in materials and piping in NPPs during operation", M-02-91 [119]. In principle, this method is divided into three parts. Defects found during ISI are schematized using a conservative approach, i.e., the equivalent defect diameter obtained from the ultrasonic tests is transformed into a fatigue-like crack with the same surface area and with a semiaxis ratio a/c equal to 0.5 for internal (subsurface) defects and to 0.4 for surface defects, respectively. Detailed rules and formulas for the evaluation of closely spaced defects or groups of defects are also given. All the defects are assumed to be in a plane perpendicular to the RPV surface as well as to the principal stresses.

Calculation of defect allowability is then performed using a complex approach, including linear elastic fracture mechanics and elastic-plastic fracture mechanics, as well as the theory of plasticity. Linear elastic fracture mechanics is used in the "brittle region" where the temperatures are below the critical temperature, T_{k2}, calculated from a plane strain condition which is valid for RPVs with a given crack size. A "quasi-brittle region" is then defined, which is 70°C wide; the calculations in this region are based on elastic-plastic fracture mechanics techniques using a pseudo-elastic stress intensity factor (calculated from Hook's stress), K_e, based on a strain intensity factor approach. The calculations in the "ductile region" where the temperatures are above the first critical temperature, $T_{k1} = T_{k2} + 70°C$ are based on an evaluation of the plastic instability of the RPV with the given defect.

The third part of the method defines how possible growth of the given defect due to the operating loads is calculated. The calculated defect growth for the remaining lifetime is added to the initially schematized defect sizes. However, the standard does not provide the coefficients for the Paris law for all materials used.

Pseudo-stress intensity factors, K_e, are calculated using a formula which takes into account the real distribution of the stresses at the deepest part of the crack. This formula is identical with the one used for the evaluation of the stress intensity factor, K_I, but with Hook's stresses instead of elastic stresses. Values of the K_e are calculated for both the deepest point of the crack as well as for the intersection with the surface (for a surface defect) or the closest point to the inner surface (for internal defects). These values are then compared with the allowable values of fracture toughness derived from the allowable stress intensity factor $[K_I]_3$ used in the standard [34]. The following safety factors are used for different operational conditions (which differ from the safety factors used for the RPV assessment during the design stage):

- for normal operating conditions:

  $n_k = 3$, $\Delta T = 30°C$

- for operational occurrences and hydraulic tests:

  $n_k = 1.5$, $\Delta T = 20°C$

- for accident (emergency) conditions:

  $n_k = 1.4$, $\Delta T = 10°C$

i.e., the allowable static fracture toughness dependence for the defect allowability evaluation is obtained as a lower boundary of two calculated curves, one obtained by dividing the $[K_I]_3$ by a safety factor n_k and the other by shifting the $[K_I]_3$ curve along the temperature axis by the value of the temperature margin ΔT.

Flaw assessment methods which are somewhat similar to the ASME Code, Section XI are used in the Czech Republic, Slovakia and Hungary. This approach can be described as follows. Schematization of the defects found during the ISI and calculation of the stress intensity factors is performed in a manner which is similar to the Russian approach. The postulated defect is defined in two ways:

- when the cladding properties are not known (as a function of neutron fluence): the postulated defect is assumed to be a surface semielliptical crack with depth, a, equal to 25% of the wall thickness, S,

- when the cladding properties are known: the postulated defect is assumed to be a subsurface (internal) underclad elliptical crack with height 2a = S/4.

Then, only linear elastic fracture mechanics methods are applied with safety factors identical to the ASME Code, but calculated with the $[K_I]_3$ static fracture toughness curve taken from Ref. [34], thus, initiation of unstable crack growth is not allowed because sufficient crack arrest fracture toughness data for the WWER materials do not exist. Research is being performed in this field and an initiation and arrest procedure, identical with the ASME Code, Section XI approach will also be applied, but with the WWER materials data.

The entire ASME, Section XI approach is used in Finland for their defect allowability evaluations, except they use the WWER material data.

7. AGEING MITIGATION METHODS

Section 4 of this report describes the age related degradation mechanisms that could impair the safety performance of an RPV during its service life. For four of these mechanisms (radiation embrittlement, fatigue, stress corrosion cracking and corrosion) mitigation methods are available to control the rate of ageing degradation and/or to correct the effects of these ageing mechanisms; thermal ageing and temper embrittlement are not addressed in this section since they are considered not to be significant.

7.1. RADIATION EMBRITTLEMENT

The radiation embrittlement can be mitigated by either flux reductions (operational methods aimed at managing ageing mechanism) or by thermal annealing of the RPV (maintenance method aimed at managing ageing effects). Flux reductions can be achieved by either fuel management or shielding the RPV from neutron exposure.

Managing ageing mechanism

7.1.1. Fuel management

The neutron flux (hence fluence) can be reduced by initiating a fuel management programme early in the life of a given plant. Such fuel management is carried out by implementing a low neutron leakage core (LLC). A LLC is a core that utilizes either spent fuel elements or dummy (stainless steel) fuel elements on the periphery of the core which reflect neutrons back into the core or absorb them rather than allowing them to bombard the RPV wall. LLCs can result in a reduction in power and/or increase in cost to the NPP owner.

Most of the western PWRs and all of the WWER plants have implemented LLC management programmes using spent fuel elements on the periphery of the core, but generally only after some period of operation. LLCs have been effective in reducing the re-embrittlement of the WWER-440/V-230 RPVs after thermal annealing.

A more drastic reduction of neutron flux can be achieved by inserting shielding dummy elements into the periphery of an active core, for example into the corners of the WWER active core hexagons. Dummy elements were inserted into most of the WWER-440/V-230 reactors in the middle of the 1980s. Dummy fuel elements were also used in some of the WWER-440/V-213 plants with RPVs with relatively high impurity content (e.g. Loviisa, Rovno). Thirty-two dummy elements are usually inserted into the core periphery. They cause not only a significant flux reduction but also a shifting of the maximum neutron flux by an angle of about 15° relative to both sides of the hexagon corners. Thus 12 new peak values of neutron flux are created on the pressure vessel wall. The original peak flux is decreased by a factor of 4.5 and the "new" peak flux is decreased by a factor of close to 2.5 — see Fig. 43. Thus, the cumulative effect of flux reduction must be calculated for both locations. Again, this method is most effective when applied during the first years of operation or just after a thermal annealing. The use of dummy elements usually results in a significantly different neutron balance in the core. The radial gradient is increased and thus the power distribution is disturbed in such a way that the peak power may exceed certain limits. Thus, a reduction in the fuel cycle length or a reduction of the reactor output are often necessary.

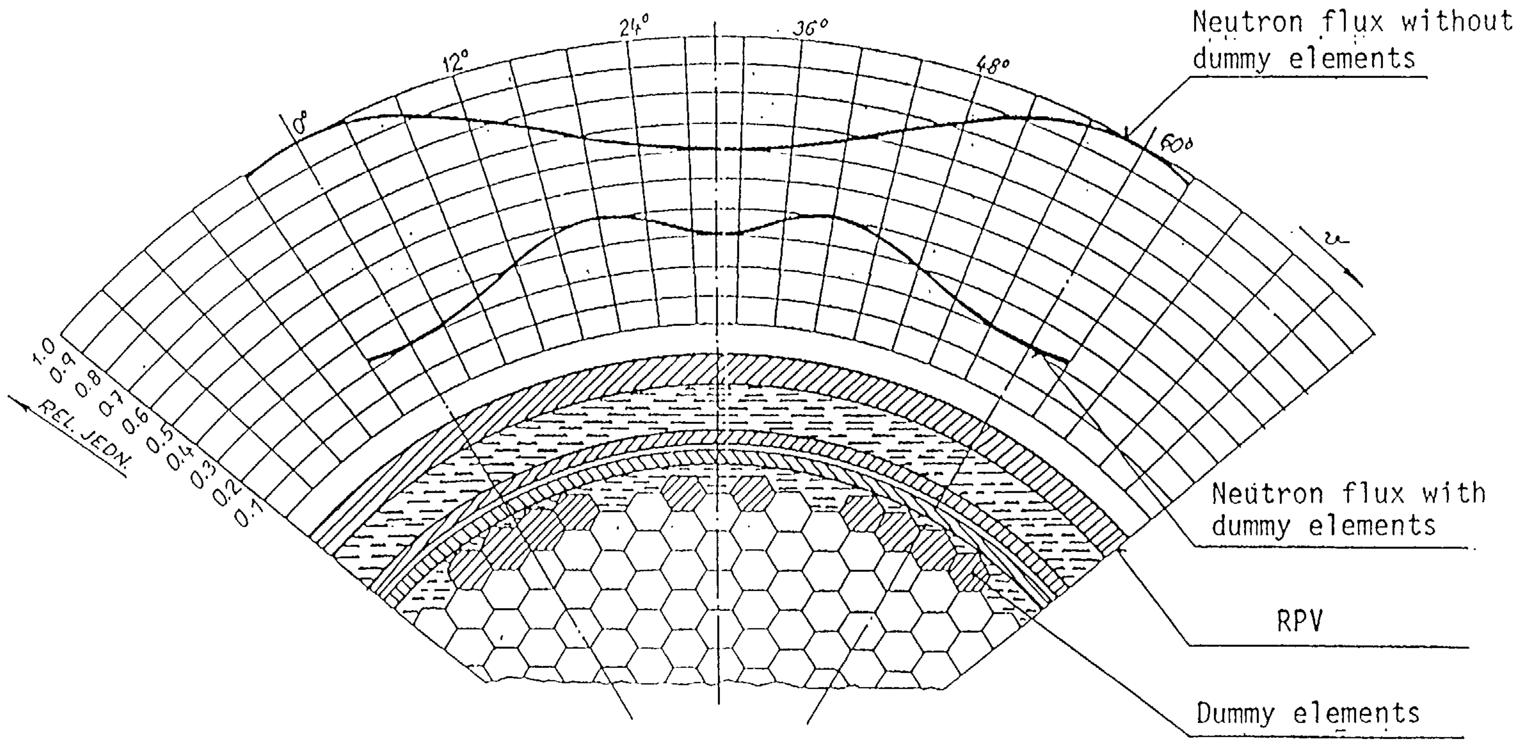

FIG. 43. WWER Flux distributions in low leakage cores.

7.1.2. RPV shielding

The flux (hence fluence) can also be reduced by further shielding the RPV wall from neutron bombardment. The reactor internals, the core barrel and thermal shield provides design basis shielding of the RPV. However, if it is judged that the design basis neutron exposure will result in significant radiation damage such that limitations are placed on the heating up and cooling down of the plant and/or accident conditions such as PTS becomes a potential safety issue, additional shielding is required. Shielding of the RPV wall from neutron exposure can be accomplished by increasing the thickness of the thermal pads that exist on the thermal shield at locations where the fluence is high or by placing shielding on the RPV wall. There are a number of alloys or elements that can providing shielding of the RPV wall by absorbing the high energy neutrons. Probably, the most effective shielding material is tungsten.

Managing ageing effects

7.1.3. Thermal annealing

Once a RPV is degraded by radiation embrittlement (e.g. significant increase in Charpy ductile-brittle transition temperature or reduction of fracture toughness), thermal annealing of the RPV is the only way to recover the RPV material toughness properties. Thermal annealing is a method by which the RPV (with all internals removed) is heated up to some temperature by use of an external heat source (electrical heaters, hot air), held for a given period and slowly cooled. The restoration of material toughness through post-irradiation thermal annealing treatment of RPVs has received considerable attention recently, due to the fact that a number of operating plants in the USA and elsewhere are approaching the PTS screening criteria during their normal license period, with several more approaching it during their license renewal period.

Experience in the USA. Thermal annealing is not without precedent; in the mid-1960s, the US Army SM-1A reactor reached a point where thermal annealing of the RPV was required after only a few years of operation because of sensitive material and a low operating temperature of 220°C (430°F). In the early 1980s, the Westinghouse Electric Corporation performed a study to examine the feasibility of thermal annealing of commercial RPVs and developed an optional, in situ, thermal annealing methodology that maximizes the fracture toughness recovery, minimizes re-exposure sensitivity and minimizes reactor downtime when thermal annealing becomes necessary. It was concluded from this study that excellent recovery of all properties could be achieved by annealing at a temperature of some 450°C (850°F) or higher for 168 hours. Such an annealing was predicted to result in a significant ductile-brittle transition temperature recovery. Further embrittlement under irradiation after the annealing was also predicted to continue at the rate that would have been expected had no annealing been performed. System limitations were identified for both wet and dry annealing methods. Several drawbacks were identified for the lower temperature wet thermal annealing that reduced its practicality. Therefore, a conceptual dry procedure was developed for thermal annealing embrittled RPVs. A follow-up study for EPRI showed that applying this procedure to two different plants resulted in acceptable stress, temperature and dimensions of the vessel and associated components.

The surveillance materials were irradiated to fluences up to 3×10^{23} n/m^2 (neutrons with energies less than 1 MeV) and at temperatures of about 290°C, which are typical of western RPVs. A good recovery of all of the mechanical properties was observed when the thermal annealing temperature was about 450°C for about 168 hours (1 week). And, the re-embrittlement rates upon subsequent re-irradiation were similar to the embrittlement rates observed prior to the thermal anneal. The dominant factors which influence the degree of recovery of the properties of the irradiated RPV steels are the annealing temperature relative to the irradiation (service) temperature, the time at the annealing temperature, the impurity and alloying element levels, and the type of product (plate, forging, weldment, etc.) [120].

In 1986, the ASTM published a guide for in-service annealing of water cooled nuclear reactor vessels [121] which basically follows procedures developed by Westinghouse.

The USNRC has issued revisions to 10 CFR 50.61 and 10 CFR 50 Appendices G and H, new section 10 CFR 50.66 (the thermal annealing rule) and new Regulatory Guide 1.162 [122] to address RPV thermal annealing. The modification to 10 CFR 50.61 explicitly cites thermal annealing as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}. The thermal annealing rule (10 CFR 50.66) addresses the critical engineering and metallurgical aspects of thermal annealing. The Regulatory Guide 1.162 on thermal annealing describes the format and content of the required report for thermal annealing.

10 CFR 50.66 requires a thermal annealing report which must be submitted at least 3 years prior to the proposed date of the annealing operation. The content of the report must include:

- A thermal annealing operating plan;
- An inspection and test programme to requalify the annealed RPV;
- A programme for demonstrating that the recovery of the fracture toughness and the re-embrittlement rate are adequate to permit subsequent safe operation of the RPV for the period specified in the application; and
- A safety evaluation identifying any unreviewed safety questions and technical specification changes.

The thermal annealing operating plan will provide the following:

- Background on the plant operation and surveillance programme results;
- Description of the RPV, including dimensions and beltline materials;
- Description of the equipment, components and structures that could be affected by the annealing operation to demonstrate that these will not be degraded by the annealing operation;
- Results from thermal and stress analyses to establish time and temperature profiles of the vessel and attached piping, and to specify limiting conditions of temperature, stress and strain, and heatup and cooldown rates;
- Proposed specific annealing parameters, in particular the annealing temperature and time, and heatup and cooldown rates, and the bounding time and temperature parameters that define the envelope of permissible annealing conditions to indicate conformance with the operating plan;

- Description of the methods, equipment, instrumentation and procedures proposed for the annealing operation;

- As low as reasonably achievable (ALARA) considerations for occupational exposure during the process; and

- Projected recovery and re-embrittlement trends for the RPV beltline materials.

Upon completion of the anneal and prior to restart of the NPP, licensee must certify to the NRC that the thermal annealing was performed in accordance with the approved application required by 10 CFR 50.66. The licensee's certification must establish the period for which the RPV will satisfy the requirements of 10 CFR 50.61 and Appendix G. The licensee must provide:

- The post-anneal RT_{NDT} and Charpy upper-shelf energy values of the RPV materials for use in subsequent reactor operation;

- The projected re-embrittlement trends for both RT_{NDT} and Charpy upper-shelf energy; and

- The projected values of RT_{PTS} and Charpy upper-shelf energy at the end of the proposed period of operation addressed in the application.

If the licensee cannot certify that the thermal annealing was performed in accordance with the approved application, the licensee shall submit a justification for subsequent operation for approval by the USNRC.

In 1994, the US Department of Energy (DOE) initiated a programme to demonstrate the feasibility of thermal annealing western-type RPVs to temperatures of about 454°C (850°F) without causing structural damage to the vessel, piping, supports, or other major components of the NSSS. A team led by the Westinghouse Electric Corporation, and including ASME, EPRI, and certain US nuclear utilities, successfully performed a demonstration thermal annealing of the Marble Hill RPV as part of this programme. (Marble Hill is a Westinghouse type PWR which was nearly completed but never operated.)

WWER experience. A high radiation embrittlement rate was found in most of the WWER-440 /V-230 RPVs (and some of the WWER-440/V-213 RPVs, e.g. Loviisa) at a point of time which was too late to ensure the planned reactor lifetime, i.e. 30 years. The only mitigation method was found to be thermal annealing of the affected RPVs. Following the publication of the Westinghouse conceptual procedure for dry thermal annealing an embrittled RPV, the Russians (and recently, the Czechs) undertook the thermal annealing of several highly irradiated WWER-440 RPVs. To date, at least 15 vessel thermal annealings have been realized. The WWER experience, along with the results of relevant laboratory scale research with western RPV material irradiated in materials test reactors and material removed from commercial RPV surveillance programmes, are consistent and indicate that an annealing temperature at least 150°C more than the irradiation temperature is required for at least 100 to 168 hours to obtain a significant benefit. The selection of the temperature regime for annealing type 15Kh2MFA steel (and its weldments) was based on a large amount of experimental work, which has been done by the various organizations involved, considering:

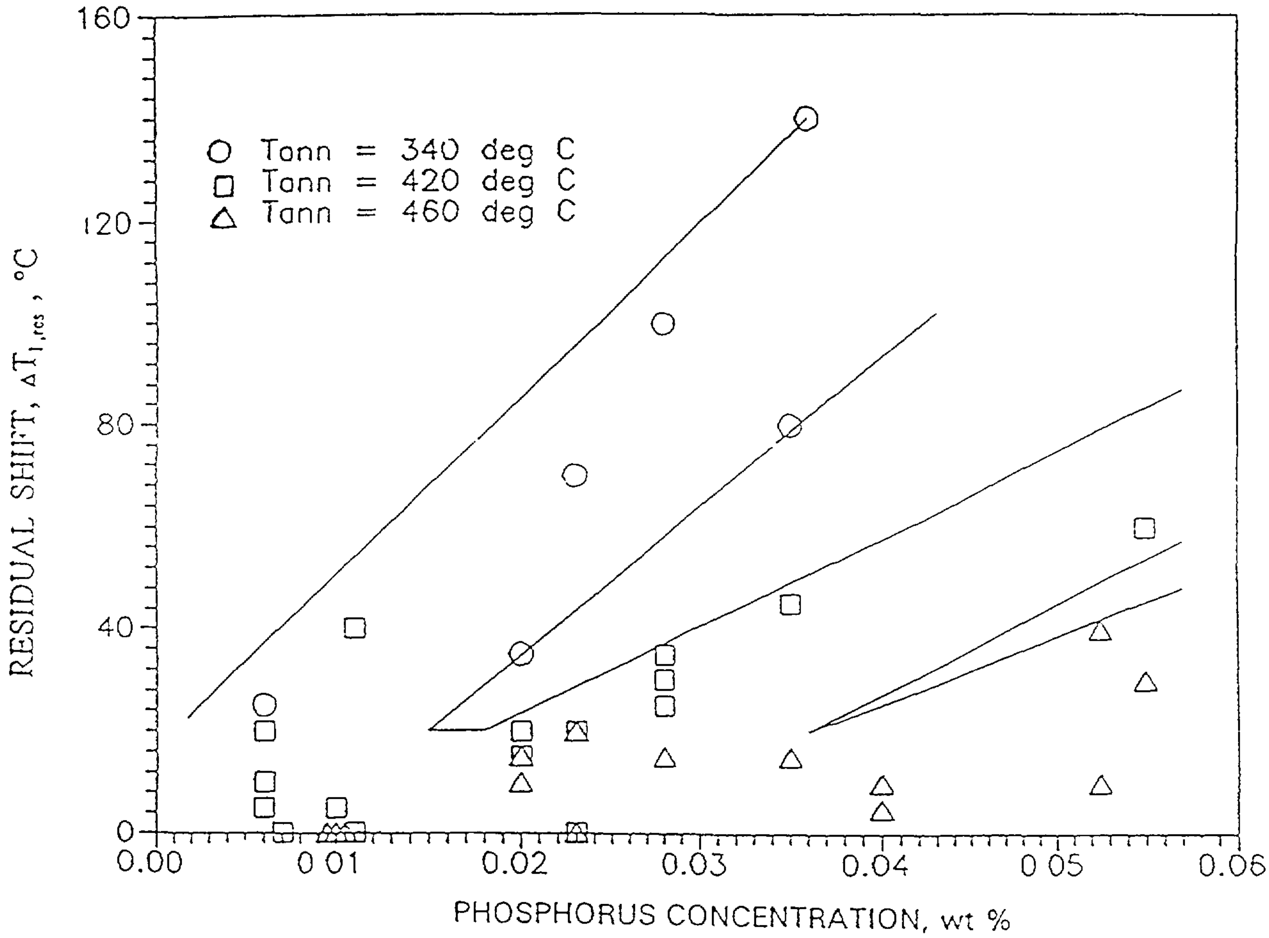

FIG 44 Residual transition temperature shift as a function of phosphorus content in 15 Kh 2 MFA steel

- optimization of the recovery of the ductile to brittle transition temperature shift, and

- evaluation of margins against the occurrence of temper embrittlement of base and weld metals

An annealing regime with a temperature above 460°C (the latest version is 475°C) during a hold period of at least 100 hours (168 hours in previous annealings) results in acceptable mechanical property recovery and a residual embrittlement which does not depend on neutron fluence (in the studied range) but mainly on phosphorus concentration — see Fig 44 The data available, obtained both from radiation experiments as well as from templates cut-out directly from the vessels, indicate that the residual transition temperature shift, ΔT_{res} is below + 20°C for steels with less than 0 04 mass % phosphorus It appears that for these steels a margin of 20°C conservatively covers the possible deviations However, this cannot be claimed for material with larger phosphorus contents without further validation — see Fig 44

An open question still remains concerning the so-called re-embrittlement rate, which is the rate of radiation embrittlement after annealing Two main models are used conservative and lateral shifts, respectively Many results show that after annealing at temperatures not lower than 425°C this re-embrittlement rate is well characterized by a "lateral shift" as is shown in Fig 45 a)

b)

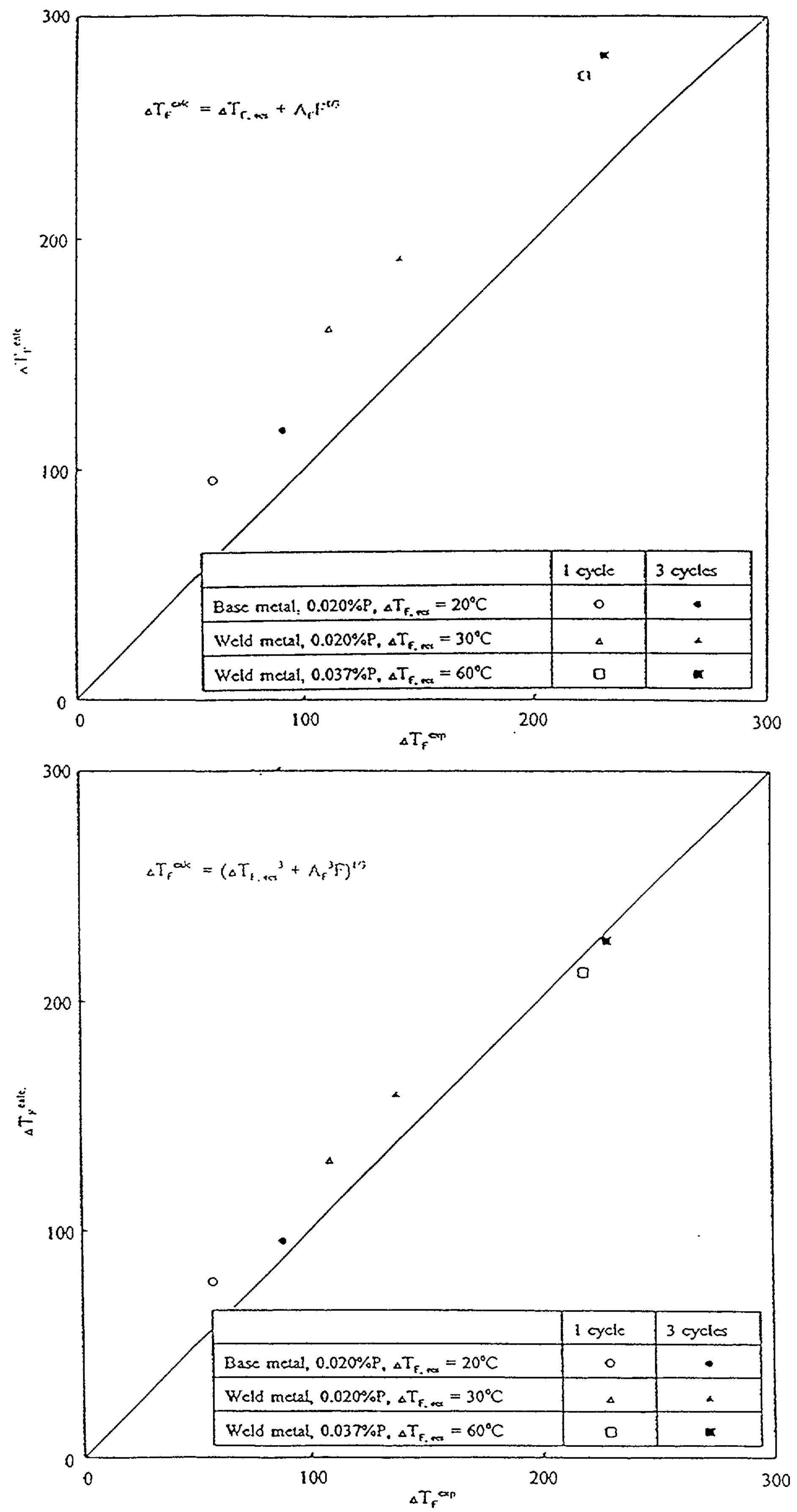

FIG. 45. Predicted vs. measured transition temperature shift due to re-irradiation up to three cycles (anneal-irradiate) for 15Kh2MFA base and weld metal. T_{Irr} = 260°C, T_{anneal} = 425°C. specimen size 5 × 5 × 27.5 mm. graph a) shows the conservative shift approach; graph b) shows the lateral shift approach.

The thermal annealing of a RPV requires the installation of monitoring and control devices and the development of procedures. The transition region at high temperatures has to be defined with respect to temperature limitations on specific components and limits on the secondary stresses in those components. A special annealing device, consisting of electrical heaters divided into sections, is inserted into the empty RPV. These heaters are controlled by thermocouples on the inner RPV wall and the required temperature gradient in the azimuthal as well as in the axial directions is achieved not only at the annealing temperature but also during the slow prescribed cooldown rate. A mock-up experiment on a model RPV of real dimensions is sometimes necessary, for example such an experiment was conducted at the ŠKODA plant as a necessary step before annealing the RPVs at Bohunice and more recently at Loviisa.

Depending on the presence of cladding (which limits access to the inner surface), two ways exist for determining the residual transition temperature shift after annealing, either:

- evaluation of delta T_{res} as a function of phosphorus content according to the existing database. In this case the knowledge of the chemical composition and T_{k0} is of importance, or

- evaluation of T_k after annealing by testing subsize Charpy specimens from templates cut from the inner surface. In this case, the use of a correlation between subsize and standard specimens results is required.

Both methods have been used, but uncertainties remain.

The large uncertainties, considerable data scatter and lack of data on material irradiated at conditions close to that of the vessel wall could be resolved by further investigations on decommissioned RPVs. The methodology for the Novovoronezh Unit 2 plant and others could be complemented by investigations on the shutdown Greifswald plant, which is typical of other WWER-440/V-230 plants in terms of operating conditions and material sensitivity to radiation embrittlement.

Instrumented hardness measurements on the cladding are recommended (they are realized in Bohunice and Dukovany plants as a part of ISI) for the evaluation of mechanical properties of the cladding. Instrumented hardness measurement at the outer surface cannot lead to an accurate assessment due to material uncertainties as discussed above.

7.2. STRESS CORROSION CRACKING OF CRDM PENETRATIONS

As discussed in Section 4.5.1, cracks have been discovered in Alloy 600 reactor vessel *head penetrations. Failure analyses have identified the cracking to be caused by* PWSCC. The operational methods that can be considered to control the rate of PWSCC include reduced upper head temperatures and coolant additives. The maintenance methods, i.e. repair and replacement technologies, which have been developed to correct the Alloy 600 reactor vessel head penetration cracking problems include: (1) surface treatments, such as special grinding techniques, nickel plating and peening; (2) stress improvement methods; (3) repair techniques such as grinding and rewelding or sleeving; and (4) replacement of either individual CRDM nozzles or an entire head. Each of the above operational or maintenance methods is designed to eliminate or reduce one or more of the three factors required to cause PWSCC (discussed in Section 4.3.1).

The head penetrations in all the WWER-type reactors, as well as the nozzle safe ends and safe-end to piping connections, are made of Type 18Cr/10Ni stabilized stainless steel. To date no cracks have been found in these locations.

Managing ageing mechanism

7.2.1. Coolant additives

The most promising coolant additive is zinc, which as been shown to reduce the radiation activity of the primary coolant as well as increase the resistance of Alloy 600 material to PWSCC. The zinc interacts with chromium in the oxide film on the Alloy 600 components and forms a more protective (stable) oxide coating, which delays initiation of PWSCC [123]. With the addition of 20 ppb of zinc, the PWSCC initiation time for Alloy 600 reverse U-bend specimens is increased by a factor of 2.8, and, with 120 ppb of zinc, the initiation time is increased by a factor of 10 [124]. With the addition of 20 ppb of zinc and a crack-tip stress intensity in the range of 40 to 50 MPa$\sqrt{m}$ (36 to 45 ksi$\sqrt{inch}$) the PWSCC crack growth rates are reduced by a factor of approximately 3.3. EPRI and the Westinghouse Owners' Group implemented zinc addition in June 1994 at Farley Unit 2 for field demonstration. The duration of this demonstration is about 39 months [125]. The zinc is being added in the form of zinc acetate, which has a high solubility in the PWR coolant at operating temperature. Adding zinc is expected to mitigate PWSCC in both new and old plants. However, it may take longer for zinc to be incorporated into the oxide film present in an older plant because the film is likely to be thicker and more stable.

7.2.2. Reduced upper head temperatures

The reactor upper head temperatures can be lowered somewhat by making minor modifications to the internals of certain RPVs to increase the bypass flow. This has been tried in France, but the results were not entirely satisfactory.

Managing ageing effects

7.2.3. Surface treatments

There are several different inside surface treatments being considered for mitigating Alloy 600 CRDM nozzle cracking, including special grinding, nickel plating and peening. Grinding techniques are being developed in France and Japan to remove the surface layer where cracks might have initiated, but remain undetected, and then produce compressive stresses on the regenerated surface [126]. Nickel plating can protect the treated surfaces from the PWR coolant, stop existing cracks from propagating and repair small cracks. Nickel plating has been qualified for steam generator tubes and has been applied to about 1100 tubes in Belgium and Sweden in the last 8 years. All of these tubes, except for the first few, are still in service, whereas unplated sister tubes are degrading [127]. The nickel plating does not provide structural strength for the CRDM nozzle. Peening with shot or other methods replaces high tensile residual stresses on the surface with compressive stresses. It has been used to prevent PWSCC initiation in steam generator tubes. However, shot peening is not effective if cracks already exist.

7.2.4. Stress improvement methods

Porowski et al. [128] have proposed a mechanical stress improvement method that redistributes the residual stresses in the nozzle and produces a layer of compressive stresses on the inside surface of the nozzle. The method consists of applying a compressive axial load at the nozzle ends, which are accessible. Analysis of the application of this method shows that the imposed axial compressive stresses interact with the residual tensile stresses on the inside surface, and the resulting plastic flow removes the residual tensile stresses from the sites on the nozzle inside surface near the partial penetration weld. The analysis results also show that the residual stresses on the inside surface are reduced, and the surface becomes near stress-free after removal of the applied axial load. This method is being considered at a few plants in the USA.

7.2.5. Alloy 600 head penetration repairs

Two options exist for the repair of Alloy 600 RPV head penetrations which contain stress corrosion cracks. The first method involves grinding out the stress corrosion crack and filling the resulting cavity with a suitable weld metal. The welding process should be such that residual stresses are minimized. Following the welding process, grinding is again performed to contour the surface of the weld repair to that of the head penetration. The weld filling material is usually Alloy 182.

The second method to repair head penetrations with stress corrosion cracks is to insert a thin liner (tube) of thermally treated Alloy 690 TT or austenitic stainless steel into the degraded head penetration. The head penetration in question is then pressurized and the liner will expand onto the head penetration tube and seal the crack.

7.2.6. Head penetration replacement

Head penetration replacement can take the form of either replacing the RPV closure head with a new closure head or replacing each head penetration; the new head penetration should be made from material other than Alloy 600. In several plants, where replacement of existing RPV closure heads has occurred, thermally treated Alloy 690 has been chosen as the material of construction for penetrations in replacing Alloy 600. Test results and limited field experience associated with other Alloy 690 components exposed to PWR primary coolant indicate that Alloy 690 material is not susceptible to PWSCC damage. In addition, new weld materials, Alloy 52 and 152, have been used in place of Alloy 82 and 182. The new materials have better resistance to PWSCC.

EdF is planning to replace all the vessel heads as a preventive measure. EdF has decided on replacement of vessel heads instead of mitigation of PWSCC damage and repair and replacement of the nozzles, for two reasons: (1) EdF found it more economical to replace the vessel head than to inspect the nozzles and repair them if cracks were found, and (b) current mitigation and repair techniques do not address the possible cracking of Alloy 182 weld metals [129]. The Kansai Electric Power co. of Japan has also decided to replace the vessel heads of three plants — Takahma 1 and 2 and Mihama 3 — as a preventive action. The vessel head temperature at these three plants is 320°C (608°F), which is 10°C (18°F) higher than the temperature at the other plants operated by this utility and, therefore, the CRDM nozzles at these three plants are considered more susceptible to PWSCC. The eddy-current inspection did not reveal any cracks in the CRDM nozzles of these three plants. However, the utility decided to replace the heads because of defence in depth considerations.

7.3. CORROSION AND PITTING OF INSIDE SURFACES AND FLANGES

Managing ageing effects

General corrosion as well as pitting corrosion has been found during visual testing of WWER pressure vessels without austenitic cladding. In some cases, only slight mechanical grinding was necessary to remove this damage. Pitting has also been discovered in the seal areas of some French PWRs (on the surfaces). The problem was addressed with grinding.

7.4. STRESS CORROSION CRACKING AND WEAR OF BOLT HOLE THREADS

Managing ageing effects

The old threads are removed by machining and a sleeve with new threads is inserted.

8. REACTOR PRESSURE VESSEL AGEING MANAGEMENT PROGRAMME

The information presented in this report suggests that radiation embrittlement of the reactor pressure vessel continues to be a significant safety and economic concern for both the western design reactor pressure vessels and the WWER reactor pressure vessels. Other age related mechanisms such as thermal ageing and temper embrittlement of the reactor pressure vessel materials, while not considered safety significant by themselves, can increase the safety significance of the radiation embrittlement of both the western and WWER reactor pressure vessels. Finally, the age related mechanisms of corrosion, stress corrosion cracking, wear and fatigue are not considered safety significant; however, they may be cost significant. Therefore, a systematic reactor pressure vessel ageing management programme is needed at all nuclear power plants.

The preceding sections of this report dealt with important elements of an RPV ageing management programme whose objective is to maintain the integrity of the RPV at an NPP throughout its service life. This section describes how these elements are integrated within a plant specific RPV ageing management programme utilizing a systematic ageing management process which is an adaptation of Deming's Plan-Do-Check-Act cycle to ageing management (Fig. 46). Such an ageing management programme should be implemented in accordance with guidance prepared by an interdisciplinary RPV ageing management team organized at a corporate or owners group level. For guidance on the organizational aspects of a plant ageing management programme and interdisciplinary ageing management teams refer to IAEA Safety Report "Implementation and Review of Nuclear Power Plant Ageing Management Programme" [130].

A comprehensive understanding of an RPV, its ageing degradation and the effects of the degradation on the ability of the RPV to perform its design functions is the fundamental basis of an ageing management programme. This understanding is derived from a knowledge of the design basis (including applicable codes and regulatory requirements); the design and fabrication (including the materials properties and specified service conditions); the operation and maintenance history (including commissioning and surveillance); the inspection results; and generic operating experience and research results. Sections 1.1, 2, 3 and 4 contain information on important aspects of the understanding of RPVs and their ageing.

In order to maintain the integrity of an RPV, it is necessary to control within defined limits the age related degradation of the RPV. Effective ageing degradation control is achieved through the systematic ageing management process consisting of the following ageing management tasks, based on understanding of RPV ageing:

- operation within operating guidelines aimed at minimizing the rate of degradation – *managing ageing mechanisms* (Sections 8.1.3 and 7);
- inspection and monitoring consistent with requirements aimed at timely detection and characterization of any degradation (Section 5);
- assessment of the observed degradation in accordance with appropriate guidelines to determine integrity (Section 6); and
- maintenance (repair or parts replacement) to correct unacceptable degradation – *managing ageing effects* (Section 7).

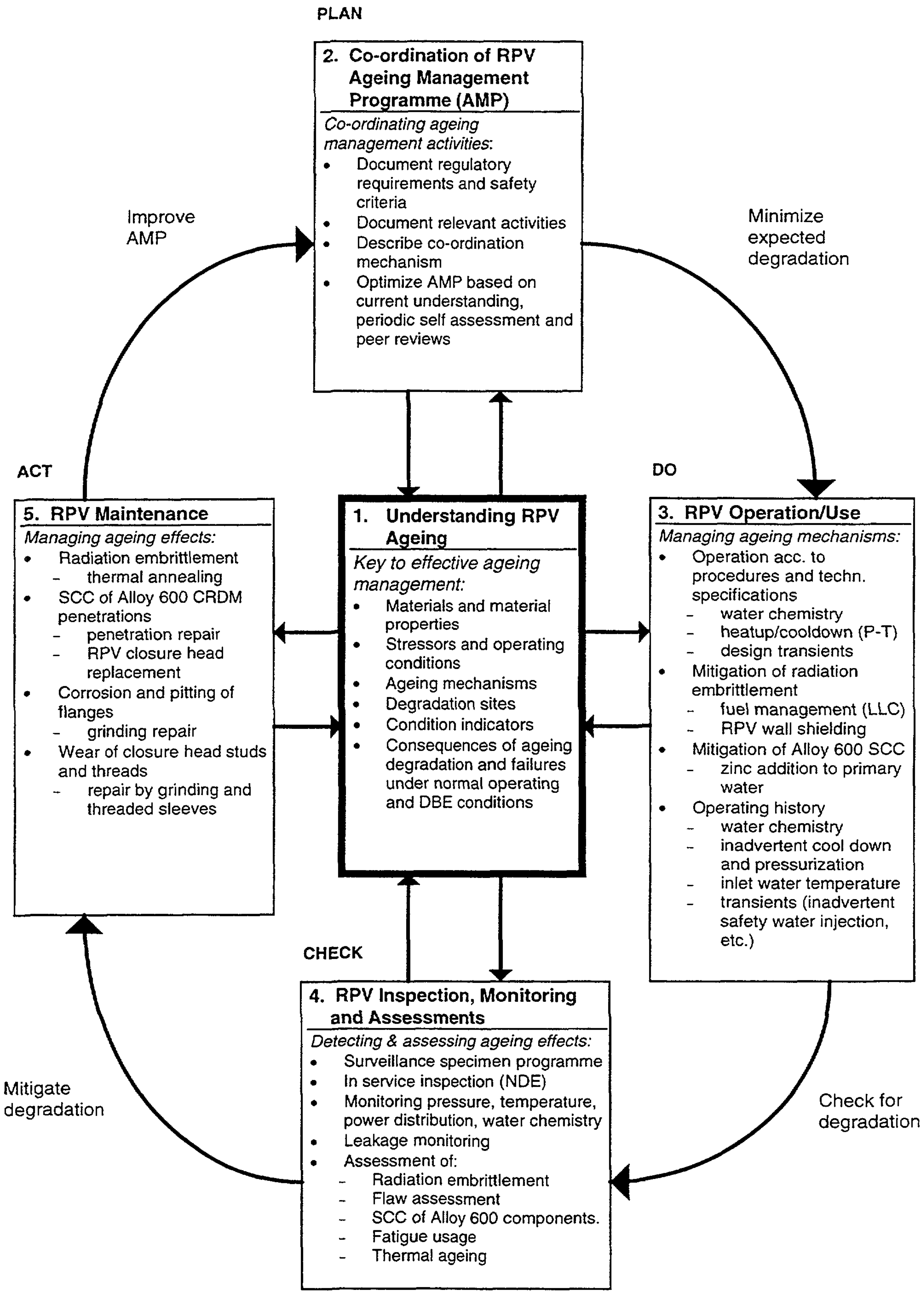

FIG. 46. Key elements of a PWR pressure vessel ageing management programme utilizing the systematic ageing management process.

An RPV ageing management programme co-ordinates programmes and activities contributing to the above ageing management tasks in order to detect and mitigate ageing degradation before the RPV safety margins are compromised. This programme reflects the level of understanding of the RPV ageing, the available technology, the regulatory/licensing requirements and plant life management considerations/objectives. Timely feedback of experience is essential in order to provide for ongoing improvement in the understanding of the RPV ageing degradation and in the effectiveness of the ageing management programme. The main features of an RPV ageing management programme, including the role and interfaces of relevant programmes and activities in the ageing management process, are shown in Fig. 46 and discussed in Section 8.1 below. Application guidance is provided in Section 8.2.

8.1. KEY ELEMENTS OF RPV AGEING MANAGEMENT PROGRAMME

8.1.1. Understanding RPV ageing

Understanding RPV ageing is the key to effective management of RPV ageing, i.e., it is the key to: co-ordinating ageing management activities within a systematic ageing management programme, managing ageing mechanisms through prudent operating procedures and practices (in accordance with procedures and technical specifications); detecting and assessing ageing effects through effective inspection, monitoring and assessment methods; and managing ageing effects using proven maintenance methods. This understanding consists of: a knowledge of RPV materials and material properties; stressors and operating conditions; likely degradation sites and ageing mechanisms; condition indicators and data needed for assessment and management of RPV ageing; and effects of ageing on safety margins.

The understanding of RPV ageing is derived from the RPV baseline data, the operating and maintenance histories, and external experiences. This understanding should be updated on an ongoing basis to provide a sound basis for the improvement of the ageing management programme consistent with operating, inspection, monitoring, assessment and maintenance methods and practices.

The RPV baseline data consists of the performance requirements, the design basis (including codes, standards, regulatory requirements), the original design, the manufacturer's data (including materials data) and the commissioning data (including inaugural inspection data). The RPV operating history includes the pressure–temperature records, system chemistry records, records on material radiation embrittlement from the surveillance programme and the ISI results. The RPV maintenance history includes the inspection records and assessment reports, design modifications and type and timing of maintenance performed. Retrievable up-to-date records of this information are needed for making comparisons with applicable external experience.

External experience consists of the operating and maintenance experience of (a) RPVs of similar design, materials of construction and fabrication; (b) RPVs operated with similar operating histories, even if the RPV designs are different; and (c) relevant research results. It should be noted that effective comparisons or correlations with external experience require a detailed knowledge of the RPV design and operation. The present report is a source of such information. However, this information has to be kept current using feedback mechanisms provided, for example, by owners groups. External experience can also be used when considering the most appropriate inspection method, maintenance procedure and technology.

8.1.2. Co-ordination of RPV ageing management programme

Existing programmes relating to the management of RPV ageing include operations, surveillance and maintenance programmes as well as operating experience feedback, research and development and technical support programmes. Experience shows that ageing management effectiveness can be improved by coordinating relevant programmes and activities within an ageing management programme utilizing the systematic ageing management process. Safety authorities increasingly require licensees to implement such ageing management programmes for selected SSCs important to safety. The co-ordination of an RPV ageing management programme includes the documentation of applicable regulatory requirements and safety criteria, and of relevant programmes and activities and their respective roles in the ageing management process as well as a description of mechanisms used for programme coordination and continuous improvement. The continuous ageing management programme improvement or optimization is based on current understanding of RPV ageing and on results of periodic self-assessments and peer reviews.

8.1.3. RPV operation

NPP operation has a significant influence on the rate of degradation of plant systems, structures and components. Exposure of RPV to operating conditions (e.g. temperature, pressure, fast neutron dose rate, water chemistry) outside prescribed operational limits could lead to accelerated ageing and premature degradation. Since operating practices influence RPV operating conditions, NPP operations staff have an important role within the ageing management program to minimize age related degradation of the RPV. They can do this by maintaining operating conditions within operational limits that are prescribed to avoid accelerated ageing of RPV components during operation. Examples of such operating practices are:

- fuel loading scheme to control the rate of radiation embrittlement;
- operation within the prescribed pressure and temperature range during startup and shutdown to avoid the risk of overpressure (this risk varies, depending on the fracture toughness) of the material;
- defining appropriate operator actions for the case of a possible PTS event to avoid critical transients;
- performing maintenance according to procedures designed to avoid contamination of RPV components with boric acid or other reagents containing halogens;
- on-line monitoring and record keeping of operational data necessary for predicting ageing degradation and defining appropriate ageing management actions.

Operation and maintenance in accordance with procedures of plant systems that influence RPV operational conditions (not only the primary system but also the auxiliary systems like water purification and injection systems), including the testing of the RPV and its components, and record keeping of operational data (incl. transients) are essential for an effective ageing management of the RPV and a possible plant life extension. Specific operational actions used to manage RPV-significant ageing mechanisms are described in Section 7.

8.1.4. RPV inspection, monitoring and assessment

Inspection and monitoring

The RPV inspection and monitoring activities are designed to detect and characterize significant component degradation before the RPV safety margins are compromised. Together with an understanding of the RPV ageing degradation, the results of the RPV inspections provide a basis for decisions regarding the type and timing of maintenance actions and decisions regarding changes in operating conditions to manage detected ageing effects.

Current inspection and monitoring requirements and techniques for RPVs are described in Section 5. Inspection and monitoring of RPV degradation falls in two categories: (1) inservice inspection and surveillance capsule testing, and (2) monitoring of pressures and temperatures, water chemistry, transients (relative to fatigue), RPV leakage and power distributions. Results of the ISI are used for flaw tolerance assessments while the surveillance capsule test results are used as input for the assessment of the radiation embrittlement. Monitoring of the power distributions provides input to the calculation of the RPV fluence from the neutron dosimeters encapsulated in the surveillance capsules. Monitoring temperature and pressure also provides input for the assessment of radiation embrittlement. Transient monitoring provides realistic values of thermal stresses as opposed to design basis thermal stress values for fatigue assessments. Finally, monitoring for leakage provides for the recognition of potential PTS transients or CRDM leakage.

It is important to know the accuracy, sensitivity, reliability and adequacy of the non-destructive methods used for the particular type of suspected degradation. The performance of the inspection methods must be demonstrated in order to rely on the results, particularly in cases where the results are used in integrity assessments. Inspection methods capable of detecting and sizing expected degradation are therefore selected from those proven by relevant operating experience.

Integrity assessment

The main safety function of an RPV is to act as a barrier between the radioactive primary side and the non-radioactive outside environment. Safety margins are part of the design and licensing requirements of a NPP to ensure the integrity of the RPV under both normal and accident conditions. An integrity assessment is used to assess the capability of the RPV to perform the required safety function, within the specified margins of safety, during the entire operating interval until the next scheduled inspection.

Integrity assessments have used a variety of methods in response to the particular conditions and circumstances present at the time of the assessment. Section 6 of this report describes the assessment methods used. Included in the RPV integrity assessments are radiation damage trend curves for comparison with surveillance capsule test results to assess radiation embrittlement and utilization of the ISI results along with fatigue crack growth models and fracture mechanics technologies to assess the flaw tolerance of the RPV. In addition to the integrity assessment relating to the RPV safety function, assessments are required of other ageing related degradations that may have an economic impact on the ageing management programme. These include assessment of the fatigue usage factors utilizing information/data from the on-line transient monitoring system, assessments of the stress corrosion cracking susceptibility of the Alloy 600 components and thermal ageing assessments.

8.1.5. RPV maintenance

Maintenance actions that can be used to manage ageing effects detected by inspection and monitoring methods in different parts of an RPV are described in Section 7. Decisions on the type and timing of the maintenance actions are based on an assessment of the observed ageing effects, available decision criteria and understanding of the applicable ageing mechanism(s), and the effectiveness of available maintenance technologies.

Maintenance actions for managing radiation embrittlement fall into two categories: (1) installing reactor vessel wall shielding to control the rate of future embrittlement; and (2) thermal annealing to recover RPV material fracture toughness. Maintenance actions for managing the stress corrosion cracking of Alloy 600 CRDM penetrations include the replacement of the RPV closure head (including new CRDM penetrations) or the replacement of CRDM penetrations in the existing RPV closure head. If there are only a small number of cracked CRDM penetrations, the cracks may be removed by grinding and the cavity filled with weld metal. For susceptible CRDM penetrations without cracking, the addition of zinc to the primary coolant has the potential to minimize or prevent future SCC.

Maintenance of the surfaces of the closure flanges may be required if corrosion or pitting occurs due to damaged O-rings. If corrosion or pitting is observed, the surfaces of the closure flanges may be repaired by grinding off any corrosion products or pitting.

Wear of the closure head studs and threads is also occasionally observed. The degradation of the closure studs and threads by wear requires that the closure holes be machined out and new threaded sleeves be inserted into the stud holes. The maintenance of the closure head studs and threads should be scheduled based on previous inspections for wear.

8.2. APPLICATION GUIDANCE

The RPV ageing management programme should address both safety and reliability/ economic aspects of RPV ageing to ensure both the integrity and serviceability of the RPV during its design life and any extended life. The following sections provide guidance on dealing with the relevant age related degradation mechanisms.

8.2.1. Reactor pressure vessel radiation embrittlement

Radiation embrittlement of the RPV is a safety concern. All RPV materials are radiation embrittlement sensitive to some degree. The ageing management programme activities which address radiation embrittlement can be identified as follows.

(a) Utilization of the radiation embrittlement databases/trend curves to predict the degree of radiation embrittlement for a given RPV.

(b) RPV materials radiation surveillance programmes: most of the RPVs have materials radiation surveillance capsules within the vessel. The western RPV surveillance capsules are located at the beltline regions of the RPVs, thereby providing a monitoring of the radiation sensitivity of the RPV materials. Some WWER surveillance capsules are located outside of the beltline region, and therefore, require methodology to assess the radiation sensitivity of beltline materials from the data obtained outside of the beltline region. The WWER-440 Type 230 plants were not supplied with a RPV material surveillance programme. Also, a few western RPVs depend on sister plants for their material surveillance data.

(c) Low leakage core (LLC) fuel management programme to reduce the neutron, flux, hence rate of radiation embrittlement

(d) Additional RPV wall shielding to reduce the rate of radiation embrittlement.

(e) Application of thermal annealing of a reactor pressure vessel fabricated from radiation sensitive material is always an option. If the reactor pressure vessel materials are highly sensitive to radiation damage, the ageing management programme should evaluate the response of the surveillance capsule materials to thermal annealing and develop a plan for thermal annealing the reactor pressure vessel.

8.2.2. Stress corrosion cracking of penetrations fabricated from Alloy-600

Although the ageing degradation of the Alloy-600 RPV penetrations, especially the stress corrosion cracking of the CRDM penetrations discussed in Section 4.5.1, is generally considered to be an economic concern, the USNRC considers this degradation also a long-term safety concern. Therefore, the ageing management programme should address this issue from both an economic and safety perspective. The ageing management programme should include:

(a) An ISI programme for the Alloy-600 penetrations which will ensure timely detection of any Alloy-600 penetration cracking. Ultrasonic and/or eddy-current technologies can be used. Criteria such as published in the ASME Code should be used to identify reportable indications.

(b) A flaw evaluation handbook should be prepared if reportable indications are found that exceed the given acceptance criteria identified by the ASME Code or other governing regulatory agency; or plant-specific criteria should be developed and documented in a flaw evaluation handbook to determine if continued operation is acceptable or repair or replacement is warranted.

(c) The ageing management programme should also have in place repair procedures and/or contingency plans for reactor pressure vessel head replacement.

(d) On-line leak monitoring systems should be installed at plants with susceptible Alloy-600 penetrations.

8.2.3. Thermal ageing of reactor pressure vessel materials

As discussed in Section 4.2, thermal ageing of the RPV material is not considered to be a safety or economic concern since the available published data does not indicate a large increase in the NDTT or RT_{NDT}. However, even a relatively small increase in the NDTT or RT_{NDT} of 20°C may in combination with the irradiation damage at the beltline region of the RPV (measured by the radiation damage surveillance program), it can be a safety issue if a flaw is located in a region outside of the beltline region. Therefore, the ageing management programme should address thermal ageing as follows:

(a) ISI of the regions outside the RPV beltline should be periodically carried out to ensure timely detection of any flaws. Current inspection requirements only require that weldments and a limited distance in the base metal be inspected. However, the critical location for flaw instability may not be in the region that is covered by the ISI. Therefore, the activities discussed below should be implemented.

(b) If a flaw (reportable indication exceeding ASME allowable) is detected during ISI, as discussed above, the flaw should be evaluated in accordance with the ASME Section XI Code or the prevailing Code or Regulatory Rules of the given country. For flaws detected within the RPV beltline region, the effect of thermal ageing is accounted for in the results from the post-irradiation testing of surveillance capsule specimens. For flaws detected in regions outside the RPV beltline, the effect of thermal ageing must be taken into consideration. An estimate of the increasing RT_{NDT} due to thermal ageing should be made from published data for the material of interest or if the given RPV surveillance program contains thermal ageing specimens outside the RPV, the results from the testing of these specimens should be taken into consideration. The increase in RT_{NDT} from the above methods must be included in the required fracture mechanics assessment of any flaws detected during the ISI.

8.2.4. Fatigue

The assessment in the ageing management programme of fatigue crack initiation caused by cyclic loadings should be carried out by either the use of delta stress (S) versus number of cycles (N) curves given in the ASME Section III B3000 rules or similar curves in the given country's code or regulatory rules. If a flaw is detected during ISI, fatigue crack growth analyses must also be performed as discussed below. Also, removal of the flaw with a boat sample and microstructural analysis should be considered.

(a) Analytical method — Miner's Rule is an analytical method which can be used to assess the possibility of fatigue crack initiation in RPVs. ASME Section III, NB-3222.4, specifies the use of Miner's Rule for calculating fatigue damage in structural components, as do the codes in a number of other countries. The use of Miner's Rule requires that the cyclic stresses and the number of cycles are known. The cyclic stresses and number of cycles are given in the RPV stress report. These values are determined from the NSSS vendor estimate of the type and number of transients. Use of Miner's Rule results in the determination of a cumulative usage factor, U, which is the total number of expected cycles at a given stress level divided by the allowable number of cycles at that stress level. The allowable number of cycles at any stress level can be determined from the stress versus number of cycles (S/N) design curve for the material of interest in the code. When more than one stress level is expected (which is usually the case), the cumulative usage factor is the summation of the ratio at each stress level. The cumulative usage factor shall not exceed 1.0 for any part of the RPV, and cumulative usage factors should be calculated for all the key components of the RPV including the closure head, nozzles, penetrations, studs and beltline region.

(b) Transient monitoring — The NSSS vendors' input to the stress report as to the number and type of transients can be overly conservative. Transient monitoring can be used to obtain more accurate estimates of both the total number of cycles and the stress ranges. For RPVs that went into operation prior to installing a transient monitoring system, a review of past operating records must be made to determine the number and type of transients prior to the installation of the monitors. Transient monitoring systems are a very valuable tool in determining the life of a RPV and should be part of the ageing management programme.

(c) Evaluation of ISI results — As discussed in Sections 3, 5 and 6 of this report, each country has specific ISI requirements. If a flaw is detected in the RPV during ISI and if the size of the flaw requires that a fracture mechanics analysis be performed to demonstrate the integrity of the component, then a fatigue analysis must also be performed. The fatigue analysis considers the growth of the flaw or crack in fracture mechanics terms using a correlation between the cyclic crack growth rate, da/dN and the stress intensity range, ΔK. The growth of the flaw can be determined using the methodology given in Appendix A to ASME Section XI, or similar methodology. Flaw Evaluation Handbooks can be obtained from the NSSS vendors that can be used as a plant specific tool to assess the growth of a flaw over the design life of the RPV, as well as to determine the critical flaw size for instability. The ageing management programme should include either a Flaw Evaluation Handbook or be prepared to perform a fracture mechanics analysis if and when a flaw is detected during ISI.

(d) Microstructural analysis of a flaw — If a flaw is detected during ISI, consideration should be given to removing the flaw by taking a boat sample that contains the flaw and performing a microstructural analysis to determine if striations are evident on the surface of the flaw. Striations on the surface of a flaw means that the initiation of the flaw or growth was due to fatigue. If it is determined that a flaw was initiated by fatigue, then one should question the fatigue analysis performed prior to service. Removal of a flaw following ISI is not normally performed once a NPP has gone into operation because Code or Regulatory approved fracture mechanics methodologies are available to assess the growth and critical size of flaws. However, the ageing management programme should consider removal and metallographic evaluation as an option.

8.2.5. Wear

Degradation due to wear may occur during maintenance operations concerned with opening and closing of the RPV head. Wear can occur in the filets of the RPV bolts (studs). And, the RPV O-ring and the surfaces of the RPV flanges may also be degraded or damaged during the opening and closing operations. The RPV bolts (studs), the surface of the flanges and the O-ring should be inspected for evidence of degradation or wear. In addition, the outside of the RPV should be visually inspected for evidence of corrosion due to leakage from the head bolts or studs, a damaged O-Ring or scared flanges. Visual inspection of components of the RPV that may be subjected to wear should be part of the ageing management programme.

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ABBREVIATIONS

ALARA	as low as reasonably achievable
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&W	Babcock & Wilcox
BWR	boiling water reactor
CANDU	Canada deuterium–uranium (reactor)
CF	chemical factor
CMEA	Council for Mutual Economic Assistance
CRDM	control rod drive mechanism
DIN	Deutsche Industrienorm (German Industrial Standard)
ECCS	emergency core cooling system
ECT	eddy-current testing
EdF	Electricité de France
EPRI	Electric Power Research Institute
ISI	in-service inspection
K_I	total stress intensity factor
K_{IR}	reference stress intensity factor
KTA	Kerntechnischer Ausschuß (German Committee for Nuclear Technology)
LEFM	linear elastic fracture mechanics
LLC	low leakage core
NDE	non-destructive examination
NDT	non-destructive testing
NDTT	nil-ductility transition temperature
NPP	nuclear power plant
NSSS	nuclear steam supply system
NUSS	IAEA Nuclear Safety Standards
PSI	pre-service inspection
P–T	pressure–temperature
PTS	pressurized thermal shock
PWR	pressurized water reactor

PWSCC	primary water stress corrosion cracking
QA	quality assurance
RPV	reactor pressure vessel
RT_{NDT}	reference nil-ductility transition temperature, sometimes also called the ductile-to-brittle transition temperature
SCC	stress corrosion cracking
SSCs	systems, structures and components
TRL	transmitter-receiver-longitudinal wave-probe
TÜV	Technischer Überwachungsverein (German Authorized Inspection Agency)
USE	upper shelf energy
UT	ultrasonic testing
WWER	water moderated, water cooled energy reactor
ΔRT_{NDT}	change in the reference nil-ductility transition temperature

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