IAEA-TECDOC-1470

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

IAEA
International Atomic Energy Agency

October 2005

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BWR pressure vessels

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The originating Section of this publication in the IAEA was:

Engineering Safety Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

ASSESSMENT AND MANAGEMENT OF AGEING OF MAJOR NUCLEAR
POWER PLANT COMPONENTS IMPORTANT TO SAFETY:
BWR PRESSURE VESSELS
IAEA, VIENNA, 2005
IAEA-TECDOC-1470
ISBN 92–0–109305–5
ISSN 1011–4289

Printed by the IAEA in Austria
October 2005

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 (caused for instance 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. Since the reports are written from a safety perspective, they do not address life or life cycle management of plant components, which involves economic considerations.

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), and 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. The guidance reports are directed at technical experts from NPPs and from regulatory, plant design, manufacturing and technical support organizations dealing with specific plant components addressed in the reports.

This report addresses the reactor pressure vessel (RPV) in BWRs. Maintaining the structural integrity of this RPV throughout NPP service life in spite of several ageing mechanisms is essential for plant safety.

The work of all contributors to the drafting and review of this report, identified at the end, is greatly appreciated. In particular, the IAEA would like to acknowledge the contributions of J.P. Higgins, J. Pachner, J. Hakala, B. Kastner, C. Dillmann, U. Blumer and Y. Motora. The IAEA officer responsible for this report was T. Inagaki of the Division of Nuclear Installation Safety.

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, thus ensuring their integrity and functional capability throughout plant service life. General guidance on NPP activities relevant to the management of ageing (operation, maintenance, examination and inspection of SSCs) is given in the International Atomic Energy Agency (IAEA) Nuclear Safety Standards (NUSS) on the Safety of Nuclear Power Plants: Operation Requirements [1.1] and associated Safety Guide on Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants [1.2], hereinafter the MS&I Safety Guide.

The Operation Requirements require that NPP operating organizations prepares and carries out a programme of maintenance, testing, surveillance and inspection of plant SSCs important to safety to ensure that their level of reliability and effectiveness remains in accordance with the design assumptions and intent throughout the service life of the plant. This programme should take into account the operational limits and conditions, any other applicable regulatory requirements, ageing characteristics of SSCs and be re-evaluated in the light of operating experience. The associated Safety Guide provides further guidance on NPP programmes and activities that contribute to timely detection and mitigation of ageing degradation of SSCs important to safety.

The MS&I Safety Guide [1.2] 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 caused by stressors such as stress, temperature, radiation, vibration and water chemistry 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 also covered in the MS&I Safety Guide [1.2]. The MS&I Safety Guide also provides 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). 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 MS&I Safety Guide provides general guidance, but does not give detailed technical advice for particular SSCs.

Programmatic guidance on ageing management is given in Technical Reports Series No. 338 Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety [1.3] and in a Safety Practices Publication Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing [1.4]. 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. This publication on boiling water reactor (BWR) reactor pressure vessels is one of such TECDOCs. TECDOCs already issued address: steam generators [1.5], concrete containment buildings [1.6], CANDU pressure tubes [1.7], PWR reactor pressure vessels [1.8], PWR reactor vessel internals [1.9], metal components of BWR containment systems [1.10], in-containment I&C cables [1.11], CANDU reactor assemblies [1.12], and primary piping in PWRs [1.13].

The first commercial BWR was Dresden unit 1 near Morris Illinois. The BWR 2 product line was the beginning of the design on which plants built by GE, Hitachi and Toshiba are based.

Boiling water reactors are operating in Finland, Germany, India, Japan, Mexico, the Russian Federation, Spain, Sweden, Switzerland, Taiwan (China), and the United States of America.

The BWR reactor pressure vessel is the most important pressure boundary component of the nuclear steam supply system (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 pressure 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 pressure 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.

BWR reactor pressure vessels experience service at 100°C–300°C. The neutron fluence is lower than that found in PWR plants due to the relatively large annulus between the shroud and the RPV wall and because BWRs generally operate at lower power densities. In general, only a small portion of the beltline region experiences fluence above the damage threshold. There have been several fabricators of RPVs, which, combined with the change of ideas over time, has resulted in variability in materials and practices. Thus an aging program, while it can draw on generic guidelines, must consider the plant specific parameters.

1.2. Objective

The objective of this report is to identify significant ageing mechanisms and degradation locations, and to document the current practices for the assessment and management of the ageing of BWR RPVs. The report emphasizes safety aspects and also provides information on current inspection, monitoring and mitigation practices for managing ageing of BWR RPVs.

The underlying objective of this report 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 IAEA Member States. The target audience includes NPP operators, regulators, technical support organizations, designers, and manufacturers.

The readers who are not interested in technical details related to ageing degradation of BWR RPVs but are interested in ageing management strategy for BWR RPVs utilizing a systematic ageing management approach should go directly to Section 8. This section presents a strategy for managing each of the three significant ageing mechanisms: radiation embrittlement, stress corrosion cracking and fatigue.

1.3. Scope

This report provides the technical basis for understanding and managing the ageing of the BWR RPV to ensure that the acceptable 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 heads. The scope of this report does not treat RPV internals, the control rod drive mechanisms (CRD), or the primary boundary piping used in BWRs. The pressurized water reactor (PWR) reactor vessels and Canadian deuteriumuranium (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 BWR reactor 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 presents the ageing mechanisms, susceptible degradation sites, their significance and operating experience. Section 5 addresses the application of various inspection, monitoring and maintenance technologies. Section 6 gives the current practices and data required in assessing degradation of an RPV. Section 7 describes methods used to mitigate stress corrosion cracking. This report concludes, in Section 8, with a description of a systematic ageing management programme for BWR RPVs.

1.5. Nation specific aspects

Regulations and practices of safety and ageing management of RPVs in different countries generally show some differences. This applies for the design codes and ISI codes as well as the approach to life extension regulation.

Most countries adopted an approach for design and ISI regulation at least very similar to the ASME Code in the United States of America. Therefore, this report reflects mostly the approaches used in the USA. Wherever distinct differences are applicable in other countries, extra sub-chapters are introduced to explain the specific national regulation or approach.

A distinct difference between countries is the practice of repeated pressure testing. This difference potentially can mean a marked difference in the significance of radiation embrittlement between countries for BWR RPVs.

The regulation of operating life licensing has distinct differences. Whereas the USA has implemented a License Renewal process, which controls procedures to extend NPP lives above 40 years, many other countries follow the path of Periodic Safety Reviews (PSRs), which typically have to be performed every 10 years without a specific threshold at 40 years. This report mainly addresses the technical aspects of ageing management and therefore does not emphasize any possible differences resulting from this different licensing approach.

REFERENCES TO SECTION 1

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[1.3] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for Management of Ageing of Nuclear Power Plant Components Important to Safety, Technical Reports Series No. 338, IAEA, Vienna (1992).

[1.4] INTERNATIONAL ATOMIC ENERGY AGENCY, Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing, Safety Series No. 50-P-3, IAEA, Vienna (1991).

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[1.6] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Buildings, IAEA-TECDOC-1025, IAEA, Vienna (1998).

[1.7] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: CANDU Pressure Tubes, IAEA-TECDOC-1037, IAEA, Vienna (1998).

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[1.10] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Metal Components of BWR Containment Systems, IAEA-TECDOC-1181, IAEA, Vienna (2000).

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[1.12] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: CANDU Reactor Assemblies, IAEA-TECDOC-1197, IAEA, Vienna (2001).

[1.13] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Primary Piping in PWRs, IAEA-TECDOC-1361, IAEA, Vienna (2003).

2. DESCRIPTION OF REACTOR PRESSURE VESSEL

This section provides the overall system description for BWR RPVs and includes design features, applicable material specifications, and differences amongst the various RPV components.

Operating BWR RPVs were fabricated by several suppliers. These include Combustion Engineering, Inc., Babcock & Wilcox Company, CBIN, Ishikawajima-Harima Heavy Industries Company, Ltd., Chicago Bridge & Iron Company, Babcock Hitachi and Rotterdam Dry-dock and Manufacturing (RDM). For the Siemens BWR RPVs main suppliers were e.g. Breda, Uddcomb and RDM. For the ABB BWR RPVs a main supplier is Uddcomb.

2.1. RPV design features

The BWR RPV is comprised of a shell and a removable top head each with flanges which accommodate the head to flange bolting, closure studs, a bottom head which is welded to the shell, multiple nozzles and safe-ends, multiple penetrations and control rod drive stub tubes, a vessel support skirt and several attachment welds. Figure 2-1 (a) to (f) shows typical BWR RPVs.

Vessel inside diameters (IDs) range from 3658 to 7112 mm (144 to 280 inches), vessel shell thicknesses from 100 to 180.3mm (4 to 7.1 inches), vessel heights (inside top head to inside bottom head) from 16408 to 22149 mm (646 to 872 inches) and vessel head thicknesses from 68.6 to 172.7mm (2.7 to 6.8 inches), more than 200 mm for Siemens BWR plants.

GE BWR Nuclear steam supply product lines, BWR/1 through BWR/6, represent an evolutionary development of RPV design concepts. For example, the BWR/2 product line uses forced recirculation. Internal jet pumps were introduced with the BWR/3 product line to improve recirculation efficiency; jet pump use was continued on later product lines. BWR/4 and BWR/5 product lines increased power density and made other product improvements. Later vessels tend to have more nozzles and penetrations than do the earlier ones, i.e. more emergency core cooling system (ECCS) nozzles and control rod drive (CRD) and neutron monitoring penetrations. BWR/2 through BWR/5 vessels have internal stub tube /CRD housing designs for the bottom head CRD penetrations. BWR/6 vessels use a straight- through design with CRD housings welded directly to the bottom head. The Advanced Boiling Water Reactor (ABWR) developed by GE, Hitachi, Toshiba and Japanese Utilities combines selected features of worldwide BWR designs. The most notable changes are the use of internal recirculation pumps and Fine Motion Control Rod Drives (FMCRD).

Siemens BWR are represented by two product lines ’69 and ’72. Remarkable improvement from ’69 to ’72 with respect to the RPV was the manufacturing of the shells from forgings instead of longitudinal welded plates. Top and bottom heads as well as adjoining segment rings are manufactured from plates in both product lines.

Nozzles and penetrations

Reactor vessels have many penetrations for piping and equipment. Vent, instrumentation, and head spray nozzles are located in the top head. CRD penetrations, flux monitoring instrument penetrations, core pressure drop, standby liquid control nozzle, and drain nozzle are in the bottom head. The remaining nozzles are in the cylindrical shell. Recirculation system nozzles and the jet pump sensing line penetrations are located in the lower shell course (below the core). Recirculation system nozzles for the BWR/3 through BWR/6 product lines are in the lower shell course; BWR/2 recirculation inlet/outlet nozzles are in the bottom head and lower shell respectively. In addition some plants have an isolation condenser return line nozzle in the lower shell. The beltline region shell course has no nozzle penetrations. The upper shell course has steam nozzles and some instrumentation nozzles. The remaining nozzles which include feedwater, high pressure core spray, low pressure core spray, control rod drive (CRD) hydraulic return line, instrumentation and low pressure coolant injection (LPCI), are typically located in the shell course just above the core region. Of these, the LPCI nozzles are located nearest to the beltline region. Some plants do not have all these nozzles.

In the case of Siemens reactor pressure vessels nozzles for instrumentation and head spray are located in the top head. Nozzles for pressure and water level measurement, main steam nozzles, feedwater nozzles and emergency core cooling nozzles are located in the cylindrical part. The bottom part contains CRD nozzles, different instrumentation nozzles (core, Δp, temperature, etc.) and recirculation pump nozzles.

Plant operating pressures are typically 6.90 to 7.24 MPa (1000 to 1050 psi); the vessel design pressure is 8.62 MPa (1250 psi), and the usual pre-service vessel hydrostatic pressure is 10.78 MPa (1563 psi: 1.25 × design pressure. It should be noted that RPVs designed to ASME Section I or Section VIII were tested at 1.5 × design pressure, i.e. 1875 psi or 12.95 MPa). The in-service hydrostatic pressure is typically 6.90 MPa (1000 psi) for leak testing and 7.59 MPa (1100 psi) for ISI. Some countries repeat hydrostatic pressure testing at 1.25 × design pressure at regular intervals. The BWR reactor pressure vessel design temperature is typically 302°C (575°F); operating temperatures range from 282 to 288°C (540 to 550 °F). The plant operating pressure for Siemens BWR is typically about 7.3 MPa, the RPV design pressure is 8.9 MPa, the hydrostatic test pressure is 11.6 MPa (1.3 × design pressure). The design temperature is 310°C for the cylindrical part and 300°C for the top and bottom region, whereas the operating temperature for Siemens BWR is about 286°C.

The ABWRs as well as most ABB BWRs and all but one Siemens' BWR plant utilize internal recirculation pumps mounted on nozzles located in the knuckle region of the lower head. The Siemens plants use pumps with mechanical seals while the ABWR and ABB plants use seal less pumps. Other features of the ABWR design that depart from past practice are the use of an ellipsoidal lower head and the different size and arrangement of nozzles. The ABWR FMCRD does not use a CRD return line.

ABB RPVs have two main generations, (1) External pump reactors and (2) Internal pump reactors. For ABB RPVs the design pressure is 8.5 MPa (abs) and the design temperature is 300^{o}C. The operation pressure is 7.0 MPa. Pre-service pressure test is done at 1.3 times the design pressure at elevated temperature.

Design analysis of ABB RPVs is done according to the ASME Code Section III NB-3000. Fatigue analysis is performed in accordance with ASME section III section NB-3216.2. Process Nozzles in ABB RPVs are designed as full forged set-in nozzles welded into the ring shell of the RPV. In internal pump RPVs all process nozzles are above the core top level.

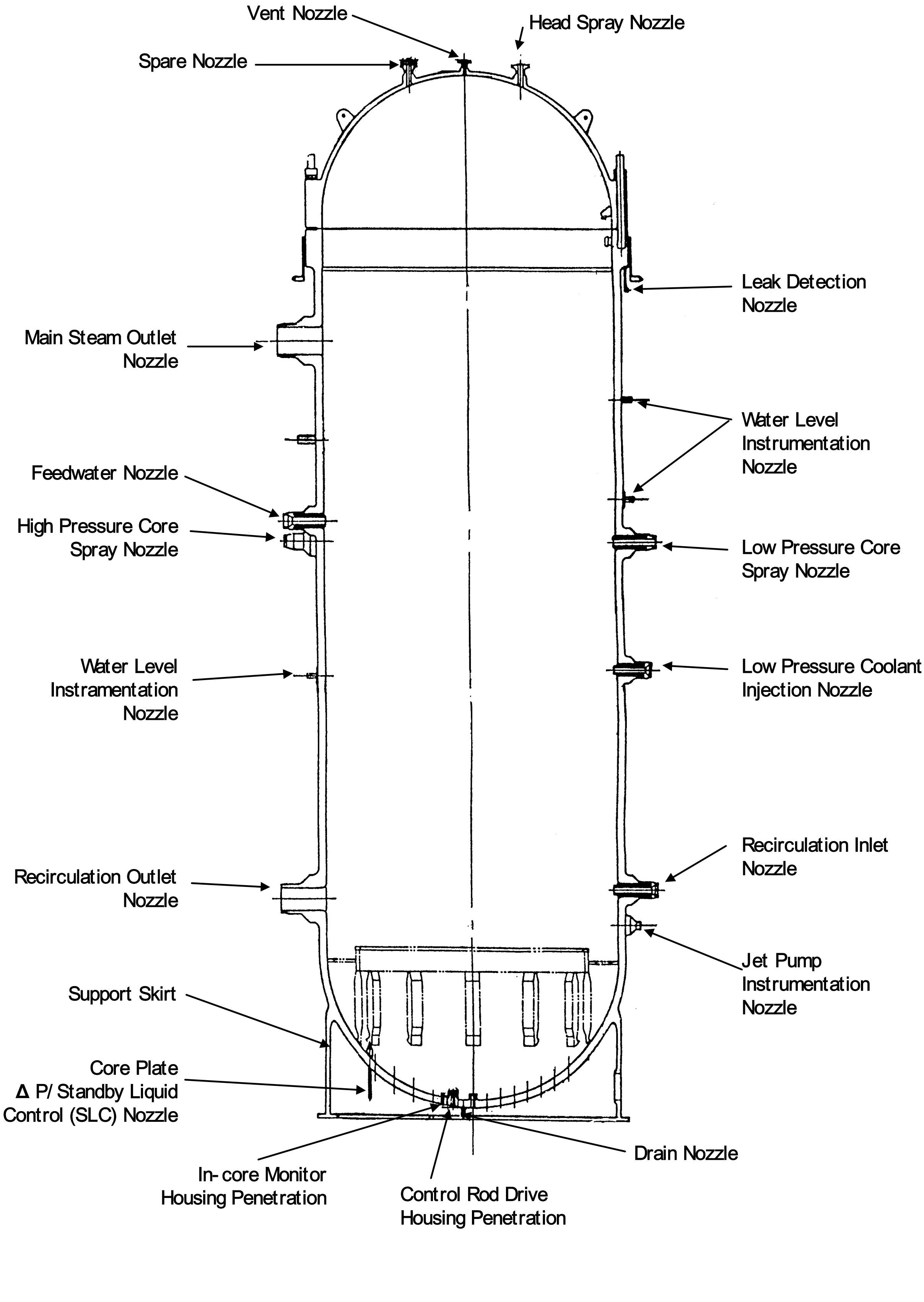

FIG. 2-1(a). Typical GE BWR-5 vessel.

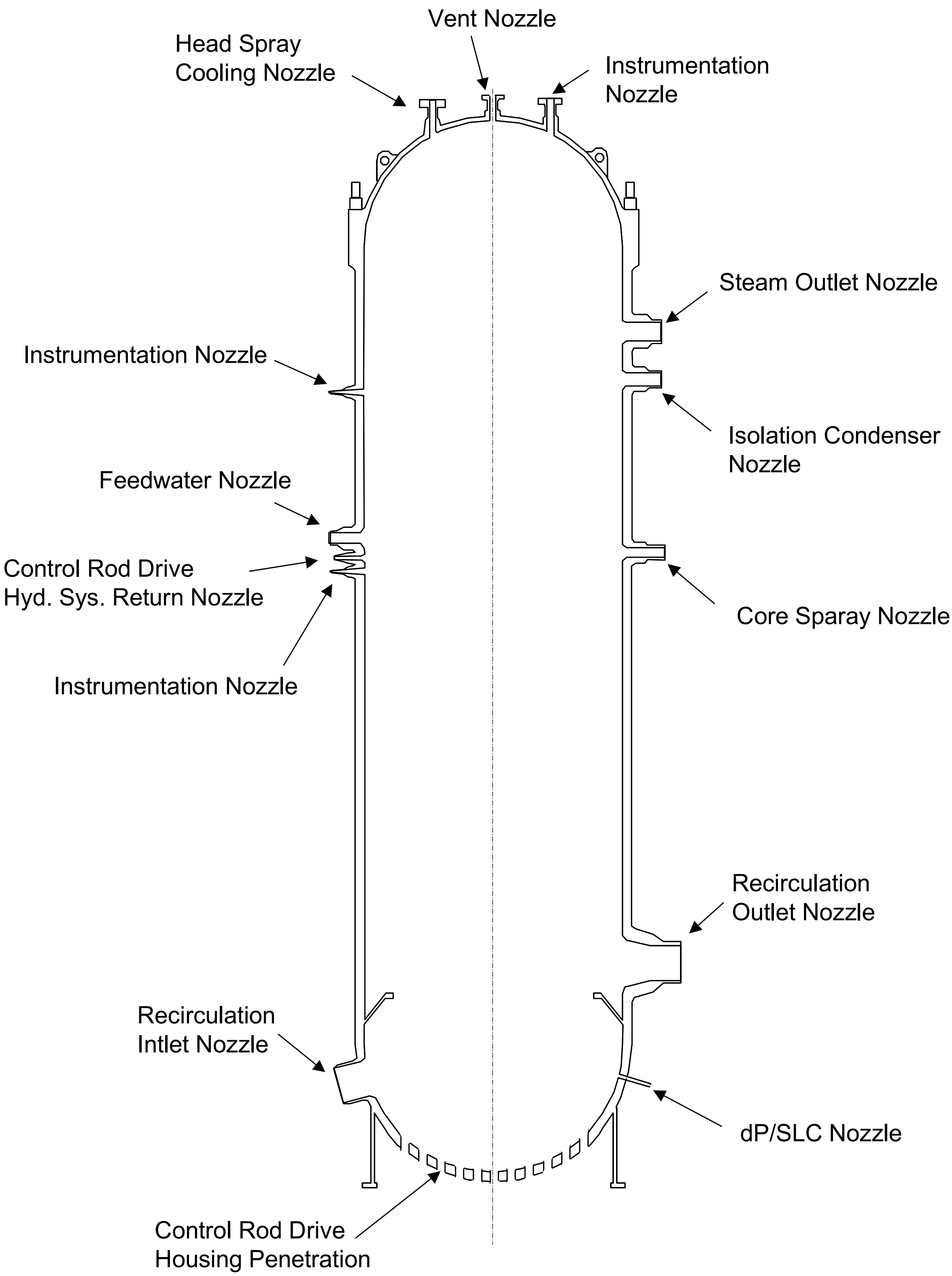

FIG. 2-1(b). Typical GE BWR-2 vessel.

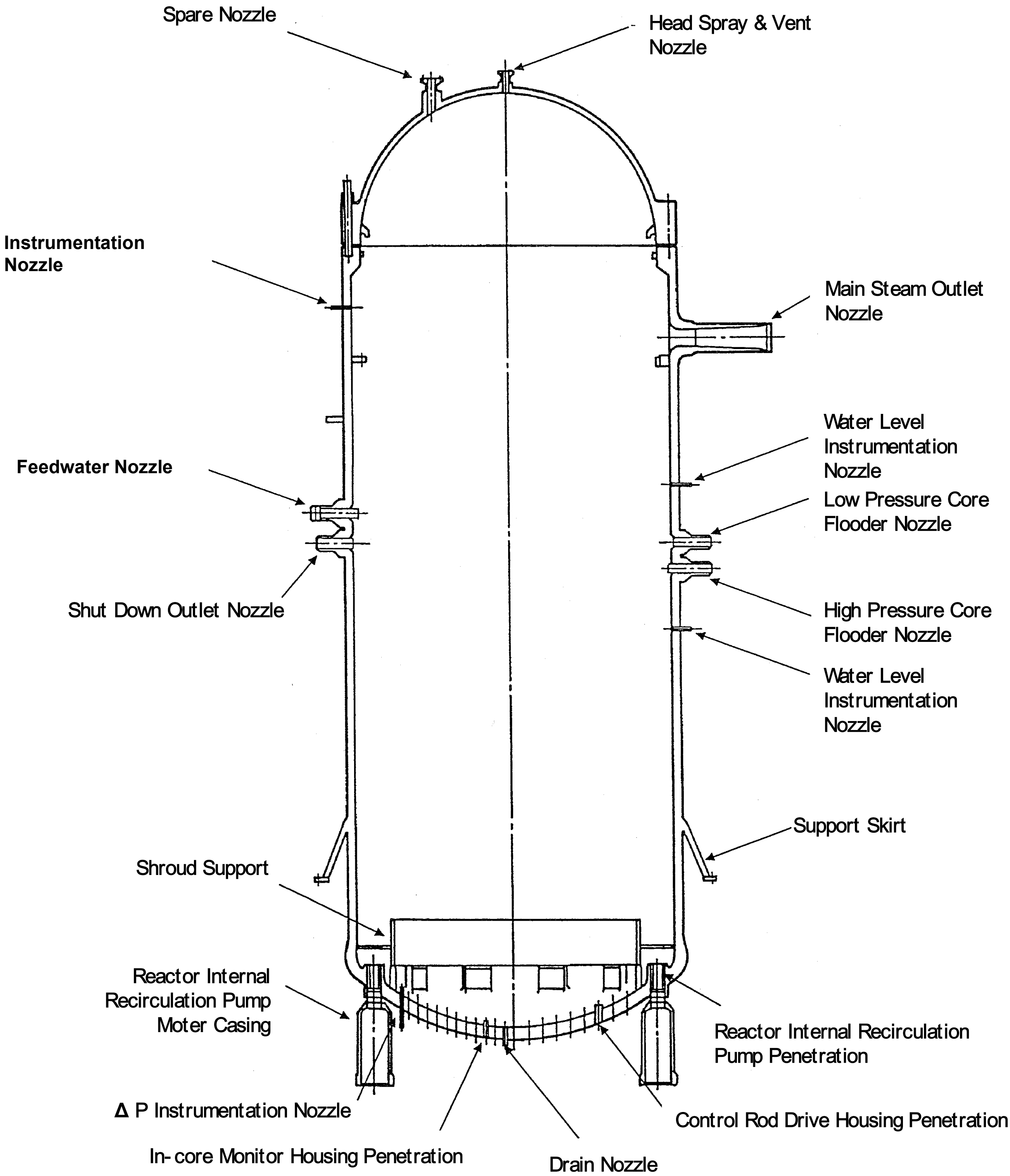

FIG. 2-1(c). ABWR vessel.

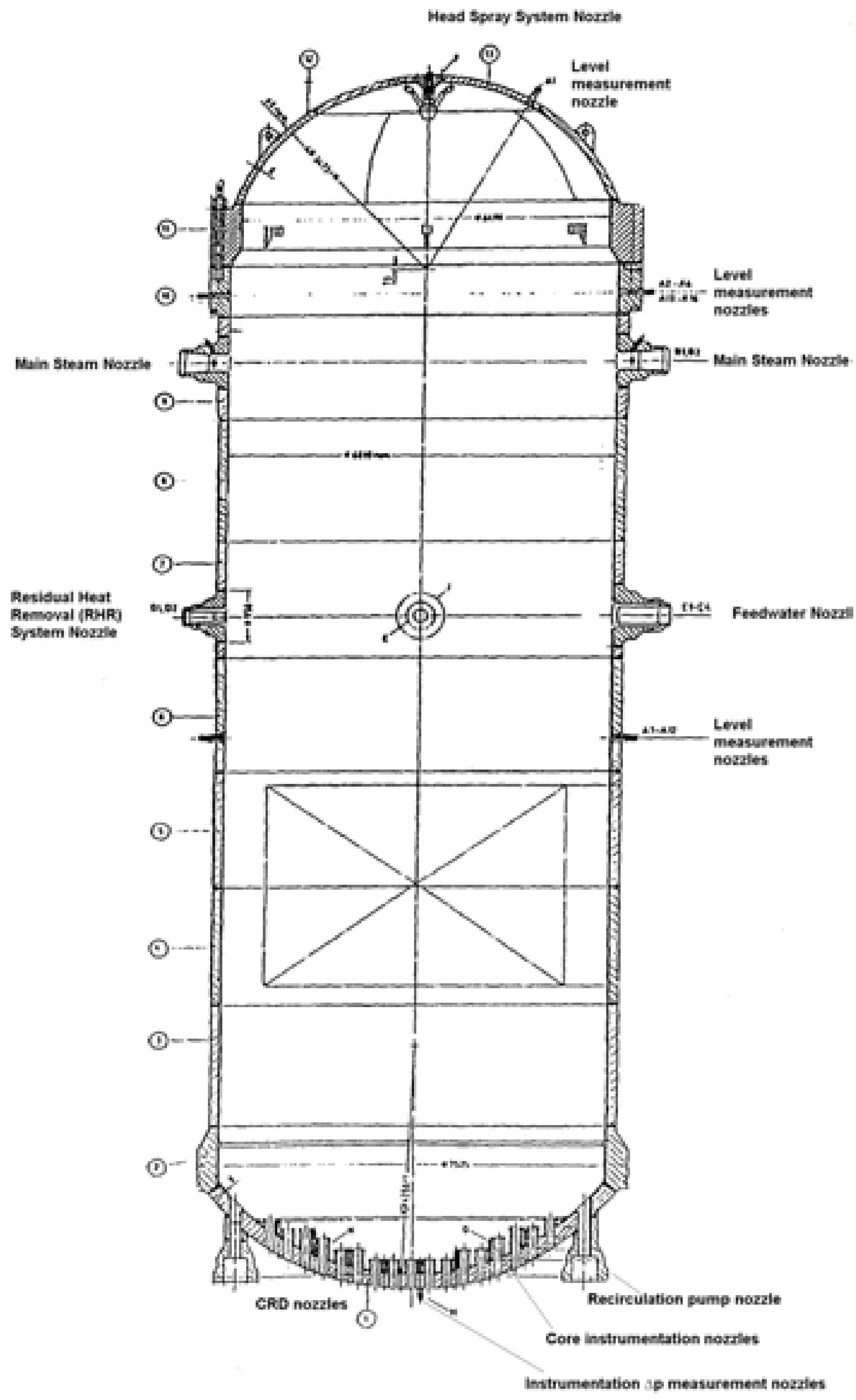

FIG. 2-1(d). Typical Siemens BWR vessel.

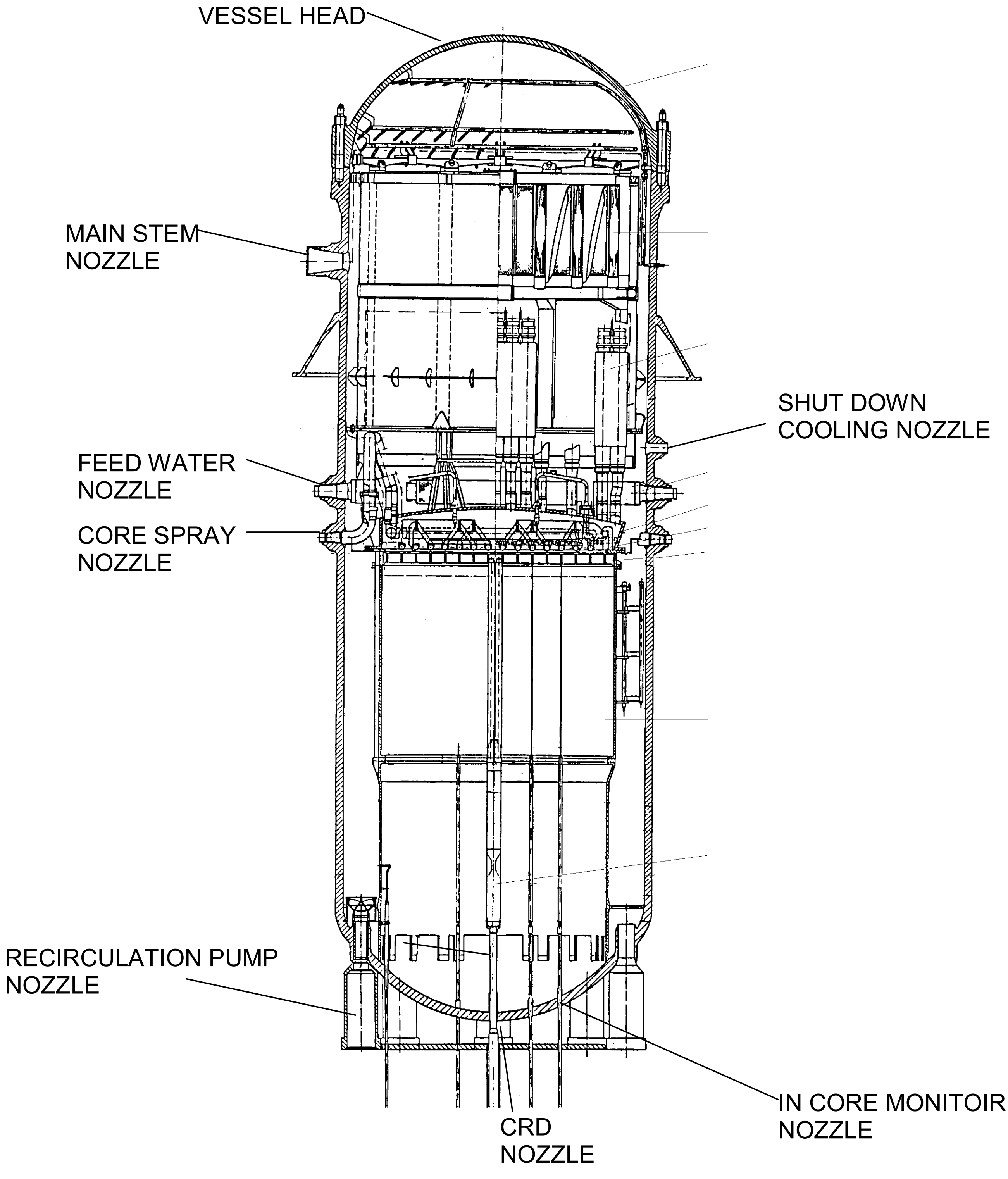

FIG. 2-1(e). Typical ABB BWR Vessel with internal pump (BWR660)

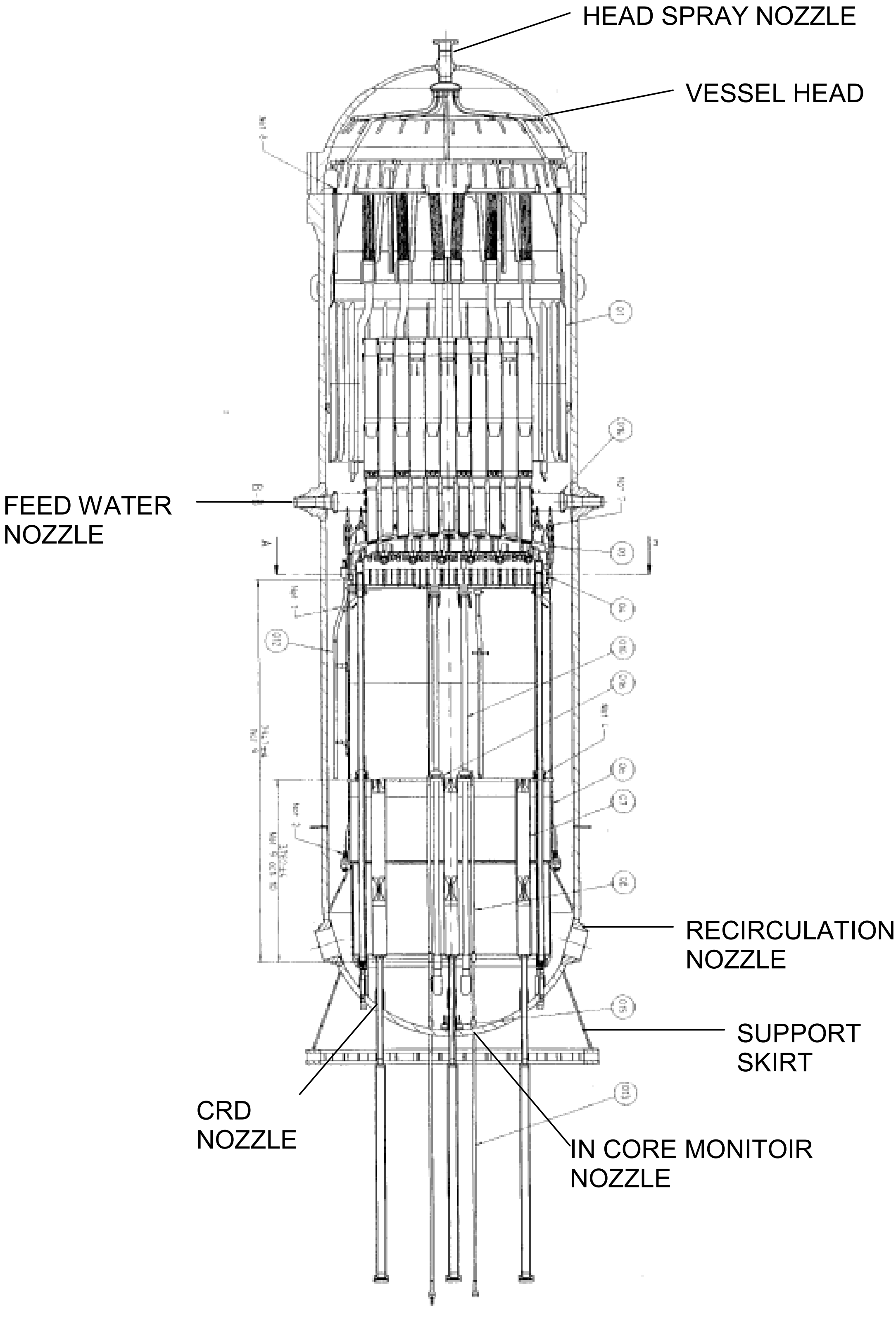

FIG. 2-1(f). Typical ABB BWR Vessel with external pump (BWR580)

2.2. Vessel materials and fabrication

Materials

BWR vessels use different materials for the different components (shells, nozzles, flanges, studs, etc.). In addition, the choices in the materials of construction changed as the BWR product line evolved. For example, shell plates changed from American Society for Testing and Materials (ASTM) A302, Grade B to American Society of Mechanical Engineers (ASME) SA 533, Grade B, Class 1. Table 2-1 (1) to (7) show the various materials used in typical BWR vessel constructions.

Siemens BWR RPV shells, flanges and many of the nozzles are manufactured from plates and later forgings of material 22 NiMoCr 3 7, German Material No. 1.6751 corresponding to ASTM 533 Grade B, Class 1 and ASTM 508 Class 2 respectively. [See Table 2-1 (4)]

Typical ABB RPV materials are presented in Table 2-1(5). ASME SA 533, Grade B, Class 1 is also used as a shell plate material. Shells and nozzles are manufactured from ASTM SA 508 Class 3.

Material properties, such as chemistry and toughness, are specific for each heat of material in each vessel. An area of particular interest is the chemistry of low alloy steel materials from which the beltline portion of the vessel shell is made. The beltline portion material directly surrounds 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). 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. As irradiation embrittlement research progressed, beltline material chemistries in the BWR changed. The trend of decreasing copper, which decreases embrittlement, can be seen in Table 2-2 (1) and (3). However, nickel in welds was increased before it was identified as a contributor to embrittlement.

TABLE 2-1 (1) TYPICAL BWR VESSEL MATERIALS (USA)

Material	Typical Use
A308/309 SS or Alloy 182	Attachment welds
Carbon Steel	Safe ends, small nozzles
SA-336	Nozzles, shell head flanges
A540, Grade 23 or 24	Studs, nuts, washers
Inconel SB166	Small nozzles, shroud support, safe ends
Inconel SB 167	Penetrations
Austenitic stainless steel	Safe ends, thermal sleeves, brackets
SA-302, Grade B (mod)	Shell courses
SA-508, Class 2 (mod)	Nozzles and flange forgings
SA-105, Grade II	Nozzle, Safe ends
Sa-533, Grade B, Class 1	Shell courses
A-308/309 SS	Cladding

TABLE 2-2 (1) TYPICAL VALUES OF RADIATION SENSITIVE CONSTITUENTS OF TYPICAL BELTLINE MATERIALS (USA)
Average Chemistry (wt. %)*

BWR Product Line	Product Form	Copper	Phosphorus	Nickel
BWR/2 and BWR/3	Plate Weld	0.19 0.21	0.010 0.018	0.54 0.65
BWR/4	Plate Weld	0.14 0.16	0.011 0.016	0.57 0.81
BWR/5 and BWR/6	Plate Weld	0.10 0.11	0.011 0.013	0.59 0.85

* Values are samples of representative units.

TABLE 2-1 (2) TYPICAL BWR VESSEL MATERIALS (MEXICO)

Material	Typical Use
SA-336, SA-508	Small nozzles, safe ends
SA-336, SA-182, SA-508, SB-166, SS-308L	Safe ends
A540, Grade B24	Studs, nuts, washers
Inconel SB166 SA-508, Class 2	Small nozzles
SA-508, Class 2	Nozzles, Main closure flanges
Sa-533, Grade B, Class 1	Shell head
SS-304	Cladding

TABLE 2-2 (2) RANGES OF RADIATION SENSITIVE CONSTITUENTS OF TYPICAL BELTLINE MATERIALS (MEXICO)
(wt. %)

BWR Product Line	Product Form	Copper	Phosphorus	Nickel
BWR/5	Plate Weld	0.11-0.15 0.02-0.06	0.006-0.014 0.01-0.02	0.49-0.57 0.82-1.08

TABLE 2-1 (3) TYPICAL BWR VESSEL MATERIALS (JAPAN)

BWR-5

Material	Typical Use
A-540 Grade 24	Studs, nuts, washers
SB-166/168	Small nozzles, shroud support, stub tubes
SA-336	Safe ends, thermal sleeves, brackets
SA-508 Class 3	Shell courses, nozzles, flange, lowerhead knuckle
SA-508 Class 1	Safe ends
SA-533 Grade B Class 1	Shell courses
A-308/309 SS	Cladding

ABWR

Material	Typical Use
A-540 Grade 24	Studs, nuts, washers
SB-166/168	Small nozzles, shroud support, stub tubes
SA-336	Thermal sleeves, brackets
SA-508 Class 3	Shell courses, nozzles, flange
SA-508 Class 1	Safe ends
SA-533 Grade B Class 1	Shell courses
A-308/309 SS	Cladding

TABLE 2-2 (3) RANGES OF RADIATION SENSITIVE CONSTITUENTS OF TYPICAL BELTLINE MATERIALS (JAPAN)
(wt. %)

BWR Product Line	Product Form	Copper	Phosphorus	Nickel*
BWR/2 and BWR/3	Plate Weld	0.21 0.09	0.017 0.015	0.57 (037 – 0.73) 0.77
BWR/4	Plate Weld	0.10 0.08	0.011 0.013	0.57 (037 – 0.73) 0.73
BWR/5	Plate Weld	0.06 0.06	0.008 0.011	0.64 (0.37 – 1.03) 0.73
ABWR	Plate Weld	0.03 0.01	0.002 -	0.89 (0.37 – 1.03) 0.68

* Values in brackets are design specifications. Other values are samples of representative units.

TABLE 2-1 (4) TYPICAL BWR VESSEL MATERIALS (SIEMENS)

Material	Typical Use
1.6751 (22 NiMoCr 3 7)	Shells, flanges, head & bottom, nozzles, safe ends
Inconel 600	Internal part of pump nozzles and jacket
Austenitic stainless steel	Nozzles, safe ends, sleeves
A-540 Grade B24	Studs, nuts, washers
A308/309, A347 (1.4551)	Cladding
Inconel 182, 82	Butterings, attachment welds

TABLE 2-2 (4) TYPICAL VALUES *OF RADIATION SENSITIVE CONSTITUENTS OF TYPICAL BELTLINE MATERIALS (SIEMENS) (WT. %)

BWR Product Line	Product Form	Copper	Phosphorus	Nickel
‘69	Plates	0.12	0.009	0.80
‘69	Welds	0.09	0.021	0.07
‘72	Plates, Forgings	0.05	0.007	0.87
‘72	Welds	0.06	0.007	1.70

* Values are samples of representative units.

TABLE 2-1 (5) TYPICAL BWR VESSEL MATERIALS (ABB)

Material	Typical Use
SA-508 Class 3	Shell rings, nozzles, flange
SB-166	Nozzle safe ends
SA-533 Grade B Class 1	Shell courses
A7-E19.9L	Nozzle cladding
A43-EniCrFe-3	Nozzle to safe ends welds and buttering

TABLE 2-2 (5) RANGES OF RADIATION SENSITIVE CONSTITUENTS OF TYPICAL BELTLINE MATERIALS (ABB)
Average Chemistry (wt. %)

BWR Product Line	Product Form	Copper	Phosphorus	Nickel
Internal Pump Plant (OL1)	Plate Weld	0,04(max 0.15) 0,11(max 0,15)	0,007 (max 0,015) 0,013 (max 0,015)	0.45-0.70

TABLE 2-1 (6) TYPICAL BWR VESSEL MATERIALS (SWITZERLAND)

Material	Typical Use
SA-533 Gr.B Cl.1, SA-508 Cl.2	Main plates, main forgings
SA-182 F304	Brackets
SA-540 Gr.B23 Cl.3	Studs, nuts, washers
SA-168	Core support structures
SFA-5.9 308L / 304L SAW ER309L, ER308L	Cladding, weldments for brackets

TABLE 2-2 (6) RANGES OF RADIATION SENSITIVE CONSTITUENTS OF TYPICAL BELTLINE MATERIALS (SWITZERLAND) (WT. %)

BWR Product Line	Product Form	Copper	Phosphorus	Nickel
BWR/3	Plates	0.15	0.009	0.87
BWR/3	Manual weld	0.11	0.015	0.17
BWR/3	Automatic weld	0.32	-	0.12
BWR/6	Plates	<0.1	<0.015	0.55-0.7
BWR/6	Welds	0.008	0.011	0.96

TABLE 2-1 (7) TYPICAL BWR VESSEL MATERIALS (SPAIN)

(BWR3)

Material	Typical Use
A-336 c.c.1332	Vessel shell (forged rings)
A-332 B c.c.1339	Bottom and top heads
SA-193 par.4 c.c.1336-1 Cl.3	Studs, nuts, washers and bushings
A-336 c.c.1332	Nozzles
A-276 Tp.304	Small nozzles
A-105 Gr II (original) SA-508 Cl.1 (replacements)	Low alloy steel safe ends
A-336 F8 (original) SA-182 Tp.316 (replacements)	Stainless steel safe ends
ER 309/308	Cladding

(BWR6)

Material	Typical Use
SA-533 Gr. B Cl.1 LAS	Vessel shell and Head plate
SA-540 Gr. B24 LAS	Studs, nuts, washers and bushings
SA-508 Cl. 2 LAS	Nozzles and closure flanges
SA-182 T304	Small nozzles
SA-508 Cl. 1, Inconel SB-166, SA-336 F8	Safe ends
SA-376 Tp 304, SB-166	Thermal sleeves
E308L	Cladding

TABLE 2-2 (7) RANGES OF RADIATION SENSITIVE CONSTITUENTS OF TYPICAL BELTLINE MATERIALS (SPAIN)
(wt. %)

BWR Product Line	Product Form	Copper	Phosphorus	Nickel
BWR/3	Plate	0.10	0.009-0.014	0.67-0.72
	Weld	0.30-0.35	0.012	0.090-0.095
BWR/6	Plate	0.03-0.06	0.005-0.008	0.62-0.65
	Weld	0.02-0.08	0.012-0.015	0.40-0.82

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 [2.1, 2.2].

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 Dry-dock and Manufacturing and by Creusot-Loire. In some cases, vessels were constructed by more than one fabricator because of scheduling problems in the shops.

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 2-2(a)). In the second method, large ring forgings are used (Fig 2-2(b)). 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 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.

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.

The shell courses and the bottom heads are clad with weld deposited stainless steel or Ni-Cr-Fe alloy. On earlier vessels the top head was clad. On later vessels the top head was left unclad because the interior surface is in contact with dry steam during normal operation. Carbon and low alloy steel vessel nozzles are normally clad if located below the water line. Recent construction practice use unclad feedwater nozzles, and cladding has been removed from the feedwater nozzles of many operating plants. On BWR/5 RPVs all nozzles except recirculation nozzles are unclad to improve the capability for inspection of the nozzle to shell welds by ultrasonic testing (UT). On BWR/6 RPVs all nozzles are unclad. This was done after corrosion studies showed minimal corrosion rates on exposed low alloy steel surfaces.

Siemens BWR RPVs have a stainless steel cladding covering almost all areas of the vessel and the ferritic nozzles. Exceptions are the main steam nozzles that are clad only up to the conical region.

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 growth 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.

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. Both of the reheat cracking and the cold cracking were an issue at the time of fabrication and they were mostly resolved. Today they are not considered an ageing issue.

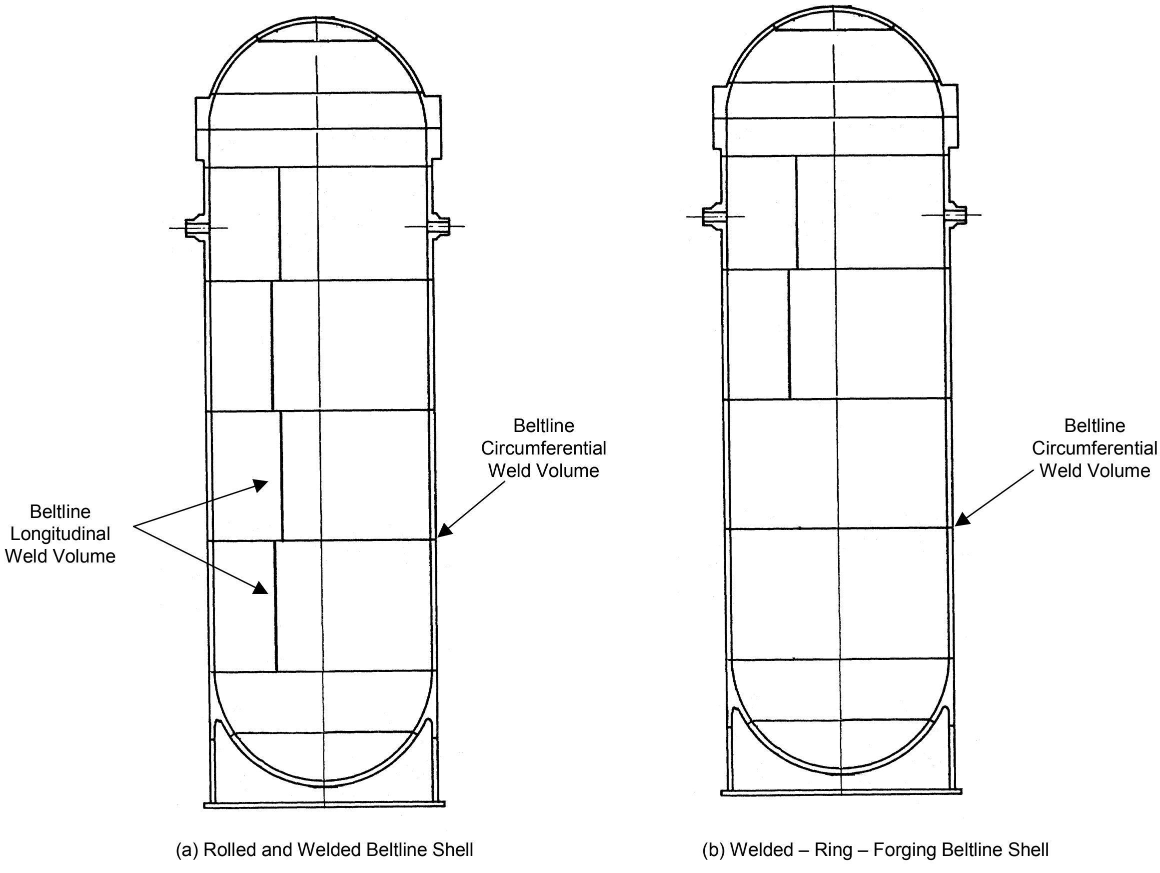

FIG. 2-2. Fabrication configuration of BWR beltline shells.

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 welds sensitivity to radiation embrittlement, RPV fabricators imposed strict limits on the percentage of copper and phosphorus in the welds as well as in plates [2.1, 2.3, 2.4]. The use of copper coated weld wire was eliminated due to the strict limits on the percentage 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 heat are needed. This becomes important when determining the properties of each individual weld in the beltline for sensitivity to neutron irradiation if different heats were used. 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 heat used in fabrication. Each weld in the vessel can be traced by the unique weld wire heat and flux lot combination used [2.2].

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.

REFERENCES TO SECTION 2

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[2.2] NUCLEAR REGULATORY COMMISSION, Reactor Pressure Vessel Status Report, Rep. NUREG-1511, USNRC, Washington, DC (1994).

[2.3] WHITMAN, D.G., et al., Technology of Steel Pressure Vessel for Water-Cooled Nuclear Reactors, ORNL-NSIC-21, Oak Ridge National Laboratory, Oak Ridge, TN (1967).

[2.4] GRIESBACH, T.J., SERVER, W.L., Reactor Vessel Embrittlement Management Handbook, TR-101975-T2, Electric Power Research Institute, Palo Alto, CA (1993).

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 [3.1]. 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 Section 3.2. Some important differences exist in the RPV design requirements of Germany and these differences are discussed in Sections 3.3.

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 [3.1], which is divided into three sub-articles:

(a) NB-3100, General Design
(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 based on minimizing cumulative damage and distortion whereas limits for very infrequent conditions allow more damage and distortion albeit still within conservative limits.

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

Later code editions clarified this nomenclature but basically retained the same stress allowables. The corresponding new categories are:

(a) Service Level A
(b) Service Level B
(c) Service Level C
(d) Service Level D.

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, 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.

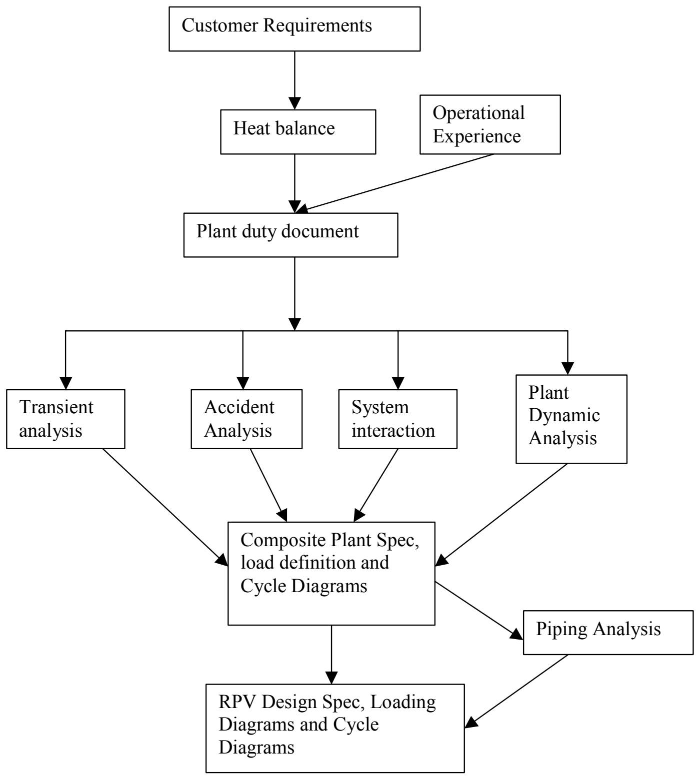

FIG. 3-1. Development of design transients.

For a BWR, 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. 3-1.

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 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. 3-2 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.

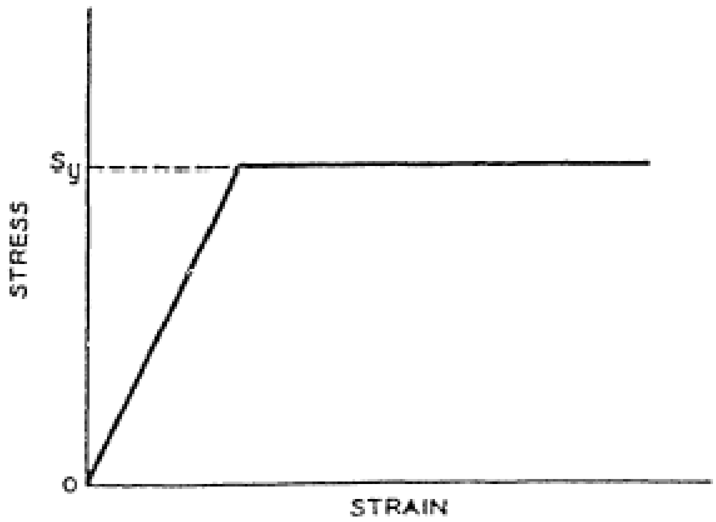

Fig. 3-2 Idealized stress-strain relationship.

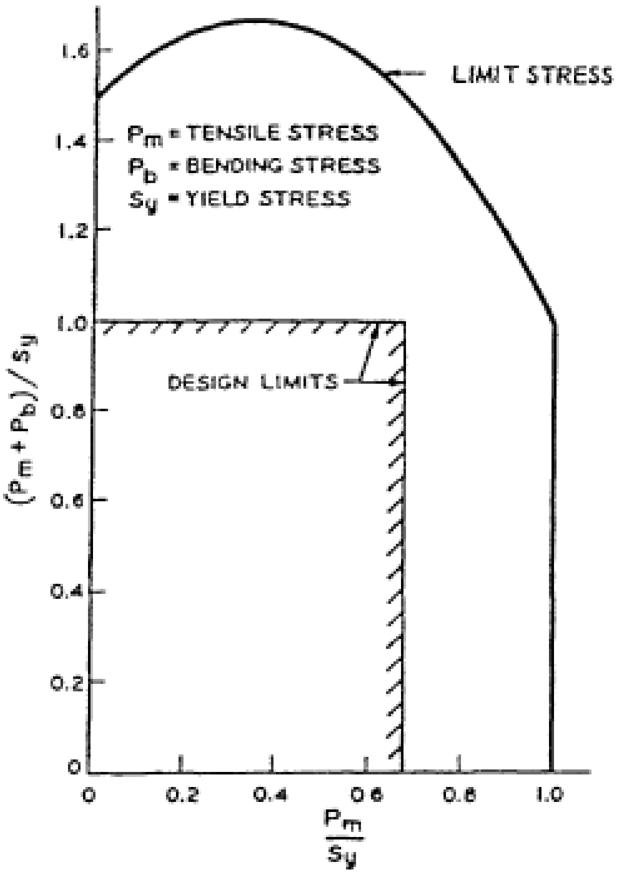

FIG. 3-3. Limit stress for combined tension and bending (rectangular section).

In a structure as simple as a straight bar in tension, a load producing yield stress, Sy, 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. Fig. 3-3 shows the value of the maximum calculated stress at the outer fiber of a rectangular 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, Sy. When the average tensile stress, Pm is zero, the failure stress for bending is 1.5 Sy. When the average tensile stress is Sy no additional bending stress, Pb, may be applied.

Fig. 3-3 was used to choose allowable values, in terms of the yield stress, for general primary membrane stress, Pm and primary membrane-plus-bending stress, Pm + Pb. It may be seen that limiting Pm to (2/3) Sy and Pm + Pb to Sy 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 Sy 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 Sm, 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) Sy and may reach 0.9 Sy at temperature. Some explanation of the use of up to 0.9 Sy for these materials as a basis for Sm is needed in view of Fig. 3-3 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 austenitic and 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 Sm 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 3-1 The basis for the allowable design stress intensity values (Sm) is shown in Table 3-2 for typical reactor vessel material.

TABLE 3-1 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 3-2 BASIS FOR THE ALLOWABLE DESIGN STRESS-INTENSITY VALUES (SM) IN SECTION III OF THE ASME CODE (THESE VALUES REFLECT FACTORS AT THE TIME OF DESIGN OF MOST VESSELS)

- 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-chronium-iron and Ni-Cr-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 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 the specified minimum yield strength at room temperature
  - 1/3 of the yield strength a temperature of 426.7 ˚C (800 ˚F)

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.

Areas of the vessel analyzed

The regions of the vessel, which are examined in order to determine compliance with the ASME Code are the areas which have potentially the highest stresses. Typically, these areas include nozzles, flanges, bolting, attachments and supports. Also a basic sizing calculation is completed for all components.

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.

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 3-3.

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 yield strength (σ_y) in the primary membrane plus primary bending stress category, the stress intensity is limited to 1.35 σ_y.

TABLE 3-3 ALLOWABLE STRESS INTENSITY LIMITS IN SECTION III OF THE ASME CODE FOR EMERGENCY CONDITIONS

Primary stress		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.

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 [3.2]. 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 as 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 as 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. 3-4. 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). (Older BWRs usually have Charpy specimens oriented parallel to the rolling direction). 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. 3-5, 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.

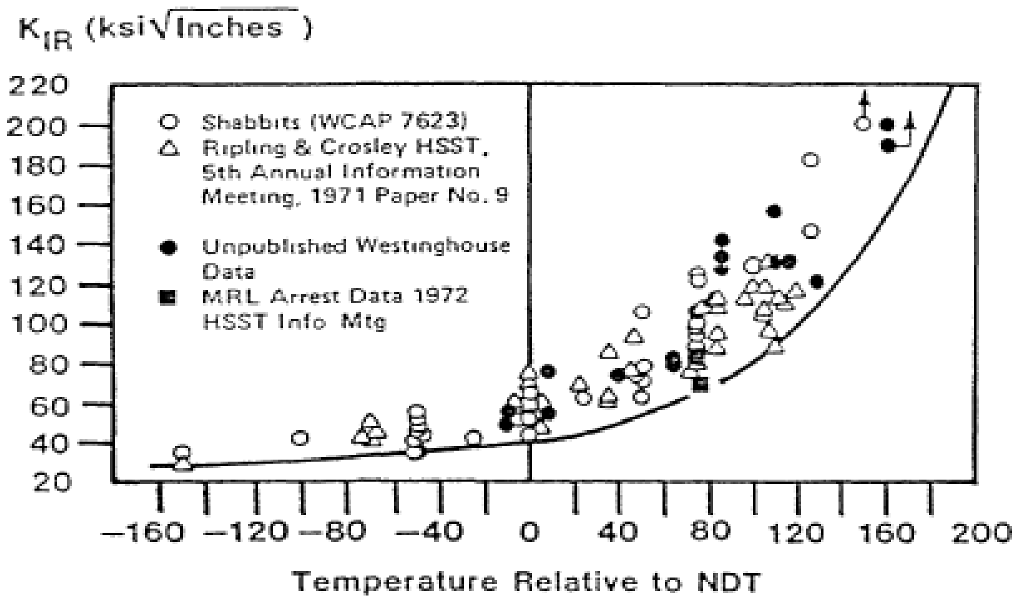

FIG. 3-4. Derivation of curve of reference stress intensity factors (K_{IR}).

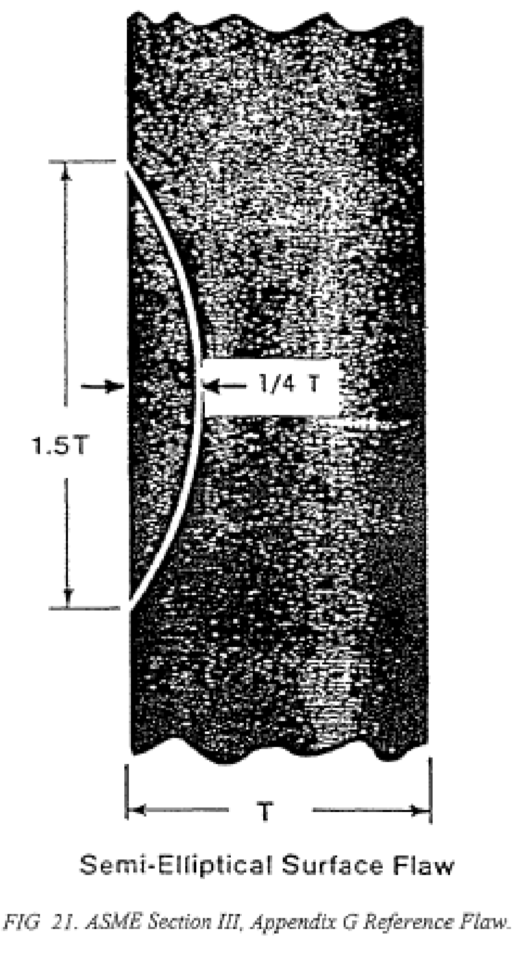

Figure 3-5. Derivation of curve of reference stress intensity factors (KIR).

G-2200 outlines the recommended procedure for protection against non-ductile 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.

Because the pressure-temperature (°F) relationship of a BWR is controlled by the steam properties, brittle fracture concerns are limited to determining the test temperature.

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 heat-up and cool-down rates, the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heat-up or cool-down 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:

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

where

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

Therefore, the governing equation for the heat-up/cool-down analysis is defined in Appendix G of Section III of the ASME Code [3.2] as follows:

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

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

For determining test temperatures, 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

Also, the 1993 Amendment to 10 CFR 50 has a rule, which addresses the metal temperature of the top 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 pre-service hydrostatic test pressure.

Vendors, owners and regulatory bodies can perform or require others to perform 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. Additional regulatory requirements for RPV in the USA

3.2.1. Codes and standards

Part 50 of the Code of Federal Regulations, Title 10 (10CFR50) [3.3] regulates construction of nuclear power plants. Section 10CFR50.55(a) defines the reactor vessel to be part of the reactor coolant boundary and requires that the vessel meet requirements for Class 1 vessels contained in the ASME Boiler and Pressure Vessel Code (B&PV Sections III [3.1] and XI [3.4].

Section III of the ASME Code is the industry standard for construction of nuclear power plant facilities, including reactor pressure vessels (RPVs), while Section XI prescribes in-service requirements, including inspection and evaluation of defects. Almost all of the states in the U.S. have adopted the ASME Code. All BWR vessels have followed the ASME Code. For plants with construction permits issued before May 14, 1984, the ASME Code edition and addenda in effect at the time of issuance applies. BWR/3 through BWR/6 plants were constructed in accordance with various editions of Section III of the Code. Earlier plants, such as BWR/2 units, were constructed to predecessors of the nuclear construction code, such as Section I (Power Boilers) [3.5] and Section VIII, Division 1 (Unfired Pressure Vessels). [3.6]

Federal Regulation 10CFR50 [3.3] also contains other regulations, such as those in Appendices A, G, and H, which are applicable to the vessel. 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 A specifies fracture toughness requirements for ferritic materials based on ASME Code, Section III. Requirements for the reactor vessel material surveillance program are based on ASTM requirements [3.7] and are specified in Appendix H of Federal Regulation 10CFR50.

Many GE BWRs were already operating when Appendices G and H were first published in 1973 so these requirements are not part of the design basis. Most other GE BWRs were in the late stages of construction by this date. The requirements of these appendices have been implemented on a plant-by-plant basis.

3.2.2. Regulatory guidance

The US NRC has issued several Regulatory Guides (RG) describing acceptable methods for implementing specific parts of 10CFR50. The following RGs are relevant to reactor pressure vessels.

U.S. NRC Regulatory Guide 1.99

Irradiation embrittlement of the beltline is calculated according to RG 1.99. Revision 1 [3.8] of this guide, in effect since 1977, has been replaced by Revision 2 [3.9] which predicts higher Reference Temperature (RT) shifts for BWRs. Rev. 3 of RG 1.99 is in preparation.

Early plants were designed before this guidance was issued. Vessel base materials and/or their welds in some older BWRs have relatively large amounts of copper. Revision 1, which influenced the design of BWR/5 and BWR/6 plants, used correlations based on copper and phosphorus. Revision 2 correlations are based on copper and nickel rather than copper and phosphorus. Furthermore, Revision 2 is based almost entirely on pressurized water reactor (PWR) surveillance irradiation shift data. As a result of Revision 2, predicted BWR irradiation shifts will increase, even for BWR/5 and BWR/6 plants, where, even though copper and phosphorus contents are kept low, nickel content can be high.

Other regulatory guides

RG 1.43 [3.10] provides guidance to assure that production of cladding complies with ASME Section III and XI requirements to prevent underclad cracking. The presence of intergranular cracking in base metal under the cladding has been observed in reactor vessels. Surface fissuring of some cladding at US BWR plants has been observed.

RG 1.65 [3.11] provides guidance on vessel closure bolting materials and inspections. BWR plants have closure bolts in compliance with ASME Section III and are inspected according to ASME Section Xl. All studs are volumetrically examined and receive a surface examination during each 10 year inspection interval.

RG 1.150 [3.12] provides guidance on ultrasonic test procedures, which supplements those provided in ASME Section XI. BWR procedures for inspection of vessels comply with this guidance.

3.3. Regulatory requirements for RPV 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 (Upper Shelf Energy) 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 a smaller size 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:

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

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 3-5. 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.

Modeling 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 the regions of interest.

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. Regulatory requirements for RPV in Japan

Design requirements for RPV are prescribed by METI Notification No.501 [3.13], and JSME Code on Code for Design and Construction for Nuclear Power Plants, JSME S NC1-2001 [3.14], which are based on ASME Boiler and Pressure Vessel Code, Section III. In addition, JEAC 4206-2000, published in 2000 by the Japan Electric Association [3.15], prescribes experimental methods to confirm the integrity of nuclear power plant components against non-ductile failure. These methods include the linear elastic fracture mechanics analysis method and the PTS evaluation method. JEAC4206 incorporates NRC 10 CFR Part 50 Appendix G (1995) and Appendix H (1995), the ASME Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components (1998); K_{IR} equations are slightly different from those of ASME Section III because they take into account Japanese experimental data.

Two equations are provided for 1 path bead drop-weight and 2 path bead drop weight tests:

K_{IR}= 29.46 + 15.16 exp [0.0274 (T-RT_{NDT})] for 1 path bead; (5)

K_{IR}= 29.43 + 1.344 exp [0.0261 (T-RT_{NDT}+88.9)] for 2 path bead (6)

where:

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

In 2003, the addendum of JEAC 4206 [3.16] was published, which allows to use K_{IC} calculated by the following equation for pressure- temperature limits of the operation condition I, II and the hydraulic pressure/ leak test condition instead of K_{IR}.

K_{IC}= 36.48 + 22.78 exp [0.036 (T-RT_{NDT})] (7)

3.5. Regulatory requirements for RPV in Finland

In Finland nuclear power plant requirements are presented in Nuclear Energy Act, Nuclear Energy Decree, Decisions of the Council of State and Regulatory Guides given by Radiation and Safety Authority, STUK. Design and analysis requirements are in accordance with ASME III.

Identical ABB-type NPP units, Olkiluoto 1 and 2, were taken into operation on 1978 and 1980. In both units the power is upgraded from original 660 Mwe to 840 Mwe. The licensing of upgrading included the updating of safety analysis of systems including RPV in accordance with current code requirements.

TABLE 3-5 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	-----	$1.1\ S_m$	$0.9\ R_{p0.2PT}$	$R_{p0.2PT}$ [3]	$0.7\ R_{mT}$
			P_l	$1.5\ S_m$	-----	$1.65\ S_m$	$1.35\ R_{p0.2PT}$	$1.5\ R_{p0.2PT}$	R_{mT}
			$P_m + P_b$ or $P_l + P_b$	$1.5\ S_m$	-----	$1.65\ S_m$	$1.35\ R_{p0.2PT}$	$1.5\ R_{p0.2PT}$	R_{mT}
		P_e		-----	$3\ S_m$ [1]	$3\ S_m$ [1, 5]	-----	-----	-----
		$P_m + P_b + P_e + Q$ or $P_l + P_b + P_e + Q$		-----	$3\ S_m$ [1]	$3\ S_m$ [1, 5]	-----	-----	-----
	$P_m + P_b + P_e + Q + F$ or $P_l + P_b + P_e + Q + F$			-----	$D \leq 1.0$; $2\ S_a$	$D \leq 1$; [6] $2\ S_a$	-----	-----	

(1) When the 3 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.

REFERENCES TO SECTION 3

[3.1] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components", ASME, New York (1998).

[3.2] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components", Appendix G, "Protection Against Non-ductile Failure", ASME, New York (1998).

[3.3] Code of Federal Regulations, Title 10 — Energy Part 50, Domestic Licensing of (Nuclear Power) Production and Utilization Facilities." Published by the Office of Federal Register, National Archives and Records Administration, Washington, DC.

[3.4] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, ASME Boiler and Pressure Vessel Code, Section XI, Division 1, "Rules for In-service Inspection of Nuclear Power Plant Components", 1998 Edition 2000 Addenda, ASME, New York.

[3.5] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, ASME Boiler and Pressure Vessel Code, Section I, "Power Boiler", ASME, New York.

[3.6] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, ASME Boiler and Pressure Vessel Code, Section VIII, "Unfired Pressure Vessel", ASME, New York.

[3.7] ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels," Annual Book of ASTM Standards, American Society for Testing and Materials 1916 Race Street, Philadelphia, PA 19103, July 1982.

[3.8] Nuclear Regulatory Guides, RG 1.99 (Revision 1), "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials", April 1987.

[3.9] Nuclear Regulatory Guides, RG 1.99 (Revision 2), "Radiation Embrittlement Reactor Vessel Materials", May 1988.

[3.10] Nuclear Regulatory Guides, RG 1.43, "Control of Stainless Steel Cladding of Low Alloy Steel Components", May 1973.

[3.11] Nuclear Regulatory Guides, RG 1.65, "Materials and Inspection for Reactor Vessel Closure Studs", October 1973.

[3.12] RG 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and In-service Inspections", June 1981.

[3.13] Notification of Establishing Technical Standards on Structures etc., of Nuclear Power Generation Facilities, Ministry of Economy and Industry Notification No. 501, Japan, October 1980.

[3.14] Japanese Society of Mechanical Engineers, JSME Codes for Nuclear Power Generation Facilities, JSME S NC1-2001, Rules on Design and Construction for Nuclear Power Plant, JSME Tokyo, August 2001

[3.15] Japanese Industrial Technical Standards: Test Methods to Confirm Fracture Toughness of Nuclear Power Plant Components, JEAC 4206-2000, Japan Electric Association, 2000.

[3.16] Japanese Industrial Technical Standards: Test Methods to Confirm Fracture Toughness of Nuclear Power Plant Components, JEAC 4206-2003 Addendum, Japan Electric Association, 2003.

4. AGEING MECHANISMS

This section describes the age related degradation mechanisms that could affect BWR 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 component operating experience, relevant laboratory data and related experience from other industries. This set consists of the following mechanisms:

1. Radiation embrittlement
2. Fatigue
3. Intergranular and irradiation assisted stress corrosion cracking
4. General corrosion
5. Erosion corrosion.

When experience has shown combining two or more of these mechanisms to be significant (e.g. erosion corrosion), any such synergistic effect has been explicitly evaluated.

The technical evaluation of a particular age related degradation mechanism and its effects on the continued safety performance of a particular BWR 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 programs 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 plant life. The significance of the RPV parts for the relevant mechanisms is addressed in this section.

4.1. Radiation embrittlement

4.1.1. Description of mechanism

Neutrons produce energetic primary recoil atoms, which displace large numbers of atoms from their crystal lattice positions by a chain of atomic collisions. Neutron exposure damage can be characterized by displacements per atom (dpa), which accounts for the neutron energy spectrum as well as the fluence. However, the dpa exposure parameter is not a direct measure of the number of residual defects; the primary defects undergo temperature dependent rearrangements both within the chain and as a consequence of long range migration. Only a fraction of the primary defects survives in the form of small clusters or cluster solute complexes.

Excess point defects created by radiation impede the motion of dislocations, hence raise the alloy’s yield stress. Shifts in Charpy V-notch (CVN) transition temperatures are quantitatively related to the yield stress increases; drops in CVN upper shelf energy also correlate with yield stress increases although the underlying mechanism for this form of embrittlement is not well understood. A variety of empirical and semi-empirical relations have been proposed to relate tensile and CVN properties to changes in fracture toughness.

Embrittlement is a function of both environmental and metallurgical variables. Fluence or dpa, and copper and nickel content have been identified as the primary contributors. Important second order variables include flux, temperature and phosphorous content. There is evidence that a number of other variables such as heat treatment may also influence embrittlement. Past research [4.1] has shown that embrittlement is affected by the combination of environmental and metallurgical variables. Hence, mathematically based statistical data correlations, such as that given in Revision 1 of Regulatory Guide 1.99 [4.2], which do not reflect the basic mechanisms of embrittlement are subject to uncertainty. For the purposes of this evaluation, Revision 2 of Regulatory Guide 1.99 [4.3], with margins added, is considered to be an acceptable basis for evaluating the effects of neutron embrittlement on BWR RPV material.

4.1.2. Significance

The core beltline region of the cylindrical wall and some BWR/5 LPCI nozzles have to be considered for significant embrittlement ageing.

For the following components the 60 year fluences will be less than 10^{21}n/m^2 (E>1MeV) (the damage threshold for neutron irradiation induced embrittlement) or the components are made of material (stainless steel or Alloy 182) that is less susceptible to neutron embrittlement:

- Attachment Welds
- Nozzles
- Top Head
- Bottom Head
- Penetrations
- Vessel Flange
- Closure Studs
- Safe Ends

The effect of low-energy neutron exposure on the reactor support skirt has been evaluated. This assessment took into account the potential effects on the high flux isotope reactor (HFIR) that were noted at the Plant Lifetime Improvement Materials workshop which was held on September 1987 in Albuquerque NM [4.4]. Using the most conservative interpretation of HFIR data and 80 year fluence levels it was found that the reference temperature (RT) shift does not cause a concern as it is less than 11 °C (20 °F). This assessment concluded that because of distance from the reactor core, thermal neutron irradiation induced embrittlement is not significant for BWR vessel support skirts.

Therefore, neutron embrittlement is potentially significant only for shell beltline and some BWR/5 low pressure coolant injection (LPCI) nozzles, which are located in a high flux region.

4.2. Fatigue

4.2.1. Description of mechanism

Fatigue is the sub-critical crack growth under the influence of fluctuating or cyclic applied stresses. Various sources exist for fluctuating stresses but the chief sources are vibration and temperature fluctuations. Fatigue is characterized as a macroscopically brittle mode of failure since there is no gross plastic deformation of the material before ultimate failure, although the crack tip is locally plastic.

Fatigue behavior of components is influenced by a variety of parameters such as stress range, mean stress, frequency, surface roughness, and environmental conditions at notches and around surface defects. The time expiring between crack initiation and detection may constitute a large proportion of component fatigue life. Fatigue initiation curves indicate how many stress cycles it takes to initiate fatigue cracks. These curves, which are materials specific, relate the allowable number of stress cycles to applied cyclic stress amplitudes. Design curves for RPV materials are given in Appendix I to Section III of the ASME Code [4.5]. The fatigue curves are obtained from best fit curves and include a factor of 2 on stress or a factor of 20 on cycles, whichever is more conservative. These factors account for the effects of different characteristics that influence susceptibility to fatigue failure. Significant variations in fatigue life can result from differences in geometry, surface finish, cyclic rate, temperature, and statistical variation in material test results.

Environment significantly affects fatigue initiation. In the BWR, the presence of an active (e.g. oxidizing) environment can accelerate fatigue crack initiation and fatigue crack propagation. Significant environmentally assisted fatigue, commonly referred to as corrosion fatigue, should be considered when dealing with components in the BWR environment. Licensing renewals in the USA might require updating fatigue analysis.

In Japan, METI notified the utilities to adopt The Guidelines for Evaluating Fatigue Initiation Life Reduction in LWR Environment (METI guidelines) in September 2000. METI guidelines define the evaluation formula for carbon steel, low-alloy steel and their welds under LWR environment and austenitic stainless steel and their welds under PWR environment. Utilities and suppliers of components established a practical method for meeting METI guidelines. The evaluation guidelines were completed and published as TENPES guidelines in June 2002 [4.6], [4.7].

For the RPV, there are two causes of thermal fatigue. One is the mixing of cold and hot fluids. The second is the change in plant operating conditions. The mixing phenomenon produces high cycle fatigue while the plant transients produce low cycle fatigue.

Cyclic stresses produced by hot and cold water mixing induces thermal fatigue. The potential for thermal fatigue depends upon the temperature difference between the mixing fluids, flow velocities and heat transfer coefficients. Cracking due to such mixing has been seen in feedwater and CRD Return Line (CRDL) nozzles.

Design improvements and cladding removal have eliminated high cycle fatigue in the feedwater nozzles but the stresses due to start up/shut down and change in feedwater temperature might potentially cause significant fatigue.

4.2.2. Significance

Based on reported in-service cracking incidents, fatigue is a significant degradation mechanism for some BWR pressure vessel components. Most BWR vessels and components have been designed against fatigue crack initiation by using conservative amplitudes and recurrence frequencies for normal and upset loading cycles together with detailed fatigue design procedures of the ASME Code, Section III, and Appendix I fatigue design curves. The remainder either considered designs against fatigue during the design process using simplified fatigue strength reduction procedures, or have addressed fatigue during implementation of the NRC Systematic Evaluation Program (SEP). The evaluation of fatigue for individual reactor components is considered next and, where possible, resolved on a generic basis.

Attachment welds

Thermal and mechanical fatigue cycling of attachment welds is low based on conservative evaluations documented in vessel design stress reports. Fatigue crack initiation is not expected during the RPV service life. Thus, fatigue damage of attachment welds is not a significant ageing degradation mechanism. This conclusion applies to all attachment weld designs.

Bottom head

The bottom head (not including end penetrations or attachment welds) is subjected to complex mixing of water and environmental interaction. This interaction results in moderate fatigue usage. Current plant technical specifications give maximums for top to bottom vessel thermal gradients to limit fatigue of the bottom head. BWR3/4/5 and ABWR maximum top to bottom temperature difference is specified as 80 °C (144 °F). BWR/6 maximum temperature gradient is 56 °C (100 °F). Maximum thermal gradients for BWR/2 have not been specified, but a temperature difference of 81 °C (145 °F) has been recommended. If these temperature differences are not exceeded, fatigue damage to the bottom head is minimal. If these temperature limits are exceeded, a plant specific analysis is required.

Since adherence to plant technical specifications results in low fatigue usage factors (0.03), fatigue crack initiation is not expected during RPV service life. Thus, fatigue damage of the bottom head of the vessel is not a significant ageing degradation mechanism. This conclusion applies to all bottom head design variations.

The problem with temperature difference is that changes in flow can cause it to diminish suddenly, thus plant procedure preclude changing flow until the temperature difference is within the limits. Following procedures will preclude any damage.

Closure studs

Vessel closure studs are subjected to low cycle loads associated with repeated pre-stressing resulting primarily from head removal and installation. The studs typically have high fatigue usage for 40 years. Fatigue is potentially significant for the closure studs during extended operation, requiring further evaluation. However, the fatigue issue can easily be resolved by replacement of the studs, which has been done already in some plants.

Nozzles

Table 4-1 (page 69) gives the degradation potential by ageing mechanism for each nozzle and safe end design. The ratings were based on the presence of susceptible conditions and field experience. As the table reflects, significance of fatigue for the various BWR vessel nozzles is confined to rapid-cycling temperature fluctuations associated with thermal sleeve seal leakage in feedwater nozzles and mixing flow in uncapped CRDRL nozzles. The rapid-cycling fatigue phenomenon results when cold water introduced into a feedwater nozzle annulus mixes with hotter downcomer water. The triple sleeve sparger with piston rings is designed to counter these mechanisms by channeling the shedding vortices and any leakage flow to the downcomer without it contacting the nozzle wall. The welded spargers do not have a leakage issue and some have a double sleeve to counter the shedding vortex issue.

The cause of cracking in the CRDRL nozzle blend area observed at several plants was attributed to mixing of cold CRD return flow with hot vessel water. NUREG-0619 [4.8] established actions to deal with this problem as well as that of feedwater nozzle blend area cracking caused by seal leakage. Most plants except BWR/2 plants have capped or plan to cap the CRD return lines. Capping reduces fatigue duty on the nozzle to the point where fatigue will not be a significant degradation mechanism. This conclusion applies to all capped CRDRL nozzles.

As stated earlier, the high cycle fatigue mechanism that caused feedwater nozzle cracks has been eliminated by design improvement. The duty on feedwater nozzles is still significant due to the effect of start up/shut down and feedwater temperature changes. This duty will be greater for older plants with lower feedwater temperature and plants using final feed water temperature reduction.

Penetrations

ASME Code Section III fatigue analysis and ASME Code Section XI in-service inspections are adequate to demonstrate that fatigue usage is within allowable limits.

In general, the instrument penetrations met the ASME Code Section III criteria to be exempt from fatigue analysis. In cases where fatigue was calculated, the usage was very low (typically less than 0.1). Operating experience and period inspections confirm that there are no unexpected fatigue mechanism accelerating fatigue degradation in the instrument penetrations and nozzles [4.9]

Except in the case of the CRD stub tubes, design evaluations confirm that all other penetrations are not subjected to significant loading. Fatigue crack initiation of these penetrations is not expected either during the initial license term or anticipated extended operation.

Thus, except for the CRD stub tubes, fatigue is not a significant ageing degradation mechanism for the penetrations. This conclusion applies to all penetrations and penetration design variations.

The CRD stub tube region is very complex in geometry and exposure to loads and temperature fluctuations. The major temperature transients that affect the stub tubes are the heat-up and cool-down cycles, which are accompanied by pressure changes that also stress the stub-tubes and the attachment weld between the stub tube and housing. Scram also produces transients affecting the region due to the sudden introduction of cold water at higher flow rates than are experienced in normal operation. There is also some low level thermal cycling that occurs as the result of the colder purge flow, although this does not produce significant fatigue damage.

One purpose of the stub tube is to provide a flexible transition between the rigid head and the CRD housing. The effectiveness of this configuration is shown by the comparison in fatigue usage factors of the stub tube configuration and the straight through design used in BWR. The Stub tube reduces usage factors by approximately 75%.

Safe ends

Table 4-1 (page 69) presents the potential by influencing mechanism for each safe end design. These degradation potentials were based on the presence of susceptible conditions and field experience.

The degradation potential of each safe end was ranked as high or low relative to each ageing mechanisms. Table 4-1 presents these rankings which were based on judgments which took into account not only the calculated 40-year fatigue usage factors, but field experience and presence of conditions which tend to Increase susceptibility as well. The limiting locations are the feedwater nozzle and uncapped CRDRL nozzles.

Fatigue is a significant degradation mechanism for safe ends and therefore requires further evaluation.

Vessel shell

Thermal and mechanical fatigue cycling of the shell is minimal based upon conservative evaluations documented in vessel design stress reports. Therefore, fatigue crack initiation is not expected during the RPV service life. Thus, fatigue is not a significant ageing degradation mechanism for the vessel shell. The conclusion applies to all shell design variations.

Top head

Thermal and mechanical fatigue cycling of the top head (excluding nozzles) is minimal. Fatigue usage for the top head is typically 0.1 for 40 years. Therefore, fatigue crack initiation is not expected during the RPV service life. Thus, fatigue is not a significant ageing degradation mechanism for the top head of the vessel. This conclusion applies to all top head design variations.

Vessel flange

Based on field experience as documented in plant specific procedures, the physical wear of flanges has been insignificant. There have been required flange repairs due to damage caused by maintenance operations. However, these situations are not common and therefore, physical wear from normal maintenance is not a concern. In addition, the calculated fatigue usage factor is well below 1.0 even for 60 years of operation. Fatigue is not a significant ageing degradation mechanism for the vessel flange. This conclusion applies to all vessel flange design variations.

Vessel support skirt

The vessel support skirt fatigue has been evaluated in accordance with the ASME Code Section III. These ASME Code analyses show that fatigue usage factors vary widely from plant-to-plant. For high design usage factors, the fatigue analyses might have to be refined for long-term operation.

4.3. Stress Corrosion Cracking (SCC)

4.3.1. Description of mechanism

Stress corrosion cracking is the sub-critical crack growth of susceptible alloys under the influence of a corrosive environment. The three factors necessary for stress corrosion to occur are tensile stress, susceptible material, and a corrosive environment as shown in Fig. 4-1.

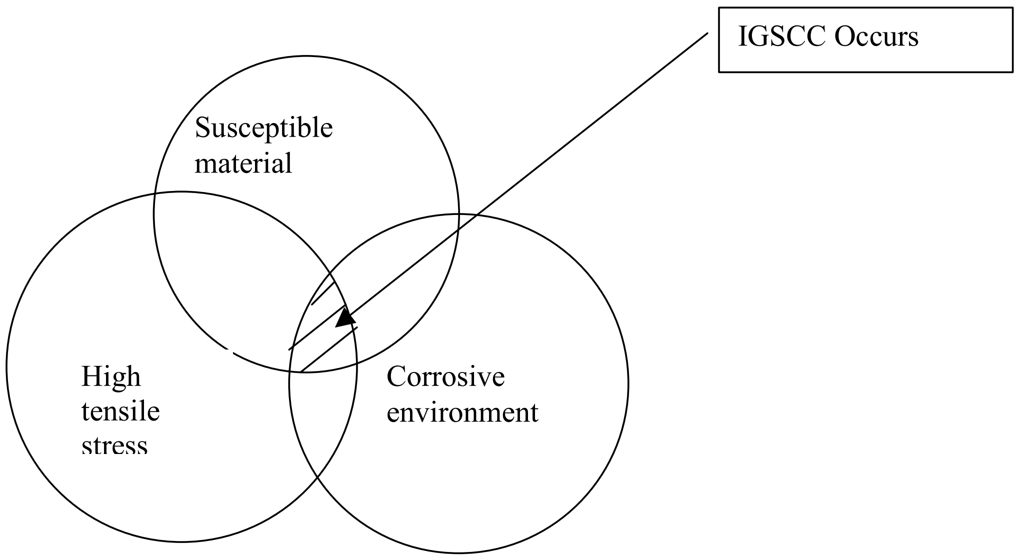

FIG. 4-1. Factors of Stress Corrosion Cracking.

Tensile stresses causing SCC are typically at material yield strength levels. Material susceptibility is related to the environment and may be influenced by the metallurgical condition of the material. For example, sensitized stainless steels are more susceptible to inter-granular stress corrosion cracking (IGSCC) in the BWR environment than fully annealed stainless steels. Exposure to high levels of neutron fluence can also cause stainless steels to become susceptible to SCC. This is a special form of SCC known as irradiation assisted stress corrosion cracking (IASCC) which has a potential to occur in BWR RPV stainless steel internal components. IASCC has not been observed in BWR pressure vessels.

*Influence of stress on S*CC

Applied stresses required to induce SCC must be tensile and of sufficient magnitude. Sources of stress include applied, residual, and thermal stresses. Numerous cases of SCC have been identified where there was no externally applied stress. For example, welded material contains self equilibrating residual stresses that may approach the material yield strength.

These stresses alone may be sufficient to induce SCC in sensitized material with an aggressive environment.

Stress corrosion cracks usually propagate perpendicular to the applied tensile stress. Cracks propagate with little or no attendant macroscopic plastic deformation and vary in degree of branching or formation of satellite cracks. Variation in crack morphology is a function of environment and microstructure as well as the type and orientation of the applied stresses.

In general, increasing the applied stress level decreases the time to crack initiation. The material may exhibit a threshold stress intensity factor (K_{ISCC}) below, which SCC does not occur. The fracture toughness threshold level is related to environment and may vary if environmental conditions change.

Influence of material on SCC

Susceptibility to SCC varies from alloy to alloy and with the metallurgical condition of the material under consideration. Type-304 stainless steel, for example, is susceptible to SCC when in the sensitized condition; that is, after exposure to high temperatures in the 500 to 833 $^{\circ}$C (900 to 1500 $^{\circ}$F) range for a required amount of time, which results in grain boundary carbide precipitation. SCC susceptibility of material in thermally sensitized stainless steel can be reduced by controlling carbide precipitation, by reducing carbon levels in the steel (e.g. Type 304L, 316L stainless steel) or by using stabilized grades (e.g. Type 347 stainless steel).

Both Alloy 82 and Alloy 182 are used in vessel attachment welds. Generic Letter (GL) 88-01 [4.10] indicates that Alloy 82 is resistant to stress corrosion cracking but that Alloy 182 is susceptible.

In stainless steel weld metal low ferrite will result in high susceptibility to SCC. After this was discovered ferrite controls were implemented but earlier RPVs may have weld metal such as cladding with a ferrite number of 0. Also it must be noted that exposure of the weld deposit to post weld heat treatment will reduce the ferrite content. Thus the ferrite controls require measurement after heat treatment.

Influence of environment on SCC

The two major variables that influence SCC aggressiveness of BWR water are conductivity and electro-chemical corrosion potential (ECP).

Conductivity measures the ionic nature of the water. In general, higher water conductivity values are associated with higher levels of an ionic species; these ionic species tend to promote SCC more aggressively. A clear correlation between plant water conductivity and cracking has been demonstrated for a number of components in operating BWRs.

ECP is a measure of the oxidizing nature of the environment with respect to the component of interest. It is controlled by the level of oxidizing species, such as oxygen. ECP may be controlled in a number of ways such as by injection of hydrogen or cathodic protection which are aimed at maintaining the ECP below the SCC threshold. Maximum benefits are attained if both water conductivity and ECP are controlled [4.11].

4.3.2. Significance

Most BWR RPV components are not subject to SCC because the combination of material sensitization, tensile stress level, and aggressive chemical environment does not simultaneously exist. Each component is assessed below.

RPV shell

In the absence of high stress fields the low alloy steel of the BWR vessel is resistant to SCC. This resistance is enhanced through application of post weld heat treatment (PWHT) following fabrication. PWHT improves material condition and reduces weld residual stresses. The PWHT of low alloy steel substantially reduces the residual stress of alloy steel weld metal and also tempers the heat affected zone thus eliminating what would be a brittle notch. However the temperatures used are not sufficient to lower the residual stress in stainless or Ni-Cr-Fe welds and that weld metal including the cladding will retain yield level residual stresses. Therefore, SCC could be significant at welded pad/bracket locations in the vessel shell and further evaluation for SCC is required. (See Attachment Welds).

Vessel cladding SCC, although not yet observed, cannot be totally dismissed. However, such cracking is of lesser concern. Even if SCC cracks initiate in cladding, they will generally not propagate into the alloy steel to any significant depth because steel is not susceptible. Furthermore the stresses in the cladding are generally compressive due to coefficient expansion. In addition, cladding has been removed near feedwater nozzles and SCC has not been observed even though this area is a highly stressed location [4.8].

Attachment welds

SCC initiation has never occurred in properly post weld heat treated pressure vessel steel. However, five isolated incidents (5 in thousands of inspections as documented in Section XI reports) of SCC in the RPV have been seen in the presence of high residual tensile stresses caused by inadequate PWHT or non-heat treated RPV attachment welds. Therefore, SCC could be significant at welded pad/bracket locations in the vessel shell.

Observations at two BWRs of SCC initiation in susceptible material adjacent to the RPV reinforces need for further evaluation. In this case the cracks extended into the low alloy steel (SA-508 Class 2) nozzle. Attachment weld residual stress was the main contributor to crack propagation into the nozzle material.

Cracking initiated in the Alloy 182 weld between the LAS nozzle and stainless steel recirculation inlet safe end. Axial cracking in the Alloy 182 weld butter was estimated by nondestructive examination (NDE) to extend 0.25 inch into the LAS nozzle in one instance, apparently due to SCC which initiated in the weld butter and propagated into the nozzle. Residual stress states decrease rapidly with distance from the weld. As residual weld stresses govern crack extension into the nozzle, crack propagation in properly post-weld heat treated vessel material is expected to be suppressed, especially as cracking extends into lower stressed regions or is suppressed altogether. Calculated stresses in the cracked nozzle at one BWR decreased from 448 MPa (65 ksi) to 207 MPa (30 ksi) in less than one inch [4.12]. In addition, circumferential cracking is also possible since the residual axial stress is also high. However, as the referenced calculations show, the axial stress is not as high as the hoop stress. Generic Letter 88-01 [4.10] addresses the issue of SCC in nozzle welds. Conformance to Generic Letter 88-01 allows one to assume that cracking is being effectively managed in nozzle welds.

The potential for low alloy steel cracking in the BWR was addressed in the 1978 to l984 time period. Recent events involving cracks in recirculation inlet nozzles made from low alloy steel (SA 508 Class 2) raised questions about the adequacy of closure of the low alloy steel cracking issue. The level of concern for stress corrosion cracking in PWHT vessel welds and butters that have not been subsequently welded to vessel internals remains low because SCC initiation and propagation into LAS is not likely with low stress levels. There is, however, a greater degree of concern about vessel attachment welds with high associated residual stresses and SCC susceptible weld material.

The following RPV attachment welds could have weld residual stress states high enough to promote cracking in adjacent LAS vessel material:

- Core Shroud Support Attachment Weld
- Core Shroud Support Leg Attachment Weld
- Core Spray Pipe Bracket Attachment Weld
- Feedwater Sparger Bracket Attachment Weld
- Guide Rod Bracket Attachment Weld
- Jet Pump Riser Bracket Attachment Weld
- LPCI Pipe Bracket Attachment Weld
- Steam Dyer Support Bracket Attachment Weld
- Steam Dryer Hold-down Bracket Attachment Weld
- Surveillance Specimen Capsule Holder (or Mounting) Bracket Attachment Weld

Although none of these attachment welds were subjected to PWHT, the wide range of stress states depends on the geometry and loading. Furthermore, likelihood of crack initiation in attachment welds or weld butters depends upon material susceptibility to SCC and upon environment as well. It is expected that some vessel attachment welds would have a low likelihood of cracking while others would have a greater chance of cracking and of experiencing subsequent crack propagation into the LAS vessel material. To determine if a given attachment is crack prone, one must check if susceptible material is present.

Therefore, stress corrosion cracking is a potentially significant degradation mechanism for these RPV attachment welds and requires further evaluation.

Bottom head

The bottom head, is made from low alloy steel (LAS) which is not expected to be susceptible to SCC. In addition, SCC has never initiated in LAS and thus is not considered a degradation mechanism in LAS.

Closure studs

During the long periods of normal reactor operations the RPV closure studs are protected by a dry atmosphere. But on occasion, such as during refueling outages, the studs can be exposed to moisture.

Two studs cracked at one BWR due to SCC. The cracking was caused by exposure of studs in the preloaded condition to oxygenated water. Normally, there is no concern for SCC in the studs but any studs that have been exposed to coolant while preloaded should be monitored.

Nozzles

SCC growth into nozzles has occurred in one nozzle at two plants. This cracking was due to growth of an IGSCC crack into the LAS material. Therefore, SCC of nozzles is a significant issue and requires further evaluation.

However, oxygen depleted water enhances SCC resistance; this and the low stress levels in the capped CRDRL nozzles combine to render SCC a non-significant degradation mechanism for the capped CRDRL nozzles.

Penetrations

SCC of CRD stub tubes that occurred at some plants was attributed to high residual stresses in sensitized weld material.

In 2002 a through wall crack caused by SCC took place at the welds which join the CRD stub tube and the bottom head in BWR/4 plant in Japan. This was attributed to high residual stresses in sensitized weld material (Alloy 182).

SCC of CRD stub studs as well as of other vessel penetrations is significant and requires further evaluation.

Safe ends

SCC of safe ends has been observed at several plants. Cracking occurred in many types of geometries and material conditions. These cracks can potentially propagate into adjacent nozzle material. Table 4-1 (page 69) shows that susceptibility of safe ends to SCC can be high in some cases. Therefore, SCC is a potentially significant degradation mechanism for safe ends and requires further evaluation.

Top head

The top head is made of low alloy steel. SCC of low alloy steel is not expected in the absence of attachment welds; thus, SCC is not a significant degradation mechanism for the top head. This conclusion applies to all top head design variations.

Vessel flange

IGSCC of the closure flange will not occur because it is dry and stress is low. SCC is not a significant degradation mechanism for the closure flange. This conclusion applies to all vessel flange design variations.

Vessel support skirt

IGSCC of the vessel support skirt will not occur because the environment is not aqueous. SCC is not a significant degradation mechanism for the vessel support skirt. This conclusion applies to all support skirt design variations.

4.4. Irradiation Assisted Stress Corrosion Cracking (IASCC)

4.4.1. Description of mechanism

IASCC is a form of SCC that can occur in components which accumulate high neutron fluence. This irradiation effect has been observed in BWR core internal components such as control blade sheaths and handles. Based on available field and laboratory data, a threshold fast neutron fluence (Energy> 1 Mev) of approximately 2 x 10^{25} n/m^2 appears to exist for IASCC of low stressed components made from stainless steel or Alloy 600 regardless of the state of material sensitization. For highly stressed components this threshold is 5 x 10^{24} n/m^2.

4.4.2. Significance

The expected fluence level at the vessel will not reach any fluence threshold for IASCC nor is the IASCC experienced by BWR stainless steel or Alloy 600 internals expected to occur in the low alloy vessel steel. Therefore, this mechanism is not significant for the pressure vessel.

4.5. General corrosion

4.5.1. Description of mechanism

Corrosion is the electrochemical reaction between a metal or alloy and its environment and is characterized by deterioration of the material or its properties. The chemical effects take two major forms. In the first form, environment influences mechanical processes, such as fatigue (as in corrosion fatigue) or SCC. In the second form, corrosion reduces the component wall thickness, either locally (crevice corrosion, pitting, etc.) or more uniformly (wastage).

Uniform corrosion frequently results in solid corrosion product (oxide) formations which, if adherent and nonporous, reduce the corrosion rate by protecting the underlying metal. Furthermore, if the oxide is self healing, localized oxide disruptions will rapidly oxidize again which effectively immunizes the metal to general corrosion. However, environmental changes, either on a general or local scale, can affect the stability of the protective oxide and render the material susceptible to attack.

4.5.2. Significance

The interior vessel surface is usually clad with an austenitic stainless steel or nickel based alloy that imparts good general corrosion resistance in the BWR environment. Even in unclad regions (e.g. feedwater nozzles), general corrosion rates in typical BWR environments are so low that general corrosion is not a significant factor in RPV service life. In tests in a simulated BWR environment, carbon steel was found to have a corrosion rate similar to that shown by two stainless steels (Types 304 and 316). At the observed rate, the metal loss would have been only 0.005 mm (0.2 mils) in 40 years [4.11]. Therefore, general corrosion is not a degradation mechanism of significance for BWR pressure vessel components.

4.6. Erosion corrosion

4.6.1. Description of mechanism

Erosion Corrosion is caused by flowing liquid impinging on component surfaces. Erosion can occur even if no abrasive particles are present in the flowing liquid. For example, vapor bubbles forming and collapsing at the liquid surface can interact with a solid surface, leading to cavitation erosion. If the surface is covered by a protective oxide, the erosion can locally disrupt the coating to expose fresh metal to the environment. This process leads to accelerated local degradation of the metal by the erosion corrosion mechanism.

Erosion by liquid impingement can also occur in components exposed to high velocity steam containing liquid droplets, in which case, material is progressively removed from the surface.

4.6.2. Significance

Coolant flow rates near the vessel walls is low (<1.5m/scc (5 ft/sec)). In the high flow regions of the vessel, erosion corrosion is not considered to be an issue. The oxygen levels at the steam outlet nozzle and the top head are sufficiently high to preclude erosion corrosion. The oxygen in the feedwater is carefully regulated to maintain a level sufficient to prevent erosion corrosion. Also, no evidence of erosion corrosion in the BWR vessel has ever been observed.

Erosion corrosion is not a significant degradation mechanism for any part of the BWR RPV.

4.7. Operating experience with the relevant ageing mechanisms

The following observations of BWR vessel and component service history are relevant to ageing degradation. These incidents offer a perspective on the design bases and their conservatism relative to operating parameters. It is particularly noteworthy that each has been resolved by qualified repair programs. Nozzle cracking, stub tube cracking, safe end cracking and closure stud cracking are all age related degradation mechanisms, which have been effectively managed.

Feedwater nozzle and control rod drive return line nozzles have experienced thermal fatigue cracking. Nuclear Regulatory Commission (NRC) findings and requirements with regard to this cracking are discussed in NUREG-0619 [4.8].

4.7.1. Feedwater Nozzle Cracking

GE conducted an extensive program to understand the feedwater nozzle cracking problem. The program report [4.13] concluded that cold feedwater leakage around the feedwater nozzle thermal sleeve must be avoided. Most GE and Japanese BWRs have installed new feedwater nozzle thermal sleeves and spargers.

NUREG-0619 [4.8] requires that surveillance be ongoing for all installations and clearly defines the frequency and extent of examinations which must be followed.

Potential feedwater nozzle cracking is a generic ageing degradation mechanism of current concern and is a plausible mechanism for evaluation with respect to life extension. Leakage must be minimized because it can soon lead to thermal fatigue cracking.

4.7.2. CRD Penetrations (Stub Tube Cracking)

4.7.2.1 Stainless steel stub tube cracking

Several GE BWRs have experienced CRD stub tube stress corrosion cracking (SCC) since start of operation. All stub tubes in these three BWRs were fabricated from Type-304 furnace-sensitized stainless steel. Furnace-sensitization of Type 304 stainless steel was eliminated from later BWR vessels.

BWR/1 CRD housings have a close fit to the stub tube inside diameter (ID). Leakage (bottom head to stub tube weld defect) was stopped by rolling (expanding) the housing into the bottom head penetration.

CRD housings in the BWR/2 design have wider gaps between the stub tube and housing than do those in the BWR/1 plants. Leakage occurred in the stub tube just below the stub tube-to-housing weld. A total of 33 housings in a BWR/2 have been ID roll expanded into the bottom head penetration to stop leakage; some of these housings were later roll expanded again.

The BWR/3 design has a relatively large gap between the housing and the stub tube ID. In the one BWR/3 plant that experienced CRD stub tube problems, three cracked stub tubes leaked; these leakages were stopped by installing sleeves on the stub tubes. Several stub tubes in this plant were later found (through ultrasonic testing) to be cracked. Although none of these stub tubes leaked, sleeves were installed over three of the largest cracks as a preventive measure.

CRD Stub tube repair programs

Several repair options stopped CRD penetration leakage. Roll expansion of the CRD housing into the bottom head penetration has been particularly successful when the gap between the housing and vessel penetration is small. Success also seems to be related to early detection and repair, which minimizes damage to the surfaces to be rolled. Mechanical sleeves are effective. At least two utilities are sponsoring and developing repair welding options. Access is a major problem for both welding and machining operations because the penetrations are closely spaced and difficult to reach. However, based on the previous ability to install mechanical sleeves, it appears likely that there is sufficient access for making these repairs.

Rolled housings are not considered to be a permanent solution because the process causes cold-working of the stub tube. Although the amount of cold working is within permitted allowables, SCC cannot be ruled out in the long term. There could also be leakage after several thermal cycles, long term solutions, such as mechanical seals or housing and stub tube replacement, are under development. Activities are currently underway within ASME Section XI to develop a Code Case to permit the roll repair method as an alternative repair technique for long term operation.

Consequences of CRD stub tube cracking

Based on limited leakage area, safety evaluations have concluded that stub tube cracking does not limit vessel life or pose a safety concern. However, when cracking occurs, plant availability is impacted and examination and repair costs are significant.

All BWRs are equipped with features that prevent the ejection of a housing even in the event of total weld failure. This both restricts leakage and avoids withdrawal of the associated control blade. Internal pumps have equivalent features.

4.7.2.2 Alloy 182 Welds SCC

In 2002 through wall crack caused by SCC took place at the welds, which join the CRD stub tube and the bottom head in BWR/4 plant in Japan. This was attributed to high residual stresses in sensitized weld material (Alloy 182). The stub tube, which had the crack was replaced with new one. Alloy 82 was utilized as weld material. [4.14] The other 88 stub tubes of the plant were visually inspected with an underwater camera and no crack was found. METI required all Japanese BWR utilities to perform inspections of CRD stub tubes at the plants, which have similar stub tube design (material) and no prior inspection experience.

4.7.3. Safe End Cracking

GE BWR reactor pressure vessel safe ends are fabricated from low alloy steel, carbon steel, stainless steel, or Ni-Cr-Fe alloy. IGSCC has been observed in many stainless steel and Alloy 600 safe ends.

To date, there has been no IGSCC of low alloy steel safe ends; however, there have been instances of crack propagation into low alloy steel nozzle material from adjacent susceptible locations. EPRI Report NP-4443 [4.15] provides a comprehensive listing of safe end materials and safe-end-to-nozzle welding materials for many operating plants. Also, considerable data are presented in this document for recirculation inlet safe ends, recirculation outlet nozzle safe ends, core spray nozzle safe ends and jet pump instrumentation safe ends.

IGSCC in stainless steel was caused by sensitization from welding or furnace post weld heat treatment. Cracking was observed in creviced low carbon stainless steel and Alloy 600 safe ends having creviced geometry. Though not directly associated with the safe ends, there has been IGSCC cracking in Alloy 182 Ni-Cr-Fe nozzle butters, which were used by several vessel fabricators.

Fatigue cracking by thermal stratification has been observed in CRDRL safe ends. This is reduced by flow increase, which might prevent thermal stratification and fatigue. The same mechanism might apply to feedwater nozzle safe ends.

4.7.3.1 IGSCC in sensitized material

In 1975, GE began an ongoing research program to identify the cause of IGSCC and develop corrective measures. It has been found that IGSCC could occur when the following three conditions exist simultaneously: (1) a sensitized microstructure, (2) the sensitized material must be in contact with a corrosive environment (oxygenated BWR water), and (3) the material must be subjected to a tensile stress, which could include residual stresses from welding or other sources.

This led to a search for a material that would not become sensitized by welding. The material of choice (though not the only material identified) is Nuclear Grade Type-316 austenitic stainless steel, which has a carbon content limited to 0.02% (wt) maximum. It should be noted that the Nuclear Grade material has nitrogen added to compensate for the loss of strength caused by the low carbon content.

4.7.3.2 IGSCC in cold worked or creviced safe ends

Field experience during the past few years has shown that IGSCC can occur in stainless steel (including nuclear grade 316) if the steel is used in a cold worked or creviced condition. Field experience also shows that Alloy 600 safe ends will suffer IGSCC if the design is creviced. The existence of cold work is not normally documented. It has been found to occur when base metal was ground near a weld or a weld repair. Several safe end-to-thermal sleeve designs used on GE vessels are creviced. These creviced safe ends were replaced in most BWR5 with noncreviced safe end designs to reduce SCC susceptibility.

4.7.3.3 IGSCC in Alloy 182 Nozzle Butter

On many GE RPVs, safe ends were replaced before plant startup to remove furnace sensitized material. The safe ends were replaced with non-sensitized stainless steel or Alloy 600. At some plants, the nozzle butter was removed and replaced with an Alloy 182 Ni-Cr-Fe butter. However, field experience has shown that Alloy 182 Ni-Cr-Fe nozzle butter also is susceptible to IGSCC cracking. Alloy 82 is resistant to IGSCC.

4.7.3.4 Consequences of safe end cracking

Cracking of nozzle safe ends has been detected and controlled before the pressure boundary has been compromised. In service inspection procedures has been effective, remedies have included repair and/or replacement of the affected materials. These remedies, while not inexpensive, indicate that IGSCC of nozzle safe ends can be managed. Nevertheless, this ageing degradation mechanism is an important generic consideration.

4.7.4. Closure stud cracking

Two RPV closure studs cracked at one BWR plant. The closure stud steel material is susceptible to IGSCC if moisture is present. However, except for brief periods such as upon removal of the top head for refueling the studs are kept dry. The cracking was caused by exposure of the studs in the preloaded condition to oxygenated water during refueling outages.

4.8. Conclusion of significance of ageing mechanisms

TABLE 4-2 summarizes the conclusions reached in the Section 4 discussions. Degradation of all RPV found to be potentially significant are identified with an '**PS**' in this table. Degradation mechanisms, which are not significant for a particular component, are identified by an asterisk (*). Degradation mechanisms, which are not significant for any RPV components are identified by a dash (-).

TABLE 4-1 NOZZLE/SAFE ENDS AGEING MECHANISM POTENTIAL

RPV Component	Nozzle			Safe Ends		
	Radiation Embrit.	Fatigue	IGSCC	Radiation Embrit.	Fatigue	IGSCC
Feedwater nozzle triple thermal sleeve	-	PS	-	-	PS	-
Feedwater nozzle triple thermal sleeve w/stainless steel inlay	-	PS	-	-	PS	-
Feedwater nozzle welded single sleeve	-	PS	-	-	PS	-
Feedwater nozzle single sleeve s/flow baffle (BWR/2)	-	PS	-	-	PS	-
Recirculation inlet nozzle w/straight sleeve	-	-	-	-	-	PS[a]
Recirculation outlet nozzle	-	-	-	-	-	PS[a]
Recirculation inlet nozzle w/ shaped sleeve	-	-	-	-	-	PS[a]
Core spray nozzle	-	-	-	-	-	PS[a]
Jet pump sensing line nozzle	-	-	-	-	-	PS[a]
CRD nozzle return capped	-	-	-	-	-	PS[a]
Main steam nozzle	-	-	-	-	-	-
LPCI nozzle	PS[X]	-	-	-	-	PS[a]
Isolation condenser return line nozzle	-	-	-	-	-	-

-: Not Significant
PS: Potentially Significant
a: Applied to Safe end with SCC susceptible material
x: Applied to some BWR/5 LPCI nozzels

TABLE 4-2 AGE RELATED DEGRADATION MECHANISM ASSESSMENT SUMMARY

		Relevant Age Related Degradation Mechanism					
RPV Components		Neutron Embrit.	Fatigue	IGSCC	IASCC	General Corrosion	Erosion Corrosion
Attachment Welds		-	-	PS[a]	-	-	-
Bottom Head		-	-	-	-	-	-
Closure Studs		-	PS	PS[x]	-	-	-
Nozzles	BWR/5 LPCI	PS	-	PS[a]	-	-	-
Nozzles	Feedwater	-	PS	PS[a]	-	-	-
Nozzles	BWR/2 CRDRL	-	PS	PS[a]	-	-	-
Nozzles	All Others	-	-	PS[a]	-	-	-
Penetrations	CRD Stub Tube	-	PS	PS[a]	-	-	-
Penetrations	All Others	-	-	PS[a]	-	-	-
Safe Ends		-	PS	PS[a]	-	-	-
Vessel Flange		-	-	-	-	-	-
Vessel Shell	Beltline	PS	-	-	-	-	-
Vessel Shell	Weldments	PS[xx]	-	-	-	-	-
Vessel Shell	All Others	-	-	-	-	-	-
Vessel Support Skirt		-	-	-	-	-	-
Top Head		-	-	-	-	-	-

-: These degradation mechanisms or component/degradation mechanism combinations are not significant.
PS: These combinations are potentially significant and require further evaluation.
a: Applied to those with SCC susceptible material.
x: Applied to those exposed to coolant while preloaded
xx: Applied to those in beltline.

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5. INSPECTION, MONITORING AND MAINTENANCE

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 [5.1]. 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. More importantly, Programme A schedules 8 percent of the fourth inspection interval to be completed in year 27; an additional 17 percent by year 30; and additional 25 percent increments by years 33, 37 and 40. Program B maintains the uniform schedule 16 percent by year 33, an additional 34 percent by year 37, and the remainder by year 40. All BWR plants follow Program B.

Extensive manufacturing and pre-service inspections are required for reactor pressure vessels. In addition, ASME Code Section XI requires four inspection intervals during the first forty years of plant life. The first inspection occurs three to ten calendar years after initial plant startup. As a result of ASME Section XI, 1988 Addenda, the following inspections are required at every interval:

> Volumetric inspection of all pressure-retaining welds (i.e. shell, head, head-to-flange, shell-to-flange, and repair welds) in accordance with Examination Category B-A of IWB-2500-1; volumetric of all full-penetration nozzle welds (i.e. nozzle-to vessel welds) in accordance with Examination Category B-D of IWB-2500-1. However, activities are underway to introduce a Code Case to establish a minimum inspection of 25% of nozzle inner radius and nozzle-to-shell welds, including at least one nozzle from each system and nominal pipe size. The technical basis for this Code Case is contained in Reference [5.2.]. This case will require that the provisions of Appendix VIII from the 1989 Addenda or later editions and addenda be used for examinations. This case excludes BWR feedwater nozzles and control rod drive return line nozzles;
>
> Visual (VT-2) inspection of the external surface of 25% of the pressure-retaining partial penetration welds in the vessel (i.e. control rod drive mechanism housing and instrumentation tubes) in accordance with Examination Category B-E of

IWB-2500-1, such examinations are performed during conduct of system hydrostatic tests, with relevant conditions defined in IWB-3522;

Volumetric and surface inspection of all pressure-retaining dissimilar metal welds in vessel nozzles (i.e. nozzle-to-safe end butt welds) in accordance with Examination Category B-F of IWB-2500-1.

Volumetric and surface inspection of all pressure-retaining bolting greater than two inches in diameter (i.e. closure studs) in accordance with Examination Category B-G-1 of IWB-2500-1.

The Boiling Water Reactor Vessel Internals Program (BWRVIP) developed a technical basis for elimination of the ASME Code section XI requirement to perform examination of the RPV circumferential welds. The US NRC issued a safety evaluation report [5.3] and Generic Letter [5.4] informing BWR licensees that they may request relief from the in-service inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RPV welds.

Additional inspections, not covered by ASME, but recommended by the BWRVIP, are shown in Table 5-1.

The Performance Demonstration Initiative (PDI) is a utility consortium that was formed in 1991 to aid utilities' implementation of the qualification requirements found in the ASME Code, Section XI, Appendix VIII [5.5]. The PDI program implementation is overseen by a steering committee comprised of 15 members representing 15 different utilities in the USA. All utilities owning nuclear plants in the USA are participating in the PDI. The PDI Performance Demonstration Guidelines are the basic operating document and include the rules for organizing and operating the performance demonstration program. This protocol document provides detailed guidance in the conduct of the actual demonstrations. The EPRI NDE Center, acting as the Performance Demonstration Administrator (PDA), is responsible for the development of detailed procedures to fulfill these requirements. In this capacity, the NDE Center is responsible for sample design and procurement, sample documentation, administering demonstrations, QA activities and audit support, and maintaining qualification records. The documentation necessary to show compliance with the requirements of Appendix VIII includes the list of qualified essential variables for each procedure-vendor combination, sample set descriptions, and the results of the qualification demonstrations.

Performance demonstrations using realistic piping and RPV samples were initiated in 1994. The current PDI programme encompasses performance demonstrations for: Supplement 2 (austenitic piping), Supplement 3 (ferritic piping), Supplement 4 (RPV shell welds, clad-to-base-metal interface), and Supplement 6 (RPV shell welds). An array of ultrasonic performance data, including detection, length sizing and depth sizing, has been generated as a result of the PDI programme. Since the demonstration phase for piping and RPV shell welds began in 1994, more than 500 candidates and 20 organizations have participated in the programme. There have been in excess of 5000 sizing measurements and 14,000 for piping flaw detection and length sizing. The PDI provides evidence that current RPV examinations are highly effective in assuring that the clad-to-base-metal region is free of unacceptable flaws [5.6]. 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 have not been inspected since beginning of their operation. However, all BWR vessels were inspected prior to being declared fit for service. That fabrication inspection is the basis for current operation in many of these older plants.

Many newer BWR vessels have been inspected since being put into service. Inspection results from these newer plants form the basis for confidence that the level of safety in vessels, which have not had in-service inspections to date is acceptable. It is expected that more BWR vessels will be inspected as advances in inspection technology are made.

Most older BWR plants have been granted exemptions by the US NRC from the ASME Section XI requirement to inspect beltline welds due to inaccessibility and because BWR vessel fracture toughness margins are high. The reasons for these exemptions constitute valid bases for concluding that BWR vessels today maintain an acceptable level of safety. Plant specific exemptions and relief will need to be reviewed by the applicant for explicit time dependencies and applicants will have to justify continuation of such exemptions and relief.

Section XI acceptance standards for Class 1 components are contained in IWB-3000, with specific evaluation requirements in IWB-3110 and IWB-3120 for Pre-service Examinations and in IWB-3130 and IWB-3l40 for In-service Examinations. For both types of examinations, components for which examinations indicate existence of flaw indications which exceed the Table IWB-3410-1 Acceptance Standards are acceptable for service provided certain administrative procedures are followed.

When evaluating flaws that exceed the Section XI flaw standards, IWB-3600 of Section XI requires the use of the original safety margins for all operating conditions. These are normally 3 for normal and upset conditions and 1.5 for emergency and faulted conditions. They may vary for specific cases, but are always conservative with respect to the design margins.

The relationship between ASME Section III and Section XI is that Section III is a design code and has specific design margins while Section XI is an inspection code and does not perform a redesign function. Section XI requires that the safety margin of the flawed structure be at least as great as the original design margin for both ductile and non ductile failure modes.

Linear elastic fracture mechanics (LEFM) methodologies are used to perform a cumulative fatigue crack growth study to determine the potential for crack growth.

TABLE 5-1 ADDITIONAL EXAMINATION GUIDANCE RECOMMENDED BY THE BWRVIP

Component	Subcomponents	Examination Guidance
Core Plate P/Standby Liquid Control Penetration [5.7]	Nozzle-to-safe end welds made of stainless steel and cold-worked safe end extensions	UT of nozzle-to-safe end weld every 10 years using qualified examination Alternate examinations: 1. Enhanced leakage inspection during each Category B-P pressure boundary leak test by means of direct visual examination with insulation removed 2. Surface examination every other refueling outage using Section XI acceptance criteria
Lower Plenum[5.8]	CRD stub tube and/or housing-to-vessel	No inspections required

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 In-service Inspection Guidelines [5.9] 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 [5.10], which today specifies the NDE requirements for ISI.

The NDE methods to be applied are specified in KTA 3201.4 as shown in Table 5-2.

TABLE 5-2 KINDS, METHODS, AND TECHNIQUES OF INSPECTION IN GERMANY

Serial Number	Type of Test	Test Procedure	Test Technique
1	Examination with regard to cracks in the surface or in near-surface regions	Magnetic particle flaw detection	Magnetic particle examination (MT), magnaflux examination
		Liquid penetrant examination (LT)	e.g. dye penetrant examination
		Ultrasonic examination procedure (US)	e.g. surface waves, mode conversion, dual search units with longitudinal waves, electromagnetic ultrasonic waves
		Eddy-current examination procedure (ET)	Single frequency, multiple frequency
		Radiographic examination procedure (RT)	X-ray Radioisotope
		Selective visual examination (SV)	With or without optical means
2	Volumetric examination	Ultrasonic examination procedure (US)	e.g. single probe technique with straight (ES) or angle beam scanning, tandem (angled pitch-catch) technique, mode conversion
		Radiographic examination procedure (RT)	X-ray Radioisotope
		Eddy-current examination procedure (ET) for thin walls	Single frequency Multiple frequency
3	Integral examination	Integral visual examination	—
		Pressure test	—
		Functional test	—

The inspection scope and the NDE-methods to be applied to a RPV are listed in Tables 5-3 and 5-4. As can be seen, the ISI includes all welds, the nozzle radii, the control rod ligaments in the BWRvessel, the bolts, nuts, washers and threaded blind holes. The inspection intervals for the RPV are 5 years; however, the scope of an inspection may be subdivided and each part carried out separately during the 5-year period, e.g. each year at the refuelling outage for BWRs.

TABLE 5-3 INSPECTION REQUIREMENTS IN GERMANY – SCOPE AND INTERVALS FOR RPVS

Item to be inspected	Test procedure / Test technique	Flaw orientation	Extent of testing	Test intervals
Longitudinal and circumferential welds	US	l, q	all weld seams, entire length, entire volume as well as the surface areas with their near-surface regions	5 years
Nozzle-to-shell welds ≥ DN 250 [1]	US			
Nozzle inside edge ≥ DN 250 [1]	US	r	surface areas with their near-surface regions of the entire inside edge of all nozzles	
Ligaments in nozzle fields	US		all ligaments, surface areas and centres of ligaments	
Cladding	SV	any	representative locations, the test extent shall be specified for the individual plant	
Screw bolts	US or MT or ET, SV	q, relative to the bolt axis	surface areas with their near-surface regions of all bolts, entire tensioned length including the threaded regions	Within 5 years [2] at least 25 % of the bolts with the corresponding threaded blind holes, nuts and washers, however, at three successive test intervals of 5 years 100 % shall be tested. Alternatively, the test may be performed at intervals of 10 years [2] where 100 % each shall be tested
Threaded blind holes	US or ET, SV	q, relative to the thread axis	surface areas with their near-surface regions of all blind holes, entire thread length	
Nuts	SV or ET or US	q, relative to the thread axis	threaded region and loaded end face (contact surface) of all nuts	
Washers	SV	any	both contact surfaces as well as the surface of the washer hole	
Attachment welds	Agreements shall be made because of the differing design details. The type and extent of the tests shall be incorporated in the test instructions.			
Auxiliary welds	MT or US	The requirements shall be specified in accordance with 5.2.1.1 (3).		

Abbreviations for the test procedures and techniques are explained in **Table 2-1**. (table 5-2)
l : longitudinal flaw q : transverse flaw r : radial flaw (e.g. for nozzle inside edges or ligaments in nozzle fields)

[1] In the case of nominal diameters of the connecting pipe < DN 250 the requirement for in-service inspections shall be reviewed from case to case.
[2] Selective visual inspection of stud bolts (where accessible), nuts and washers after each unbolting of bolted joints

TABLE 5-4 INSPECTION REQUIREMENTS IN GERMANY — SCOPE AND INTERVALS FOR CONTROL ROD DRIVES' PRESSURE RETAINING PARTS

Item to be inspected	Test procedure / Test technique	Flaw orientation	Extent of testing	Test intervals
Circumferential welds PWR [1]	ET or RT or US	l	Inner surface of representative welds on 10 % of pipes in due consideration of accessibility	10 years
Circumferential welds BWR	LT or ET or US	l	Inner surface of circumferential welds of 4 rod drive housing pipes	

Abbreviations for the test procedures and test techniques are explained in **Table 2-1**. l : longitudinal flaw (table 5-2)

[1] These welds also include in-core instrumentation and control rod nozzle welds

The inspection technique mainly used is UT. 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. For a wall thickness of equal or more than 100 mm, additional inspection with techniques for detecting planar flaws perpendicular to the surface shall be performed.

The German Nuclear Safety Standard KTA 3201.4 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 an 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, the data for evidence of the kind, position and size of the flaw have to be analyzed. 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 Japan

The basic inspection requirements are given in the JEAC-4205 [5.11], the Japan Electric Association Code for ISI of light water cooled nuclear power plant components and also in JSME Code on Fitness-for-Service for Nuclear Power Plants [5.12], JSME S NA1-2002. Requirements in them are the same. Examination Categories B-A to B-D, B-F to B-H, B-J, B-N (in JSME Code JP-1), B-O and B-P (Section 2, Class 1 Components) prescribe the methods, inspection area and frequencies for the RPV in-service inspection. The basic examination required is a periodic volumetric examination of the reactor pressure vessel weld lines. The following volumetric examination is required at every inspection interval (10 years): 7.5 % of each core belt region weld which receives neutron fluence of 10^{23} n/m^2 (E>1 MeV) or less and each pressure-retaining weld of the shell and heads other than core belt region weld (i.e. shell, head and repair welds) in accordance with Examination Category B-A and B-B; all core belt region welds which receive neutron fluence larger than 10^{23} n/m^2 (E>1 MeV), all head-to-flange welds, all shell-to-flange-welds, all full-penetration nozzle welds (i.e. nozzle-to vessel welds) and all pressure-retaining dissimilar metal welds (i.e. nozzle-to-safe end welds) whose diameter is 100 mm and over in accordance with Examination Category B-A, B-C, B-D and B-F. From the 4th inspection (after 30 years of operation), the inspection interval for Class 1 components becomes 7 years.

5.1.4. Regulatory requirements in Finland

ISI requirements are specified in Regulatory Guides given by Radiation and Safety Authority STUK. Guides follow ASME XI requirements. Additional inspections are performed if technical reasons appear or service experiences indicate some reason to increased frequency of inspection.

A new Regulatory Guide concerning ISI will be published in near future. This Guide provides the qualification of NDT according to principles of European working group (European Network for Inspection Qualification, ENIQ) documents.

5.1.5. Requirements in India

Atomic Energy Regulatory Board’s (AERB) Code on Safety in Nuclear Power Plant Operation mandates implementation of an ISI programme for all items important to safety. The extent of ISI requirements shall be appropriately related to the importance to safety of the items to be examined. AERB’s Safety Guide on ISI of NPPs gives further guidance. The detailed ISI manual prepared for a specific plant is reviewed by AERB. This specifies the areas to be examined, examination methods, inspection interval and areas exempted from examination. For BWRs, the requirements of ASME Section XI are followed for establishing the ISI programme.

For the two operating BWR units in the country, based on national and international experience, the requirements of ASME Section XI and limitations specific to these units, the inspection requirements for RPV components have been specified in the stations ISI Manual.

5.2. NDE techniques

5.2.1. Ultrasonic examination methods

Smooth, sharp-edged flaws oriented in a plane normal to the vessel surface located the cladding/base-metal interface are the most critical type of flaws since they may occur in highly stressed area. 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 (DAC) curve, which can be obtained from an ASME reference block with one 3-mm (0.125-in.) side-drilled hole [5.13]. 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 [5.14, 5.15].

The amplitude-based technique uses the decibel-drop method to determine *flaw sizes* much larger than the width of the sound field [5.13, 5.16]. 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 [5.17].

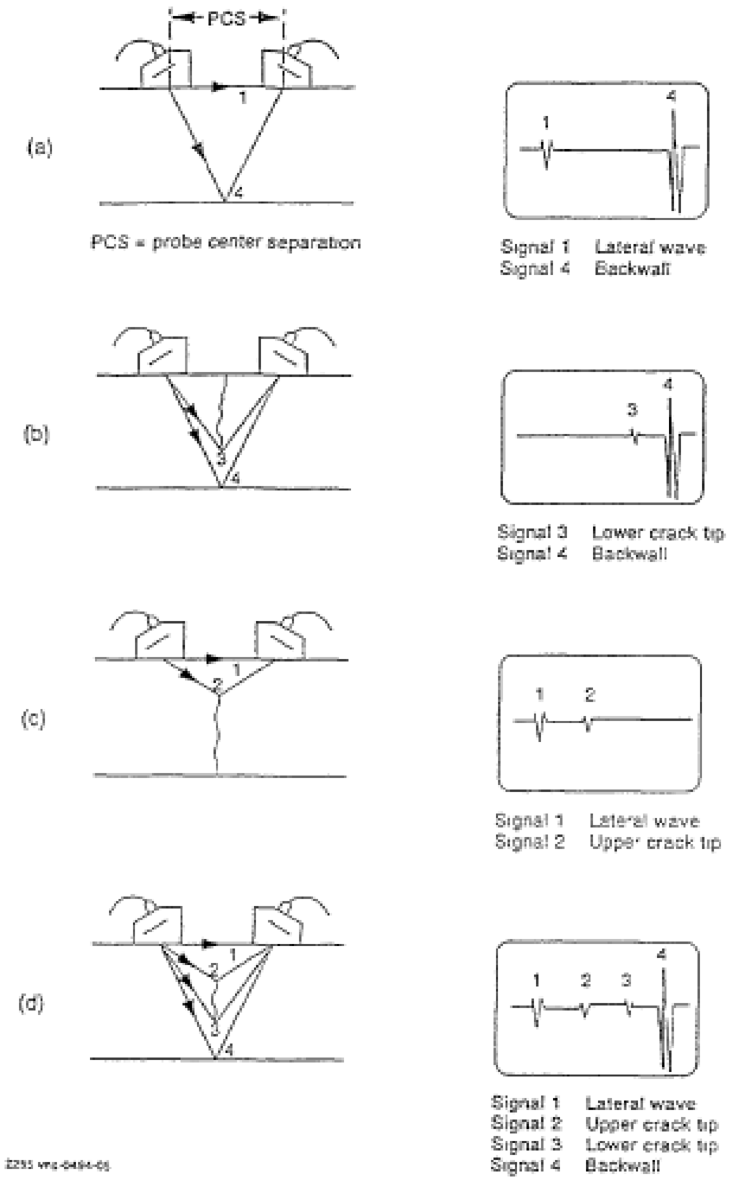


FIG. 5-1 Examples of time-of-flight diffraction (TOFD) signals (Pers-Anderson 1993) Copyright TRC, reprinted with permission.

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 (TOFD) technique, the difference in the travel times of ultrasonic waves diffracted from each of the flaw tips is measured to estimate the flaw size [5.18]. Examples of time-of-flight diffraction are depicted in Fig. 5-1 [5.19]. The technique consists of a separate transmitter and receiver oriented in opposite directions, as shown in Fig. 5-1 (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 of "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 5-1 (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. 5-1 (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. 5-1 (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 [5.16, 5.20]. 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 [5.16]. 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 [5.21].

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 [5.22]. 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 [5.23].

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. 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 vessels than currently used distributions such as the Marshall distribution [5.24].

5.2.2. Acoustic emission monitoring

Acoustic emission methods may be used to monitor potential flaw growth in welds and base metal if the outside surface of the vessel is accessible. Some BWR 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 [5.25]. Acoustic emission was also used to monitor possible crack growth during the 1987 hydro test of the High Flux Isotope Reactor (HFIR) located at the Oak Ridge National Laboratory; no evidence of crack growth was detected [5.26].

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 in-service monitoring of crack growth in thick wall, geometrically complicated components such as RPV nozzles [5.27]. 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 [5.28]. 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-3132.3.

5.2.3. Visual inspection

Visual inspection requirements are specified in ASME Section XI, IWB-2500. Table IWB-2500-1 specifies the components for which visual examinations are permitted. In some cases, these inspection requirements have been supplemented in the U.S. by BWRVIP guidelines. These supplemental inspections are necessary for detection of IGSCC.

Visual inspection methods contained in the ASME Code are VT-1, VT-2 and VT-3. For detection of IGSCC, the BWRVIP recommends the use of an enhanced visual technique. Definitions of these visual methods are discussed below.

- ASME VT-1: a visual inspection method capable of achieving 1/32 inch (0.79 mm) resolution. VT-1 is conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion.

- ASME VT-2: a visual inspection method capable of detecting evidence of leakage from a pressure retaining component, with or without leakage collection systems, as required by the system pressure test.

- ASME VT-3: a visual inspection method for assessing the general mechanical and structural condition of components and their supports. Parameters such as clearances, settings, and physical displacements must be verified to detect discontinuities and imperfections such as loss of integrity at bolted or welded connections, loose or missing parts, corrosion, wear, or erosion.

- BWRVIP EVT-1: a visual inspection method capable of achieving ½ mil wire resolution. This technique is necessary for detection of IGSCC.

In many cases, cleaning of surfaces using brushes or hydrolazing is required for detection and sizing of IGSCC.

Note that for plants in the USA, substitution of a more sensitive inspection technique (e.g. EVT-1 in lieu of a VT-1) does not satisfy a plant's ASME Section XI program inspection requirements unless concurrence for the substitution is approved by the plant's Authorized Nuclear In-service Inspector in accordance with a plant's specific ASME Section XI Program. However, substitution of a more sensitive inspection technique (e.g. UT in lieu of EVT-1) for a BWRVIP recommended inspection is permitted. If UT is used, the technique shall be qualified in accordance with BWRVIP-03 [5.29] requirements.

5.2.4. Eddy current inspection

In-service inspection manipulators, which are used in Finnish and in most Swedish BWR's contain both ultrasonic transducers and eddy current transducers. Eddy current inspection is done simultaneously with ultrasonic inspection.

Eddy current testing is suitable to find and size indications up to a depth of about 3 mm from the inner surface. The sizing of surface indications in longitudinal direction is accurate. However the sizing in depth direction is not possible.

5.3. RPV material surveillance programmes

5.3.1. Requirements in the USA

Every BWR pressure vessel operating in the western world has an ongoing RPV material radiation surveillance programme. To date, a large number of surveillance capsules have been removed from their host RPV and tested. The results of these specimen tests have been used to confirm the design predictions. In a BWR, the only concern relative to radiation embrittlement is the hydrostatic test temperature.

Fracture toughness requirements

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 [5.30]. 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.
- 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 (KI) shall be lower than the reference stress intensity factors (KIR) 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 KIM can be reduced from 1.5 to 1.0.

10 CFR Part 50 Appendix H, 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 programmes
[5.31]. 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 El85 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.

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

US NRC Requirements and the BWRVIP Integrated Surveillance Program (ISP)

The US NRC has established specific criteria in 10CFR50 Appendix H for an integrated surveillance program. The requirements for an integrated surveillance program, as specified in 10CFR50 Appendix H, are as follows [5.31]:

(1) In an integrated surveillance program, the representative materials chosen for surveillance for a reactor are irradiated in one or more reactors that have similar design and operating features. The Director, Office of Nuclear Reactor Regulation, on a case-by-case basis, must approve integrated surveillance programmes. Criteria for approval include the following:

    (a) The reactor in which the materials will be irradiated and the reactor for which the materials are being irradiated must have sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage.

    (b) Each reactor must have an adequate dosimetry program.

(c) There must be adequate arrangement for data sharing between plants.

(d) There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

(e) There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

(2) No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted.

(3) No reduction in the amount of testing is permitted unless previously authorized by the Director, Office of Nuclear Reactor Regulation.

Each plant in the U.S. BWR fleet has an existing vessel surveillance programme that consists of a set of surveillance capsules that were installed when the plant was licensed. The surveillance capsules typically include specimens for plate, weld, and heat affected zone (HAZ) materials. The test results from the specimens are used for monitoring radiation embrittlement of the beltline materials for that plant. However, many plants do not have a surveillance material that represents the limiting plate and/or weld material of the plant vessel. Instead of using the plant-specific surveillance data from a given plant, the data from across the fleet could be used. Material data from another plant surveillance programme or other source could be used to better represent the limiting material for the target plant. Integrating the existing surveillance programmes together is called the Integrated Surveillance Programme (ISP).

Because U. S. BWRs were licensed over a period of years, the requirements and content of the individual surveillance programmes vary. For example, as a result of changes to industry standards and NRC regulatory guidance, some plants do not have surveillance specimens for the limiting RPV plate or weld material. In 1998, the EPRI managed BWR Vessel and Internals Project (BWRVIP) and developed an Integrated Surveillance Programme (ISP) using similar heats of materials in the surveillance programmes of BWRs to represent the limiting materials in other vessels and improve the monitoring of embrittlement in BWR vessels. The ISP combines all the separate U. S. BWR surveillance programmes into a single integrated programme and adds data from a supplemental surveillance programme. The ISP has been designed to meet the criteria for an integrated surveillance programme in 10CFR50 Appendix H. The BWRVIP submitted the BWRVIP-78 report [5.32] in 1999 to describe the technical basis of the ISP related to material selection and the testing matrix outlining the ISP plan and documenting the design of the test matrix. The BWRVIP-86-A report [5.33] was submitted in 2000 and revised in 2002 addresses the implementation plan for the ISP, testing schedule and test matrix.

A test matrix was developed to identify those specimens that best meet the needs of each BWR. The materials for the ISP were specifically chosen to best represent the limiting plate and weld materials for each plant using specimens from the entire BWR fleet. Specimens that provide little or no added value are not included and need not be tested because other materials in the integrated programme provide better quality and more representative data. The

BWRVIP ISP is intended to substitute the existing plant surveillance capsule programmes with representative weld and base materials data from host reactors. A representative material is a plate or weld material that is selected from among all the existing plant surveillance programmes or the SSP to represent the corresponding limiting plate or weld material in a plant. Under the ISP, representative capsule data will be provided to each BWR vessel owner for limiting vessel weld and base materials. These data should be evaluated using the methods in Regulatory Guide 1.99, Revision 2, in accordance with 10CFR50, Appendix G, for determination of Adjusted Reference Temperature (ART) values.

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 [5.34 –5.36] 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 belttime 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:

$$\Delta RT_{NDT} = (CF)\ f^{(0.28\text{-}0.10 \log f)} \tag{23}$$

where CF is a chemistry 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, $f^{(0.28\text{-}0.10 \log f)}$ is determined by calculation or from a figure presented in the regulatory guide.

Regulatory Guide 1.99 Revision 0 and 1 [5.34, 5.35] 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

Reg. Guide 1.190 provides state-of-the-art calculations and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence. This guide is intended to ensure the accuracy and reliability of the fluence determination required by General Design Criteria (GDC) 14, 30, and 31 of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50. The guide describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence. These methods are directly applicable to the determination of RT_{NDT} and RT_{PTS}.

5.3.2. Requirements in Germany

According to the stipulations of the German Nuclear Safety Standard KTA 3203 [5.37], radiation embrittlement can be neglected when neutron fluences are lower than 1 x E21 n/m² (E > 1 MeV). KTA 3203 is valid up to fluences of 5 x E23 n/m². However, the German Reactor Safety Commission (RSK) Guidelines include a recommendation for a maximum allowable fast neutron fluence of 1 x E23 n/m². The number of surveillance sets and the withdrawal schedule relative to the RPV fluence (50% and 100% of the design fluence) is also fixed in KTA 3203 depending on the RPV design fluence. KTA 3203 allows lead factors > 3 on the radiation capsules. This is to ensure, that the results for the first set of irradiated specimens withdrawn at approximately 50% of the fluence predetermined for the vessel at end-of-life are available prior to the first in-service pressure test of the RPV. The surveillance specimens are located in a device welded to the RPV inner wall approximately at the level of the maximum axial fluence as shown in Fig. 5-2. The requirements from KTA 3203 regarding the configuration and quantities of specimens for each surveillance set (materials to be included, number of Charpy-V and tensile specimens) as well as withdrawal schedule is described in Fig. 5-3. A sketch of the surveillance programme capsule is shown in Fig. 5-4.

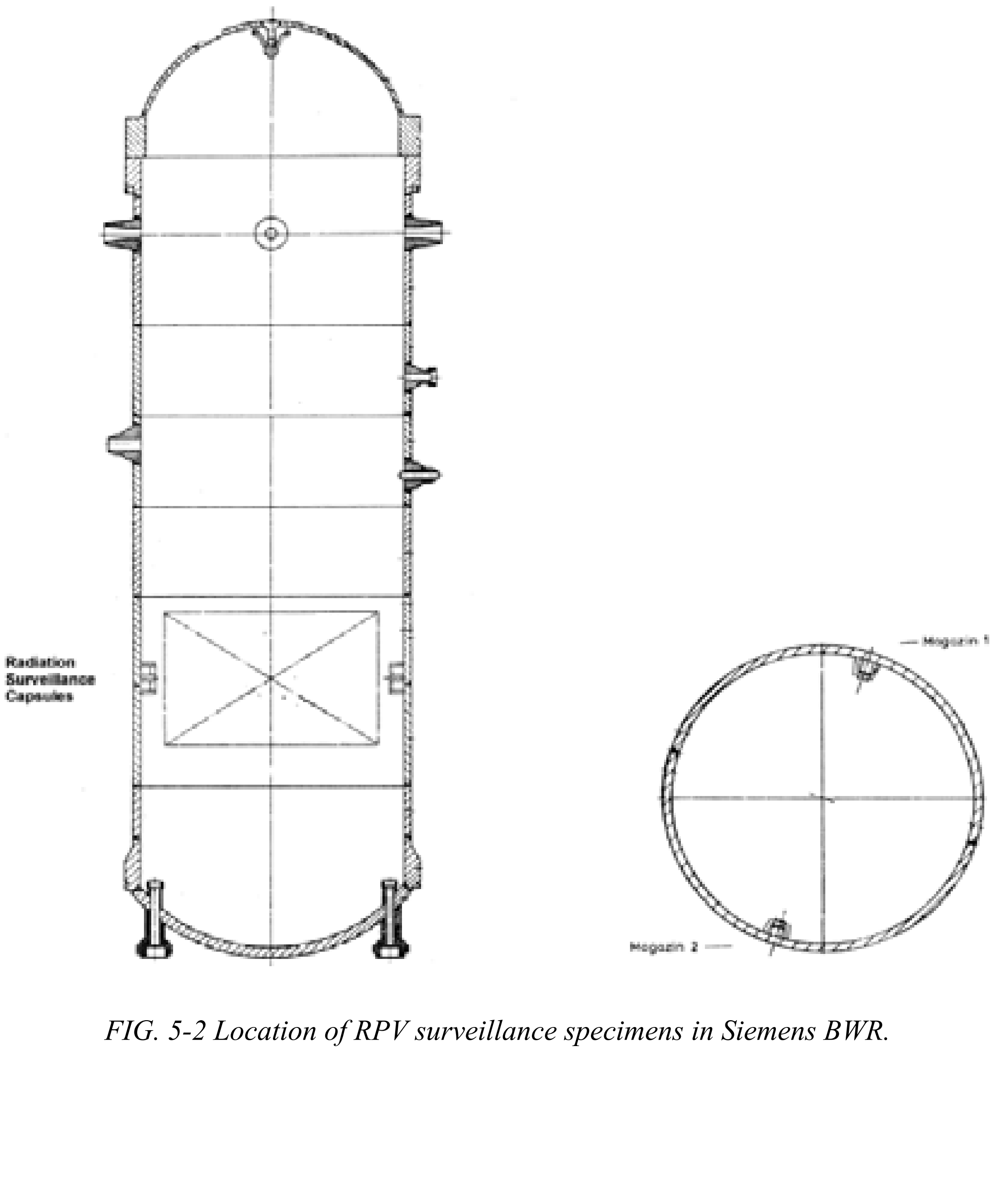
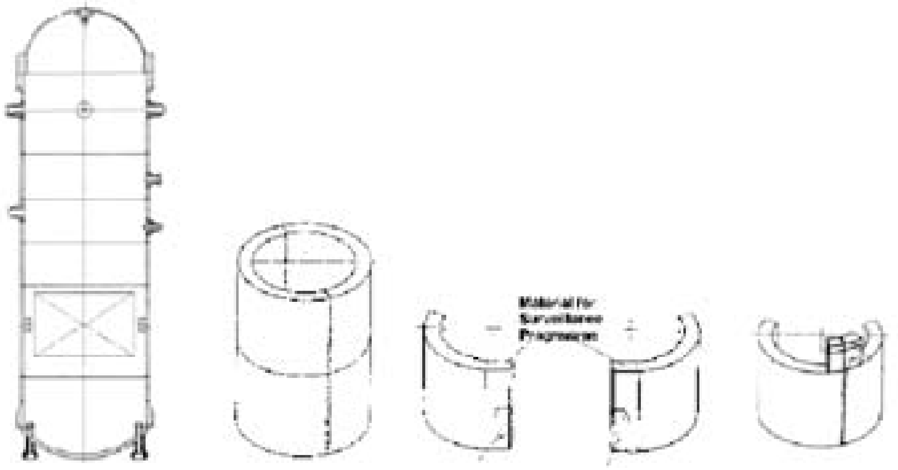
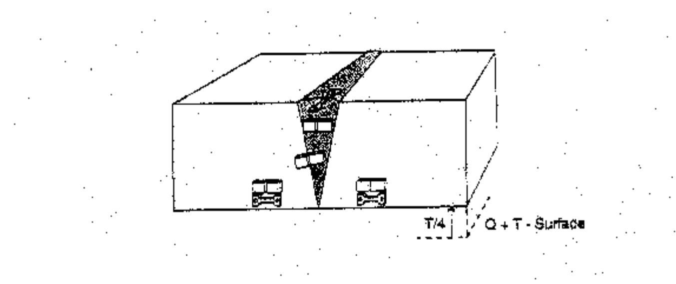


FIG. 5-2 Location of RPV surveillance specimens in Siemens BWR.





No. of specimen set	Charpy-V-notch specimens		Tensile specimens		Time of withdrawal
	BM	WM	BM	WM	
1	12	12	3	3	unirradiated
2	12	12	3	3	≈ 50 % AF
3	12	12	3	3	≥ 100 % AF

BM : base metal
WM : weld metal
AF : assessment fluence

Table 3-1: Number of test specimens for each irradiation exposure set for assessment fluences equal to or smaller than $1 \cdot 10^{19}$ cm^{-2} at neutron energies E > 1 MeV

No. of specimen set	Charpy-V-notch specimens			Tensile specimens			Time of withdrawal
	BM I	BM II	WM	BM I	BM II	WM	
1	12	12	12	3	3	3	unirradiated
2	12	12	12	3	3	3	≈ 50 % AF
3	12	12	12	3	3	3	≥ 100 % AF

BM : base metal
WM : weld metal
AF : assessment fluence

Table 3-2: Number of test specimens per irradiation exposure set for assessment fluences greater than $1 \cdot 10^{19}$ cm^{-2} at neutron energies E > 1 MeV

FIG. 5-3 Configuration and type of specimens used in the surveillance programme in Siemens RPVs.

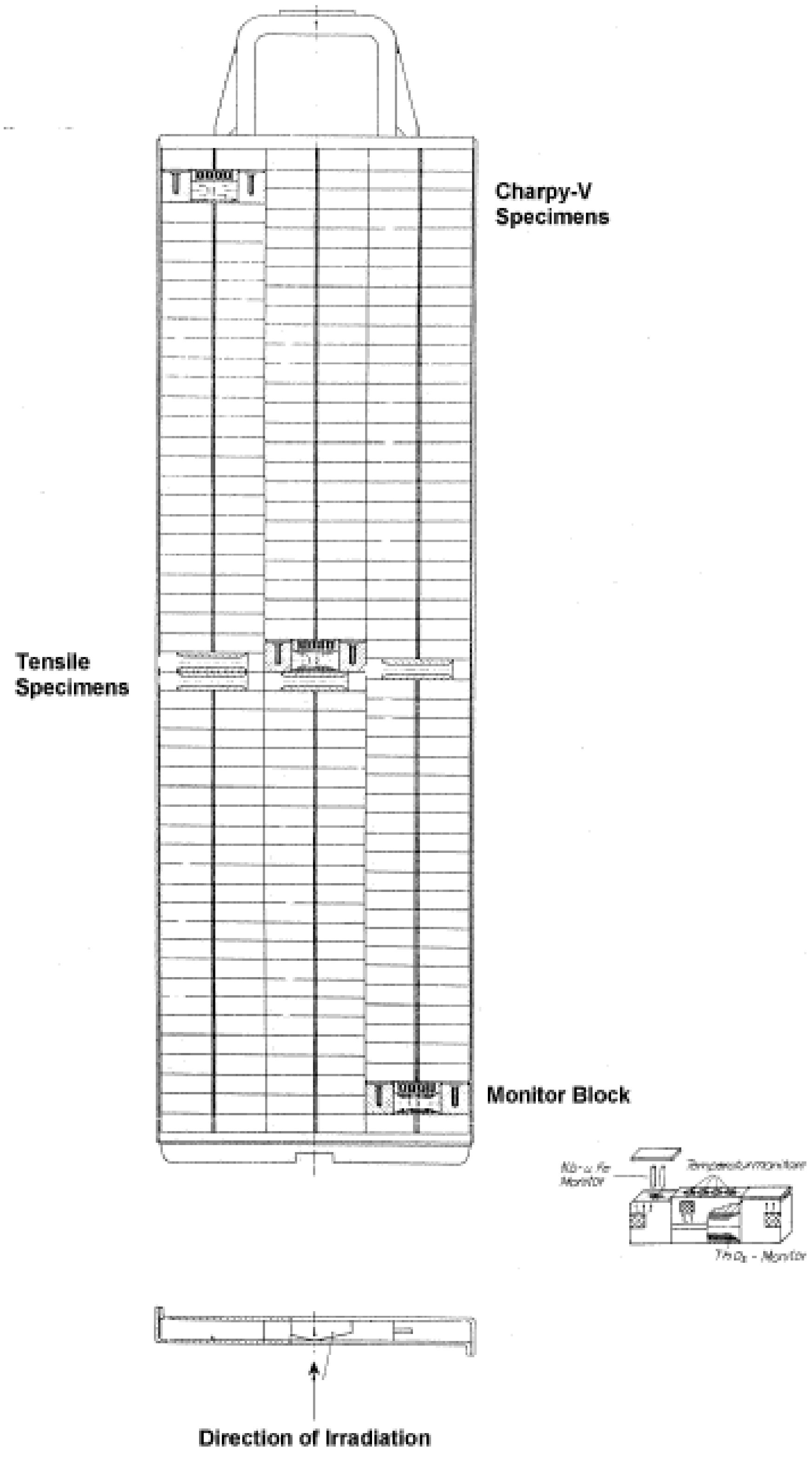


FIG. 5-4 Irradiation capusules used in the RPV surveillance programme in Siemens BWRs.

5.3.3. Requirements in Japan

The METI Notification 501 published by METI requests RPV material surveillance programme. And the Japan Electric Association describes detail procedure of surveillance programme in industrial technical standards, JEAC 4201-2000 [5.38]. This standard specifies quantity of Charpy V-notch test specimen and tensile test specimen, removal period of surveillance capsules and the method of the calculation of changes of RT_{NDT} throughout the service life of RPV materials caused by neutron irradiation. And the requirements of fracture toughness throughout the service life is specified in industrial technical standards, JEAC 4206-2000 [5.39], as not less than 68J (Charpy upper-shelf energy).

JEAC 4201-2000 prescribes the reactor vessel material surveillance programme which, based on NRC 10 CFR Part 50 Appendix G (1995) [5.30] and Appendix H (1995) [5.31], and ASTM E185-94; it also includes the Japanese embrittlement predictive equation which is mentioned in Section 6.1.8.

JEAC 4206-2000 provides experimental methods to confirm the integrity of nuclear power plant components against non-ductile failure. These methods include the linear elastic fracture mechanics analysis method. JEAC4206 is based on NRC 10 CFR Part 50 Appendix G (1995) [5.30] and Appendix H (1995) [5.31], the ASME Boiler and Pressure Vessel Code Section III.

The surveillance specimens are located between the shroud and RPV along the entire core length as shown in Fig 5-5. The configuration and type of specimens are shown in Fig 5-6.

5.3.4. Requirements for RPV in Finland

Requirements are presented in Finnish Regulatory Guides. In those brittle fracture analysis, fracture toughness tests and surveillance programme are provided. Requirements are in accordance with ASME.

In Olkiluoto 1 and 2 RPV surveillance specimens (Cha–py V – and tensile test specimens , base metal, weld metal and HAZ) are located on two distances from the core. Nearest specimens are near to the out side surface of core shroud wall. Other specimens locate about on midpoint between the core shroud and inner surface of RPV. In addition to these RPV specimens Olkiluoto 1 has core shroud specimens (stainless steel) inside core shroud.

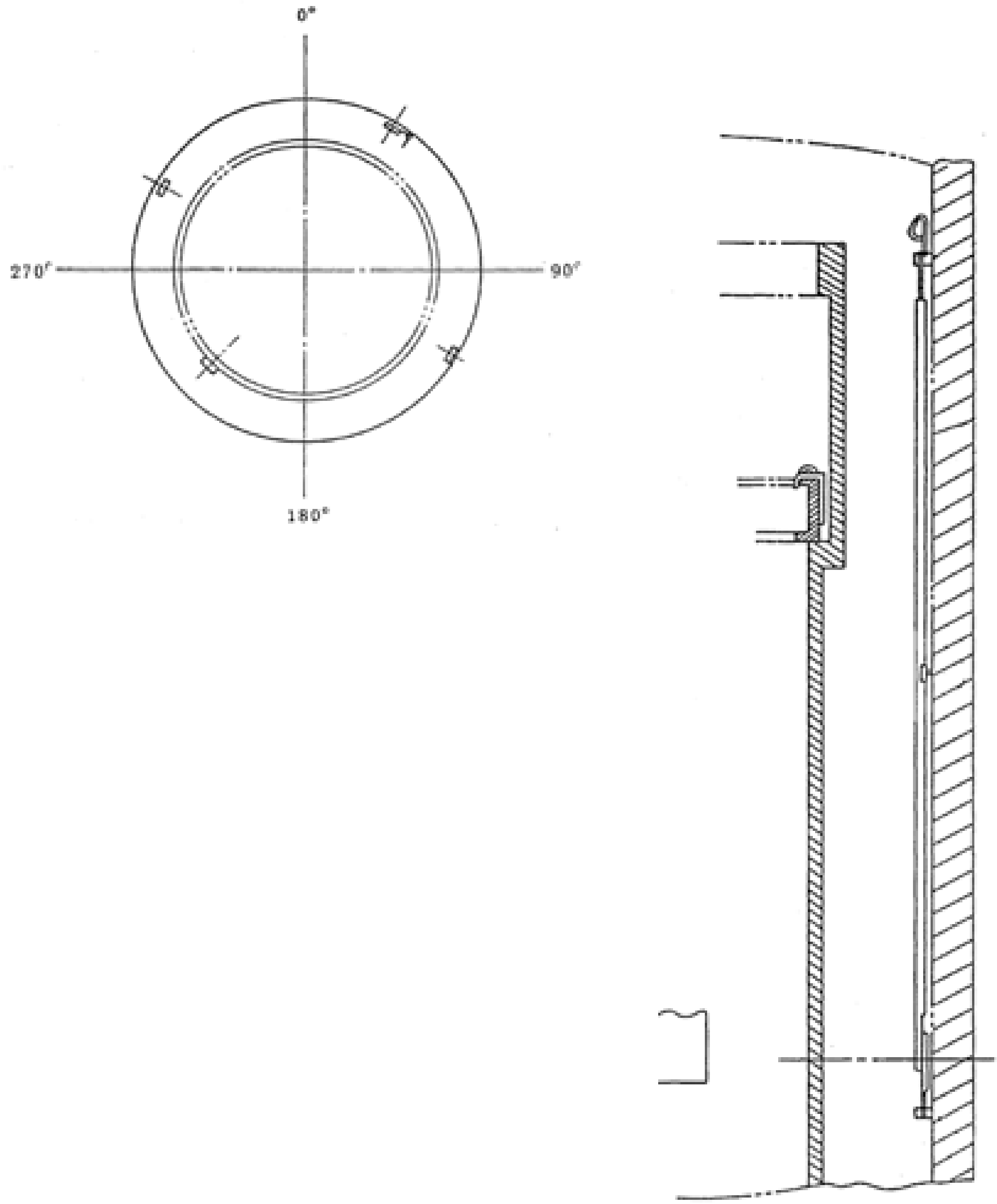


FIG. 5-5 Surveillance specimens are located between the shroud and RPV along the entire core length.

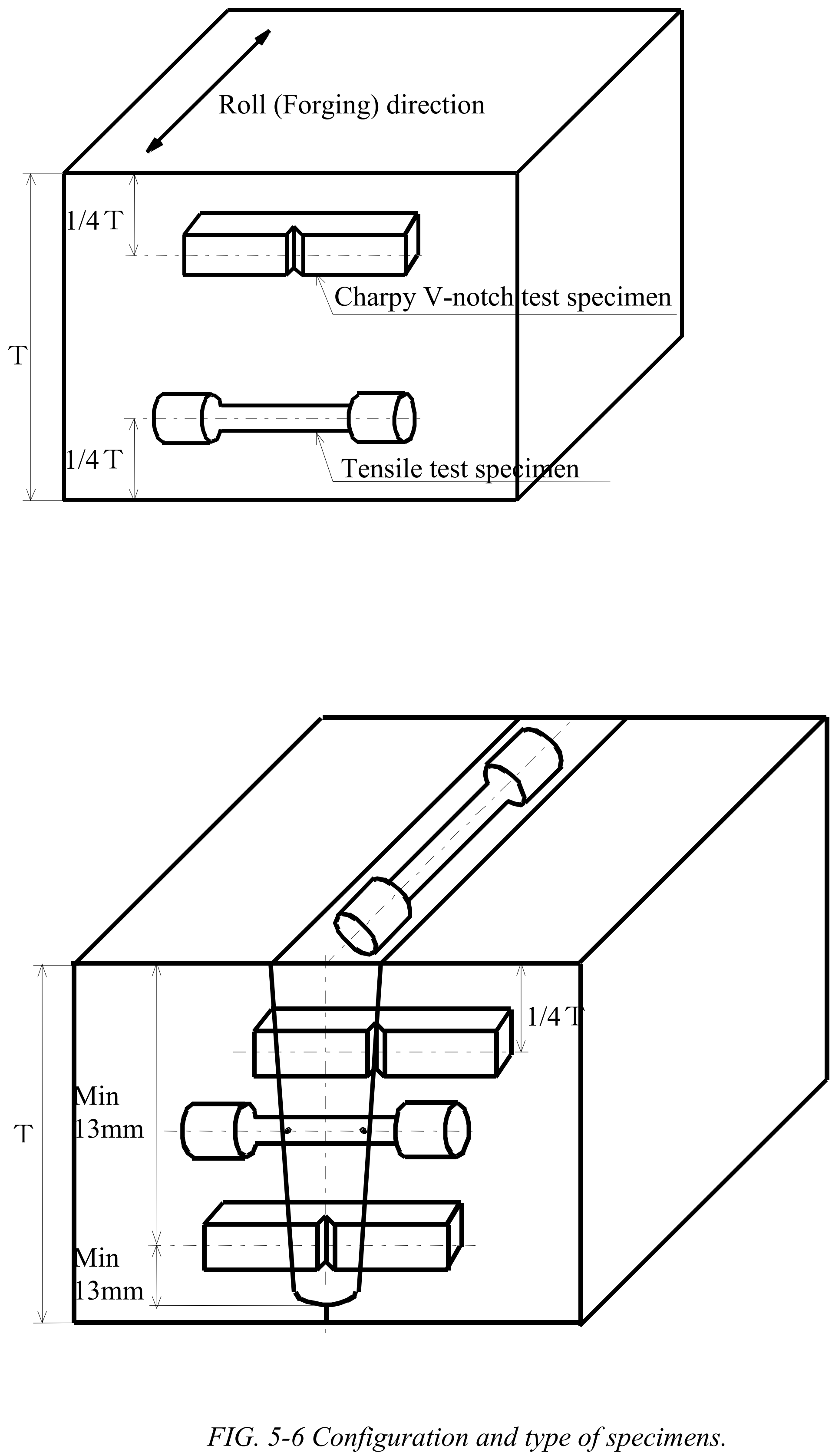


FIG. 5-6 Configuration and type of specimens.

5.3.5. IAEA RPV surveillance database

Recently, the IAEA International Working Group on Lifetime Management of NPPs started the creation of a worldwide database which would store the results from the RPV surveillance programmes. The primary purpose shall 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 (including different source of metallifc charges etc.) are slightly different in some countries, even though the manufacturing is performed according to the same standards and general requirements. As a result, vessels from each of the manufacturers represent a family, which can be slightly different from the others. Thus, results from one database may not be fully applicable to RPVs from other manufacturers. Creation of this database started in 1996 under the coordination of the aforementioned International Working Group on Lifetime Management of Nuclear Power Plants. Up to now, results from surveillance programmes from 10 countries are included in this database.

First use of this database is included in the IAEA Co-ordinated Research Project on "Evaluation of Radiation Damage of WWER RPV Using the IAEA Database on RPV Materials" with the main goal in preparation of "Guidelines for Prediction of Radiation Embrittlement of Operating WWER-440 Reactor Pressure Vessels" [5.40].

Additionally, this database has also the second part that collects data from all IAEA Coordinated Research Projects in the field of radiation damage in RPV materials, mainly:

- CRP-1 on "Irradiation Embrittlement of Reactor Pressure Vessel Steels" [5.41]
- CRP-2 on "Analysis of the Behaviour of Advanced Reactor Pressure Vessel Steels under Neutron Irradiation" [5.42]
- CRP-3 on "Optimising Reactor Pressure Vessel Surveillance Programmes and their Analysis" [5.43]
- CRP-4 on "Assuring Structural Integrity of Reactor Pressure Vessels" [5.44]
- CRP-5 on "Surveillance Programmes Results Application to RPV Integrity Assessment" [5.54, 5.46]
- CRP on "Nickel Effects in Radiation Embrittlement of RPV Materials" [5.47]
- CRP on "Guidelines for Prediction of Radiation Embrittlement of operating WWER-440 Reactor Pressure Vessels" [5.40]
- Round-robin Exercise (RRE) on "WWER-440 RPV Weld Metal Irradiation Embrittlement, Annealing and Re-embrittlement"

Access to both parts of databases is controlled by agreements with the IAEA.

5.4. Transient and fatigue cycle monitoring

5.4.1. Practice in the USA

As discussed in Section 4.4, the only RPV components likely to experience significant fatigue damage are the RPV studs. In the worse case, the studs can be replaced, eliminating any concern.

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 BWR RPVs. This is most likely a plant specific issue, since it is difficult to judge, in advance, where an indication might be detected.

In the case of life extension, the limiting components, other than Nozzles, Penetrations, Safe ends, Vessel support skirt. It is probable that these can be shown to have a significant capability for a longer life via combination of removing the conservatisms applied in the original analyses and taking credit for actual operating history.

5.4.2. Practice in Germany

All German BWRs 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. Practice in Finland

In Finland transient and fatigue cycle monitoring is followed using operational data.

Loading cases are reported yearly to Radiation and Safety Authority, STUK.

5.4.4. Practice in Japan

The Japan Atomic Power Company (JAPC) equipped fatigue monitoring system at Tsuruga NPP Unit1 in 1991. The feedwater nozzle was selected as an evaluation (representative) point because it received the highest thermal load among the RPV parts. This system automatically calculates usage factors using process data, e.g. temperature, pressure and flow rate, provided from the exiting instrument elements. The main objectives of this system are:

(1) to show that the design base fatigue analysis is conservative;
(2) to show that environmental effects do not cause actual problems;
(3) to quantitatively demonstrate soundness of the component against fatigue;
(4) to enable an efficient fatigue evaluation;
(5) to contribute the long term operation consideration and planning.

The evaluation results show that the feed water nozzle has enough margin with fatigue strength even though environmental effects based on the Higuchi- Iida equation [5.48] are taken into account.

5.5. Current inspection, assessment and maintenance practices

The section describes current inspection, assessment and maintenance practices for each vessel component for which ageing degradation is shown to be potentially significant.

5.5.1. Attachment Welds: IGSCC

Stress corrosion cracking (SCC) initiation in attachment welds or weld butter not only depends on IGSCC material susceptibility but upon environment as well. Hence, it is expected that some attachment welds would have lower probability of cracking while others have a greater chance of experiencing SCC and subsequent crack propagation into the low alloy steel (LAS) vessel material.

Since there is an increased level of concern over the potential for cracking due to IGSCC initiation in attachment welds, it is considered prudent to implement an inspection programme that would ensure that vessel integrity is maintained despite potential IGSCC. This inspection programme should take into account the likelihood of IGSCC initiation in each attachment weld (excluding ones associated with vessel penetrations), the potential for subsequent propagation into LAS, and accessibility of the weld locations. Specific details such as material, geometry, stress level, fabrication history and water chemistry history need to be considered in setting inspection priority. For this reason prioritization of attachment welds is a plant-specific action.

The BWRVIP evaluation of attachment welds is contained in BWRVIP-48 [5.49]. It contains a definition of the inspection strategy for the vessel interior attachments to the RPV. For the core spray piping bracket and jet pump riser brace attachments, enhanced visual inspections (1/2 mil wire resolution) are recommended. For the other bracket attachments, the standard VT- 1 (beltline) and VT-3 (beyond beltline) visual inspections and schedule specified in ASME Section XI for reactor vessel interior attachments (B-N-2) are recommended. However, for the steam dryer support and feedwater sparger bracket attachments, which use furnace-sensitized stainless steel (E 308/309 or 308L/309L) or Alloy 182 material, the visual inspections should be performed with enhanced visual inspections (1/2 mil wire resolution). The basis for these recommended inspections, as well as inspection frequencies and scope expansion, is included in the report.

For all indications, which are detected during the enhanced visual inspections, ultrasonic inspections should be performed to determine if the indication has propagated into the reactor vessel base material. For any flaw which is found to have propagated into the vessel base material, an evaluation should be performed in accordance with the requirements of ASME Section XI.

The propagation of SCC cracks into low alloy steel near attachment welds is limited due both to the absence of a driving residual stress and to an inherent material resistance. Only the attachment welds and material immediately adjacent to the weldment are of concern.

5.5.2. Closure Studs: Fatigue/SCC

Fatigue and SCC are significant degradation mechanisms for the closure studs. The circumstances of SCC caused stud failures, which occurred at a BWR/3 plant, are described in Section 4. Closure stud degradation due to both fatigue and SCC is addressed by current ASME code guidelines and GE RICSIL 055R1 [5.50], which responded to the closure stud cracking incidents, the effect of SCC of closure studs can be bounded within acceptable degradation limits and consequences of excessive degradation can be managed by inspection (ASME Section XI, GE RICSIL 055R1) and analytical evaluation. This management programme can be applied to all closure stud designs. The concern for fatigue/SCC of the studs is lessened by the fact that the studs are replaceable.

5.5.3. Nozzles: Fatigue/IGSCC

SCC is a significant degradation mechanism for nozzles. Standard ASME Code evaluations may be used to determine appropriate inspection intervals during the extended license period. In addition, in-service inspection and flaw evaluation procedures are given in Section XI Subsection IWB of the ASME Code for detecting and SCC determination. Additional proactive measures are implemented by Generic Letter (GL) 88-01 [5.43]. Adequate methodology is available to bound the effects of SCC within acceptable limits and acceptable techniques exist to manage consequences of excessive degradation.

5.5.3.1 Replaceable Feedwater Sparger (Fatigue)

Rapid cycle fatigue cracking has been identified as a significant degradation mechanism for replaceable feedwater spargers. BWR plants currently operate with augmented programmes conforming to regulatory instruments that were developed when feedwater and CRDRL nozzle fatigue became a generic BWR issue. NUREG-0619[5.52] addressed the feedwater/CRDRL nozzle cracking issue. Programmes that conform to NUREG-0619 effectively manage this issue.

One method of determining the potential for such cracking would be to monitor bypass leakage in the thermal sleeve using thermocouples to measure temperature fluctuations. An alternative to leakage monitoring would be fatigue crack initiation evaluation. For a replaceable sparger design, plant-specific feedwater thermal sleeve geometry, flow rate, and temperature are combined with the established feedwater duty map and the nozzle low-cycle fatigue usage evaluation to decide whether refurbishment might become necessary.

Since seal replacement is expensive and involves radiation exposure, leak detection is recommended.

5.5.3.2 Welded Feedwater Sparger (Fatigue)

Fatigue has been identified as a significant age-degradation mechanism for welded feedwater spargers requiring further evaluation.

Earlier analyses, which were used to resolve the feedwater nozzle high cycle fatigue issue, showed that fatigue usage factors for the welded design ranged from 0.46 to 0.77 for 40-years operation. These calculations used conservative assumptions for frequency of temperature oscillation, magnitude of stress reversals and duration of operation with high-cycle thermal oscillation. The fatigue usage factor is likely to be well below 1.0 if actual transient events and event severities are considered. Fatigue reevaluation based on actual events and severity of events can be used to demonstrate that fatigue usage is below 1.0 even when considering the extended operations. Thus, it is necessary to verify, using the methods discussed above, that fatigue usage is less than 1.0 even when considering the extended operation.

The effect of SCC and fatigue can be bounded within acceptable degradation limits and consequences of excessive degradation can be managed by inspection, monitoring and analysis. This programme can be applied to all nozzles and nozzle designs.

Welded design with a double thermal sleeve have significantly more margin.

5.5.3.3 Control Rod Drive Return Line (CRDRL) Nozzle (Fatigue)

Fatigue cracking has been identified as a significant degradation mechanism for CRDRL nozzles, which have not been capped. Fatigue usage ranges from 0.6 to 1.0 over 40 years for this condition. For plants with uncapped nozzles, CRDRL nozzle fatigue should be monitored since fatigue damage cannot be shown to be bounded by conventional analytical procedures.

If the CRDRL nozzle has been capped the fatigue mechanism is no longer present.

5.5.3.4 BWR/5 LPCI Nozzles (Radiation Embrittlement)

LPCI nozzles in some BWR/5 plants are close enough to the beltline to require some adjustment to RT due to irradiation. However, analyses for these plants have shown that, after a few years of operation, the more highly irradiated beltline shell material becomes more limiting. Therefore, LPCI nozzle embrittlement is addressed by plant-specific pressure-temperature limit adjustments made for the beltline shell.

5.5.4. Safe Ends: Fatigue/IGSCC

Fatigue and IGSCC have been identified as significant degradation mechanisms for safe ends. However, both of these degradation mechanisms can be managed by existing procedures. Standard ASME Code Section III evaluations may be used to determine appropriate inspection intervals for an extended license period. In addition, in-service inspection and flaw evaluation procedures are given in Section XI, Subsection IWB of the ASME Code for detecting and assessing any deterioration from fatigue or SCC. Additional inspection procedures are described in Revision 2 of NUREG-0313 [5.53] and implemented by Generic Letter 88-01 [5.51].

Any concerns are accentuated at the feedwater and CRDRL safeends.

5.5.5. Vessel Shell; Radiation Embrittlement

Embrittlement by fast neutrons has been identified as a potentially significant ageing degradation for the vessel beltline weldments. The proximity of the nuclear fuel causes the fluence to be high enough to affect the low steel (LAS) material properties.

In addition to meeting the technical requirements of Appendix G to 10 CFR50, there is the practical question of operating the plant as irradiation effects increase minimum permissible vessel temperatures. The inherent pressure-temperature relationship for the BWR during core critical operation and heatup/cooldown, typically on the saturated steam curve, is such that increasing beltline shift is not a concern. However, difficulty in heating the vessel to high temperature for a pressure test before operation could be economically important for some BWRs. In any event, it will be possible to perform pressure tests throughout the vessel life.

5.5.6. Vessel Support Skirt: Fatigue

Vessel support skirt fatigue was evaluated in accordance with ASME Code Section III requirements. Usage factors vary widely for each plant, depending on methods and details of fatigue analysis. Fatigue usage in the support skirt is primarily caused by startup/shutdown (heat-up/cool-down) thermal events. Based on an assessment using a realistic stress cycle loading, fatigue is not a significant degradation mechanism for the support skirt.

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6. ASSESSMENT METHODS

In this section component/age related degradation mechanism combinations found to require further evaluation are re-examined in terms of the capability of maintenance, in-service inspection, surveillance, testing and analytical assessment programmes to effectively manage potentially significant degradation effects. Degradation mechanism/component combinations for which generic program elements effectively manage the age-related degradation are considered to be adequately addressed.

Combinations of mechanisms and components for which generic effective programme elements cannot be shown to manage potentially significant age-related degradation require evaluation as discussed in the following sub-sections.

6.1. Radiation embrittlement assessment

Requirements for assuring vessel integrity against brittle fracture are specified in Appendices G and H of 10CFR50 [6.1]. Appendix G deals with operating requirements and Appendix H with surveillance requirements. The current approach to evaluating pressure-temperature limits is to use the guidelines in Appendix G of Section III of the ASME Code [6.2] along with Regulatory Guide 1.99 (Revision 1 of Regulatory Guide 1.99 [6.3] was updated to Revision 2 [6.4] in May 1988 to account for irradiation shift in RT_{NDT}). Surveillance data is collected according to the guidelines of ASTM E185 [6.5], which is referenced in Appendix H to 10CFR50. Surveillance data is factored into the allowable pressure-temperature limits in accordance with guidelines set forth in Regulatory Guide 1.99.

10CFR50 appendix G places two requirements on the low alloy steel vessel materials to assure ductile behavior. First, requirements are based on pressure-temperature limits for various operating conditions. These requirements include consideration of the reference temperature (RT) of the beltline materials, the RT is the initial RT_{NDT} plus the shift in RT_{NDT} resulting from accumulated neutron fluence. Second, Paragraph IV.A.1 of Appendix G to 10CFR50 states “Reactor vessel beltline materials must have charpy upper-shelf energy of no less than 75 ft-lb (102 J) initially and must maintain upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J) 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 USE equivalent margins analyses instead of taking the following three tasks.

— Volumetric examination of the beltline
— Supplemental fracture toughness testing
— Fracture mechanics analysis showing equivalent margins of safety

If these steps do not provide satisfactory results, the beltline materials must be annealed to recover fracture toughness.

The ability to meet the 10CFR50 Appendix G requirements depends on the irradiation embrittlement experienced during vessel life. Revision 2 of Regulatory Guide 1.99 generally predicts higher shifts for BWRs than did Revision 1. In addition, revision 1 was based on correlations for copper and phosphorus content. As a result, shifts calculated by the Revision 2 correlation are as much as 100 °F higher for some BWR/6 plants where copper and phosphorus were controlled to be low but nickel content was high. Methods used to estimate reference temperature and USE are discussed below.

Revision to Reg.Guide 1.99 is in process.

6.1.1. Initial reference temperature

Initial RT_{NDT} data for the vessel beltline materials must be collected and retained to provide the starting point for the vessel toughness evaluation. This information has been documented in the Final Safety Analysis Reports (FSARs) for most BWR/5 and BWR/6 plants. The data generally consist of Charpy test and drop weight test results for plates, longitudinal welds and circumferential welds. After 1973, RT_{NDT} values are calculated according to ASME Code Section III (NB-2300).

For older plants, updates to the current 10CFR50 Appendix G requirements have usually been made when the first surveillance capsule was removed and tested. Vessel quality assurance records typically have adequate data to estimate initial beltline plate RT_{NDT} values.

For plants constructed before the 1973 ASME Code requirements became effective, Charpy tests on plate material involved only longitudinal specimens, and the requirement was to meet 41 J (30 ft-lb) impact energy at designated temperature. The post 1973 requirements involve transverse specimens where 68 J (50 ft-lb) impact energy and 0.89 mm (35 mils) lateral expansion must be met 33 °C (60 °F) above the RT_{NDT}. In this case, 10CFR50 Appendix G (Section III. A) requires equivalence with the current ASME Code requirements be demonstrated. GE has developed correlations, based on the research documented in the NRC 1975 Heavy Section Steel Technology (HSST) programme [6.6], to estimate an RT_{NDT} value given 41 J (30 ft-lb) longitudinal Charpy data. Branch Technical Position MTEB 5-2 [6.7] also contains correlations for establishing equivalent RT_{NDT} values for older plants.

The NRC approved BWR OG report NEDC-32399-P, “Basis for GE RT_{NDT} Estimation Method, in a safety evaluation dated December 16, 1994.

Charpy and drop weight test data for plates in older plants were generally better documented than were weld data. Sometimes even vessel fabricators cannot produce useful weld toughness data. In such cases, industry data bases, such as the EPRI surveillance data base [6.8], can be examined to see whether the same weld material was used in another plant. As an upper bound estimate, the requirements of the vessel purchase specification can be used.

6.1.2. Material chemistry

Most BWRs in the U.S. were fabricated with plate from Lukens Steel. Fabrication records sent to GE by Lukens generally did not include copper content, but Lukens has kept copper information in internal records. As part of the recent BWR Owners’ Group RG 1.99 Impact study [6.9], Lukens provided GE with copper content information on most BWR beltline plates.

Weld fabrication records, when available usually have copper and nickel data. However, in some plants, especially older plants, weld chemistry is unknown. When chemistry data are unavailable, an upper bound is mandated by Regulatory Guide 1.99 Revision 2 [6.4] which specifies an upper bound chemistry of 0.35 Cu (% wt) and 1.0 Ni, which may be much higher than the unknown chemistry. In 1998 the BWR Vessel and Internals Project (BWRVIP) surveyed all US BWR reactor vessel plate and weld chemistry information.

The purpose of this effort was to resolve missing data and establish best estimate chemistry values for the limiting plate and welds materials for each US BWR. This chemistry data and other related material property data was then used to judge the adequacy of the materials in each plant surveillance programme. For each vessel limiting weld and limiting plate, a best representative surveillance material was assigned, based on heat number, similar chemistries, common fabricator, and the availability of unirradiated data. The results of this work are reported in BWRVIP-86-A [6.10].

The following provides the procedures adopted by the BWRVIP for the determination and use of best chemistry information.

6.1.2.1 Determination and use of best estimate chemistry

Analysis of the embrittlement behavior of surveillance materials requires knowledge of the material's chemistry content, so that measured embrittlement shifts can be compared to the shifts that would be expected based on chemistry. Historically, there has been little guidance regarding the proper techniques for characterizing a material's chemistry when various chemistry measurements are available for a heat of material. To ensure consistency of analysis, the BWRVIP ISP has adopted the protocol provided below for estimating a surveillance material's best estimate chemistry.

6.1.2.2 Vessel weld vs. surveillance weld material chemistries

It is important to differentiate between the best estimate chemistry for a vessel weldment and the best estimate chemistry of surveillance weld material. For vessel welds, the U.S. NRC has established that the best estimate chemistry is obtained from the average of all industry-wide data for the specific heat [6.11]. Because of the wide variations in chemistry that are often present in weldments, the average of all industry-wide data is regarded as the best representative for a vessel weld.

For surveillance weld chemistry, the U.S. NRC has held that the best estimate should be based on the chemistry data for that specific weld rather than the heat best estimate chemistry [6.11].

Chemistry measurements

As a standard practice, chemistry measurements should be made on one or more surveillance specimens of each surveillance material (e.g. plate and weld) whenever a surveillance capsule is tested. The new chemistry data should be documented in the report, and the data should be averaged with any previous data to obtain a refined best estimate chemistry. The new, revised best estimate chemistry should then be used in the analytical evaluation of the surveillance material embrittlement behavior.

Standard practice for calculating best estimate chemistry

In general, there are two types of chemistry data available: material certification data, to qualify a material for use in the reactor vessel; and specific chemistry measurements made on surveillance specimens fabricated, in many cases, years after the original material certifications are performed. For surveillance plates, both types of data are useful for estimating the chemistry. For surveillance welds, only the specific measurements on the surveillance specimens are preferred as the basis for calculating best estimate chemistry.

In the case of surveillance plates, the surveillance materials are taken from the actual vessel material and the certification chemistry measurements generally reflect the chemistry of the surveillance plate specimens. Therefore, any certification chemistry data for the plate heat should be averaged with chemistry measurements that have been made on specific surveillance plate specimens in order to determine the best estimate chemistry for the surveillance plate material heat.

In the case of welds, most surveillance welds were made well after the vessel weld certification test was performed. Weld certification data preceded vessel fabrication, so in many cases there is a period of years between weld certification chemistry measurements and fabrication of the surveillance weld. As a result, the surveillance weld was typically made from a different coil of weld metal wire, even though it bears the same heat number as the qualification weld. Due to the variability of chemistry in different coils of weld wire, chemistry reported for the certification weld can differ significantly from the chemistry of the surveillance weld. Therefore, weld certification data should not be used in the calculation of the surveillance weld best estimate chemistries (unless no other data is available). In the past, weld certification chemistry data was sometimes reported as the estimate because no measurements existed specifically for the surveillance weld specimens. For all ISP surveillance welds, however, specific chemistry measurement data for surveillance specimens now exist.

The guidelines for handling the combining of chemistry data were provided by the Nuclear Regulatory Commission (NRC) at a workshop in 1998 [6.11]. When there are multiple specimens with individual chemistry measurements, each measurement is added into the average for that material heat. In the cases where multiple measurements exist for one specimen, an average is first determined for that specific specimen, and then that specimen's average value is averaged with the other specimens' measurements to yield the overall average for the material.

In the example below, the average chemistry for Specimen JJ1 (two data sets) is determined first because it is a single specimen having multiple data points. This data point is then averaged in with the other two chemistry sets to determine the overall average for the material heat.

Source		Specimen ID	Cu	Ni	P	S	Si
Baseline	**CMTR**		**0.5**	**0.8**	**0.001**	**0.002**	**0.2**
Capsule X		JJ1	0.4	0.6	0.001		0.2
		JJ1	0.5	0.7	0.002		0.1
		Avg. JJ1	**0.45**	**0.65**	**0.0015**		**0.15**
		J25	**0.4**	**0.7**	**0.003**	**0.003**	**0.1**
Best Estimate Average →			**0.5**	**0.7**	**0.002**	**0.003**	**0.2**

It should be noted that there might be multiple sources for surveillance chemistry data. Most information comes from capsule test reports or surveillance programme documentation. In some cases, supplemental chemistry and Charpy testing has been performed by other organizations (e.g. national laboratories such as Oak Ridge), and that data should also be considered if its provenance is adequate. Care should be taken, however, when the inclusion of such data is considered; testing conducted at a laboratory for research purposes may not have been conducted with intent (and necessary quality assurance) for results to be used in a nuclear surveillance programme.

6.1.3. Fluence

Flux distribution calculations are performed to determine peak fluence on the vessel inside surface and at the ¼ thickness depth. The flux distribution evaluation combines two-dimensional calculations to establish the azimuthal variations and one-dimensional calculations to establish the axial variations. The one-dimensional calculations must be validated and documented.

The same calculations are used to develop lead factors, the ratio of flux between the vessel peak location and the surveillance dosimetry location. Lead factors are used with surveillance dosimetry data, when available, to compute peak vessel fluence versus effective full power years (EFPY). Industry convention has been to consider 32 EFPY as the design condition based on 40 years of operation at 80 % capacity factor. Depending on capacity factor, fluences for 40 or 50 EFPY should be considered for extended service operation.

Welds are often limiting because of high copper content. One refinement in the fluence evaluation that might provide some relief in pressure-temperature limits is to compute the maximum fluence specific to the weld locations. If no welds are at the peak locations, a decrease in predicted reference temperature RT_{NDT} shift could be realized.

In the USA, the requirements are to be followed by using the methodology in Reg. Guide 1.190.

6.1.4. Adjusted reference temperature

The RT is calculated as the initial RT_{NDT} plus the irradiation shift for a given fluence. Revision 2 to Regulatory Guide 1.99 [6.4] includes a margin term that compensated for the statistical uncertainty of the prediction correlation and the uncertainty on the initial RT_{NDT} determination if actual NDT is not available.

$$RT = \text{Initial } RT_{NDT} + \text{Shift} + \text{Margin} \tag{6-1}$$

where $$\text{Shift} = [CF] f^{(0.28 - 0.10 \log f)} \tag{6-2}$$

CF = chemistry factor [6.4].

F = fluence/10^{19}

M(argin) = 34°F (19°C) for plate material or 56 °F (31°C) for weld metal for initial measured values of RT_{NDT}. If generic values of RT_{NDT} are used, M = 48°F (27°C) for plate and 66 °F (37°C) for welds, unless the shift is less than these values. For such cases margin = shift.

The Chemistry Factor (CF) is based on the copper/nickel combination for the material. If more than two credible surveillance data are available the CF should be calculated by curve fitting as explained in RG 1.99, Rev. 2.

6.1.5. Upper shelf energy (USE)

In addition to increasing the beltline RT, the upper shelf energy (USE) of low alloy steel materials decreases with neutron exposure. 10CFR50 appendix G stipulates that the USE be 50 ft-lb or greater. The decrease in USE may be calculated by the guidelines of Regulatory Guide 1.99, Revision 2, [6.4] based on correlations for copper and fluence.

Initial USE data for plates can often be conservatively approximated from fabrication data. However, for older plants, where Charpy plate specimens were longitudinal, an adjustment must be made to obtain results equivalent to those of transverse specimens. NRC Branch Technical Position MTEB 5-2 [6.7] suggests taking 65 percent of the longitudinal value to obtain a transverse equivalent USE. Following this recommendation, plate USE predictions for some BWRs may decrease below 50 ft-lb before 40 EFPY. This situation has been reviewed by the BWROG and found to still be acceptable [6.12]. Procedures according the BWRVIP programme [6.13]can be used to predict radiation effects on USE.

6.1.6. Surveillance data application

Irradiation of ferritic steels at high energy (E > 1 MeV) neutron fluences above a threshold level of approximately 10^{21} n/m^2 increases the material yield strength with an attendant decrease in ductility. From the standpoint of operation and design margin (as related to RPV fracture prevention), the toughness properties are of principal interest. Conventionally, the Charpy impact test is used to monitor toughness. An alternative approach according to ASTM-E-1921 (Master Curve) [6.14] can be used to reduce conservatisms in radiation predictions in, as allowed according to ASME Code Case N-629 [6.15].

The BWR has three surveillance capsules, which contain Charpy, tensile, and flux dosimetry specimens. According to the current version of ASTM E 185 [6.5], the first two capsules should be withdrawn at 6 EFPY and 15 EFPY for testing. The third is to be kept in the vessel as a spare. It should be noted that in order to conform to ASTM E185, additional capsules may need to be added.

Surveillance capsules, containing materials from plates, welds and heat affected zones, are normally located at positions near the vessel wall. Evaluation of service induced property changes in the surveillance specimens irradiated in an environment similar to that experienced by the vessel provides a basis for plant specific property predictions.

Revision 2 of Regulatory Guide 1.99 [6.4] requires two sets of surveillance capsule test results before the surveillance data can be used to predict RT_{NDT} shift. These two sets of surveillance data must pass the “credibility criteria” in Regulatory Guide 1.99, Rev. 2 to be used in the calculation.

All BWRs have surveillance programmes, which are designed to various versions of ASTM E 185 [6.5]. 10CFR50 Appendix H requires that the latest version of ASTM E 185 issued at the date of construction be met. In 2002 the U.S. NRC approved an integrated surveillance programme for the BWRs. A summary of the BWRVIP ISP is provided in Section 5.3.1.

New reductions of conservatisms have been introduced by the Code Cases N-588 (defects in circumferential welds) [6.16] and N-640 (use of fracture toughness curves for p-T limit curves) [6.17]. These 2 changes have already been implemented into the newer Code editions.

6.1.7. Radiation embrittlement assessment methods in Germany

The German Nuclear Safety Standards KTA 3201.2 [6.18] and KTA 3203 [6.19] require that the USE must remain above 68 J (50 ft-lb) during operation. If an upper shelf energy value of more than or equal to 68 J cannot be proven by the surveillance programme, further measures have to be undertaken to confirm the safety of the RPV. Such measures shall be defined in accordance with the authorized expert.

The fracture toughness during operation is determined by use of the adjusted reference temperature. The procedure to calculate the fracture toughness curve is given in the German Nuclear Safety Standard KTA 3201.2. The determination of the adjusted reference temperature itself is described in the German Nuclear Safety Standard KTA 3203. The adjusted reference temperature may be determined either according to the reference temperature RT_{NDT} concept or the Master Curve concept.

Up to a fluence of 1 x 10^{23} n/m², a fixed value of 40 K may be used as adjusted reference temperature for the materials where no data from surveillance sets are available. For fluences of more than 1 x 10^{23} n/m², the value for the limiting adjusted reference temperature given by KTA 3203 must be proven by surveillance data for all beltline materials from the sets as required in KTA 3203 (see Fig. 5-3 in page 92).

6.1.8. Radiation embrittlement assessment methods in Japan

The Japan Electric Association describes method of calculation of changes of RT_{NDT} throughout the service life of RPV materials caused by neutron irradiation in industrial technical standards, JEAC 4201-2000 [6.20]. The change of RT_{NDT} is calculated using following equation:

$$RT = \text{initial } RT_{NDT} + \text{Shift} + \text{Margin}$$

where

Base metal

$$\text{Shift} = [CF] \times f^{(0.29 - 0.04 \log f)}$$

$$[CF] = -16 + 1210 \times P + 215 * Cu + 77 * (Cu * Ni)^{0.5}$$

Welds

$$\text{Shift} = [CF] \times f^{(0.25 - 0.10 \log f)}$$

$$[CF] = -26 - 24\times Si - 61xNi + 301\times CuxNi)^{0.5}$$

$f = f_0 \times e \times p\ (-\ 0.24a / 25.4)$

where

f: fluence ($\times 10^{19}$ n / cm^2)

f_0: fluence of vessel inside surface

The approach for prediction of USE is regulated in JEAC 4201-2000 [6.20] and based on Regulatory Guide 1.99 Rev.2. [6.4]

6.2. Fatigue assessment

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, Subsection NB-3200, while the latter is exemplified by the flaw evaluation/acceptance procedures of the ASME Code Section XI, Subsection IWB. Environmental effects might be addressed in considering fatigue.

6.2.1. Crack initiation

Crack initiation is estimated by determining the fatigue usage at a specific location that 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 have to be treated with caution.

For 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. The sum of the ratios of these quantities gives the cumulative fatigue usage factor. The best source of information for plants with less than five years of operation 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 anticipated 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 conservatisms 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 smooth-bar laboratory test data 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.

6.2.2. 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 [6.20] is generally used to calculate crack growth:

$da/dN = C(\Delta K)^n$ where

da/dN = fatigue crack growth rate (in/cycle);

ΔK = stress intensity factor range (ksi/in) = $(K_{max} – K_{min})$;

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 below which fatigue damage is highly unlikely. At higher ΔK values, lessening importance is responsible for leveling of the growth rate curve.

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 (Kc), local crack instabilities occur with increasing frequency. Increasing the R ratio (K_{min}/K_{max}) causes an increase in cycle crack growth.

Knowing the history of stress cycle events in conjunction with the appropriated crack growth correlation allows prediction of the crack growth 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 critical crack size of the component, and thus time to failure, or residual life.

6.2.3. Time-dependent cyclic crack growth

The time dependent crack growth resulting from cyclic loading can be determined by:

$$da/dt = f(da/dN) = f[C(\Delta K)^n], \qquad (6\text{-}5)$$

where da/dt = crack growth rate, in/year, and

f = stress cycle or load frequency (e.g. cycles/year)

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.

6.2.4. Fatigue damage management programs

It is recommended that the fatigue usage is monitored for critical parts. For long term operation the following criteria should be met (see also 6.2.6):

(1) For plants with an ASME section III design basis, the expected transients for long term operation must fall within the original design basis by cycle counting or the analysis should be refined;

(2) 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 extended operation, an effective programme for managing the effects of potentially significant fatigue damage in components is adherence to ASME Section XI procedures. Formal in-service examination requirements are provided for each plant in its plant In-service Inspection (ISI) and In-service Testing (IST) programmes and are referenced to an applicable edition of the ASME Code Section XI Rules for In-service Inspection of Nuclear Power Plant Components. The plant in-service inspection programme, including any commitments to enhanced or augmented inspections as the result of plant operating experience or regulatory enforcement, provides an acceptable basis for continued operation of the component with indications that are within established limits. 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.

6.2.5. Fatigue reanalysis

If the confirmation of the current fatigue design basis for the 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 design cycles be tracked during service, relative to actual operating transients, to assure satisfaction of fatigue design requirements. However, since Technical Specification transient tracking requirements vary widely from plant to plant, the demonstration that the design-basis transients remain valid for the 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 design-basis transients extended through the whole vessel life. 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 predict cumulative fatigue usage factors which approach a value of 1.0. Unless the excessive conservatisms can be removed, more frequent in-service inspections may be required or, in the worst case, replacement or refurbishment may be recommended prematurely.

One way to remove conservatisms 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 the 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.

6.2.6. In-service examination

For component locations and parts having a history of fatigue damage, or as an alternative to analytical verification of the adequacy of the fatigue design basis throughout the extended operation, potentially significant fatigue damage can be managed through a programme of periodic in-service examinations. It will be necessary to supplement the measurements with analysis to assure operating margins. These examinations should be directed toward the detecting and characterizing fatigue crack initiation and growth to assure that the detected and sized indications do not compromise component structural integrity during the period intervening between examinations.

6.2.7. Fatigue assessment methods 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.

6.2.8. Fatigue assessment methods in Japan

General requirement

The fatigue evaluation method and (S,N) fatigue curves stipulated in the METI notification No.501 [6.22] and JSME Code and Standards “Code for Design and Construction”, JSME SNA2-2002 [6.23] are similar to the ASME Section III B3000 rules.

Environmental fatigue evaluation

In September 2000, Japanese regulatory body, MITI (current METI), published a notification, which required electric utilities to perform fatigue evaluation for the plant life management evaluation taking into account environmental effect. It was attached to the notification.

The parameter for evaluating environmental effects is the fatigue life reduction factor for environmental effects, F_{en}. F_{en} represents the reduction in fatigue life resulting from the high-temperature water LWR environment. As shown in Equation (6-6) below, F_{en} is defined as the fatigue life obtained from fatigue tests under room temperature in air, divided by the fatigue life obtained from fatigue tests under high-temperature LWR water conditions with the same strain amplitude.

$$F_{en} = \frac{N_A}{N_W} \tag{6-6}$$

where, N_A is fatigue life under room-temperature atmosphere and N_w is fatigue life under water environment.

The cumulative usage factor allowing for environmental effects, UF_{en}, can be expressed by the following equation, using F_{en}.

$$UF_{en} = \sum_{i=1}^{n} U_i \times F_{en,i} \tag{6-7}$$

Where U_i and $F_{en,I}$ represent the cumulative usage factor and fatigue life reduction factor for environmental effects, respectively, for the I-th load set pair (individual transient cycle evaluations) among the n-number of load set pairs (all evaluated transient cycles).

In the above equation, F_{en} is calculated according the following two F_{en} equations for carbon steel/ low alloy steel and for austenitic stainless steel. They are provided in the MITI environmental fatigue evaluation guideline.

F_{en} for carbon steel/ low alloy steel:

$$\ln(F_{en}) = -(0.199 \times T^* \times O^* + 0.112) \times S^* \times \dot{\varepsilon}^*$$

where

$$\begin{aligned}
&\dot{\varepsilon}^* = 0 \quad (\dot{\varepsilon} > 1.0\% / \mathrm{sec}) \\
&\dot{\varepsilon}^* = \ln(\dot{\varepsilon}) \quad (1.0\%/\mathrm{sec} \geq \dot{\varepsilon} \geq 0.0004\%/\mathrm{sec}) \\
&\dot{\varepsilon}^* = \ln(0.0004) \quad (\dot{\varepsilon} < 0.0004\% / \mathrm{sec}) \\
&T^* = 0.00531 \times T - 0.7396 \quad (T \geq 180^\circ C) \\
&T^* = 0.216 \quad (T < 180^\circ C) \\
&O^* = \ln(DO/0.03) \quad (0.03 \leq DO \leq 0.5\mathrm{ppm}) \\
&O^* = 0 \quad (DO < 0.03\mathrm{ppm}) \\
&O^* = \ln(0.5/0.03) \quad (DO > 0.5\mathrm{ppm}) \\
&S^* = 17.23 \times S + 0.777
\end{aligned}$$

F_{en} for austenitic stainless steel:

$$\ln(F_{en}) = 1.233 - P \times \ln(\dot{\varepsilon}^* / 0.4)$$

where

$P = 0.04 \quad (T \le 100°C)$

$P = 9.33 \times 10^{-4} \times T - 0.053 \quad (100°C < T < 325°C)$

$P = 0.25 \quad (T \ge 325°C)$

$\dot{\varepsilon}^* = 0.4 \quad (\dot{\varepsilon} \ge 0.4\%/sec)$

$\dot{\varepsilon}^* = \dot{\varepsilon} \quad (0.0004\%/sec \le \dot{\varepsilon} \le 0.4\%/sec)$

$\dot{\varepsilon}^* = 0.0004 \quad (\dot{\varepsilon} < 0.0004\%/sec)$

$\dot{\varepsilon}^*$ is strain rate-dependent parameter, T^* is temperature-dependent factor and O*is dissolved oxygen dependent factor.

The MITI environmental fatigue evaluation guideline mentions that these two equations were acquired from experiment data of PWR coolant environment (DO≤0.005 ppm). F_{en} for BWR coolant environment (DO>0.005 ppm) is under preparation. It is possible to apply the above two F_{en} equations to BWR coolant environment but evaluation results could be a little too conservative.

The MITI environmental fatigue evaluation guideline does not specify a detailed evaluation procedure for evaluating actual plant conditions. “Guidelines on Environmental Fatigue Evaluation for LWR Component” [6.24] published by Thermal and Nuclear Power Engineering Society (TENPES) in June 2002 provides a detailed procedure and specific and practical techniques. The document is used as a guidance documents for evaluating environmental fatigue for LWR components.

6.3. Stress corrosion cracking assessment

The technique used for determining SCC effects on component life is in-service inspection by volumetric means, such as ultrasonic imaging to detect and size flaws, and subsequent fracture mechanics evaluation to predict life remaining after initiation of the detected flaw.

Analytical evaluation is a useful tool for dispositioning detected and sized flaws, this method most often used for component life prediction is crack growth assessment, using laboratory crack growth data and analytical fracture mechanics stress intensity calculations. SCC crack growth rates under various environmental conditions have been generated for the austenitic steels used for internals. The BWRVIP programme has also provided recommended IGSCC crack growth rates for use in dispositioning detected and sized flaws [6.25 – 6.28]. These data, in conjunction with geometry specific stress intensity solutions, can determine subsequent crack growth, assuming that a crack has been detected and has reached a measurable initial crack size. The method does not consider the time to crack initiation; it relies on inspection to detect cracks or on historical data to predict the time to initiate cracking.

For component life to be determined, a limit on the amount of crack growth (allowable crack size) for the particular component must be established. Crack growth calculations can then be performed to verify that the predicted crack size, for some period of continued operation, is less than the allowable crack size. Component life can then be estimated by calculating the time at which the allowable crack size is reached. Repair or replacement options can be implemented at the predicted end of allowable component life.

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7. MITIGATION TECHNOLOGIES

This section addresses mitigation methods for SCC since SCC is the prevalent degradation mode for RPV. Mitigation methods for other known sources of degradation (e.g. fatigue and thermal) are not addressed.

IGSCC has been a concern in the BWR community since first detected in the late 1950’s in annealed stainless steel fuel cladding and in the mid-1960’s Type 304 stainless steel recirculation piping. In the late 1980’s IGSCC was detected in reactor internal components, i.e. shroud head bolts, core shrouds, access hole covers, etc. More recent experience from the world wide BWR fleet indicates that cracking of vessel internals has become more widespread than previously thought. For RPVs, parts like attachment welds, CRD stub tubes and safe ends have been subject to IGSCC.

Radiolysis of the coolant in the BWR core produces an oxidizing environment that is very aggressive in nature. The oxidant concentrations are a key factor in the initiation and propagation of IGSCC. A number of mitigation strategies have been developed to combat stress corrosion in BWRs [7.1]. The most prominent strategy to mitigate IGSCC is to change from an oxidizing to a more reducing environment. Laboratory and in reactor tests in the late 70’s and early 80’s using hydrogen addition to control the oxidant concentration and subsequent Electrochemical Corrossion Potential (ECP) proved to be a practical method to control IGSCC.

In the early 1990’s, GE introduced a program known as Optimum Water Chemistry to establish chemistry standards that would address the IGSCC concern and integrate the requirements to reduce person-Sievert exposure and radiation waste while protecting the BWR fuel. This programme was later expanded into a Reactor Internals Management (RIM) programme. As a result of the increased occurrence of shroud cracking, in late 1994 BWR plant owners formed the BWRVIP (Vessel Internals Project) committee. The BWRVIP is a utility driven committee with programme management supplied by the Electrical Power Research Institute (EPRI). The BWRVIP is developing several approaches for issue resolution, including more aggressive inspection, monitoring, and repair efforts, and modification of reactor water chemistry to reduce the probability of cracking.

7.1. Mitigation via water chemistry control

Several SCC mitigation methods by coolant chemistry control offer significant potential to mitigate SCC. These include:

- lowering ECP in the bulk water and locally in the crack opening by producing less acidic electrolytes or,
- improving the protecting effect of the oxide layer on the metal surface.

These controls may be implemented by appropriate consideration of the following actions:

- lowering the reactor water conductivity to extremely low values,
- increasing the pH-value and,

- adding conditioning agents such as a corrosion inhibitor at a low concentration level to the bulk water.

The two most common methods are currently being used to mitigate IGSCC/IASCC in BWR internals through water chemistry control. These are Hydrogen Water Chemistry (HWC) and Noble Metal Chemical Application (NMCA). Additional information about experience with these mitigation methods and their effects on IGSCC/IASCC, radiation dose and fuel integrity is provided in BWR Water Chemistry Guidelines –2000 Revision [7.2]. These Guidelines are an industry consensus document and are updated periodically. The mitigation methods are summarized below:

7.1.1. Hydrogen water chemistry

In the United States, as a part of the overall strategy to mitigate these phenomena, Hydrogen Water Chemistry (HWC) was first tested in the early 1980's at Dresden Unit 2. Hydrogen was introduced into the feedwater in order to change the recirculation water from an oxidizing to a reducing environment and to mitigate IGSCC in recirculation piping.

HWC is effective by reducing oxidant (oxygen and hydrogen peroxide) concentrations to low levels, < 2 ppb. Subsequently, this condition results in an environment that has an ECP less than -230 mV (SHE). Laboratory test and in-reactor constant extension rate tests have shown that initiation and propagation of IGSCC is mitigated when the ECP is below -230 mV (SHE). The concentration of feedwater hydrogen required to mitigate IGSCC in BWR internals varies from 1-2 ppm. One of the drawbacks of HWC is an increase in the main steam line radiation levels caused by N-16. The NMCA technology is discussed in the next section was developed.

7.1.2. Noble Metal Chemical Application (NMCA)

NMCA involves injecting platinum and rhodium compounds into the reactor water during an outage. The noble metals deposit on the surfaces in contact with reactor water. Platinum and rhodium catalyze the recombination of hydrogen and oxygen produced by radiolysis in the core. This leads to a decrease in the local oxygen concentration at the surfaces and a reduction in ECP to values below –230 mV at feedwater hydrogen concentrations of 0.2-0.4 ppm in contrast to 1-2 ppm required with HWC. NMCA increases the effectiveness of hydrogen in mitigating and allows a reduction of radiation exposure to plant personnel. A cooperative effort to demonstrate NMCA at the Duane Arnold Energy Center was undertaken by GE, IES Utilities, BWRVIP and EPRI in 1996. The demonstration showed that NMCA treated piping and reactor internals in lower and upper core could be protected at a feedwater hydrogen concentration of 0.25 ppm without increasing the main steam line radiation levels. BWRVIP conducted an extensive surveillance program of NMCA effectiveness over two cycles and fuel surveillance over 3 cycles. The results of this demonstration were documented in a series of BWRVIP reports. After the successful demonstration at Duane Arnold the NMCA process was applied at many BWRs in the US and a few in Europe and Japan. Currently there are 28 BWRs which have used NMCA. Ongoing BWRVIP activities are evaluating the durability, fuel and SCC-related performance characteristics of NMCA.

During operation there is a depletion of noble metal from reactor internal and piping surfaces. Consequently, every 3 to 5 years a re-application of NMCA is necessary. The proof of effectiveness of NMCA to date is based on laboratory results using crack growth specimens and in-reactor corrosion potential measurements and noble metal deposition measurements. The demonstration of effectiveness to mitigate crack propagation using in-reactor UT crack size measurements is on-going. The BWRVIP is planning to evaluate data from core shroud re-inspections to assess the effectiveness of NMCA and HWC in mitigating cracking. A fuel surveillance program at the NMCA demonstration plant was conducted and showed no adverse effect of NMCA on cladding corrosion and hydriding.

7.1.3. Deposition of noble metals by plasma spray

Noble metals can be deposited on surfaces by plasma spray as discussed in the following section.

Noble metal coatings can be applied underwater remotely using the plasma spray coating process. This application is particularly suitable for components such as the core shroud. An underwater welding process is also being developed to apply noble metal cladding; results from preliminary test programmes show a high quality and uniform application.

7.2. Mitigation via surface treatment

7.2.1. Shot peening/water jet peening

Techniques, which introduce a compressive surface residual stress have been shown to be effective SCC mitigators. The following peening techniques have been developed and already applied underwater remotely to core shrouds in Japan.

- Laser Peening (LP)
- Water Jet Peening (WJP)
- Shot Peening (SP)

The peening process introduces a compressive stress in the peened surface layer by constraint of surrounding material.

The LP process utilizes water-penetrable green light of a frequency-doubled Nd:YAG laser delivered with an optical fiber and generates the high pressure plasma of several Gpa on the surface[7.3 – 7.5]. The WJP process relies on the pressure derived from the cavitation collapse at the surface under the high pressure water jet [7.6 — 7.8]. The SP process utilizes spherical Type 304 stainless steel shots (diameter < 2mm) hardened during production process to have Vickers hardness about 500, which are projected by highly pressurized water (~ 1MPa) on the surface [7.9, 7.10].

The effectiveness of these processes to mitigate SCC of Type 304 reactor internal material have been demonstrated by laboratory testing. Compressive residual stress of several hundred MPa to a depth of 300 ~ 1000 μm is obtained. SCC susceptibility can be significantly reduced or eliminated by peening processes.

7.2.2. Surface melting/Solution annealing

Solution annealing of Type 304 stainless steel is a well established method for eliminating sensitization, thereby reducing SCC susceptibility. This principal has been extended to surface treatment to desensitize Type 304 by surface melting and solution annealing utilizing appropriate heat input controls.

Studies to evaluate the effects of heat input and quenching on the surface of sensitized 304 material utilizing YAG laser and CO_2 demonstrated that a remelted zone with a duplex austenitic/ferritic microstructure could be achieved to controlled depth of about 200 μm. SCC susceptibility relative to the sensitized 304 material as evaluated by laboratory bent beam testing in simulated BWR environment was substantially reduced or eliminated. Solution annealing and desensitization of a region near the surface was also achieved with appropriate laser heat input controls [7.10–7.16]

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8. REACTOR PRESSURE VESSEL AGEING MANAGEMENT PROGRAMME

The primary age degradation mechanisms for the RPV are cracking due to fatigue and cracking due to IGSCC. Radiation embrittlement is not an issue other than its impact on the temperature at which the pre-start up leak test or periodic hydrotest is performed. Fatigue and SCC interacts with each other and both are impacted by water chemistry. The cracking resulting from either cause can impair the safety functions of the RPV. Also both mechanisms are affected by time. For these reasons a systematic RPV ageing management is needed.

The previous sections of this report dealt with important elements of a BWR RPV ageing management 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 for ageing management, Fig. 8-1. Such an ageing management programme should be in accordance with guidance prepared by an interdisciplinary ageing management team for RPV organized at the corporate level or owner level. For guidance on the organizational aspects of a plant ageing management programme and interdisciplinary ageing management team, refer to IAEA Safety Report Series, "Implementation and Review of Nuclear Power Plant Ageing Management Programme" [8.1].

A comprehensive understanding of RPVs, their ageing degradation and the effect of the degradation on the ability of the RPV to perform their design functions is a fundamental basis for an ageing management programme. This understanding is derived from the knowledge of the design basis (including the applicable codes and regulatory requirements), the operating and maintenance history (including commissioning and surveillance), the inspection results, and generic operating experience and research results. Sections 1.1, 2, 3and 4 contain information on important aspects of the understanding of RPVs and their ageing.

In order to maintain the integrity or fitness for service of RPV, it is necessary to control within defined limits the ageing related degradation of the RPV. Effective degradation control is achieved through a systematic ageing management process consisting of the following ageing management tasks, based on understanding of RPV ageing:

- operation within specified operating conditions aimed at minimizing the rate of degradation
- inspection and monitoring consistent with requirements aimed at timely detection and characterization of any degradation and validating the ageing prediction; (Section 5)
- assessment of the expected or observed degradation in accordance with appropriate guidelines to determine integrity (Section 6); and
- maintenance, (repair or replacement) to correct or eliminate unacceptable degradation-managing ageing effects. (Section 5)

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 RPV safety margins are compromised. This programme reflects the level of understanding of the RPV ageing, the available technology, the regulatory licensing requirements, and the plant life management consideration/objectives. Timely feedback of RPV ageing degradation and measurement of the effectiveness of the ageing management programme should be implemented. 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. 8-1 and discussed in Section 8.1 below. Application guidance is provided in Section 8.2.

8.1. Key elements of the ageing management programme

8.1.1. Understanding ageing

Understanding RPV ageing is the key to effective management of RPV ageing, i.e. it is the key to: integrating ageing management activities within a systematic ageing management programme, managing ageing mechanisms through prudent operating procedures and practices (in accordance with technical specifications); detecting and assessing ageing effects through effective inspection, monitoring, and assessment methods and managing effects using proven maintenance methods. This understanding consists of: a knowledge of RPV materials and material properties, stresses 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 continuously 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 manufacturers data (including material data), and the commissioning data (including pre-service inspection data). The RPV operating history includes the pressure-temperature records, number of transients, system chemistry records, records on material embrittlement from the surveillance programme and the ISI results. The RPV maintenance history includes 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 comparison 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 with similar operating histories, even if the RPV designs are different and (c) relevant research results. It should be noted that effective comparisons or correlation with external experience requires 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 up to date using feedback mechanisms provided, for example, by owner groups. External experience can also be used when considering the most appropriate inspection method, maintenance procedure, and technology.

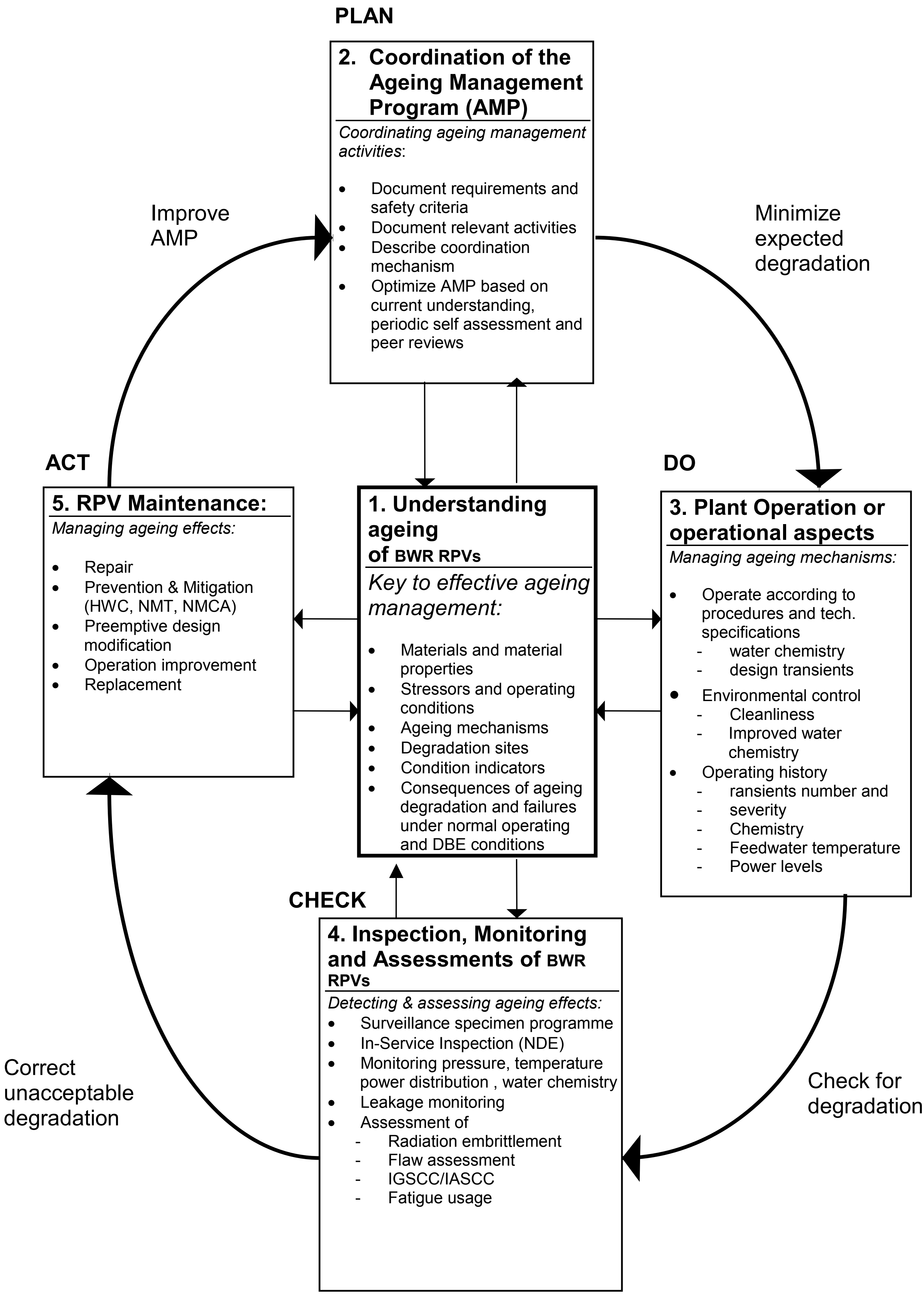

FIG.8.1. Key elements of BWR RPV ageing management programme utilizing the systematic ageing management process.

8.1.2. Coordination of the ageing management programme

Existing programmes relating to the management of RPV ageing include operations, surveillance and maintenance programmes, operating experience feedback, research and development and technical support programmes. Experience shows that ageing management effectiveness can be improved by co-ordinating 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 systems, structures, and components important to safety. The coordination 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 co-ordination 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-assessment and peer reviews.

8.1.3. Operation/use of reactor pressure vessel

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 programme 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:

- Assuring that water chemistry is always at the optimum
- Minimizing transients through sound operational practices
- Installation of on-line monitoring systems
- Immediately assess and address any anomalous observations
- Implement HWC
- Implement cleanliness and contaminant elimination program

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 SCC 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) in-service inspection and surveillance capsule testing, and (2) monitoring of pressures and temperatures, water chemistry, transients (relative to fatigue), 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 margins against brittle fracture. Transient monitoring provides realistic values of thermal stresses as opposed to design basis thermal stress values for fatigue assessments.

It is important to know the accuracy, sensitivity, reliability and adequacy of the nondestructive 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. Perform assessment of fatigue and IGSCC life based on plant history and materials. Based on this assessment implement actions to prolong life and assure safety margins.

8.1.5. RPV maintenance

The primary issue of BWR RPVs is the potential for cracking due to fatigue and SCC. Other problems that have occurred are damage due to personnel errors or tool failures during maintenance and introduction of foreign materials during operation due to failures of components in the RPV or the fluid system that is attached to it.

Detailed discussion of operation and maintenance controls is not in the scope of this document.

Thus the primary task becomes the prevention and mitigation of cracking. Assuming that the assessment recommended earlier has been completed, the following information is available:

- Detailed information relative to the materials and fabrication of the RPV and prediction based on that information of potential future problems,
- An assessment of the fatigue life of all critical locations including consideration of actual past operation and predicted future operation until end of planned life,
- A tabulation of problems that have occurred in other plants and an assessment of their applicability to the plant being addressed,

With this information, decisions can be made relative to pre-emptive repair or replacement, enhanced inspection or more detailed analysis to increase margins.

This programme combined with optimum water chemistry and implementation of online SCC monitoring should assure meeting life objectives.

Example of an AMP in Switzerland

Switzerland has an overall Ageing Management Programme in place which is harmonized between all reactor operators in the country and is split into 3 groups each for mechanical, electrical and civil SSC (Systems, Structures and Components). In the mechanical component ageing group, SSC's are subdivided into Safety classes 1, 2 and 3 and analyzed in detail. Special AM reports are being established for each component. [8.2–8.4]

Fig. 8-2 shows the concept for the ageing management of the RPV in KKM, a 355 MWe BWR in Switzerland, as an example.

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

While this issue is generally not a BWR concern, it is prudent to assure that the plant specific assessment of leak tests and hydro tests is based on the latest data and requirements and that the planned sample removal schedule is appropriate.

8.2.2. Stress corrosion cracking

This is probably the most critical issue. The first line of defense is water chemistry. There are techniques available to lower the aggressiveness of BWR coolant. A way to improve results is an integrated water chemistry program with hydrogen and noble metal chemical addition . (see chapter 7). The program should be complimented by installation of on line SCC monitoring, both to confirm effectiveness and to provide early warning of any future problems.

To some extent SCC and fatigue are not separable and any assessment of flaws initiated by SCC must include fatigue crack growth and cracks initiated by fatigue must be assessed considering SCC crack growth.

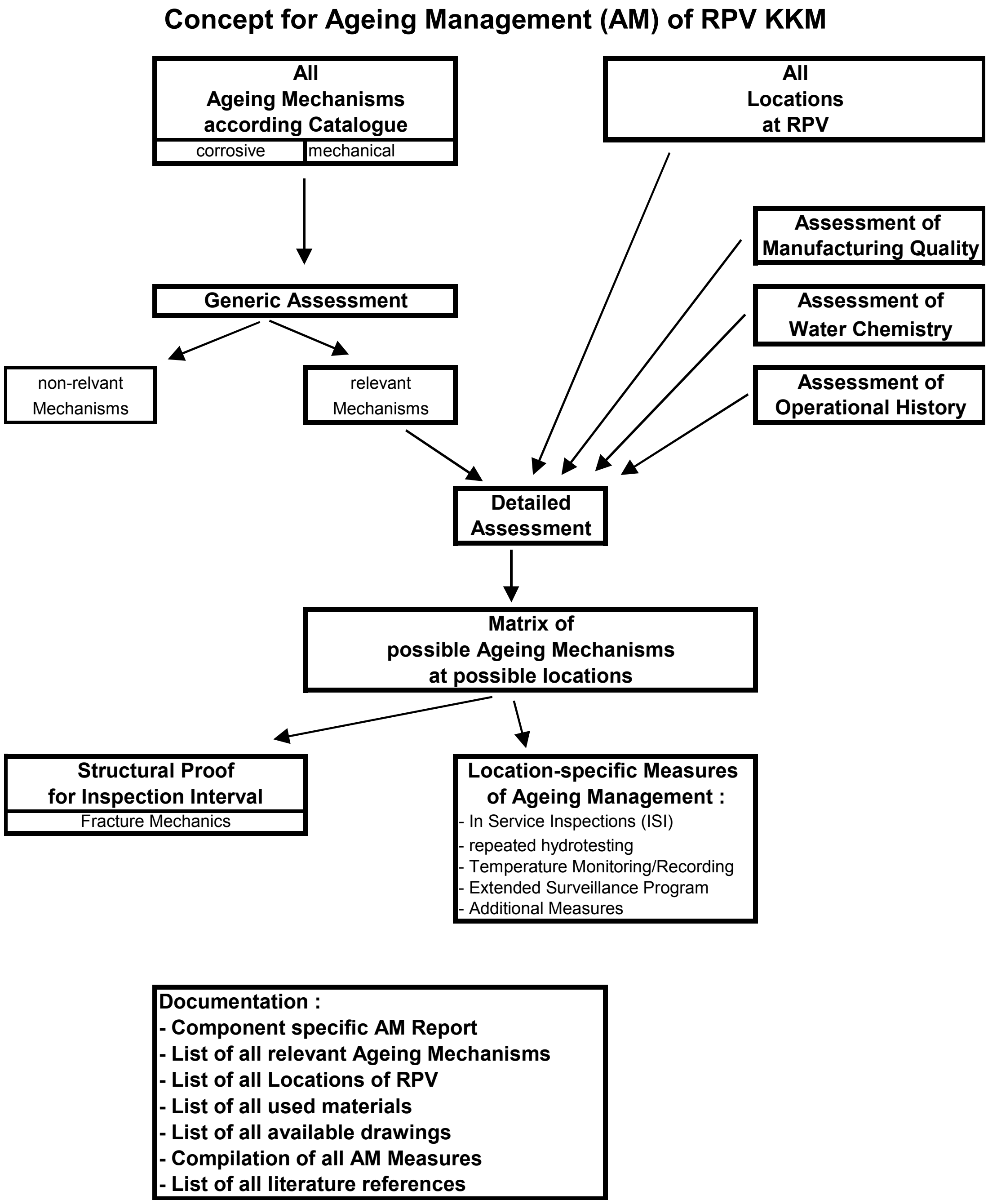

FIG. 8-2 Ageing management concept of a BWR RPV in Switzerland in the framework of the Swiss overall ageing management program.

8.2.3. 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 (Sa) versus number of cycles (N) curves given in the ASME Section III NB3000 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.

(a) Analysis — An updated analysis that reflects actual past plant operation and projected future operation based on the past 2-3 years of operation will provide the basis for ISI planning and assessment of ISI results. If low or inadequate margins are determined to exist, the use of more realistic transients may provide a solution. Analysis must meet the requirements of the appropriate codes for the country in which the plant is located. The stress vs allowable cycles curves used for the analysis might be adjusted for the environmental effects.

Where existing defects are present, the analysis should assess the crack growth using LEFM and appropriate da/dN correlations.

Some plants also prepare or have prepared by others allowable flaw size analyses of critical areas for use in accepting indications that may be found by ISI.

(b) Transient monitoring — The design input 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 should 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) Sampling of flaws — when a flaw is detected, especially if in an area where such flaws would not be anticipated, consideration should be given to removing a boat sample containing the flaw. If possible the sample should include the crack tip. Evaluation of the sample can provide information relative to the cause of the flaw as well as the condition of the material.

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ABBREVIATIONS

ABWR	advanced boiling water reactor
AERB	Atomic Energy Regulatory Board (India)
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Material
BWR	boiling water reactor
BWR VIP	boiling water reactor vessel and internals programmes
CRD	control rod drive
CRDRL	control rod drive return line
CVN	Charpy V-notch
ECCS	emergency core cooling system
ECP	electrochemical corrosion potential
FMCRD	fine motion control rod drive
FSARs	final safety analysis reports
HFIR	high flux isotope reactor
HSST	heavy section steel technology
HWC	hydrogen water chemistry
IASCC	irradiation assisted stress corrosion cracking
IGSCC	inter granular stress corrosion cracking
ISI	in-service inspection
ISP	integrated surveillance programme
JSME	Japan Society of Mechanical Engineers
KTA	Nuclear Safety Standard Commission (Germany)
LAS	low alloy steel
LEFM	linear elastic fracture mechanics
LP	laser peening
METI	Ministry of Economy, Trade and Industry (Japan)
MS& I	maintenance, surveillance and inspection
NDE	non destructive examination
NMCA	noble metal chemical application
NSSS	nuclear steam supply system
NUSS	(IAEA) Nuclear Safety Standards

NWC	normal water chemistry
PDA	performance demonstration administrator
PDI	performance demonstration initiative
PWHT	post weld heat treatment
RPV	reactor pressure vessel
RPVIs	reactor pressure vessel internals
QA	quality assurance
RT	reference temperature
SCC	stress corrosion cracking
SP	shot peening
SSCs	systems, structures and components
USE	upper shelf energy
WJP	water jet peening

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Technical Committee Meetings

Vienna, Austria: 5–9 September 1994

Vienna, Austria: 30 June–3 July 2003