

IAEA Nuclear Energy Series

No. NF-T-2.1

Basic
Principles

Objectives

Guides

Technical
Reports

Review of Fuel Failures in Water Cooled Reactors



IAEA

International Atomic Energy Agency

IAEA NUCLEAR ENERGY SERIES PUBLICATIONS

STRUCTURE OF THE IAEA NUCLEAR ENERGY SERIES

Under the terms of Articles III.A and VIII.C of its Statute, the IAEA is authorized to foster the exchange of scientific and technical information on the peaceful uses of atomic energy. The publications in the **IAEA Nuclear Energy Series** provide information in the areas of nuclear power, nuclear fuel cycle, radioactive waste management and decommissioning, and on general issues that are relevant to all of the above mentioned areas. The structure of the IAEA Nuclear Energy Series comprises three levels: **1 — Basic Principles and Objectives; 2 — Guides; and 3 — Technical Reports.**

The **Nuclear Energy Basic Principles** publication describes the rationale and vision for the peaceful uses of nuclear energy.

Nuclear Energy Series Objectives publications explain the expectations to be met in various areas at different stages of implementation.

Nuclear Energy Series Guides provide high level guidance on how to achieve the objectives related to the various topics and areas involving the peaceful uses of nuclear energy.

Nuclear Energy Series Technical Reports provide additional, more detailed, information on activities related to the various areas dealt with in the IAEA Nuclear Energy Series.

The IAEA Nuclear Energy Series publications are coded as follows: **NG** — general; **NP** — nuclear power; **NF** — nuclear fuel; **NW** — radioactive waste management and decommissioning. In addition, the publications are available in English on the IAEA's Internet site:

<http://www.iaea.org/Publications/index.html>

For further information, please contact the IAEA at PO Box 100, Vienna International Centre, 1400 Vienna, Austria.

All users of the IAEA Nuclear Energy Series publications are invited to inform the IAEA of experience in their use for the purpose of ensuring that they continue to meet user needs. Information may be provided via the IAEA Internet site, by post, at the address given above, or by email to Official.Mail@iaea.org.

REVIEW OF FUEL FAILURES IN WATER COOLED REACTORS

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN	GHANA	NORWAY
ALBANIA	GREECE	OMAN
ALGERIA	GUATEMALA	PAKISTAN
ANGOLA	HAITI	PALAU
ARGENTINA	HOLY SEE	PANAMA
ARMENIA	HONDURAS	PARAGUAY
AUSTRALIA	HUNGARY	PERU
AUSTRIA	ICELAND	PHILIPPINES
AZERBAIJAN	INDIA	POLAND
BAHRAIN	INDONESIA	PORTUGAL
BANGLADESH	IRAN, ISLAMIC REPUBLIC OF	QATAR
BELARUS	IRAQ	REPUBLIC OF MOLDOVA
BELGIUM	IRELAND	ROMANIA
BELIZE	ISRAEL	RUSSIAN FEDERATION
BENIN	ITALY	SAUDI ARABIA
BOLIVIA	JAMAICA	SENEGAL
BOSNIA AND HERZEGOVINA	JAPAN	SERBIA
BOTSWANA	JORDAN	SEYCHELLES
BRAZIL	KAZAKHSTAN	SIERRA LEONE
BULGARIA	KENYA	SINGAPORE
BURKINA FASO	KOREA, REPUBLIC OF	SLOVAKIA
BURUNDI	KUWAIT	SLOVENIA
CAMBODIA	KYRGYZSTAN	SOUTH AFRICA
CAMEROON	LATVIA	SPAIN
CANADA	LEBANON	SRI LANKA
CENTRAL AFRICAN REPUBLIC	LESOTHO	SUDAN
CHAD	LIBERIA	SWEDEN
CHILE	LIBYAN ARAB JAMAHIRIYA	SWITZERLAND
CHINA	LIECHTENSTEIN	SYRIAN ARAB REPUBLIC
COLOMBIA	LITHUANIA	TAJIKISTAN
CONGO	LUXEMBOURG	THAILAND
COSTA RICA	MADAGASCAR	THE FORMER YUGOSLAV REPUBLIC OF MACEDONIA
CÔTE D'IVOIRE	MALAWI	TUNISIA
CROATIA	MALAYSIA	TURKEY
CUBA	MALI	UGANDA
CYPRUS	MALTA	UKRAINE
CZECH REPUBLIC	MARSHALL ISLANDS	UNITED ARAB EMIRATES
DEMOCRATIC REPUBLIC OF THE CONGO	MAURITANIA	UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
DENMARK	MAURITIUS	UNITED REPUBLIC OF TANZANIA
DOMINICAN REPUBLIC	MEXICO	UNITED STATES OF AMERICA
ECUADOR	MONACO	URUGUAY
EGYPT	MONGOLIA	UZBEKISTAN
EL SALVADOR	MONTENEGRO	VENEZUELA
ERITREA	MOROCCO	VIETNAM
ESTONIA	MOZAMBIQUE	YEMEN
ETHIOPIA	MYANMAR	ZAMBIA
FINLAND	NAMIBIA	ZIMBABWE
FRANCE	NEPAL	
GABON	NETHERLANDS	
GEORGIA	NEW ZEALAND	
GERMANY	NICARAGUA	
	NIGER	
	NIGERIA	

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

IAEA NUCLEAR ENERGY SERIES No. NF-T-2.1

REVIEW OF FUEL FAILURES IN WATER COOLED REACTORS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2010

COPYRIGHT NOTICE

All IAEA scientific and technical publications are protected by the terms of the Universal Copyright Convention as adopted in 1952 (Berne) and as revised in 1972 (Paris). The copyright has since been extended by the World Intellectual Property Organization (Geneva) to include electronic and virtual intellectual property. Permission to use whole or parts of texts contained in IAEA publications in printed or electronic form must be obtained and is usually subject to royalty agreements. Proposals for non-commercial reproductions and translations are welcomed and considered on a case-by-case basis. Enquiries should be addressed to the IAEA Publishing Section at:

Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 2600 29302
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
<http://www.iaea.org/books>

© IAEA, 2010

Printed by the IAEA in Austria

June 2010

STI/PUB/1445

IAEA Library Cataloguing in Publication Data

Review of fuel failures in water cooled reactors. — Vienna : International Atomic Energy Agency, 2010.

p. ; 29 cm. — (IAEA nuclear energy series, ISSN 1995-7807; no. NF-T-2.1)

STI/PUB/1445

ISBN 978-92-0-102610-1

Includes bibliographical references.

1. Nuclear power plants — Safety measures. 2. Nuclear power plants — Management. I. International Atomic Energy Agency. II. Series.

IAEAL

10-00635

FOREWORD

In spite of the low fuel failure rate in currently operating water cooled nuclear power reactors, there is a continued high level of interest in fuel failures, for two reasons. First, the problems and inconvenience caused by fuel failures in plant operations can still be significant. Second, the generally accepted goal of achieving a zero failure rate requires detailed knowledge of existing failure mechanisms, their root causes and remedies.

Against this background, and following a recommendation by the Technical Working Group on Fuel Performance and Technology (TWGFPT), the IAEA decided to update an earlier study on fuel failures, Review of Fuel Failures in Water Cooled Reactors (Technical Reports Series No. 388), study data from different water cooled reactor types, including LWR, PWR, BWR, WWER and heavy water CANDU/PHWR types, and to gather as much experience as possible. Fuel failure statistics were derived from a questionnaire distributed to all TWGFPT members and through analysis of available publications. This review provides information on all aspects of fuel failures in current nuclear power plant operations.

The IAEA is grateful to all of the experts who contributed to the present study.

The IAEA officer responsible for this publication was V. Inozemtsev of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

SUMMARY	1
1. INTRODUCTION	3
1.1. Background.....	3
1.2. Objective.....	3
1.3. Scope.....	3
1.4. Structure	4
References to Section 1	4
2. EVOLUTION OF FUEL OPERATING ENVIRONMENT AND FUEL DESIGN CHANGES	5
2.1. Evolution of fuel operating environment	5
2.1.1. Economic drivers	5
2.1.2. Safety	7
2.1.3. Water chemistry	8
2.2. Fuel design evolution	8
2.2.1. BWRs	9
2.2.2. PWRs	9
2.2.3. WWERs	10
2.2.4. CANDU/PHWRs	10
2.3. Fuel types in operation	11
References to Section 2	12
3. WORLD OVERVIEW OF FUEL FAILURES FROM 1994 TO 2006	13
3.1. Introduction.....	13
3.2. Coolant activity levels.....	13
3.3. Methodology of fuel failure rate evaluation.....	13
3.3.1. Fuel rod failure rate calculation methodology used in Technical Reports Series No. 388.....	13
3.3.2. Fuel failure rate calculation methodology used for the present review (Part 1).....	14
3.4. Collection and treatment of fuel failure data	15
3.4.1. Involvement of the IAEA Technical Working Group on Water Reactor Fuel Performance and Technology (TWGFPT)	15
3.4.2. Questionnaire	15
3.4.3. Completeness and quality of data sets	16
3.4.4. Fuel failure rate calculation methodology used for present review (Part 2).....	17
3.5. Evaluation of PWR fuel ‘leakers’	20
3.5.1. PWR fuel failure rates.....	20
3.5.2. Distribution of failure causes in PWRs.....	22
3.5.3. Number of reactors free of defect evolution.....	24
3.5.4. Major observations regarding PWR fuel failures	24
3.6. Evaluation of BWR fuel ‘leakers’.....	25
3.6.1. BWR fuel failure rates	25
3.6.2. Distribution of failure causes in BWRs	26
3.6.3. Number of reactors free of defect evolution.....	26
3.6.4. Major observations regarding BWR fuel failures	28

3.7.	Evaluation of WWER fuel ‘leakers’	28
3.7.1.	WWER fuel failure rates	28
3.7.2.	Distribution of failure causes for WWER fuel	29
3.7.3.	Number of reactors free of defect evolution	31
3.7.4.	Major observations regarding WWER fuel failures	31
3.8.	Evaluation of fuel ‘leakers’ in CANDUs/PHWRs	32
3.8.1.	CANDU/PHWR fuel failure rates	32
3.8.2.	Distribution of failure causes for CANDUs	33
3.8.3.	Number of reactors free of defect evolution	33
3.8.4.	Major observations regarding CANDU/PHWR fuel failures	34
3.9.	Multiple failure incidents/defect excursions	34
3.9.1.	Incidents of massive and significant fuel failures	34
3.9.2.	Impact of multiple failure incidents on failure statistics	34
3.10.	Fuel rod failure rates and fuel failure causes between 1987–2006	36
3.10.1.	Evolution of fuel failure rates between 1987–2006	36
3.10.2.	Causes of fuel rod failure from 1987 to 2006	37
3.11.	Concluding remarks	38
	References to Section 3	38
4.	DETECTION, EXAMINATION AND ANALYSIS OF FUEL FAILURES	40
4.1.	Introduction	40
4.2.	Evaluation of coolant activity	40
4.2.1.	Background	40
4.2.2.	Leaker diagnostics	42
4.2.3.	Leaker estimates	46
4.2.4.	Burnup estimates: The $^{134}\text{Cs}/^{137}\text{Cs}$ activity ratio	48
4.2.5.	UO ₂ /MOX fuel discrimination: The $^{85\text{m}}\text{Kr}/^{135}\text{Xe}$ activity ratio	49
4.3.	Localization of fuel failures	49
4.3.1.	Sipping procedures	49
4.3.2.	Locating and identifying failed fuel in CANDUs	52
4.3.3.	Locating and identifying failed fuel in WWERs	53
4.3.4.	Ultrasonic testing	53
4.3.5.	Flux tilting (BWRs)	54
4.4.	Fuel examination	54
4.4.1.	Poolside inspection techniques	54
4.4.2.	Hot cell examinations	56
4.5.	Fuel failure analysis	56
4.5.1.	Background	56
4.5.2.	Manufacturing and handling	57
4.5.3.	Operation and design	57
	References to Section 4	58
5.	MECHANISMS AND ROOT CAUSES OF FUEL FAILURE	60
5.1.	Introduction	60
5.2.	Grid to rod fretting	62
5.2.1.	General background	62
5.2.2.	Failure mechanism	64
5.2.3.	Corrective actions	67
5.3.	Fretting due to debris	69
5.3.1.	General background	69
5.3.2.	Debris characteristics	70
5.3.3.	Corrective actions	70

5.4.	Corrosion.	73
5.4.1.	General background	73
5.4.2.	Excessive corrosion.	73
5.4.3.	Crud induced localized corrosion (CILC).	75
5.4.4.	Crud induced corrosion.	76
5.4.5.	Corrosion failure caused by cladding susceptibility and irregular water chemistry	80
5.4.6.	Corrosion due to carbon containing crud	81
5.4.7.	Local concentration of corrosion hydrogen (suspect)	82
5.4.8.	Exceptional crud after plant decontamination, WWER-440	82
5.4.9.	Shadow corrosion (BWRs)	83
5.5.	Pellet-cladding interaction	84
5.5.1.	General background	84
5.5.2.	Failure mechanism	86
5.5.3.	PCI failure during abnormal transients	89
5.5.4.	Other potential defect causes	90
5.5.5.	Corrective actions	91
5.6.	Manufacturing defects	92
5.6.1.	General background	92
5.6.2.	Leaking end plugs	93
5.6.3.	Welding defects	93
5.6.4.	External hydriding due to Ni contamination of the cladding surface	95
5.6.5.	Cladding flaws	96
5.7.	Cross-flow/baffle jetting	96
5.7.1.	Fuel failure due to baffle jetting	96
5.7.2.	Corrective measures	97
5.8.	Primary hydriding	97
5.8.1.	General background	97
5.8.2.	Failure mechanisms.	98
5.9.	Delayed hydride cracking (DHC)	99
5.9.1.	Introduction.	99
5.9.2.	Observations	100
5.9.3.	Failure mechanisms.	102
	References to Section 5	107
6.	FUEL STRUCTURAL DAMAGE AND OTHER FUEL ASSEMBLY ISSUES.	112
6.1.	Introduction.	112
6.2.	Assembly bow.	112
6.2.1.	PWRs	112
6.2.2.	BWRs — Channel bow.	118
6.2.3.	WWERs	119
6.2.4.	CANDUs.	123
6.3.	Mechanical damage during handling	124
6.4.	Crud and axial offset anomalies (AOA)	125
6.5.	Other miscellaneous damage	126
6.5.1.	Hold-down spring screws fracture, PWRs	126
6.5.2.	Resonance vibration, Darlington 2	126
6.5.3.	Lower end cap fretting, Oskarshamn 1	126
6.5.4.	Hydrogenation of Zry guide thimbles, Ringhals 2	127
6.5.5.	Deformation of upper grid rim, WWER-440	127
6.5.6.	Displacement of spacer grids in WWER-1000 FAs with Zr based alloy spacer grids and guide tubes	127
	References to Section 6	128

7.	SECONDARY FUEL FAILURES	130
7.1.	Background	130
7.2.	Observations from experience	130
7.2.1.	BWRs	130
7.2.2.	PWRs	133
7.3.	Degradation characteristics	133
7.4.	Mechanisms	140
7.4.1.	Axial split	140
7.5.	Circumferential fractures	146
	References to Section 7	147
8.	FUEL FAILURE PREVENTION AND MANAGEMENT IN PLANT OPERATION	150
8.1.	Existing operating limits and recommended practices	150
8.1.1.	Coolant activity	150
8.1.2.	Guidelines for power changes and load following operation	150
8.1.3.	Optimizing the loading pattern	152
8.1.4.	Guidelines for prevention of severe degradation	152
8.1.5.	Water chemistry controls	153
8.1.6.	Fuel loading/unloading practices	156
8.1.7.	Foreign material exclusion	156
8.1.8.	Current practice for handling, storage and disposal of failed fuel	156
8.1.9.	Reuse of failed LWR fuel	157
8.2.	Improvement of quality during manufacturing	157
8.2.1.	Evolution of a quality management system	157
8.2.2.	Fuel failure experiences related to fabrication defects and QMS	160
8.2.3.	QMS and good practice for prevention of fuel failure related to fabrication defects	163
8.2.4.	Fuel reliability in Japan	163
	References to Section 8	164
9.	CONCLUSIONS	167
	ABBREVIATIONS	169
	CONTRIBUTORS TO DRAFTING AND REVIEW	173
	STRUCTURE OF THE IAEA NUCLEAR ENERGY SERIES	175

SUMMARY

This report provides an overview of fuel failures in water cooled reactors and presents in detail their mechanisms and mitigation measures. It updates an earlier publication, Review of Fuel Failures in Water Cooled Reactors (Technical Reports Series No. 388) published in 1998, and describes fuel failures which have occurred between 1994 and 2006. The reactors covered in this study are pressurized light water reactors (PWRs and Russian WWERs), BWRs and heavy water reactors (CANDUs/PHWRs).

To minimize operating costs and increase cost competitiveness, utilities are changing fuel designs and reactor operating conditions, including higher fuel burnups, longer cycles, increased enrichment and new water chemistry. This increase in 'fuel duty' and these environmental changes directly impact fuel performance, and new performance issues have appeared.

Fuel cladding is a key barrier in containing fission products and it is essential that this barrier is robust and remains intact. Fuel failure occurs when this barrier is degraded and breached. It contributes to increasing plant background radiation, which impacts planned outages and increases workers' exposure. It can also contribute to the release of radioactive fission products to the environment. Thus, fuel performance should be sufficient to limit radiological releases into the environment and to be able to cope with ALARA issues. Finally, reactor fuel failure does not create confidence in the nuclear power industry and can influence public acceptance of nuclear power generation. For all of these reasons, it is a general goal of modern nuclear utilities to operate with a core free of defects.

The current Publication provides statistical data on fuel failures and proposes a new methodology for failure rate calculation on the basis of discharged fuel, creating a more realistic assessment of fuel reliability compared to calculations made on the basis of full fuel inventories. Despite continuous upgrading of fuel materials, as well as design and quality assurance procedures implemented within the last decade, which were assumed to lead to a general improvement in fuel reliability, there remain oscillations in failure rates in most countries (the only exception being Japan, with a stable and very low failure rate). Fuel failure statistics appear to reveal a balance between incentives to operate fuels under more challenging conditions, which may increase failure probability, and the aspiration to have fully reliable fuels with 'zero failure rates'. This balance varies from country to country, and is dependent both on the achieved level of technological maturity and on the local perception of economically and publicly acceptable risk. The fuel rod failure rate in LWRs has been significantly (but not monotonically) reduced since 1987, on average to levels of 10^{-5} during the period 2003–2006. For CANDUs, the element failure rate has been near 10^{-5} and has remained low at this level over the reporting period, reaching 5×10^{-6} from 2003 to 2006. However, the fuel failure rate has not markedly decreased over the last decade; a relatively large number of failures still occurred in a few plants. Moreover, signs of an increasing number of failures were observed in the early 2000s.

Fuel failure analysis involves several steps: fuel failure assessment, localization of fuel failures, and identification of failure mechanisms. This report provides information on the different techniques used in each step of an analysis. Fuel assessment is based on monitoring the release of fission products from defective fuel rods into reactor coolant. Several computer models have been developed to obtain an evaluation of the number of failed fuel rods in a core. The use of inspection techniques to localize failed fuel and to identify root causes of fuel failures is the second step in fuel failure analysis. These inspections may be performed on-site or in a hot cell laboratory.

Concerning fuel failure mechanisms, it is interesting to see that a number of mechanisms identified many years ago are still active, mostly in combination with new contributing factors such as higher burnup and power ramps. According to the classification commonly used in literature, ten mechanisms for fuel rod failure are identified and described in detail. No new power reactor failure mechanisms have been observed during the reporting period of this review. However, during ramp tests a new failure mechanism was described, when a crack initiated at a massive hydride layer at the clad outer surface and propagated through the whole cladding thickness (delayed hydride cracking or DHC); this experimentally revealed mechanism could potentially limit high burnup operations in the future. Grid to rod fretting was the major mechanism of fuel failure for PWRs in the reporting period. The main root causes of these failures have been identified as insufficient fuel rod support in the assembly due to improper design and/or fabrication, fuel rod vibration due to fluid elastic instability caused by crossflow in the assembly, and flow induced assembly and rod vibration. Debris fretting continues to be a common mechanism for fuel failure in all types of power reactors. In PWRs, corrosion is uniform on the cladding surface; excessive

uniform corrosion leading to cladding failure is very rare under normal operating conditions. Observed failures were due either to abnormally high heat flux exceeding heat flux/burnup corrosion limits or to water chemistry problems leading to excessive crud deposits. In BWRs, crud induced localized corrosion (CILC) has progressively decreased with the use of heat treated cladding, which is more resistant to nodular corrosion. However, crud induced corrosion involving high nodular corrosion resistant cladding was more recently observed in several BWRs. In these cases, the corrosion mechanism seems to be connected to unusually aggressive water chemistry conditions. Pellet-cladding interaction (PCI) fuel failure has been significantly reduced, mainly as a result of the use of operating guidelines and changes in fuel design. Despite continuous upgrades in quality assurance procedures during manufacturing implemented within the last decade which are assumed to generally improve fuel reliability, some manufacturing defects, other than those leading to primary internal hydriding (which has been historically treated separately) have still been identified. The major types of defects discussed in this review are end plug defects, and several types of end plug weld deficiencies. PCI failures were also recently observed in some reactors due to pellet chips in the fuel-clad gap or the presence of missing pellet surface, associated with significant changes in the local fuel rod power level.

In the field, apart from traditional rod failures, new fuel assembly related issues have appeared, such as fuel assembly bow and its consequences, including incomplete control rod insertion (IRI), axial offset anomaly (AOA), and crud deposits on fuel. Handling damage affecting PWR fuel is also often related to assembly bow with the consequence of spacer damage during loading or offloading. An equal variety of incidents with fretting wear have been reported. These and other observed unexpected fuel issues leading to assembly damage — but not actual fuel failure — can seriously affect plant operations, and it is now clear that these concerns are of similar significance to fuel rod failures, at least for LWR fuel.

Degradation of failed fuel rods through secondary hydriding, e.g. due to blisters, holes, cracks, failed or separated end plugs, and rod fractures in an advanced stage, have always been observed in Zr alloy clad fuel. Significant secondary degradation has been a major issue for BWR fuel in cases when primary failures developed long axial splits. Axial splits became a major concern because of extremely high activity release and fuel washout, which in some cases forced utilities into early shutdown just to remove a leaker. Failures in PWR fuel have generally been less severe, although there have been infrequent cases of axial cracking. More recently, there has been an increased frequency in circumferential fractures in some BWR and PWR fuel types. The fractures generally appear at pellet to pellet interfaces, where the cladding is typically cooler than in adjacent areas, and hydride precipitates preferentially, thus forming hydride rings. No severe degradation of this type has been reported in WWERs or CANDUs.

Fuel failures have detrimental impacts on plant operating activities and costs, and increase workers' exposure. The last section of this report describes existing operating limits due to fuel failure, and offers recommendations for operating policies and practices required to prevent fuel failures and to minimize their effects or propagation. At the same time, broad efforts have been made by the industry to improve quality during manufacturing and to reduce defects.

1. INTRODUCTION

1.1. BACKGROUND

Water reactor fuel continues to perform well around the world. However, fuel failures still occur in all countries operating nuclear power reactors. The current fuel rod failure rate varies in different countries with an average around 10^{-5} , except in Japan where reactors have operated practically free of defects for more than ten years. Further improvements in fuel performance will require an increasingly deeper understanding of fuel in-pile behaviour and failure mechanisms, as well as implementation of advanced remedies at the design, manufacturing and operating levels.

Since the late 1970s, the IAEA has published several documents on the subject [1.1–1.5]. The present publication is an updated and modified version of Review of Fuel Failures in Water Cooled Reactors, published in 1998 [1.3].

1.2. OBJECTIVE

As with the previous edition, this review has a double objective: to disseminate fuel failure statistics internationally, and to present an in-depth analysis including mitigation measures.

An important difference from the earlier study is a new methodology of failure rate calculation made on the basis of the number of discharged (or freshly reloaded) assemblies (not of the whole fuel inventory in all operating reactors or the inventory in reactors with outage during a specific year), which allows statistics to reflect failure rate more realistically.

Another difference is the detailed consideration of all fuel damage situations, including not only traditional fuel rod failure leading to fission product release, but also other fuel assembly related issues affecting plant operations.

1.3. SCOPE

No comprehensive world review of fuel failures has been published since the IAEA's publication [1.3] covering the period 1987–1994. Nevertheless, more recent fuel performance and operating experiences related to individual reactor types, regions or vendors have been discussed at a number of international meetings and presented in separate papers.

The major sources of published information in this review are the following:

- Technical meeting on fuel failures in water reactors held in Bratislava, Slovakia, 17–21 June 2002 [1.4];
- Technical meeting on structural behaviour of fuel assemblies for water cooled reactors held in Cadarache, France, 22–26 November 2004 [1.5];
- International conferences on LWR fuel performance and TopFuel;
- International conferences on WWER fuel performance, modelling and experimental support;
- International conferences on CANDU fuel.

Other important data sources are national and international proceedings and publications, which are referenced in individual sections below.

In addition, the published information has been supplemented by extended contributions from the IAEA expert group and members of the Technical Working Group on Fuel Performance and Technology (TWGFPT), prepared specifically for this review as a basis for statistical calculations.

With respect to terminology, it should be noted that there are some differences between CANDU and LWR related definitions. In particular, for CANDUs, fuel element is used instead of fuel rod, bundle instead of assembly, and burnup values are given in MW·h/kg U instead of MW·d/kg U (or GWd/t U) owing to the lower exposure of

CANDU fuel. The term ‘defect excursion’ stands for a multiple failure event stemming from a common cause. These differences are maintained throughout this review.

Most information on fuel failure mechanisms given in earlier IAEA publications [1.1, 1.2] is still valid. Therefore, all relevant information from the previous edition [1.3] has been kept, with new emphasis on additional features and issues which have been experienced during the past decade.

1.4. STRUCTURE

The reactors covered in this study are pressurized light water reactors (PWR and the Russian WWER), BWRs and heavy water reactors (CANDU/PHWR). The abbreviation LWR (light water reactor) refers to the first three reactor types: PWR, WWER and BWR.

This publication covers a number of subjects that are addressed in different sections:

- Major changes in operation and design since the mid 1990s that can influence failure rates (Section 2);
- The world overview of statistical failure data (Section 3);
- Modern methods of failure detection, examination and analysis (Section 4);
- Failure mechanisms and root causes (Sections 5, 6, 7);
- Failure mitigation and prevention measures (Section 8).

REFERENCES TO SECTION 1

- [1.1] GARZAROLLI, F., VON JAN, R., STEHLE, H., The main causes of fuel element failure in water-cooled power reactors, *At. Energy Rev.* 17 1 (1979) 31–128.
- [1.2] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management (Proc. Tech. Comm. Mtg, Dimitrovgrad, 1992), IAEA-TECDOC-709, IAEA, Vienna (1993).
- [1.3] INTERNATIONAL ATOMIC ENERGY AGENCY, Review of Fuel Failures in Water Cooled Reactors, Technical Reports Series No. 388, IAEA, Vienna (1998).
- [1.4] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Failure in Water Reactors: Causes and Mitigation (Proc. Tech. Mtg Bratislava, 2002), IAEA-TECDOC-1345, IAEA, Vienna (2003)
- [1.5] INTERNATIONAL ATOMIC ENERGY AGENCY, Structural Behaviour of Fuel Assemblies for Water Cooled Reactors (Proc. Tech. Mtg Cadarache, France, 2004), IAEA-TECDOC-1454, IAEA, Vienna (2005).

2. EVOLUTION OF FUEL OPERATING ENVIRONMENT AND FUEL DESIGN CHANGES

2.1. EVOLUTION OF FUEL OPERATING ENVIRONMENT

2.1.1. Economic drivers

The most important determinant of nuclear power's future is cost competitiveness compared to other alternatives. Nuclear power plants have a 'front-loaded' cost structure, i.e. they are expensive to build and comparatively cheap to operate. There is, therefore, a strong economic incentive to maximize utilization of the asset. For plants with batch reloading, this has meant longer operational cycles and shorter outages; the load factor of modern nuclear plants is often over 95% and two year fuel cycles are becoming common. For plants with on-load refuelling capabilities, it has meant longer fuel dwell to ensure that the fuel route does not limit output. For all power plants there is also a need to minimize the amount of waste arising, resulting from limited on-site storage facilities or the cost of waste removal and treatment.

For nuclear fuel, this has meant a need to endure longer operational periods and demonstrate increased reliability; fuel failure in operations is expensive for an operator, particularly if it limits output or increases the duration of outages. Figure 2.1 shows how average cycle length has evolved for BWR and PWR plants in the United States of America [2.1].

The cost of nuclear fuel is a relatively small fraction of the overall cost of generating electricity in a nuclear power station, but it is an expensive item and thus there is also a simple driver to reduce the cost of fuel. Increasing the energy obtained from fuel incurs both additional costs and savings; costs come from the need to add additional enrichment and to manage high burnup waste material, and savings from the need to manufacture and dispose of fewer fuel assemblies. The choice of a fuel cycle, including for example recycling, also strongly impacts fuel costs. The balance of these costs and savings to date has favoured an increase in fuel burnup, though economic studies suggest that additional costs for further burnup increases may not be as favourable as in the past [2.2].

The effect of these drivers has been an increase in the average burnup of fuel used in all reactor types. The burnup, measured in gigawatt days per tonne of heavy metal (GW·d/t), is a measure of the energy extracted from a given weight of fuel. Figure 2.2 shows the trends in fuel discharge burnup since 1970, and it can be observed that

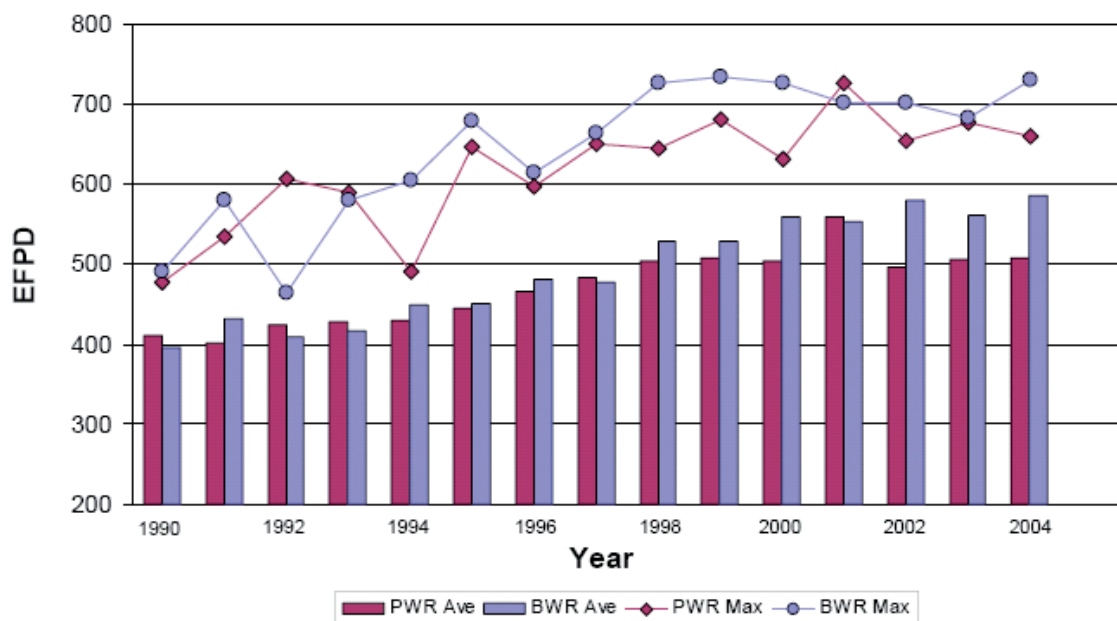


FIG. 2.1. Average cycle length in US reactors [2.1].

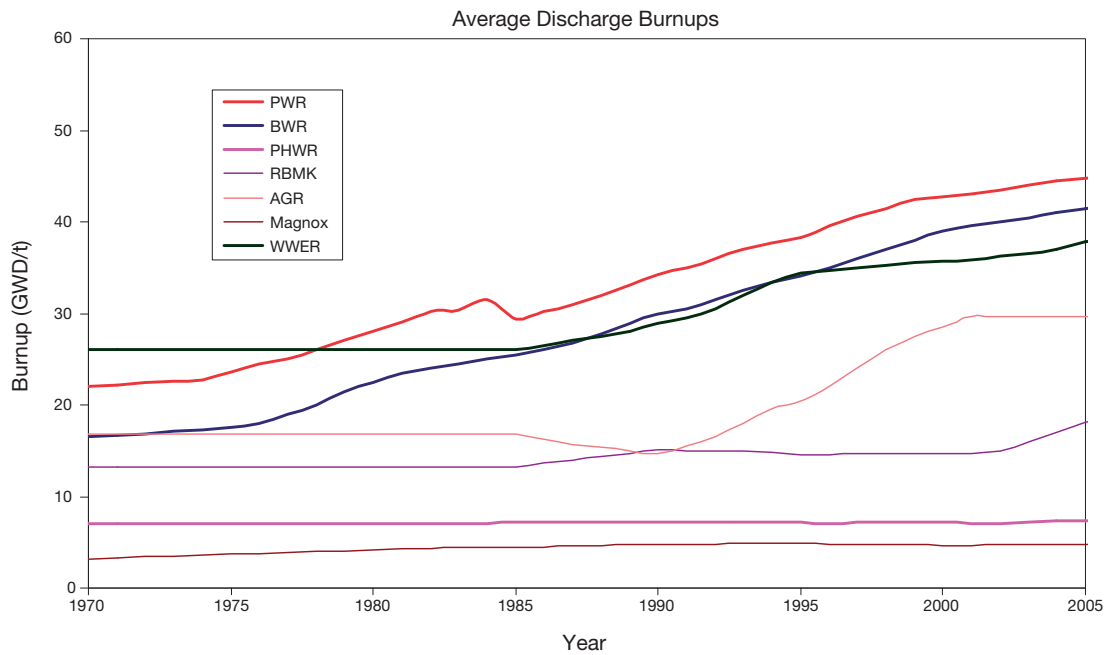


FIG. 2.2. Average discharge burnups [2.3].

the average burnup for light water reactor types has doubled. The main change implemented to achieve this increase in burnup has been to increase the ^{235}U enrichment of fuel, typically from 2.5% to around 4.5%, with a current maximum of 4.95%.

Burnup increase has been least marked in reactor types which use natural uranium as fuel, and while the two remaining Magnox stations are near final closure, PHWR vendors and reactor operators are investigating and starting to use slightly enriched uranium (SEU), which will enable them to double or triple average burnup.

It was noted above that the main change required in nuclear fuel to obtain high burnup is an increase in fuel enrichment, but that alone is not sufficient. Nuclear fuel operates in a demanding environment of high radiation fields, high temperatures and high coolant flow. Early fuel designs were adequate for initial low burnups, when fuel was in a reactor for a limited time; but as burnup has increased, it has been necessary to improve fuels in many ways. The properties of materials change with time when exposed to the intense radiation fields inside a reactor, and proper account of these changes must be made.

Another way to improve the cost competitiveness of nuclear power plants has been to increase their nominal power. Over the last years, numerous plants have increased their nominal power; about 5% for PWRs and up to 20% for BWRs. These new operating conditions have resulted in greater demands on fuel.

Fuel reliability also impacts the economics of nuclear generation. Indeed, fuel failures can increase plant operation costs in many areas, such as:

— *Losses of power production*

Severe fuel failures can sometimes create a need for mid-cycle outage to replace the failed fuel. Fuel failure can also impact outage duration due to time required for additional fuel inspection, potential delays to repair failed fuel, and necessary air or cooling activity treatment before containment entry and primary circuit opening. In BWRs, reactor power is generally reduced to perform power suppression testing by inserting control rods adjacent to the failed assemblies. Once the defect location has been identified, one or more controls rods may be permanently inserted to reduce local power in the region of the defect assembly and thereby suppress secondary degradation. Ramp rate limitations are also often imposed to limit additional degradation of failed fuel. Economic evaluation requires knowledge of the large spectrum of cost parameters impacted by fuel failures (see Refs 2.4, 2.5).

— *Fuel cycle costs*

Fuel inspection, repair or reconstitution increase fuel cycle costs. Failed fuel assemblies are repaired or definitively discharged according to the cost or complexity of repair. In this last case, they are replaced by fresh fuel assemblies or partially burned fuel assemblies stored in the spent fuel pit. These changes require core reanalysis, lead to non-optimized core design and result in a loss of fuel usage. Failed fuel assemblies or rods may also require specific storage conditions or canisters.

— *Increase in operational costs due to higher coolant activity levels and potential contamination of plant system and components*

Although the vast majority of workers' exposure stems from activation products, fuel failures can also contribute to increasing the component coming from fission products, see Ref. [2.6]. In particular, fuel failures increase potential hazards from gaseous iodine releases. There is also an increased risk of alpha contamination in cases of severe fuel degradation and fuel wash-out. In this area, the economic consequences of fuel failure are more severe in BWRs than in PWRs; steam generators in PWRs have a beneficial effect, acting as radiation barriers between primary and secondary circuits. In BWRs, severe secondary degradation may lead to unacceptable dose rates in the turbine building, forcing the utility to an anticipated shutdown of the reactor before the end of the cycle. Higher coolant activity also requires more extensive treatment of radioactive waste and adds cost to the waste disposal of resins, filters and evaporator concentrates.

These cost increases are difficult to quantify, and depend on many parameters, such as severity of the failures, number of failures, exposure of failed fuel assemblies, and impact on plant operation. Not all fuel failures have the same economic impact. For example, some failures occurring in third burned fuel assemblies at the end of a cycle can have a low impact, while a single failure occurring at the beginning of a cycle with severe fuel degradation and fuel wash-out can require a mid-cycle outage for fuel replacement and have a high adverse cost impact due to outage and production loss. The sensitivity of plant operating costs to these parameters varies greatly, but the major adverse effect on cost is due to events that result in a loss of energy production, such as derates or necessary shutdowns due to high coolant activity.

Some examples of costs related to fuel failures are given in Refs [2.1, 2.7]. A single failed BWR rod can cost more than US \$1 000 000 (2004) in outage time, fuel and power replacement costs. Failures affecting the larger fraction of a reload — such as crud or corrosion failures — can easily run into the tens of millions of dollars. In one recent example, a utility incurred more than \$50 000 000 in costs related to fuel failures.

Besides rod failures, fuel assembly related concerns such fuel assembly bow and its consequences on incomplete control rod insertion (IRI) or axial offset anomaly (AOA), can also have an adverse economic impact on nuclear power generation. In one case, a severe axial offset anomaly (AOA) caused one PWR unit to operate at reduced power for the final six months of an operating cycle. AOA has caused other units to operate with less efficient core designs at an expense ranging from \$500 000 to \$2 000 000 per fuel cycle [2.7]. In another potentially problematic area, the cost of BWR control blade interference due to channel bow can easily be in the millions of dollars (channel replacement costs, critical path time to rechannel, non-optimized core designs, etc.) [2.7].

2.1.2. Safety

Fuel cladding is a key barrier to containing fission products and it is essential that this barrier is robust and stays intact. Fuel failures represent degradation of this barrier. They contribute to increasing plant background radiation, which impacts plant outage and increases workers' exposure. They also contribute to the release of radioactive fission products to the environment. Fuel performance should be sufficient to limit radiological waste release to the environment, and to cope with ALARA issues. In addition, the presence of fuel failure in a reactor does not raise confidence in the nuclear power industry and can influence public acceptance of nuclear power.

Nuclear fuel should be sufficiently robust not only to allow it to operate normally without incident, but so that it can withstand any transient or accident that could occur in a plant in a manner that ensures safety is not compromised. This is supported through a licensing process that oversees not only operation, but also checks that

the design and manufacture of nuclear fuel is carried out to extremely high standards, and that design requirements are codified and performance is demonstrated experimentally.

In the 1970s, a large programme of experimental work was undertaken to demonstrate how fuel would behave under transient and accident conditions, and safety criteria were defined that provided limits on operation, so that fuel would not cause an unacceptable release of radioactivity. The intent was that in normal operation and frequent transient conditions, fuel cladding would not fail, and that under severe accident conditions any fuel failure could be contained and controlled by plant safety systems. Examples were the testing of fuel under transient high power conditions (power ramps) or under severe Loss of Coolant Accidents (LOCA) conditions. As burnup has increased, it has been necessary to demonstrate that changes in design or materials do not challenge the limits set by these safety criteria.

In response to the large number of incremental changes made to fuel since the early 1970s, the OECD/NEA initiated a review of fuel safety criteria for PWR and BWR plants, which was published in 2000 [2.8]. This review concluded that the current framework of fuel safety criteria remained generally unaffected by ‘new’ or modern design elements, but that the numerical values of design parameters may need to reflect the nature and properties of new materials or design elements, for example RIA/LOCA criteria. Following this review, the IAEA published a report in 2003 which takes into consideration differences in fuel safety criteria between WWER and PWR power plants [2.9]. This report noted differences in approach between East and West but was able to demonstrate that the two approaches were broadly similar and that there were no significant gaps in either approach.

The need to demonstrate compliance with safety requirements means that improvements to fuel design and operation must be carefully considered and implemented incrementally, with experimental demonstration — typically through the use of lead test assemblies — following extensive testing and research. The incremental approach to burnup extension has been a feature of nuclear fuel development as limitations on burnup extension have been identified and overcome.

2.1.3. Water chemistry

The relationship between fuel cladding and its operating environment, and in particular water chemistry, is very important. Changes in water can profoundly influence fuel oxidation rates and the migration of corrosion products from steam generators to the fuel, where they can be deposited as crud. A consequence of using fuels with higher enrichment and longer cycles is that the distribution of power in a reactor core becomes less uniform, and local power can rise within an assembly. This has led to deposition of crud from coolant at high power locations, causing power distortions and even fuel failure because of enhanced oxidation. This problem is being addressed through careful core design and control of water chemistry.

Recommended water chemistry has evolved over the years. For example, for United States of America PWRs, the diagram (Fig. 2.3) provides a simplified overview of the introduction of significant changes in water chemistry. These started with the introduction of lithium for pH control in the 1980s, zinc addition for steam generator corrosion control, and elevated pH and fuel assembly cleaning to help control crud. The influence of these changes must be monitored for any effect on fuel performance.

2.2. FUEL DESIGN EVOLUTION

The most significant development for LWR fuel has been a continuous increase in average discharge burnup to about 45MW·d/kgU, with further increases being planned and many plants operating with discharge burnups at around 60MW·d/kgU. Several CANDU programmes in progress involve new fuel cycles that will ultimately lead to burnups in excess of those achieved for natural uranium. Numerous modifications in fuel designs and materials have been made to adapt fuel performance to higher burnup goals or to improve neutron economy and simultaneously reduce failure risk. Some details are given below.

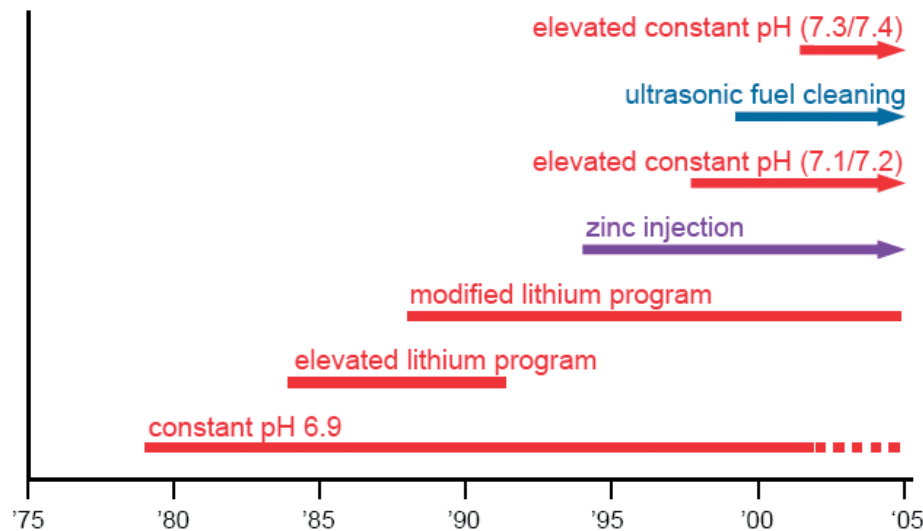


FIG. 2.3. Water chemistry changes in US PWRs [2.1].

2.2.1. BWRs

In BWR plants, there has been a steady evolution in fuel design, with the early 6×6 designs being steadily supplanted by 7×7 , 8×8 , 9×9 and, since the mid-1990s, 10×10 has become the standard except in Japan [2.10, 2.11]. Early failures due to pellet-cladding interaction (PCI) have been largely eliminated since the 1980s and liner fuel is standard.

The main failure mechanism in BWRs today is debris related and the industry is responding by using improved debris filters, coupled with an improved understanding of the mechanisms causing failure. It is common today for grids to be designed so that any small debris passing through the filter will not be trapped in the grid. This challenges grid designs, especially for 10×10 fuel, since there is also a drive to increase fuel loading, using larger diameter rods and reduced space in the lattice [2.12].

There continues to be isolated failures due to other mechanisms, such as PCI or corrosion effects. Minor changes in design have occurred and advanced corrosion resistant claddings are being considered. Empirical operating guidelines have been used to avoid the operating conditions which caused a few PCI failures in 2003, but the root cause of these remains uncertain.

2.2.2. PWRs

Since the early 1970s, many improvements have been made to PWR fuels to improve neutron economics and fuel burnup. Early changes led to reductions in fuel rod diameter and increases in the number of rods in a fuel assembly, and the standard design of today typically includes a 17×17 array of fuel rods with diameters ranging from 9.1–9.8 mm. The main improvements in the last 15 or so years include a trend to replace stainless steel components in assemblies, particularly grids, with zirconium alloys for neutron efficiency and changes in cladding alloys to improve fuel rod corrosion resistance, initially by optimizing the composition of Zr-4 alloys, and more recently through the introduction of Zr1%Nb alloys such as M5 (Areva) or Zirlo (Westinghouse). Duplex fuel, with a corrosion resistant zirconium alloy surface, is also being used.

Over the past 15 years, the main fuel failure mechanisms have remained debris fretting and grid to rod fretting, and assembly design has evolved to improve resistance to these problems. Debris resistance has been addressed through the introduction of advanced debris filters built into bottom nozzles (e.g. FUELGUARD, TRAPPER, Areva [2.13]) or through additional protective grids (such as the P-grid from Westinghouse) combined with long solid metal end plugs and protective oxide coating at the lower end of fuel rods. The change to zirconium based grids and the introduction of new grid designs, intended to improve the thermal-hydraulic performance of a fuel assembly, have occasionally led to significant grid to rod fretting with fuel failure. Grid designs have been

continually improved to reduce this problem. For example, no grid to rod fretting has yet been experienced with the Areva HTP fuel design.

Recent changes in assembly designs and materials have not led to new fuel failure mechanisms, but there have been new problems with assembly performance that have led to further design modifications. One major issue was the impact of assembly distortion during operation, which caused control rod friction inside the assemblies to increase, leading to occasional incomplete control rod insertion (IRI) during shutdown. The causes included high burnup assembly growth and cross assembly power tilts, and the remedy has been to increase the rigidity of guide tubes through increased wall thickness and detailed changes at the dashpot end of guide tubes, with designs such as MONOBLOC from Areva. Another issue was the failure of screws attaching springs to the top nozzles of some Westinghouse assemblies through stress corrosion cracking, which led to loss of hold down. Improved design of the top nozzle with a more robust method to retain the springs has now eliminated this problem.

The latest PWR fuel designs now offer a wide range of options, providing ease of use and flexibility in operation. Features such as intermediate flow mixing grids to improve thermal performance, integral poisons such as gadolinium to help achieve high burnup, removable top nozzles to facilitate repair and variable enrichment options within the assembly for optimum neutron efficiency are now readily available.

2.2.3. WWERs

WWER plants can be divided into the original version WWER-440 with channelled fuel assemblies containing 126 fuel rods, and the WWER-1000 series, in operation since the mid-1980s using unchannelled fuel (with one exception) and containing 312 fuel rods, both with hexagonal rod matrix and honeycomb type spacer grids. Fuel rods with 9.1 mm diameter Zr1Nb cladding and annular pellets are used in both types. Additional important differences in comparison to Western PWRs include the use of potassium–ammonia water chemistry, and reactivity control in the WWER-440 through special fuel assemblies containing boron steel tubes in the upper part.

In addition to the introduction of ZrNb alloy spacer grids in place of stainless steel (since 1987 in the WWER-440 and the mid 1990s in the WWER-1000), recent developments in WWER fuels comprise the use of advanced Russian Federation Zr1%Nb/Sn/Fe alloys with higher resistance to irradiation induced growth, creep and corrosion for guide tubes and for fuel rod cladding in applications with extended residence time (5–6 years).

During the past 10 years, WWER-1000 fuel assemblies have had similar issues to PWR assemblies, with assembly distortion and bow being a significant problem for both. Two different design solutions have been introduced; the TVS-A assembly includes a cage on the outside to provide structural rigidity, and the TVS-2 design uses thimble tubes of increased thickness, similar to the solution found for PWR assemblies. Both designs also include features such as debris filters and demountable top nozzles. Other changes being introduced to improve fuel burnup include advanced fuel pellet designs in which the central hole is reduced or eliminated to allow increased uranium loading and grid changes to improve thermal performance.

2.2.4. CANDU/PHWRs

CANDU reactors use strings of short fuel bundles (50 cm in length) in horizontal fuel channels and perform on-line refuelling. The fuel bundles have a circular geometry and elements with thin collapsible cladding, generally using graphite coating (CANLUB). No structural components such as spacer grids, support rods or end fittings are required, since the fuel elements themselves and the thin end plates serve as structural components. Bundle types with 19 and 28 elements (15 mm diameter) and with 37 fuel elements (13 mm diameter) are in operation. The 37 element bundle exists in two versions with small differences in end cap profiles and bearing pad positions to account for different fuel handling systems and channel configurations. (In Canada, the Bruce and Darlington reactors have ‘fuel against flow’ loading machines, the other plants have ‘fuel with flow’ machines.)

As the CANDU fuel types are mature products, their main design features have remained essentially unchanged. Nevertheless, extended technology programmes to further improve fuel performance and evaluate the use of advanced fuel (enriched uranium, recovered uranium from PWR rods, and thorium) are being carried out by AECL as well as through national programmes in India, the Republic of Korea, Argentina and Romania. The most recent fuel design is composed of a 43 element bundle with two different rod diameters, called CANFLEX. This design is at an advanced stage of development, being created through a joint AECL/KAERI programme.

2.3. FUEL TYPES IN OPERATION

An overview of the distribution of fuel types (in terms of rod lattice) for plants connected to the grid up to 2006 is given in Table 2.1.

The total number of LWRs connected to the grid by 2006 was 402 (399 by 31 December 2006) [2.14], with over 53% (214 plants) being PWRs. Currently, over 57% of BWRs operate with a 10×10 type core constitution. In operating PWRs, around 62% of the fuel has a 17×17 lattice owing to the large number of plants with this fuel type, which is built by Westinghouse, Areva and Mitsubishi. The most frequent fuel type in WWERs is a fuel assembly with 126 rods (WWER-440), but in terms of fuel rods, the fuel type with 312 rods (WWER-1000) makes an approximately equal contribution. In CANDUs the 37 element bundles dominate, since all large plants (600 MW(e) and above) use this design, while smaller units use either 28 element (Canada) or 19 element bundles (India and Pakistan).

TABLE 2.1. FUEL TYPES IN OPERATION

Type of plant	Fuel lattice	No. of plants in operation		Comments
		1993	2006	
BWR	6×6	1	—	Dodewaard (closed before 2005)
	7×7	2	2	Tarapur 1 + 2, derived from 6×6 fuel
	8×8	69	5	
	9×9	15	33	estimated >50% 9×9 fuel in core
	10×10	3	53	estimated >50% 10×10 fuel in core
	Others	1	—	Big Rock Point (closed before 2005)
Total BWR		91	93	
PWR	14×14	26	24	plants of several vendors
	15×15	29	26	plants of several vendors
	16×16	8	28	Siemens plants
	16×16	10		Westinghouse and CE plants
	17×17	122	133	plants of several vendors
	18×18	3	3	Siemens plants
Total PWR		197	214	
WWER	126 hex	26	27	WWER-440
	312 hex	19	27	WWER-1000
Total WWER		45	54	
CANDU/PHWR	19 circ	8	13	12 in India, 1 in Pakistan
	28 circ	8	6	Pickering plants in Canada
	37 circ	16	22	12 in Canada, 4 in the Republic of Korea, 2 in China, 2 in India, 1 each in Argentina and the Republic of Korea
Total CANDU/PHWR		32	41	
Overall total		365	402	

For BWRs and PWRs, the rapid introduction of new fuel designs and increasing diversification of fuel supplies to different vendors has led to many plants operating with mixed cores.

In 2006, 33 PWR and 2 BWR plants were operating with batch loadings of MOX fuel [2.14].

REFERENCES TO SECTION 2

- [2.1] YANG, R., et al., “Fuel R & D to Improve Fuel Reliability”, Water Reactor Fuel Performance Meeting, Kyoto, Japan, 2005.
- [2.2] NUCLEAR ENERGY AGENCY OF THE ORGANISATION FOR ECONOMIC COOPERATION AND DEVELOPMENT, Very High Burnups in Light Water Reactors (2006).
- [2.3] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Technology Review, IAEA, Vienna (2007).
- [2.4] EPSTEIN, R., SHEPPARD, K., STRASSER, A., “COFFEC: Cost of Fuel Failure Evaluation Code”, Report EPRI NP-5458-CCML, Vol. 1 August (1988).
- [2.5] STRASSER, A., SUNDERLAND, D., “A review of recent LWR fuel failures”, Proc. IAEA TM on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management, Dimitrovgrad, 1992, IAEA-TECDOC-709, IAEA, Vienna (1993) 17–25.
- [2.6] EPPERSON, K.R., YANG, R.L., “Economic evaluation of operation with defective fuel in PWRs”, Proc. Int. Topical Mtg on LWR Fuel Performance, Avignon, 1991, pp. 295–304.
- [2.7] YANG, R., et al., “An Integrated Approach to Maximizing Fuel Reliability”, Int. Mtg on LWR Fuel Performance, Orlando, 2004.
- [2.8] NUCLEAR ENERGY AGENCY OF THE ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, Fuel Safety Criteria Technical Review, CSNI/R (99) 25, OECD, Paris (2001).
- [2.9] INTERNATIONAL ATOMIC ENERGY AGENCY, Analysis of Differences in Fuel Safety Criteria for WWER and Western PWR Nuclear Power Plants, IAEA-TECDOC-1381, IAEA, Vienna (2003).
- [2.10] SCHARDT, J., “GNF2 Operating Experience”, Int. Mtg on LWR Fuel Performance/Top Fuel 2007, San Francisco, 2007, Paper 1087.
- [2.11] GARNER, N.L., RENTMEISTER, T., LIPPERT, H-J., MOLLARD, P., “Upgraded Fuel Assemblies for BWRs”, *ibid.*, Paper 1091.
- [2.12] RYTTERSSON, K., HELMERSSON, S., WRIGHT, J., HALLSTADIUS, L., “Westinghouse BWR Fuel Reliability — Recent Experience and Analyses”, *ibid.*, Paper 1035.
- [2.13] GOTTUSO, D., CANAT, J. N., MOLLARD, P., “High Mechanical Performance of AREVA Upgraded Fuel Assemblies for PWR in United States of America”, *ibid.*
- [2.14] ELECNUC, “Les centrales nucléaires dans le monde”, édition 2007, CEA.

3. WORLD OVERVIEW OF FUEL FAILURES FROM 1994 TO 2006

3.1. INTRODUCTION

This section attempts to give a world overview of fuel failure rates in water cooled reactors during the period 1994–2006. Cladding breaches lead to fission product release to coolant, and even to fuel washout in most severe cases. Such an event requires cleaning of the primary coolant circuits, and increases in outage time, and can have significant operating and financial impacts on affected plants. Other fuel failures without leaking rods or damage — e.g. excessive FA bow in PWRs and WWERs — which require corrective actions, may also become very costly for a utility and may offset cost savings from optimized designs [3.1]. Unfortunately, the latter category is still not quantitatively covered in literature and so the review of statistical failure data in this section is limited to failures with ‘leakers’. ‘Other’ failures are considered in Section 6.

Thus, the major purpose of this section is to provide a library of fuel failures so that any failure can be compared to previous experience, and to present fuel failure (leakage) and failure statistics.

3.2. COOLANT ACTIVITY LEVELS

Trends in fuel reliability are frequently characterized by the time evolution of coolant activity levels (mostly ^{131}I and noble gas) in plants or off-gas release rates for BWRs. This approach is sometimes used to demonstrate improvements over the years, and has been undertaken for example by several US PWR fuel vendors [3.2–3.4].

Beyond this, worldwide application of so-called ‘fuel reliability indicators’ used by utility organizations INPO (Institute of Nuclear Power Operations) and WANO (World Association of Nuclear Operators) has already taken place. These indicators are defined using steady state primary coolant ^{131}I activity for PWRs and combined steady state off-gas activity for BWRs, and are corrected for tramp uranium in both cases. Annual indicator values are evaluated and compared for different plants and countries respectively.

Another indicator used to trend fuel reliability is the number of reactors free of defect.

Although coolant activity levels do indicate gross differences in fuel behaviour and can be useful in several other aspects, they are not a good quantitative measure of the number of leaking fuel rods, as fission product release from a leaker may vary by factors of five or more, depending on the type, location and size of a failure as well as on the local linear heat generation rate (LHGR), and because tramp uranium can create additional uncertainties in spite of applied corrections. Therefore, coolant or off-gas activity data is not used for quantitative evaluation of failure rates given below, but the evolution of the number of defect free reactors will be given.

3.3. METHODOLOGY OF FUEL FAILURE RATE EVALUATION

3.3.1. Fuel rod failure rate calculation methodology used in Technical Reports Series No. 388

More accurate tracking of fuel reliability is aided by using the results of sipping in combination with fuel inspections and examinations. Sipping in the mast of fuel discharge machines (in the case of BWRs and PWRs) or in special canisters located in at-reactor pools (cladding integrity control in the case of WWERs) provides information on leaking FAs. In-pool ultrasonic testing (UT) can provide information on a number of leaking rods in an FA without requiring its dismantling, but is less frequently used. Hot cell post-irradiation examination (PIE) of defective FAs is very expensive and very limited. The rod failure rate is not very exact; for several regions or periods, the percentage of examined fuel was very small — the exact numbers of failed fuel rods are simply not known, although the number of failed assemblies is fairly accurately known. Some ‘straight’ data in literature relates to the number of leaking FAs, some only provide FR failure rates. When comparing literature data, one should be aware that different definitions for failure rates are used. Evaluation of fuel failure rates for water cooled reactors in Technical Reports Series No. 388 was made on the basis of published information supplemented by fuel failure information requested and received for the study from the following TWGFP Member States (see

Section 3.4.1): Canada, China, France, Germany, Japan, the Russian Federation, Sweden, Switzerland and the United States of America. To calculate the fuel rod failure rate, the following expression was used:

$$R \approx r \times D/N \quad (3.1)$$

where

- R is an annual rod failure rate (leaking rods/rods in core);
- N is the number of rods in core for all operating reactors with or without refuelling in the respective year;
- D is the number of leaking assemblies found and discharged from all operating reactors in a year;
- r is the average number of leaking rods per leaking assembly, equal to 1.1 for BWR, WWER-440 and CANDU fuel, and 1.3 for PWR and WWER-1000 fuel.

r values were selected on the basis of LWR experience in the United States of America [3.5]. Average rod failure rates over several years (and/or regions) were calculated by summing up annual (regional) values of D and N respectively, not by averaging annual (regional) R values.

3.3.2. Fuel failure rate calculation methodology used for the present review (part 1)

Technical Reports Series No. 388 covered fuel failure experience in water cooled reactors for the time period 1987–1994, while the present review is for 1994–2006. Data on fuel performance in 1994 were added to the present review to create succession between these two documents.

The principal difference between Technical Reports Series No. 388 and its revision lies in two aspects:

- Data for the revised version were collected by the IAEA using a questionnaire sent to all TWGFPT members, and only an insignificant part of literature data was used to fill in gaps;
- The basic fuel failure rate was defined as the ratio of failed assemblies upon discharge divided by the number of assemblies that are discharged, and normalized for 1000 discharged FAs. Two methodologies were used in the calculation of fuel rod failure rates in 1994–2006 — the above described ‘core’ methodology used in Technical Reports Series No. 388 and a ‘reload’ methodology described in detail in Section 3.4.4.

These changes were convened for several reasons. Literature data on fuel rod failures have been calculated by different authors in different manners, thus the number of failed rods have been divided either by: (1) rod core inventory of all operating reactors, or (2) only reactors with outages in the year, or, (3) by rod number in discharged batches. For the same fuel failure events, the fuel rod failure rate calculated using these three methods may differ by a factor of five or six. Calculation in one simplified case demonstrates this. For example: if there are four LWRs with 160 FAs/core each and 250 rods/FA, two reactors had outages during the year (discharge batch, 60 FAs each), and two operated without outages. During fuel reload, two FAs were identified as having leaks, one with leaking FRs in each FA. The annual FR failure rate calculated using the first method adds up to a rate equal to 12.5 ppm; the second results in 25 ppm and the third ~67 ppm. If the number of leaking FRs in a FA is unknown, it may increase the difference by a factor of 10 or more.

It is necessary to recognize the uncertainty of the fuel failure rate evaluation, which stems from the fact that some utilities and organizations prefer to use the relative number of failed FAs as a measure of fuel failure rate. These calculations can be also made in different manners, by dividing the number of failed FAs either on the FA core inventory of all operating reactors, or only on reactors with outages in the current year, or on FA numbers in discharged batches. For example, EPRI and EDF use the FA failure rate (number of FAs with failures/number of FAs in cores of reactors with outages this year) as a measure of the fuel failure rate. This method may underestimate the defect rate by a factor of about three to five (depending on the batch size of reload), because undischarged intact assemblies are counted as long as they stay in the reactor, but failed assemblies are counted only once (upon discharge).

It is clear that different methods of fuel failure rate evaluation used by different utilities, fuel vendors and organizations make fuel failure analysis and identification of general tendencies in fuel performance evaluation difficult. For this reason a uniform fuel failure evaluation methodology should be developed and applied.

A solution to this problem has been to collect data for each year on the number of failed FAs, the number of failed rods in these FAs, the respective sizes of fresh reload batches and the total number of FAs in cores of all operating reactors (with and without outages). Availability of such data allows flexibility in selection of the most appropriate calculation model and will provide an opportunity to recalculate old data and recalculate new data using the old 'core' methodology.

3.4. COLLECTION AND TREATMENT OF FUEL FAILURE DATA

3.4.1. Involvement of the IAEA Technical Working Group on Water Reactor Fuel Performance and Technology (TWGFPT)

Twenty-six IAEA Member States and two international organizations sent representatives to the IAEA TWGFPT, including 23 Member States (of 30) operating nuclear power plants (Argentina, Belgium, Brazil, Bulgaria, Canada, China, the Czech Republic, Finland, France, Germany, India, the Republic of Korea, Japan, the Netherlands, Romania, the Russian Federation, Slovakia, Spain, Sweden, Switzerland, UK, Ukraine and the United States of America). The functions of the TWGFPT include facilitation of information exchange on the status and perspectives of national, multilateral and international programmes, new technological developments and experience in areas related to nuclear power reactor fuels.

In accordance with this mission, the IAEA has been using the TWGFPT mechanism since 1995 to collect data on:

- Fuel failure experience;
- Burnup extension;
- MOX fuel use and performance.

Throughout the period 1995–2006, data on fuel failures were collected without a standard format, in particular without data on reloads. Recognizing the need for these data, a specially created IAEA Expert Group developed a questionnaire in February 2007 for TWGFPT members and other organizations in order to organize and receive data in a format that allowed for calculation of fuel failure rates as noted above. Among these organizations were: EPRI (United States of America), ENUSA (Spain) and EdF (France).

3.4.2. Questionnaire

The questionnaire, in EXCEL format, requested annual (1994, 1995 ... and 2006) data including:

- Plant name or number of plants reloaded during each year;
- Fuel cycle characteristics (number and duration);
- Number of leaking FAs and fuel rods;
- Failure type;
- Number of fresh FAs loaded;
- Total number of FAs in core/cores of reactors with and without refuelling in the respective year;
- Enrichment and burnup of failed FAs in discharged batches;
- Plants without fuel failures;
- Other damage without cladding breach (spacer grids, skeleton, FA bow, AOA etc.).

These data were requested for LWRs with fuel reload during each specific year and for CANDUs/PHWRs with on-line refuelling. For LWRs operating throughout the year without an outage, only information on plant name, cycle characteristics, enrichment and number of FAs in the core was requested.

To facilitate the task of respondents answering the questionnaire, it was recommended that an identifying number be used, e.g. PWR-1, BWR-3 or CANDU-2 instead of the plant name to allow confidentiality. If data were given for multiple plants, discrimination by reactor type was requested. For example, in the year 2000, 30 PWRs of 900 MW, 13 PWRs of 1300 MW and 4 PWRs of 1450 MW operated with outages in France (EdF data), while 42 PWRs and 19 BWRs operated with outages in the United States of America (EPRI data). Fuel failure data for these groups of plants were provided in summary ('integral form'). For CANDUs/PHWRs, the number of identified failed bundles and the total number of bundles loaded during the year was requested.

For 'failure type' and 'other damage', the following mechanisms were included in the questionnaire:

Mechanisms of fuel failure with rod leakage:

- Grid to rod fretting;
- Crud/corrosion;
- PCI-SCC — pellet cladding interaction/stress corrosion cracking (or EAC — environmentally assisted corrosion, a term used in Canada);
- Debris;
- Handling damage;
- Fabrication;
- Baffle jetting.

Other damage (without rod leakage, which is considered in Section 6):

- FA bow/incomplete control rod insertion;
- Axial offset anomaly;
- Spacer grid damage;
- FA fretting at buffer plate;
- Channel bow in BWRs.

A separate description of any major unusual or multiple failure events and examples of unusual or new failure mechanisms, if applicable, was also requested.

It is worthwhile to emphasize that presentation of data at the plant level is preferable, but if data was only available for a fleet of reactors, it was also acceptable.

It is also important to note that the IAEA's intention was not publication of a fuel failure database on a plant by plant basis. This review simply represents calculation results of fuel failure rates for countries with significant reactor fleets and regional fuel failure rates for other countries. In addition, the review presents only integrated calculation results, not original fuel failure tables submitted by countries, as was agreed upon with the data providers.

The receipt of fuel failure data sets from providers (TWGFPT members, representatives of EdF and EPRI) was planned for the 2007 TWGFPT plenary meeting. Most responses were received by this date, some arrived later. Only Bulgaria, China and the Netherlands did not send answers to the questionnaire. For Bulgaria, China and some non-TWGFPT countries, e.g. Armenia, some data were found in published reports. Completeness and quality of data sets received or found in literature are discussed below.

3.4.3. Completeness and quality of data sets

Between 1994–2006, 49 new NPP units were connected to the grid (21 PWRs, 6 BWRs, 8 WWER-1000s and 2 WWER-440s, 5 CANDUs and 7 PHWRs), and 18 units were permanently shutdown (7 PWRs, 6 BWRs and 5 WWER-440s) [3.6, 3.7]. By 31 December 2006, there were 399 nuclear power units in operation (213 PWRs, 93 BWRs, 51 WWERs and 42 CANDUs/PHWRs). Including the 18 units which were shut down, a total of 417 nuclear power units operated during the 1994–2006 period. Excluding the year when the unit was connected to the grid and the year of unit shutdown for LWRs, these 417 units accumulated 4881 years' experience with fuel reload. This means that the data received covered 93% of all reactors operating between 1994–2006 and 95.6% of reactor years of experience.

Fuel failure data are completely missing for the following countries: 6 PWRs (China), 1 PWR (the Netherlands), 1 PWR (Pakistan), 2 PWRs (South Africa), 1 PWR (Slovenia), 2 PWRs (Taiwan; not an IAEA Member State), 2 BWRs (Mexico) and 4 BWRs (Taiwan, China), 4 CANDUs (Canada) and 2 new PHWRs (India). No mechanism was found to obtain data from Mexico, Pakistan, South Africa and Slovenia, which are not members of the IAEA TWGFPT. RBMK-1000 type reactors (LWGRs) were not considered in Technical Reports Series No. 388 and are not covered by this review. With respect to reactor years of experience covered by the present review, data sets for water cooled power reactors from 1994–2006 cover 99% of PWRs (only data for the Quinshan-1 unit for 2002–2006 are missing), 100% for BWRs, CANDU/PHWRs, WWER-1000s and WWER-440/213, and 78% for WWER/230/179s (data for Armenia-2, Kola-2, Novovoronezh-3, 4 and Kozloduy-3, 4 are missing for 2003–2006).

The quality of presented data is not always uniform. Some countries did not provide information on identification of the causes of fuel failure, mostly due to missing PIE data in these countries. The Republic of Korea reported the causes of fuel failure throughout the period 1994–2006 without a link to specific years.

Completeness and quality of fuel failure data sets, as well as basic data on reactor years of experience, are provided in Table 3.1.

3.4.4. Fuel failure rate calculation methodology used for present review (part 2)

After collecting all tables (Table 3.1) and extracting and treating fuel failure data published in literature (finding the number of discharged FAs, core inventories and FR numbers in FAs), final excel tables were created for each country and year. This final table includes the following data for each country, year (from 1994 to 2006) and reactor type (PWRs or BWRs or WWERs or CANDUs/PHWRs):

- Number of units with and without outages during the specific year;
- Number of discharged FAs;
- Number of failed FAs and, if known, FRs;
- Total number of FAs in cores with and without outages;
- Average number of FRs in FAs.

Average FA failure rates over several years or/and regions are calculated using a similar approach to the previous Technical Reports Series No. 388. Numbers of discharged FAs and failed FAs are summed up and the total number of failed FAs are divided by the total number of discharged FAs. Number of failed FAs per 1000 discharged FAs is selected as the first key index of fuel reliability according to IAEA Expert Group recommendations.

The new ‘reload’ methodology of FR failure rate definition requires knowledge of the following data for each reactor (or group of reactors) per year of operation: the number of failed FAs and average number of failed FRs in an FA (multiplication provides the total number of leaking FRs in discharged batches) as well as the number of discharged FAs and average number of FRs in an FA (multiplication provides the total number of FRs in discharged batches). Dividing the total number of leaking FRs by the total number of FRs in discharged batches supplies the FR failure (leaker) rate. Average FR failure rates over several years or/and regions are calculated using an approach similar to that taken in Technical Reports Series No. 388.

Data on the size of discharge batches, the number of FAs in reactor cores and the number of failed FAs were received from data providers as Excel tables. Sometimes data were also provided on the number of FRs in discharge batches and in reactor cores. If not, calculations were made using data found in literature.

The average number of leaking FRs per FA was calculated for PWRs and BWRs using data received from Excel tables. For PWRs, it was 1.6 (an average from about 600 failed FAs) and for BWRs it was 1.1 (an average of 170 failed FAs) (see Fig. 3.1). Unfortunately, no statistical data of this type were received from WWER fuel vendors/reactor operators. In Technical Reports Series No. 388, the FR failure rate for WWER-1000s was 1.6 and 1.1 for WWER-440s.

TABLE 3.1. BASIC DATA ON REACTOR YEARS OF EXPERIENCE AND COMPLETENESS AND QUALITY OF FUEL FAILURE DATA USED IN THE REVIEW

Country	Units in operation 31 Dec. 2006	Units shutdown between 1994–2006	Reactor years of experience in 1994–2006 ¹	Number of units and years covered by the data received	Source	Quality and completeness of data sets
Argentina	2-PHWR ²		13	1/13	Q	B
Armenia	1-WWER		13	1/5	[3.8]	C
Belgium	7-PWR		91	7/91	Q	A
Brazil	2-PWR		19	2/19	Q	A
Bulgaria	2-WWER	2-WWER, 2002/12 2-WWER, 2006/12	66	6/60	[3.9–3.11]	C
Canada	18-PHWR		234	14/182	Q	A
China	7-PWR 2-PHWR 1-WWER		59	3/14	[3.12-3.13]	C
Taiwan, China ³	2-PWR 4-BWR			0/0		
Czech Rep.	6-WWER		62	6/62	Q [3.8, 3.14–3.17]	A
Finland	2-WWER 2-BWR		52	4/52	Q	A
France	58-PWR		737	58/737	Q	A
Germany	11-PWR 6-BWR	1 BWR, 1994/8 1 PWR, 2005/5 1 PWR, 2003/11	241	20/241	Q	A
Hungary	4-WWER		52	4/52	[3.6, 3.15] Q	A
India	2-BWR 14-PHWR		158	14/155	Q	B
Japan	23-PWR 32-BWR		660	55/660	Q	A
Korea, Rep. of	16-PWR 4-PHWR		193	20/193	Q	A
Mexico	2-BWR		24	0/0		
Netherlands	1-PWR	1 BWR, 1997/3	16	0/0		
Pakistan	1-PHWR 1-PWR		19	1/13	[3.16]	C
Romania	1-PHWR		11	1/11	Q	A
Russian Federation	15-WWER		176	15/164	Q [3.6, 3.10, 3.11, 3.14]	B

TABLE 3.1. BASIC DATA ON REACTOR YEARS OF EXPERIENCE AND COMPLETENESS AND QUALITY OF FUEL FAILURE DATA USED IN THE REVIEW (cont.)

Country	Units in operation 31 Dec. 2006	Units shutdown between 1994–2006	Reactor years of experience in 1994–2006 ¹	Number of units and years covered by the data received	Source	Quality and completeness of data sets
South Africa	2-PWR		26	0/0		
Slovakia	5-WWER	1 WWER, 2006/12	66	6/66	Q	A
Slovenia	1-PWR		13	0/0		
Spain	6-PWR 2-BWR	1 PWR, 2006/4	116	9/116	Q	A
Sweden	3-PWR 7-BWR	1 BWR, 1999/11 1 BWR, 2005/5	146	12/146	Q	A
Switzerland	3-PWR 2-BWR		65	5/65	Q	A
United Kingdom	1-PWR		11	1/11	Q	A
Ukraine	15-WWER		171	15/171	Q	A
United States of America	69-PWR 34-BWR	1 BWR, 1997/8 1 BWR, 1998/7 1 PWR, 1996/12 1 PWR, 1997/8 2 PWR, 1998/1	1356	108/1356	Q	A
Total	PWR-213 BWR-93 WWER-51 PHWR-42 All-399	PWR-7 BWR-6 WWER-5 PHWR-0 All-18	4866	388/4655	Q-21 P-4 Missing-5	A-18 B-3 C-4
	417					
Total in %				93.0/95.7		

¹ Years of connection to the grid and years of shutdown are not counted (no fuel reload); sum of b and c.

² PHWR Atucha-1 of specific MZFR design is not included in fuel failure statistics.

³ Units in Taiwan, China (2 PWRs and 4 BWRs) are not included in fuel failure statistics (not an IAEA Member State).

Abbreviations for Table 3.1:

A. Data sets are complete, received as Excel tables;

B. Data sets are not fully complete, received as Excel tables;

C. Some data were found in literature, no answers to the IAEA questionnaire received;

Q. Answer to the IAEA questionnaire, received as an Excel table;

P. Country for which data were only found in literature.

To be consistent with Technical Reports Series No. 388, the FR failure rate for LWRs was also calculated for 1994–2006 using the ‘old’ methodology described in Section 3.2.1. In this case, the number of failed FRs was divided by the total number of FRs in all units operated with and without outages. Of course, these values will be significantly smaller than those calculated using the ‘new’ methodology.

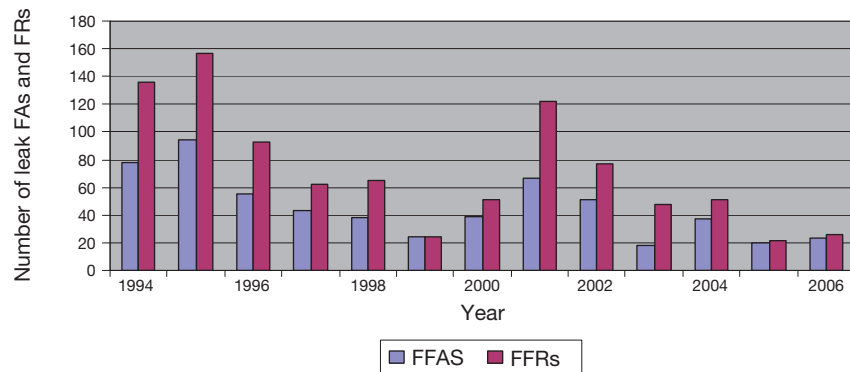


FIG 3.1. Ratio of FR/FA leaks in PWRs.

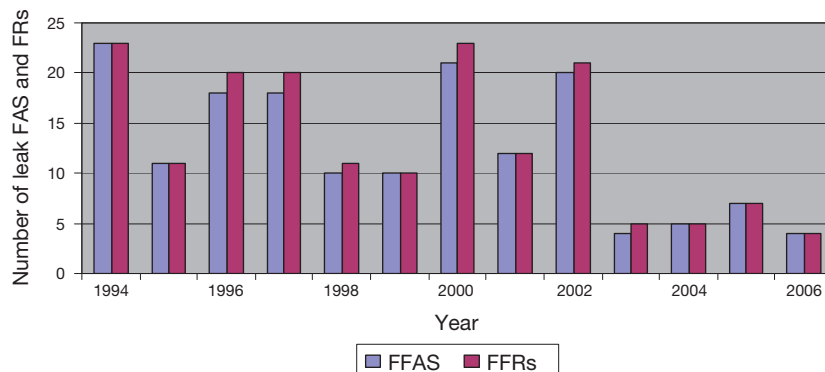


FIG 3.2. Ratio of FR/FA leaks in BWRs.

Regarding on-line reloading of CANDUs/PHWRs, ‘new’ and ‘old’ methodologies of fuel rate failure calculation are identical. AECL confirmed that the average number of leaking FRs per bundle was 1.1, the same number used in Technical Reports Series No. 388.

Below, the calculation of fuel failure (leak) rates is given as an example for LWRs in Japan in 2006 (Table 3.2).

3.5. EVALUATION OF PWR FUEL ‘LEAKERS’

3.5.1. PWR fuel failure rates

FA failure rates for the United States of America (69 units), France (58 units), and the worldwide average are presented in Fig. 3.3, and those for Europe, with the exception of France (31 units), Japan (23 units) and the Republic of Korea (16 units) are presented in Fig. 3.4. The world curve in Fig. 3.3 shows the average result for countries and region (Europe–France), including Brazil and one PWR from China, thus summarizing fuel performance for 95% of PWR units worldwide.

Figures 3.3 and 3.4 show that there is an overall downward trend number of fuel leaks, with the exception of two increases. These are observed in 1995 and 2001 for PWRs in Europe, including France, the Republic of Korea, the United States of America and worldwide. Only Japan has a leak level of practically zero for the entire period 1994–2006. Average FA failure rates (number of failed FAs per 1000 discharged FAs) for 1994–2006 are:

- World average — 13.8
- United States of America — 20.9
- France — 8.8
- Europe-France (Belgium, Germany, Spain, Sweden, Switzerland, UK) — 16.0
- Japan — 0.5
- Republic of Korea — 10.6

TABLE 3.2. CALCULATION OF FUEL FAILURE RATES IN JAPAN IN 2006 (EXAMPLE)

Type	PWR	BWR
Units	23	32
Number of FAs in all cores	3579	20828
Number of FAs in reloads	816	1952
Number of FRs in all cores	834 316	1 479 602
Number of FRs in reloads	197 394	144 112
Number of failed FAs	2	1
Number of failed FAs per 1000 discharged FAs (FA failure rate)	2.4	0.5
Number of leaking FRs, new 'reload' model	3.2	1.1
Number of leaking FRs, old 'core' model	2.6	1.1
FR failure rate, ppm, new 'reload' model	16.2	7.6
FR failure rate, ppm, old 'core' model	3.1	0.7

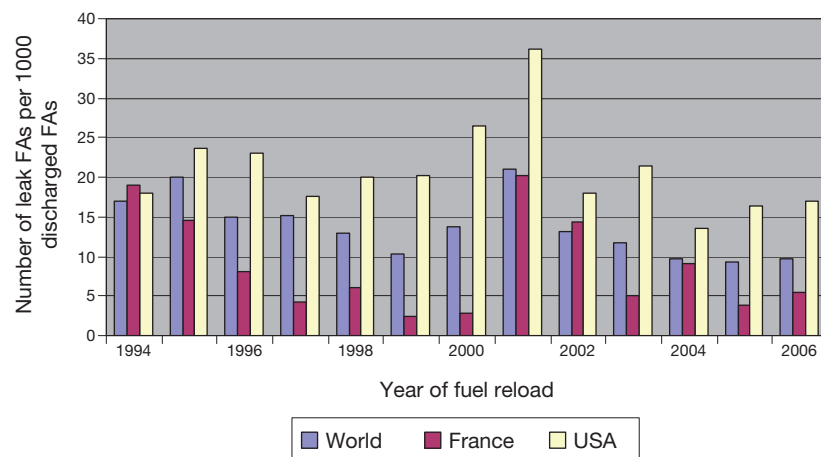


FIG. 3.3. PWR FA leak rate for France, the United States of America and worldwide.

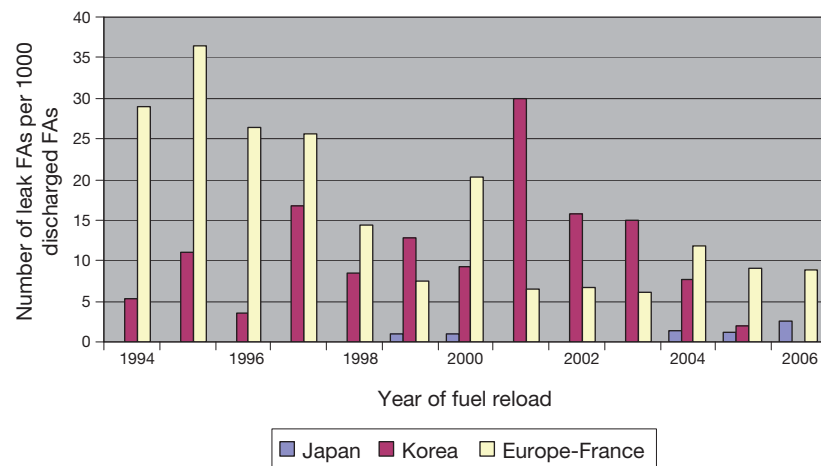


FIG. 3.4. PWR FA leak rate for Europe with the exception of France, Japan and Republic of Korea.

The world average PWR fuel rod failure rates calculated for reload batches (using the ‘new’ methodology proposed for present study) and for core inventories (using the ‘old’ methodology in Technical Reports Series No. 388) are presented in Fig. 3.5.

By definition, the curves in Fig. 3.5 look similar, with the same tendency in rises and declines. However, the fuel rod leak rate is lower for the last three years reviewed (2004–2006) in comparison to the whole time period 1994–2006. Average fuel rod leaker rate values for 1994–2006 are given in Table 3.3. This table indicates that rates calculated for reloads are around five times higher than those calculated for core inventories. This fivefold increase in the fuel rod failure rate derives from the average reload batch size, with a factor of approximately four, a rise due to the increased average number of failed fuel rods in an assembly (from 1.3 to 1.6) and further small changes due to variation in the average number of rods per assembly.

3.5.2. Distribution of failure causes in PWRs

The causes of PWR fuel leaks in the United States of America, France and other European countries (with exception of France) are presented in Figures 3.6–3.8, respectively. A worldwide summary of fuel leak causes for the period 1994–2006 can be seen in Fig. 3.9.

Grid to rod fretting is the dominant fuel rod leaker mechanism in PWRs worldwide, reaching up to 65% in the United States of America, 39% in France and 37% in Europe (excluding France), which is in agreement with Refs [3.18, 3.19]. The second most common cause is debris related failures, which add up to 6%, 11% and 18%, respectively, for the three regions. The third greatest cause is fabrication related failures (~5%), which are more or less uniformly distributed throughout regions and time. Crud/corrosion related failures are not typical for PWRs. Currently, the world average for this type of failure is 4%, primarily due to a large number of such failures in the Republic of Korea between 1994–2006, where 40 of a total of 90 rods failed due to Zry-4 cladding corrosion (thus Republic of Korea’s figures add up to 44% of the total) [3.20]. Some random corrosion related failures were also observed in the United States of America, though no failures of this type were observed in other countries. Two isolated cases of PCI–SCC type failures were observed in US PWRs, and there were five handling related failures worldwide.

TABLE 3.3. AVERAGE PWR FUEL ROD LEAKER RATES FOR 1994–2006

	World	United States of America	France	Europe—France	Japan	Korea, Rep. of
Rate (‘new reload’), ppm	86.8	131.6	56.9	108.1	3.7	40.5
Rate (‘old core’), ppm	18.2	25.8	11.4	22.2	0.7	8.7

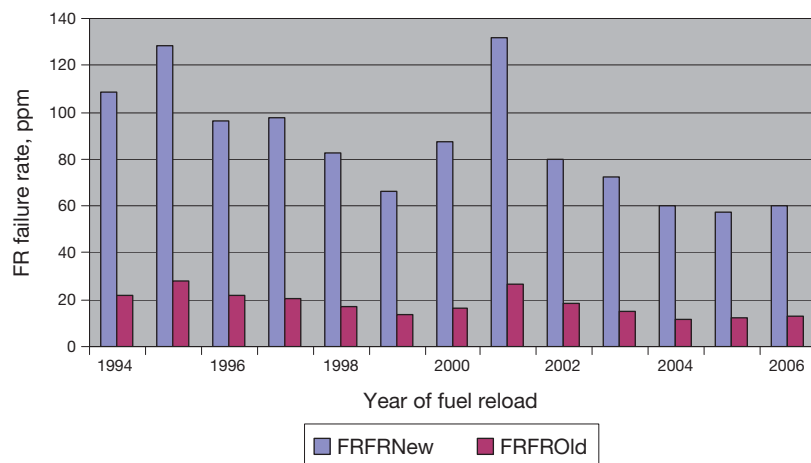


FIG. 3.5. PWR world average fuel rod leak rate calculated using new ‘reload’ and old ‘core’ methodologies.

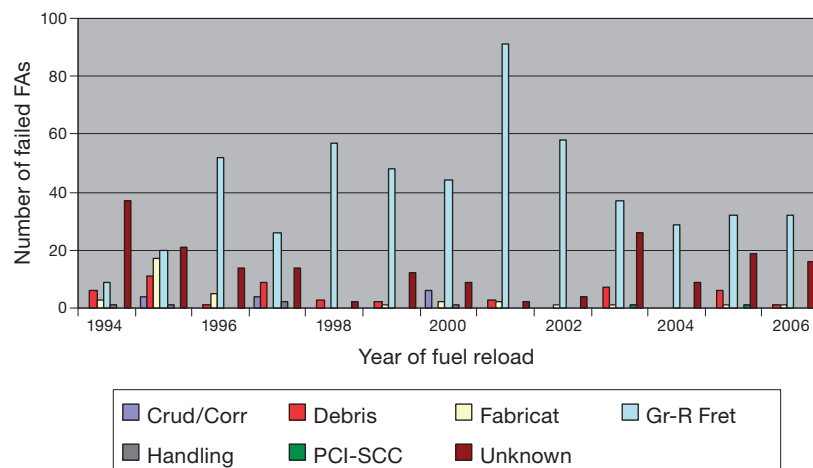


FIG. 3.6. PWR fuel leak causes in the United States of America.

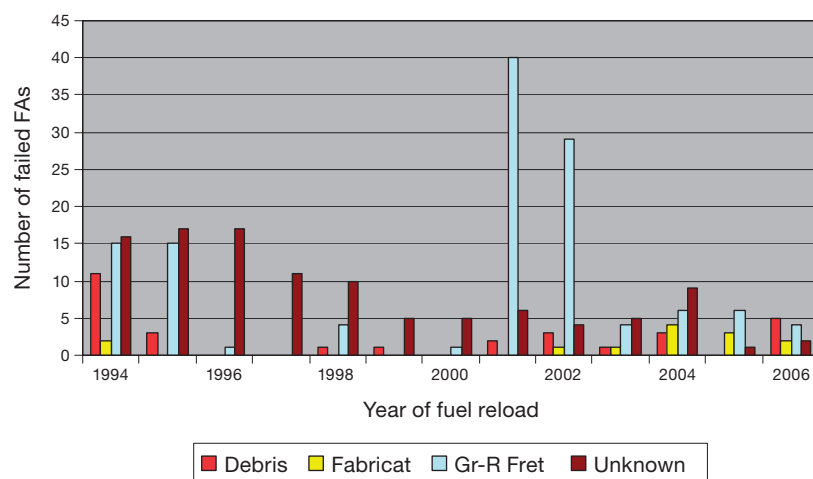


FIG. 3.7. PWR fuel leak causes in France.

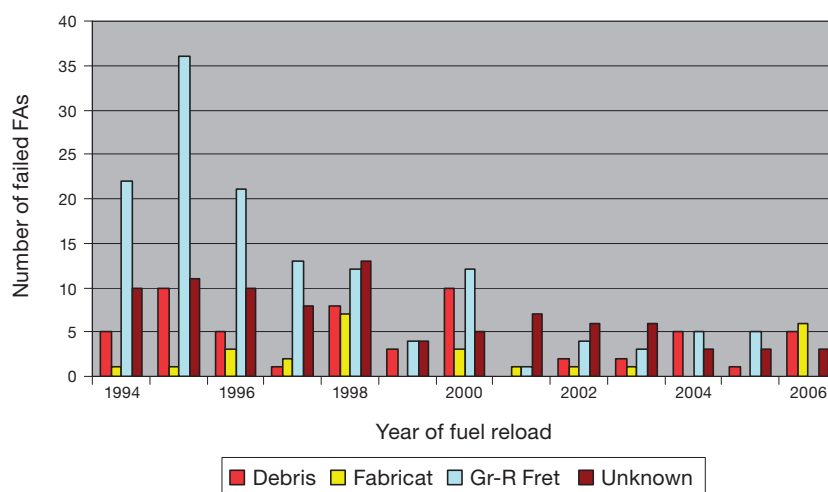


FIG. 3.8. PWR fuel leak causes in Europe with the exception of France.

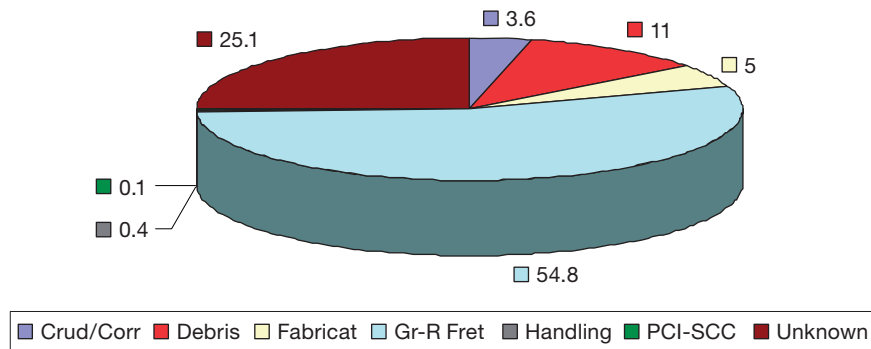


FIG. 3.9. PWR fuel leak causes worldwide in 1994–2006.

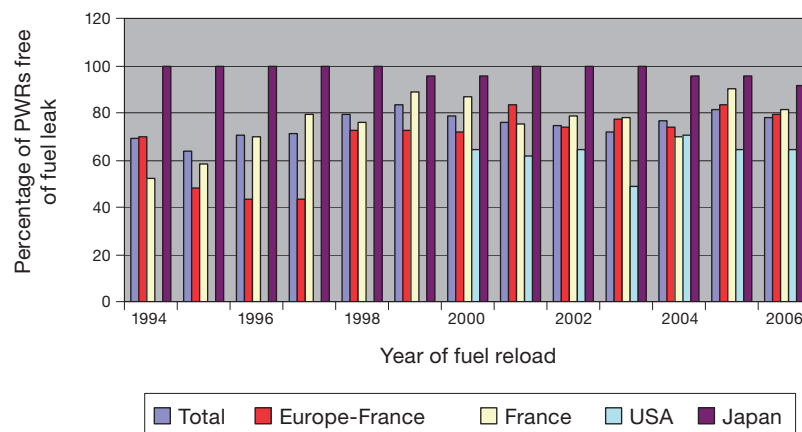


FIG. 3.10. Percentage of PWR units with zero fuel leakers.

3.5.3. Number of reactors free of defect evolution

Another way to present the fuel reliability trend is to evaluate the percentage of units experiencing an outage during each year and by reporting zero fuel failures. The percentage of ‘reactors free of defect’ for a specific year is the number of reactors free of defect with an outage in that year divided by the total number of reactors with an outage in the same year. Again, calculations were done using TWGFPT and other data sources for France, Europe (excluding France) and Japan (Fig. 3.10). EPRI only reported data on plants free of defects in the United States of America for the period 2000–2006 (EPRI’s FRED database began in 2000); thus the US data only cover the period 2000–2006. For 1994–1999, the total curve reflects the percentage of defect free PWR units worldwide, with the exception of the United States of America. Average values are: Total — 76.6%, United States of America — 62.7%, France — 75.6%, Europe (excluding France) — 68.6 and Japan — 98.0%.

Figure 3.10 shows a tendency toward fuel reliability improvement in 1994–2000 for Europe as well as worldwide. However, no significant changes in fuel reliability have been observed between the years 2001–2006.

3.5.4. Major observations regarding PWR fuel failures

- The world average (1994–2006) fuel failure rate is 13.8 leaking FAs per 1000 discharged FAs;
- Failures range from 17 in 1994–1996 to 9.5 in 2004–2006, rising in 1995 and 2001 to 20.5:
 - the rate increase in 1995 was due to grid to rod fretting fuel failures in several plants in Europe, and a massive fabrication fuel failure in one plant in the United States of America;
 - the increase in 2001 was also due to massive fuel failures (grid to rod fretting in one plant in the United States of America and specific fretting at the bottom grid level in one plant in France);
 - Grid to rod fretting failures were highest (~75%) in 1999–2002.

3.6. EVALUATION OF BWR FUEL ‘LEAKERS’

3.6.1. BWR fuel failure rates

Data presented in this section describe fuel leaker rates in the United States of America (34 units), Japan (32 units), Europe (Finland — 2, Germany — 6, Spain — 2, Sweden — 9 and Switzerland — 2, total — 21 units) and India (2 units) for 1994–2006. Only two units located in Mexico are not included in this section. Figure 3.11 shows BWR FA leak rates for 1994–2006 in the United States of America, Japan and the European region.

In the last decade, BWR fuel failure rates have been three times lower than those for PWR fuel. Similar to an effect observed with PWRs, Fig. 3.10 shows a tendency for recurrent peaks in the leaker rate. The maximums are observed in 1994, 2000 and 2002 for Europe, and in 1998 and 2003 for the United States of America, with a leaker level of practically zero for Japan between 1994–2006. Average FA failure rates (number of failed FAs per 1000 discharged FAs) for the entire period 1994–2006 are:

- World average — 4.4;
- United States of America — 5.4 (4.3 if a massive fuel failure at the BF-2 in cycle 12 is excluded; see below);
- Japan — 0.4;
- Europe (Finland, Germany, Spain, Sweden, Switzerland) — 6.8.

BWR fuel rod failure rates calculated for reload batches (using the methodology proposed for the present study) and for core inventories (using the methodology from Technical Reports Series No. 388) are presented in Fig. 3.12.

Analysis of initial data shows that a sharp increase in the US BWR failure rate in 1998 and 2003 contributed to the increased world average fuel failure rate observed in these years. Average rod leaker rate values for 1994–2006 are shown in Table 3.4. Surprisingly, the average fuel rod failure rate for reloads is slightly higher for Europe than the United States of America, but the rate calculated for core inventories is lower. This is explained by differences in fuel reload schemes for BWRs in the United States of America and Europe. In the United States of America, the BWR fuel cycle length is 17–21 months (data provided by EPRI for this study, though in published American papers PWR and BWR cycles are often indicated to be two year cycles), European BWRs generally operate on 11–12 month cycles (except Spain, which has a 15–24 month long cycle). Due to the relatively smaller sizes of reload batches in European BWRs, the fuel failure rate calculated for core inventories is smaller than the reload related fuel failure rate for the same number of failed FAs. In the old ‘core’ calculation scheme, the share of failures in the whole core inventory depended on reload batch size, approaching the real failure rate only when the whole core was reloaded, while in the new ‘reload’ scheme, calculations will provide statistically equivalent results irrespective of batch size.

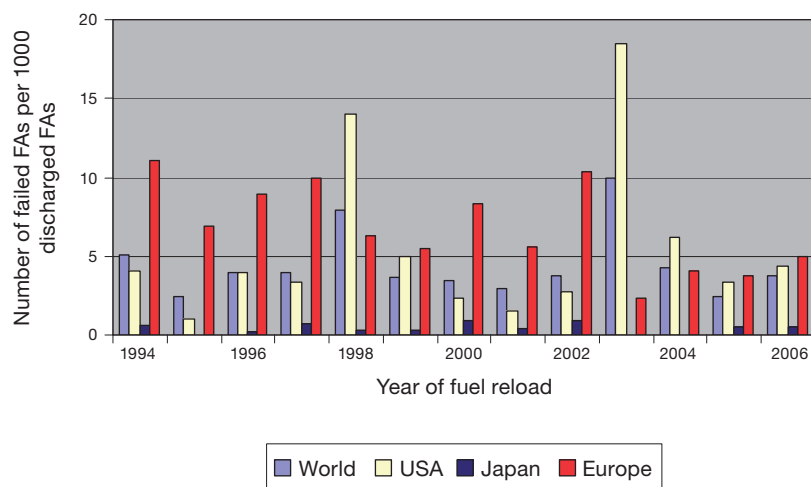


FIG. 3.11. BWR FA failure rates calculated as the ratio of leaking FAs per 1000 discharged FAs.

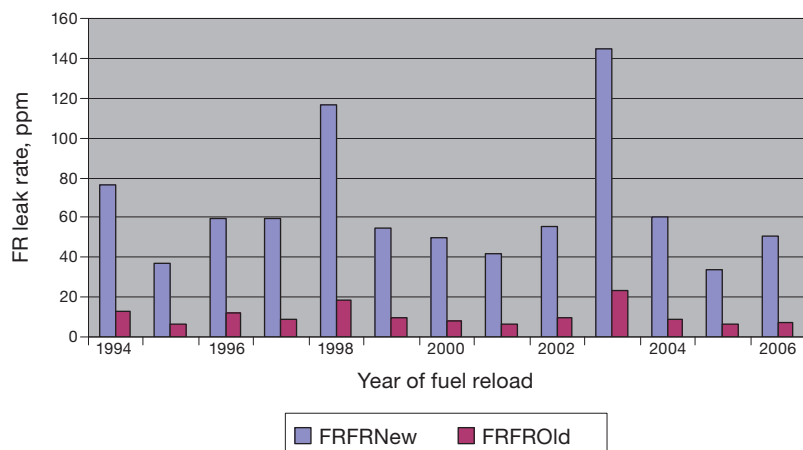


FIG. 3.12. BWR world average fuel rod leaker rates calculated using 'new' and 'old' methodologies.

TABLE 3.4. AVERAGE BWR FUEL ROD LEAKER RATES FOR 1994–2006

	World average	United States of America	Japan	Europe
Rate (new, 'reload'), ppm	64.7	78.9	6.9	101.4
Rate (old, 'core'), ppm	10.7	14.2	1.3	14.0

A comparison of Tables 3.3 and 3.4 shows that the BWR fuel rod leaker rate in 1994–2006 is lower than that for PWRs. The same tendency was observed in previous years (1987–1994) as indicated in Technical Reports Series No. 388.

3.6.2. Distribution of failure causes in BWRs

The distribution of failure causes in United States of America BWRs is shown in Fig. 3.13. Accelerated cladding corrosion and tenacious crud deposition were recognized as being behind massive failures in 1998 (46 failures were due to crud induced localized corrosion-CILC [3.21]) and in 2003 (63 crud/corrosion induced failures in the Browns Ferry Unit 2 during cycle 12 [3.22]). If BF-2 failures are excluded from the statistics, corrosion/crud and debris related fuel failures in US BWRs are at the same level.

Debris related failures dominate in European BWRs with a significant amount of PCI-SCC failure types before 2002; see Fig. 3.14.

Figure 3.15 reveals an estimate of BWR fuel failure causes worldwide. It is seen that crud/corrosion and debris related failures occur at the same frequency. PCI-SCC related failures are more or less uniformly distributed throughout the period 1994–2006, practically independent of the implementation of barrier cladding. Fabrication related failures were not observed during the last three years (2004–2006).

3.6.3. Number of reactors free of defect evolution

Figure 3.16 shows the percentage of leaker free BWR units worldwide and in Japan for the period 1994–2006, and in the United States of America for the 2000–2006 time frame, much like Fig. 3.10 for PWRs. The two figures show very similar trends. Notably, the average value for Japan throughout 1994–2006 was 96.0%. The world total percentage with the exception of the United States of America was 73.7%, while the United States of America and worldwide averages for 2000–2006 were, respectively, 63.3% and 77.6%.

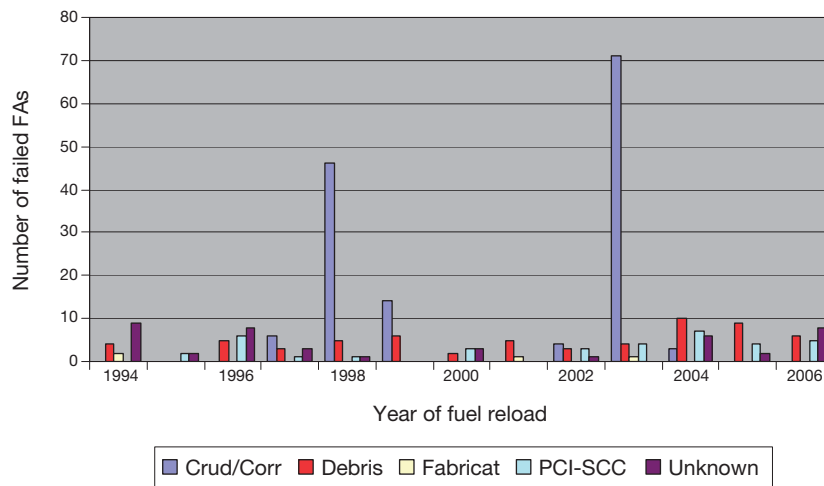


FIG. 3.13. BWR fuel failure causes in the United States of America.

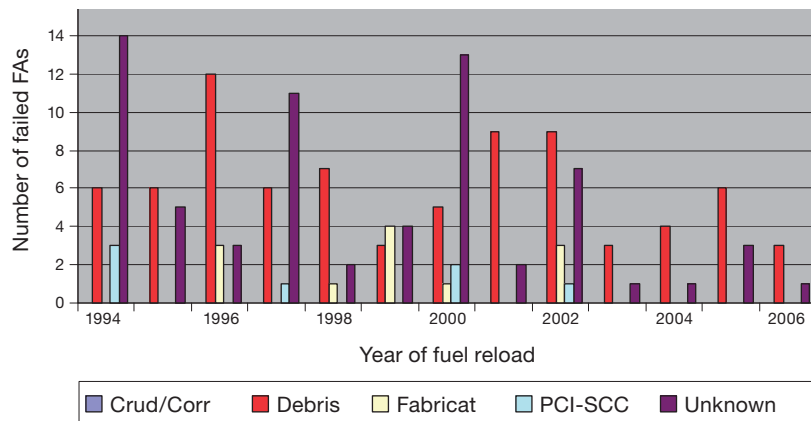


FIG. 3.14. BWR fuel failure causes in Europe.

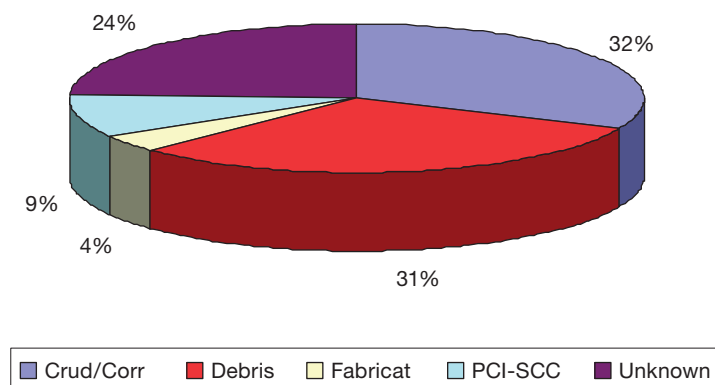


FIG. 3.15. Estimated world distribution of BWR fuel failure causes.

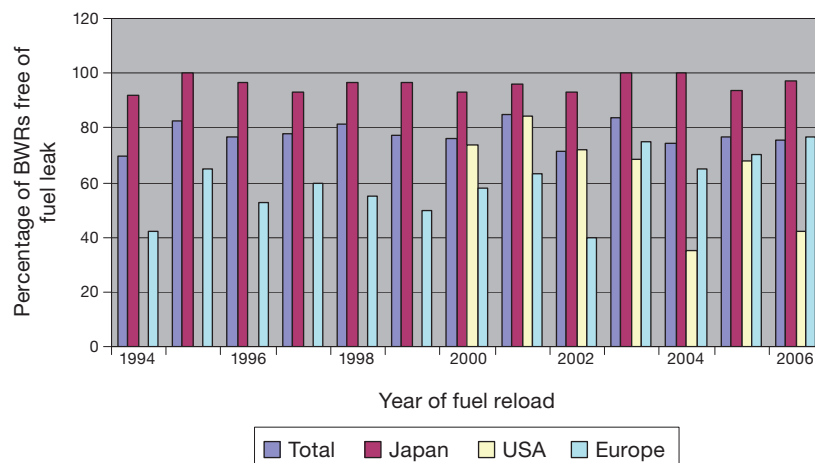


FIG. 3.16. Percentage of BWR units with zero fuel 'leakers'.

3.6.4. Major observations regarding BWR fuel failures

- The worldwide average (1994–2006) fuel failure rate is 4.4 leaking FAs per 1000 discharged FAs;
- This value is lower by a factor of about three than that for PWR fuels;
- The span of the failure rate ranges from 2.5 to 5, with two spikes: 7.9 in 1998 and 10 in 2003;
- These two increases were due to massive crud/corrosion failures in one plant in 1998 and in a second plant in 2003 in the United States of America.

3.7. EVALUATION OF WWER FUEL 'LEAKERS'

3.7.1. WWER fuel failure rates

This section reviews fuel failure data for WWER-1000s and 440s. WWER-1000s operate in Bulgaria (2 units), the Russian Federation (9 units) and the Ukraine (13 units), adding up to a total of 24 units. Two Temelin units in the Czech Republic with WWER-1000 reactors possessing some Westinghouse modified safety systems and fuel are not included and will be considered separately. WWER-440s are divided into two groups: 18 units of the newer 213 design, abbreviated WWER-440/213, (Czech Republic, 4 units; Finland, 2 units; Hungary, 4 units; the Russian Federation, 2 units; Slovakia, 4 units; and Ukraine, 2 units) and 11 units of the 'older' 230/179/270 designs, abbreviated WWER-440/230, (Armenia, 1 unit; Bulgaria, 4 units, now all shutdown; the Russian Federation, 4 units; and Slovakia, 2 units).

Figure 3.17 presents the number of leaking FAs per 1000 discharged FAs observed during outages in WWER reactors. The average number was: Bulgaria, 45; the Russian Federation, 27.5; and Ukraine, 33.9. The average for all WWER-1000s operated between 1994–2006 is 32.1. The maximum FA failure rate was observed in Bulgaria and the Russian Federation in 2001 and in Ukraine in 2004. It is worth noticing that fuel reliability significantly improved for units in the Russian Federation between 2003–2006, dropping to 12.3 failed FAs per 1000 discharged FAs. Progress was also obviously seen for Ukrainian units in 2005–2006 (with an average rate of 17 failed FAs per 1000 discharged FAs).

Two Temelin units use Westinghouse fuel of Vantage 6 design with Zry-4 claddings, intermediate grids and guide tubes, and Inconel top and bottom grids. Grid to rod fretting was a major cause of leaking in one FA in Temelin 1 in 2004, 5 and 3 FAs in 2005 and 6 and 10 FAs in 2006, in Units 1 and 2 respectively (see Fig. 3.17 and answers to the questionnaire).

The average annual FA failure rate in WWER-440/213s has been kept rather low annually for the entire 1994–2006 period (4.7 failed FAs per 1000 discharged FAs per year). Increases in the FA failure rate in 1995 (12 failed FAs per 1000 discharged FAs) and in 2001–2002 (7 failed FAs per 1000 discharged FAs annually) were due to 9 failed FAs in Loviisa-2 in 1995 and 6 and 4 failed FAs in Kola-3 in 2001 and 2003 respectively.

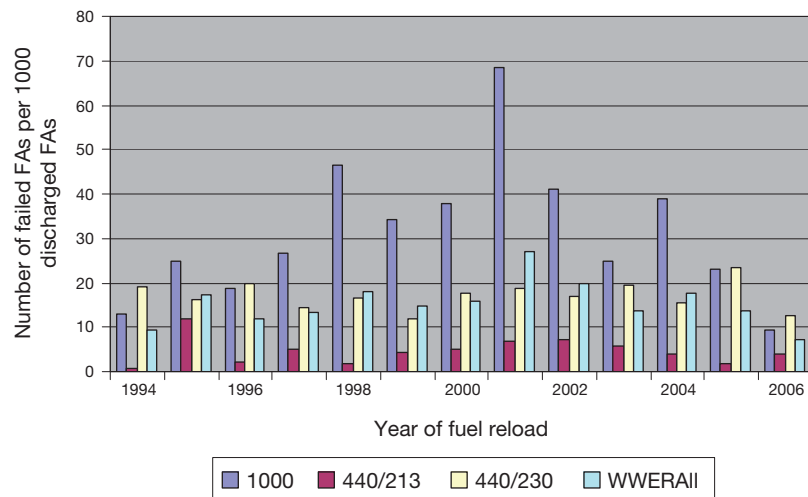


FIG. 3.17. Annual FA failure rate for WWERs.

The average annual FA failure rate in WWER-440/230s for 1994–2006 was 17 failed FAs per 1000 discharged FAs. This failure rate was steady throughout the period. However, fuel failure statistics are rather poor for 2003–2006 — from the nine reactors operating in this period, data were only available for three (Kola-1 and Bohunice-1 & 2).

Figure 3.18 shows the WWER fuel rod leaker rate for reload batches (using ‘new’ methodology). A maximum fuel rod leaker rate was observed in 2001 (286 ppm for WWER-1000s and 149 ppm for all WWERs). The rate increase in 2001 was due to massive fuel failures at WWER-1000s: Balakovo-1 and 2 (9 FAs each) and Rovno-3 (11 FAs), where failure was suspected to be due to debris. The average values for 1994–2006 are: 134 ppm for WWER-1000 fuel, 34.5 for WWER-440/213, with an overall average for all WWERs of 94 ppm. An evaluation of the WWER fuel rod leak rate using ‘core inventory’ (the ‘old’ methodology used in Technical Reports Series No. 388) was also made (Fig. 3.19). The average values for 1994–2006 are: 39.8 ppm for WWER-1000 fuel, 8.7 ppm for WWER-440/213, with an overall average for WWERs of 25.2 ppm. Maximum fuel rod failure rates were observed in 2001 and were recorded to be 84.5 ppm for WWER-1000s and 40.2 ppm for all WWERs. As mentioned above, significant decreases in FR failure rates were observed for all WWER fuel types over the last 2–3 years.

3.7.2. Distribution of failure causes for WWER fuel

According to an official paper by the Russian Federation representative to the IAEA Technical Working Group on Water Reactor Fuel Performance and Technology (TWGFPT), cladding leaktightness tests and visual inspection at NPP spent fuel storage pools and PIE in RIAR hot cells revealed the following major causes of leak failure for WWER-1000 fuel:

- Debris damage to FR claddings;
- Fretting wear on FR plugs in the bottom support grids;
- Displacement of FRs during transportation.

Grid to rod fretting, crud/corrosion, PCI/SCC and manufacturing issues were not identified as causes of failure. As noted in this paper [3.11], debris type failures in WWER-1000s accounted for 14.2% over the 2002–2006 time period, while mechanical damage of lower plugs reached 5.6%, and the remaining 80.2% of failures stemmed from undetermined causes. The Ukrainian representative to the IAEA TWGFPT, reported that debris fretting was a suspected cause of failure for WWER-1000 fuel in Ukrainian WWER-1000s.

The Russian Federation’s report to the IAEA 2007 TWGFPT Plenary Meeting identified the following major causes for leaking of WWER-440/230 fuel:

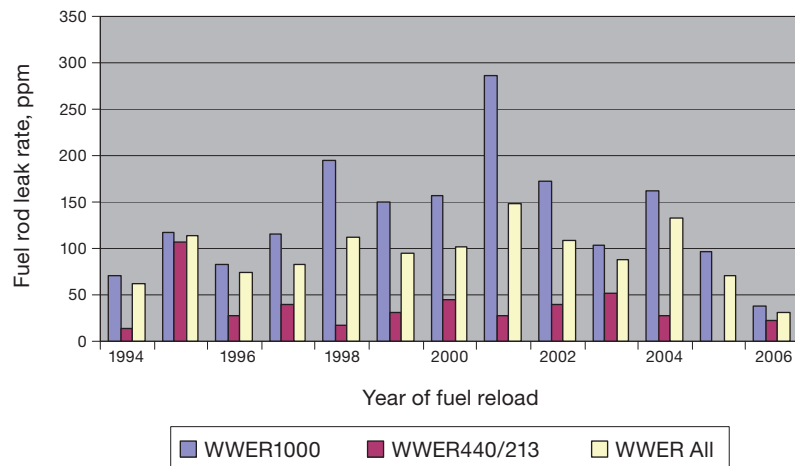


FIG. 3.18. WWER fuel rod 'leaker' rate calculated for reload batches ('new' methodology).

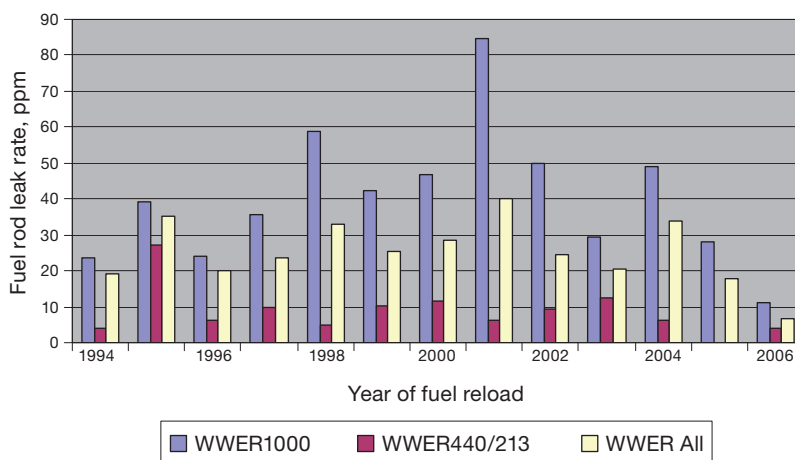


FIG. 3.19. WWER fuel rod 'leaker' rate calculated for core inventories ('old' methodology).

- Fretting wear (claddings, spacer grids, FR plugs);
- Debris induced damage to FR claddings;
- Deposits in FR bundles.

WWER/213s fuel reliability is quite high — the FR failure rate, calculated using Technical Reports Series No. 388 methodology, is lower than 10 ppm. Only one significant fuel failure in these reactors is known; 9 FAs failed in Loviisa-2 in 1995 due to deposits in the fuel bundle; details are available in Section 5. A dismantlable WWER-440 FA design was developed and implemented in the beginning of 2000, initially to serve Loviisa NPP, Finland. The Finnish Safety Authority, STUK, required a pool side inspection of failed FAs. Failure causes at Loviisa NPP Units 1 and 2 were well investigated, however no failures were observed starting in 2000, thus Finnish fuel failure experience covers only the years prior to 2000 [3.23]. Earlier causes of failure are listed in the following order: grid to rod fretting — 39%, crud/corrosion — 23%, manufacturing — 3%, unknown — 35%.

The difference in fuel failure rates between WWER reactors with 213 and 230 designs can be explained by several factors. For example, there are four units with WWER-440 reactors at Bohunice NPP in Slovakia — Units 1 & 2 (230 design) and Units 3 & 4 (213 design). Between 1986–2001, there were 47 leaking FAs in Units 1 (15) & 2 (32) and 2 leaking FAs in Units 3 & 4 loaded with the same fuel. The difference between the two 230 units is also noticeable (15 and 32 leaking FAs). Analysis shows [3.24] that:

- In-core component vibrations are more intensive in the 230 units compared to the 213 units;
- The evaluation of analysis confirmed more significant fuel vibrations with 230 reactors than 213 reactors;
- Long term monitoring of 230 units and comparison between both units confirmed higher vibration amplitudes of the basket at Unit 1, while Unit 2 monitoring drew attention to more intensive frequency vibrations of in-core components. An explanation of the differing failure rates between the two units could be that Unit 1 has a basket rigidly connected with its bottom, while at Unit 2, there is no such connection.

According to the Russian Federation contribution, in 2000–2002, an increased FA vibration load was also seen at Kola-2 with the WWER-440/230 reactor, which resulted in an FR failure increase greater than 50 ppm (using the ‘old’ methodology). After implementation of a vibration resistant FA design, the failure rate decreased. Units 3 & 4 of the Novo-Voronez NPP, which has WWER-440 reactors of the ‘old’ 179 design, operated in 1996–2002 with a fuel failure rate of ~ 50 ppm (using the ‘old’ methodology); this rate has since increased due to deposit accumulation on fuel bundles. At present, a programme on FA cleaning is underway to enhance further use of these FAs.

3.7.3. Number of reactors free of defect evolution

Figure 3.20 presents the percentage of all WWERs units which ran with zero fuel ‘leakers’. The average values for all WWER units with zero ‘leakers’ between 1994–2006 were 43.4% (WWER-1000s), 79.4% (WWER-440s/213) and 57.6% (all WWERs). Distribution by country for WWER-1000s during 1994–2006: 31.9% for Bulgaria, 39.6% for Ukraine and 52.7% for the Russian Federation.

3.7.4. Major observations regarding WWER fuel failures

- The world average (1994–2006) fuel failure rate is 15.1 leaking FAs per 1000 discharged FAs;
- For all WWERs, the range is from 6 to 18 leaking FAs for the whole time span with the exception of 2001 (27);
- A rate increase in 2001 was due to debris suspected massive fuel failures at WWER-1000: Balakovo-1 and 2 (9 FAs each) and Rovno-3 (11 FAs);
- Failure causes for WWER-1000s were attributed to debris damage, fretting wear of FR plugs in the bottom support grids, and displacement of FRs during transportation. For WWER-440s, fretting wear (claddings, spacer grids, FR plugs), debris induced damage to FR claddings and deposits in the FR bundle were mentioned as causes of failure.

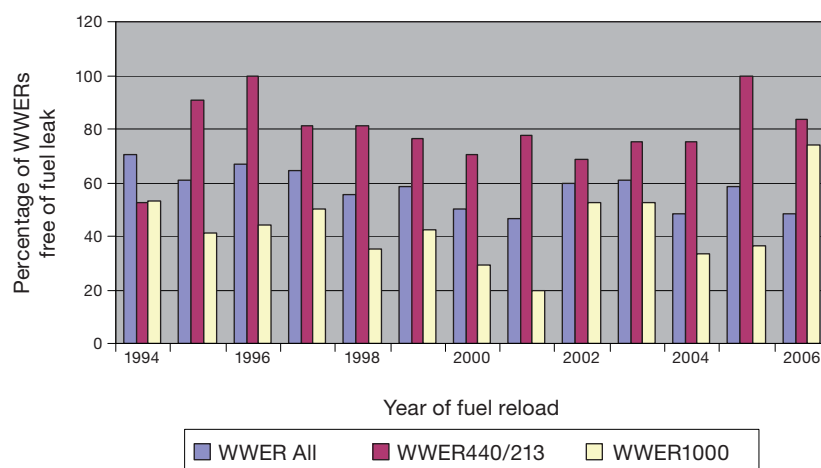


FIG. 3.20. Percentage of all WWER units operated experiencing zero fuel ‘leakers’.

3.8. EVALUATION OF FUEL ‘LEAKERS’ IN CANDUs/PHWRs

3.8.1. CANDU/PHWR fuel failure rates

Data presented in this section describe fuel leaker experience for 22 CANDU-6 units (1 — Argentina, 14 — Canada, 2 — China, 4 — Republic of Korea, and 1 — Romania) and 13 PHWRs (12 — India, including only 220 MWe units; there are no data for two new 540 MWe units, and 1 — Pakistan) for 1994–2006. Some data for the Atucha-1 plant (MZFR design) were also received, but their interpretation was difficult and is not included in these fuel failure statistics. As was mentioned in Section 3.3.2, data for two Chinese units and one Pakistani unit were taken from literature, while ‘normal’ Excel tables from TWGFPT members were received for all other plants.

Results of the fuel failure rate calculation for CANDUs and PHWRs (number of leaking fuel bundles per 1000 discharged bundles) are presented in Fig. 3.21. Traditionally, failure rates are very low in comparison to other reactor types. The average number of leaking bundles per 1000 discharged bundles accounted for 0.1 (CANDUs in Canada), 0.35 (for all 22 CANDU units) and 1.5 for PHWRs. During the 1994–2006 time period, only one defect excursion was observed in CANDU plants, namely in Wolsong-1 in 1996, when 96 bundles failed because of insufficient baking of the CANLUB graphite coating [3.25].

Fuel rod failure rates for CANDUs operated in Canada and all CANDUs are presented in Fig. 3.22. The average fuel rod failure rate for 1994–2006 was 3.5 ppm for units in Canada and 10.4 ppm for units operated in Argentina, Canada, China, the Republic of Korea and Romania.

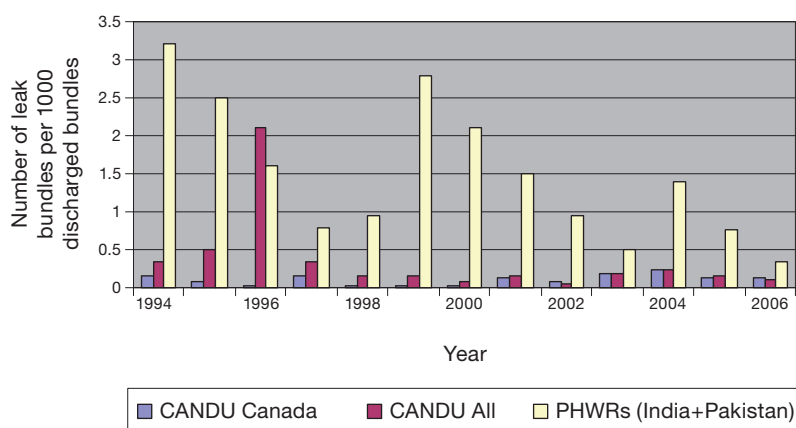


FIG. 3.21. Number of leaking fuel bundles per 1000 discharged bundles.

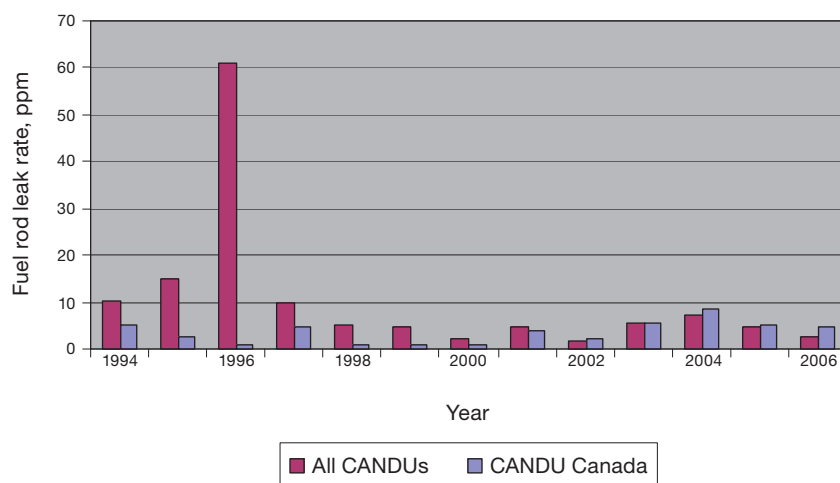


FIG. 3.22. Fuel rod failure rates for CANDUs operated in Canada and all CANDUs.

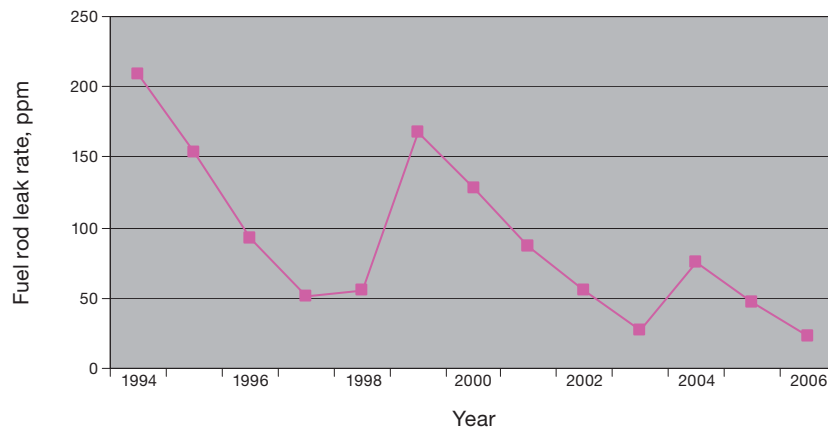


FIG. 3.23. Fuel rod failure rates for Indian PHWRs.

The PHWR fuel rod failure rate for 1994–2006 is presented in Fig. 3.23. During this period, the rate generally decreased from 209 to 23 ppm. Of course, because of a smaller number of rods in PHWR bundles (19–22 rods/bundle in 220 MWe units) in comparison to CANDUs (Pickering station has 28 rods/bundle, other have 37 rods/bundle), a factor of ~ 1.7 for the fuel rod failure rate should be taken into account. However, the fuel failure rate in CANDUs is definitely lower than in PHWRs.

3.8.2. Distribution of failure causes for CANDUs

CANDU stations employ a variety of equipment and techniques to detect and discharge failed fuel at-power if and when a failure occurs. Moreover, CANDU stations have developed specialized capabilities to interpret the on-power ‘signatures’ of various types of fuel defects, such as single element failures (e.g. from debris) vs. multiple element failures (e.g. possibly from excessive unintentional power ramps). Based on the above information and available post-irradiation examinations (PIEs), a 1.1 ‘conversion factor’ (number of failed FRs per FA) was identified in Technical Reports Series No. 388 (1998); Canadian data providers have advised that it should remain so.

Despite the low defect rate, many discharged fuel bundles have been studied via in-pool examinations at operating reactors as well as via more detailed PIEs in hot cells. These examinations have been undertaken not only as part of ‘root cause’ investigations of fuel defects, but also as part of a routine ‘fuel surveillance’ programme [3.26] which monitors the ‘health’ of CANDU fuel to look for ‘early warning’ signs of potential fuel performance issues. The examinations have yielded the following causes for fuel damage in CANDU fuel during the reported period:

- Debris damage;
- Fabrication flaws;
- Unknown.

Identified fuel failure causes in Indian PHWRs were related to:

- Debris (mainly in initial cores);
- Fabrication (end cap welding defects and hydriding near the end cap);
- Power ramps.

3.8.3. Number of reactors free of defect evolution

Not much can be said about Canadian CANDUs regarding the number of units reporting zero ‘leakers’. Canadian fuel failure data are available on a ‘station by station’ basis. Some stations have multiple reactors; the breakdown for each reactor within a multi-unit station was not always consistent. The share of units reporting defects in Argentina, the Republic of Korea and Romania (6 units in total) is $\sim 78\%$ for 1998–2006.

The share of Indian PHWRs reporting zero leaker operations for 1994–2006 was ~36%.

3.8.4. Major observations regarding CANDU/PHWR fuel failures

- The world average (1994–2006) fuel failure rate is 0.35 leaking FAs per 1000 discharged FAs for all CANDUs worldwide, e.g. 0.1 for Canadian CANDUs and 1.5 for PHWRs in India and Pakistan;
- The failure rate of CANDU fuel is rather stable and low for the 1994–2006 period with only one exception; 96 fuel bundles failed in Wolsong-1 in 1996, because of insufficient baking of the CANLUB graphite coating;
- Fuel defect mechanisms in CANDUs tend to be the same. They are: debris damage, fabrication flaws, and unknown;
- Identified fuel failure causes in Indian PHWRs were related to: debris (mainly in initial cores), manufacturing (end cap welding defects and hydriding near the end cap), and power ramps.

3.9. MULTIPLE FAILURE INCIDENTS/DEFECT EXCURSIONS

3.9.1. Incidents of massive and significant fuel failures

Regarding reported incidents of multiple failures (≥ 10 FAs in one cycle or one year for CANDUs), it is difficult to establish a comprehensive list because not all were reported and only a few have been published. As shown in Technical Reports Series No. 388, from 1987 to 1994 (8 years) there were 16 such occurrences in LWRs (10 in PWRs, 5 in BWRs and 1 WWER-440) including 4 in CANDUs. Major massive fuel failure causes were grid to rod fretting and debris in PWRs, crud/corrosion induced failures and debris in BWRs, and crud deposition in WWER-440s.

Table 3.5 gives analogous data for 1994–2006 (13 years): there were 16 occurrences (10 in PWRs, 2 in BWRs, 3 in WWER-1000s and one in a CANDU). Major fuel failure causes were grid to rod fretting in PWRs, crud/corrosion in BWRs, and debris in WWER-1000s. Fabrication (insufficient baking of the CANLUB graphite coating) caused the CANDU fuel failure.

Thus, the tendency for a decrease in multiple failure frequency from the 1960s–1970s to the beginning of 1990s, reported in Technical Reports Series No. 388, has not been confirmed over the last decade. With all incidents, the bulk of failures were due to a common failure mechanism and root cause, mainly debris fretting, crud/corrosion or grid–rod fretting in LWRs. Fuel failure mechanisms are considered in more detail in Section 5 of the present review.

Additionally, Table 3.6. provides information on significant, but not massive fuel failures.

3.9.2. Impact of multiple failure incidents on failure statistics

Accumulated data sets on fuel rod ‘leakers’ worldwide allowed for a rough estimation of the contribution of multiple and significant failure incidents (Tables 3.5 and 3.6) on the average fuel leak rate, especially for WWERs and CANDUs. This contribution amounted to ~10% for WWERs and ~30% for CANDUs (this is affected by the fuel failure in Wolsong-1 in 1995–1996). For PWRs and BWRs, this contribution is not so easy to evaluate because of missing data on a number of failed FAs in a number of these incidents. Nevertheless, identification of the failure mechanism in a majority of these incidents and well documented information on these incidents in two PWRs and one BWR (Table 3.5) leads to the conclusion that, for these types of water cooled power reactors, this contribution constitutes between 10% and 20% of all failures. This means that the exclusion/elimination of multiple fuel failures will not substantially reduce fuel failure rates.

It is easy to achieve a 10^{-6} (1 ppm) rod failure rate, even in light of multiple failure incidents. From the above it is clear that much should still be done to reach a ppm fuel failure level. A rod failure rate of 10^{-6} would mean that the total worldwide annual number of leaking fuel rods per year must not exceed:

TABLE 3.5. INCIDENTS OF MULTIPLE FUEL FAILURES (≥ 10 FAILED FAs IN ONE CYCLE)

Year	Plant	Main failure mechanism	References
1994	1 PWR — United States of America	Unknown	N/A
1995	1 PWR — United States of America	Fabrication	N/A
1996	96 bundles in CANDU-6 Wolsong-1 — Republic of Korea	Hydriding (fabrication)	[3.25]
1996	1 PWR — United States of America	Grid to rod fretting	N/A
1998	1 PWR — United States of America	Grid to rod fretting	N/A
1998	WWER-1000 Rovno-3 — Ukraine, 15 FAs	Debris suspected	Questionnaire
1998	1 BWR — United States of America	Crud/corrosion (CILC)	[3.21]
1999	WWER-1000, Rovno-3, 10 FAs	Debris suspected	Questionnaire
2000	1 PWR — United States of America	Grid to rod fretting	N/A
2001	1 PWR — United States of America	Grid to rod fretting	N/A
2001	1 PWR Cattenom-3, Cycle 8, 28 FAs	Fretting at the bottom grid level	[3.27]
2001	WWER-1000, Rovno-3 — Ukraine, 11 FAs	Debris suspected	Questionnaire
2002	1 PWR Nogent-2, Cycle 11, 22 FAs	Fretting at the bottom grid level	N/A
2003	1 BWR Browns Ferry Unit 2, Cycle 12, 63 FAs	Crud/corrosion	[3.22]
2003	1 PWR — United States of America	Unknown	N/A
2006	1 PWR — United States of America	Grid to rod fretting	N/A

TABLE 3.6. INCIDENTS OF SIGNIFICANT, BUT NOT MASSIVE, FUEL FAILURES

Year	Plant	Main failure mechanism	References
1995	WWER-1000 Kalinin-1, 9 FAs	Debris suspected	Questionnaire
1995	PWR TMI-1, Cycle 10	Crud/corrosion (distinctive crud pattern)	[3.28]
1995	WWER-440 Loviisa-2, 9 FAs, >28 rods	Grid to rod fretting/crud/corrosion	Questionnaire
1996	PWR Seabrook, Cycle 10, 1996, several FAs	Crud/corrosion	[3.29]
1996–97	PWR — Brazil, Angra-1, Cycle 6, 9 FAs	Grid to rod fretting	[3.30]
1998	16x16 PWR — Germany, 23 rods	Inconel spacer fretting	TWGFPT, 1999
1999	BWR River Bend, Cycle 8, 7 FAs	Accelerated cladding oxidation under crud (rich with copper and zinc) deposits	[3.29]
2001	PWR Cattenom-4, cycle 8, 7 FAs	Fretting at the bottom grid level	N/A
2001	WWER-1000 Balakovo-1 and 2, 9 FAs each	Debris suspected	Questionnaire
2001–2003	BWR River Bend, Cycle 11, 6 FAs	Accelerated cladding oxidation under crud (rich with copper and zinc) deposits	[3.31]

- 8–9 rods in cores and 2–3 rods in the reload of all operating PWRs;
- 4–5 rods in cores and 0.6–0.7 rods in the reload of all operating BWRs;
- 2 rods in cores and 0.5–0.6 rods in the reload of all operating WWERs;
- 4–5 elements in the reload of all operating CANDU reactors.

3.10. FUEL ROD FAILURE RATES AND FUEL FAILURE CAUSES BETWEEN 1987–2006

3.10.1. Evolution of fuel failure rates between 1987–2006

Using the data presented in Technical Reports Series No. 388 (1987–1994) and data obtained for the present review (1994–2006), the fuel rod failure curve for PWRs (all units worldwide), BWRs (all units worldwide except India), WWERs (all units, all types) and CANDUs (only Canadian units, as in Technical Reports Series No. 388) for 1987–2006 is given in Fig. 3.24. For this reason, 1994 is placed on the horizontal axis twice. The calculation was made using the ‘core inventory model’ developed and utilized in Technical Reports Series No. 388. It is worthwhile to repeat that data for 1987–1994 were taken from available publications, except in the case of WWERs and CANDUs, for which data were provided by AECL; data for the present review were provided in answers to an IAEA questionnaire. Reasonable convergence and comparison of these two sets of results can be seen in Fig. 3.23. For example, for 1994, ‘Technical Reports Series No. 388’ and ‘present review’ data (in brackets) are as follows: PWRs — 20.5 (21.9), BWRs — 10 (12.9), WWERs — 14.3 (17.6) and Canadian CANDUs — 8 (5.1). The difference in CANDU data (8 and 5.1 ppm, respectively) may be explained by a provider induced data adjustment (1994: in Technical Reports Series No. 388 — 18 failed bundles and 9 bundles were reported for that year in the present review).

This graph confirms again that the tendency is for a reduction in the fuel failure rate for the examined time span. Recurrent spiking of the leaker rate is linked mainly to massive fuel failures for all types of reactors. If these failures are excluded, a slight decrease in the fuel rod failure rate is perceptible, despite more and heavier fuel duties, including increasing burnup, cycle length and thermal up-rate.

To limit the effect of massive failures, average FR failure rates were also calculated over four year periods using the old ‘core’ method for the period 1987–2006. The results are presented in Table 3.7 and Fig. 3.25.

This figure shows:

- A continuous decrease in the fuel rod failure rate in PWRs;
- A constant fuel rod failure rate for WWERs;
- A decrease in the fuel rod failure rate in BWRs and CANDUs at the beginning of the 1990s, which stabilized at a lower level over the last decade;
- The overall decreasing trend for LWRs is driven by PWRs, which are in the majority.

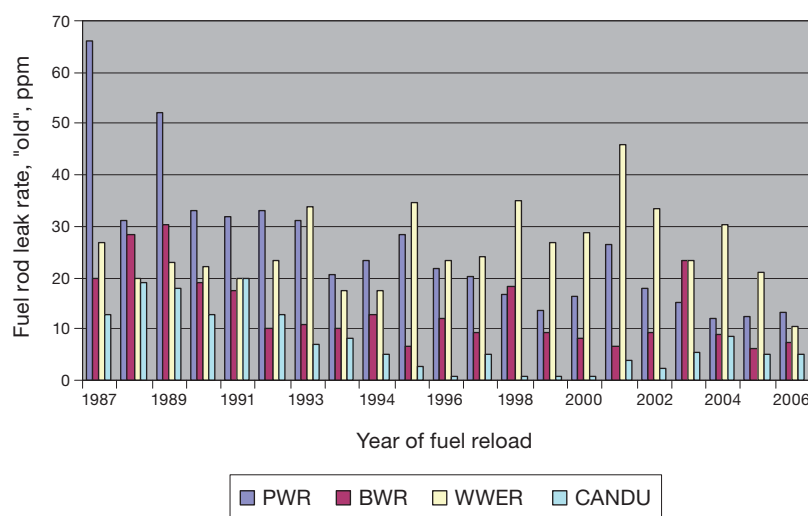


FIG. 3.24. Fuel rod leaker rate calculated using the ‘old’ methodology for the years 1987–2006.

TABLE 3.7. AVERAGE FUEL ROD FAILURE RATES CALCULATED USING THE OLD ‘CORE’ MODEL, PPM

Reactor Type	Years				
	1987–1990	1991–1994	1995–1998	1999–2002	2003–2006
PWR	45.5	29.1	21.8	18.7	13.1
BWR	24.5	12.1	11.6	8.5	11.5
WWER	22.2	22.9	29.3	34.1	22.6
CANDU	15.8	12	2.3	1.9	5.5
LWR	36.2	23.4	20.2	18.3	13.7

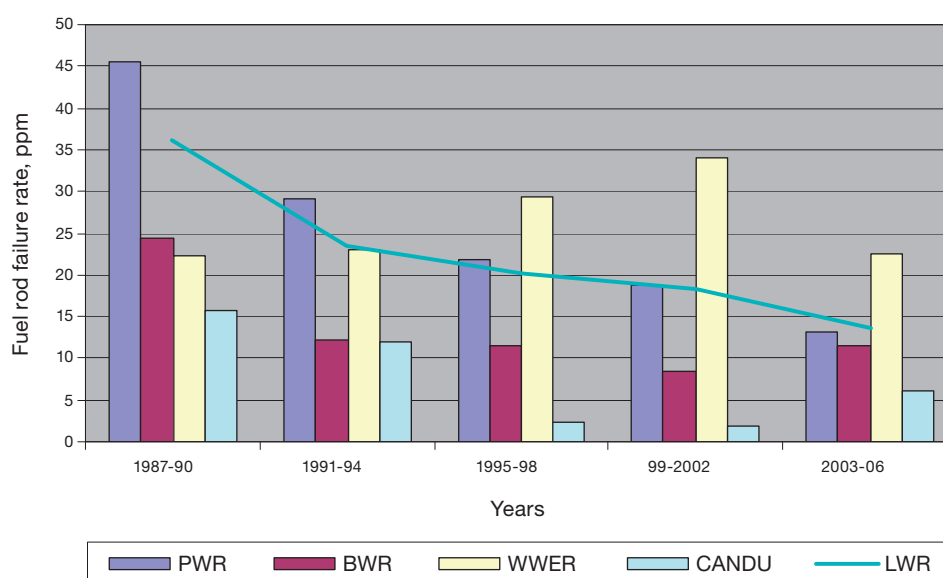


FIG. 3.25. Fuel rod leaker rate calculated using the ‘old’ methodology for the period 1987–2006.

3.10.2. Causes of fuel rod failure from 1987 to 2006

In PWRs, grid to rod fretting (Table 3.8) is the dominant fuel rod leaker mechanism, the occurrence of which increased between the 1980s and the 2000s, accounting for 52% in the period 2003–2006. Debris related failures, which contributed roughly equally to failure causes in the 1980s, were halved in the 2000s. Manufacturing related failures accounted for about 10% and 5%, respectively, in the 1980s and in the 2000s. Corrosion related failures were not seen in the 1980s; they first appeared between 1995 and 2000, with some isolated incidents in the USA. However, the higher figures are mainly due to a significant number (40 failed rods) of this type of failure in the Republic of Korea.

Debris fretting, crud corrosion and pellet–cladding interaction/stress corrosion cracking (PCI/SCC) related failures dominated by the end of the 1980s and the beginning of the 1990s in BWRs (Table 3.9), accounting for 20–30% each. Currently, corrosion by itself or in combination with crud deposits is an important issue for BWR fuel performance (~46%). Debris fretting firmly occupies the second position with a 28% share of PCI/SCC related failures — reduced to 12% — and remains at a significant level in spite of the widespread use of barrier claddings.

TABLE 3.8. ESTIMATED WORLD DISTRIBUTION OF PWR FUEL FAILURE CAUSES IN PER CENT

Mechanism of PWR fuel failure	1987–1990	1991–1994	1995–1998	1999–2002	2003–2006
Grid to rod fretting	8.3	22.2	53.5	74.8	52.1
Debris	27.8	24.3	10.6	6.1	9.3
Manufacturing	10.4	3.5	7.0	2.9	4.8
Crud/corrosion	0	0	1.6	1.3	0
PCI/SCC	0	0	0	0	0.6
Handling	1.4	2.0	0.6	0.3	0
Baffle jetting	2.1	0	0	0	0
Unknown	50.0	48.0	26.7	14.6	33.2

TABLE 3.9. ESTIMATED WORLD DISTRIBUTION OF BWR FUEL FAILURE CAUSES IN PER CENT

Mechanism of BWR fuel failure	1987–1990	1991–1994	1995–1998	1999–2002	2003–2006
Debris	13.7	34.1	30.1	38.9	27.8
Crud/corrosion	33.2	3.0	35.6	16.7	45.7
PCI–SCC	21.7	23.0	7.5	8.3	12.3
Manufacturing	7.9	7.4	2.8	8.3	0.6
Handling	1.9	0	0	0	0
Unknown	21.6	32.5	24.0	27.8	13.6

3.11. CONCLUDING REMARKS

- The current study proposes a methodology for failure rate calculation based on discharged fuel, which creates a more realistic assessment of fuel reliability when compared to calculations on the basis of full fuel inventories;
- The FA failure rate is more representative of fuel reliability than the FR failure rate because the number of failed rods in failed FAs is often unknown and can vary with the causes of failure;
- Despite continuous upgrading of fuel materials, designs and quality assurance procedures implemented over the last decade, which are assumed to lead to a general improvement in fuel reliability, the failure rates vary in most countries (with the only exception being Japan, which has a stable and very low failure level).

REFERENCES TO SECTION 3

- [3.1] STRASSER, A., et al., “Reliability — The first priority in fuel performance”, Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, FL, 1994, ANS, LaGrange Park, IL (1994) 3.
- [3.2] WILSON, H.W., et al., “Westinghouse fuel performance experience”, Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management, IAEA-TECDOC-709, IAEA, Vienna (1993) 133.
- [3.3] WILLSE, J.T., et al., “Recent results from the B&W Fuel Company’s fuel performance improvement program”, Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, FL, 1994, ANS, LaGrange Park, IL (1994) 45.
- [3.4] PAINTER, C.L., et al., Summary of fuel performance annual report for 1990, J. Nucl. Saf. **34** 2 (1993) 259.

- [3.5] YANG, R.L., et al., “Fuel performance evaluation for EPRI program planning”, Proc. Int. Top. Mtg on LWR Fuel Performance, Avignon, 1991, Vol. 1, SFEN, Paris (1991) 258.
- [3.6] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Power Reactors in the World, Reference Data Series No. 2, 2007 Edition, IAEA, Vienna (2007).
- [3.7] COMMISSARIAT À L'ÉNERGIE ATOMIQUE, ELECNUC 2007 “Les centrales Nucléaires dans le monde”, Status on 2006-12-31, CEA Edition, www.cea.fr/content/download/4667/27759/file/Elecnu2007.pdf.
- [3.8] KIRUSHKIN, A., TCHIRKOV, V., SIDOROVA, E., NOVIKOV, V., “Evaluation of Operation performance of WWER fuel manufactured by JSC MSZ, the Russian Federation”, Proc. 4th Int. Conf. on WWER Fuel-Performance, Modelling and Experimental Support, Albena, 2001, BAS, INRNI, Sofia, Bulgaria (2002) 27–32.
- [3.9] TZOCHEVA, V., ASSENOV, I., NESHOVSKA, G., “Kozloduy NPP WWER-440 fuel performance review”, Proc. 3rd Int. Conf. on WWER Fuel Performance, Modelling and Experimental Support, Pamporovo, 1999, BAS, INRNI, Sofia, Bulgaria (2000) 57–60.
- [3.10] ROZHKOVA, V.V., ENIN, A.A., SINNIKOV, Y.G., SHCHUCHKIN, V.V., “Development and design of WWER-1000 FA and facilities for transportation to NPP”, Proc. 6th Int. Conf. WWER Fuel Performance, Modelling and Experimental Support, Albena, 2005, BAS, INRNI, Sofia, Bulgaria (2005) CD-ROM, paper 2.3.
- [3.11] ENIN, A., PLUZHNIKOV, D., BEZBORODOV, Y., “Improvement of WWER-1000 FA design and the basic results of operation”, Proc. 7th Int. Conf. on WWER Fuel Performance, Modelling and Experimental Support, Albena, 2007, BAS, INRNI, Sofia, Bulgaria (2007) CD-ROM, paper 2.4.
- [3.12] INTERNATIONAL ATOMIC ENERGY AGENCY, “CANDU fuel post-irradiation examination in China”, IAEA Workshop on Experience Exchange on CANDU Fuel Defect Investigation, TQNPP, Haiyan, 2006, CD-ROM.
- [3.13] XINCAI, X., “Fuel failure and reconstitute in Qinshan nuclear power plant”, Proc. Rod Instrumentation and In-pile Measurement Techniques, IAEA TM, Halden, 2007.
- [3.14] TSIBULYA, V., et al., “Analysis of results of the WWER-440 fuel assemblies operation at different NPPs and the WWER-440 assembly design modernization related aspects”, ELECNUC 2007 “Les centrales Nucléaires dans le monde”, Status on 2006-12-31, CEA Edition, www.cea.fr/content/download/4667/27759/file/Elecnu2007.pdf, pp. 99–103.
- [3.15] NEMES, I., POS, I., PATI-SZABO, S., “Some specific features of NPP Paks fuel utilization”, Proc. Int. Top. Mtg on Water Reactor Fuel Performance, Kyoto, 2005, CD-ROM, paper 1118, JAES (2005) 421–430.
- [3.16] KHAWAJA, G.Q., et al., “Fuel management experience at KANUPP”, Proc. IAEA Techn. Mtg on Pressurized Heavy Water Reactor (PHWR) Fuel Modelling, Nub, NPCIL, Mumbai, 2006, IAEA.
- [3.17] MECIR, V., “Temelin NPP fuel experience”, Proc. 7th Int. Conf. on WWER Fuel Performance, Modelling and Experimental Support, Albena, 2007, BAS, INRNI, Sofia, Bulgaria (2007) CD-ROM, paper 1.12.
- [3.18] YANG, R., CHENG, B., DESHON, J., EDSINGER, K., OZER, O., Fuel R & D to improve fuel reliability, J. Nucl. Sci. Technol., **43** 9 (2006) 951–959.
- [3.19] EDSINGER, K., “Zero by 2010: EPRI’s fuel reliability program”, 2007 LWR Fuel Performance Mtg, TopFuel, San Francisco, CA, 2007.
- [3.20] SOHN, D.-S., “Status and Development: Plan in Korea”, Opening Plenary, *ibid*.
- [3.21] YANG, R., OZER, O., ROSENBAUM, H.S., “Current challenges and expectations of high performance fuel for the millennium”, Proc. Int. Top. Mtg on LWR Fuel Performance, Park City, 2000, ANS (2001) 756–765.
- [3.22] KEYS, T.A., LEMONS, J.F., OTTENFELD, C., SCHADT, J., “Fuel corrosion failures in the Browns Ferry nuclear plant”, Proc. Int. Top. Mtg on LWR Fuel Performance, Orlando, 2004, Paper 1036, ANS, 229–236.
- [3.23] TERASVITA, R., PIHLATIE, M., “Recent experience on operating WWER-440 fuel in Loviisa NPP, pool-side inspections and planned research programmes”, Proc. 5th Int. Conf. on WWER Fuel-Performance, Modelling and Experimental Support, Albena, Bulgaria, 2003, BAS, INRNI, Sofia (2004) 91–95.
- [3.24] KACMAR, M., BENA, J., SMIESKO, I., “Fuel reliability of Bohunice NPP”, Proc. IAEA Tech. Mtg on Fuel Failure in Water Reactors: Causes and Mitigation, Bratislava, 2002, IAEA-TECDOC-1345, IAEA, Vienna, (2003) 257–264.
- [3.25] PARK, H.T., et al., “Fuel defect root cause investigation at Wolsong-1”, Proc. 5th Int. Conf. on CANDU Fuel, Toronto, 1997, CNS, Vol. 1 (1997) 219.
- [3.26] MONTIN, J., JUDAH, R., SCOTT, B., “OPG Fuel Surveillance Program”, Proc. of the 9th International Conference on CANDU Fuel, 2005, ISBN 0-919784-84-4.1.
- [3.27] BAILLON, N., “Grid to rod fretting wear in EDF PWR — From operating problems to new design qualification method”, Proc. Techn. Comm. Mtg on Structural Behaviour of Fuel Assemblies for Water Cooled Reactors, Cadarache, 2004, IAEA-TECDOC-1454, IAEA Vienna (2005) 209–220.
- [3.28] TROPASSO, R., WILLSE, J., CHENG, B., “Crud-induced cladding corrosion failures in TMI-1 Cycle 10”, Proc. Int. Top. Mtg on LWR Fuel Performance, Orlando, 2004, Paper 1070, ANS, 339–346.
- [3.29] CHENG, Bo, et al., “Water chemistry and fuel performance in LWRs”, Proc. Int. Top. Mtg on LWR Fuel Performance, Park City, 2000, ANS.
- [3.30] CHAPOT, J.L.C., et al., “Fuel reliability at Angra 1: cause and mitigation”, Proc. IAEA Tech Mtg on Fuel Failure in Water Reactors: Causes and Mitigation, Bratislava, 2002, IAEA-TECDOC-1345 (2003) 92–116.
- [3.31] RUSAUSKAS, E.J., SMITH, D.J., “Fuel failures during cycle 11 at River Bend”, Proc. Int. Top. Mtg on LWR Fuel Performance, Orlando, 2004, Paper 1016, ANS (2004) 221–228.

4. DETECTION, EXAMINATION AND ANALYSIS OF FUEL FAILURES

4.1. INTRODUCTION

Continuous fuel failure evaluation during operation is undertaken for two major reasons:

- To detect and remove leaking fuel assemblies which could lead to high activity release or significant fuel loss during operation;
- To find the root causes of failures in order to provide feedback and take corrective actions, if necessary, in operation, design, manufacturing or development.

Before corrective action can be taken, there are four basic questions which need to be answered:

- (1) When did the fuel element fail?
- (2) What is the location of the defect in the core?
- (3) When should the defective fuel element be discharged?
- (4) What is the cause of the failure(s)? Is the cause related to manufacturing, operation or design?

The corresponding procedure and a typical sequence to address these questions are summarized in Figs 4.1 and 4.2. The different sequences will be described in more detail in the following sections.

4.2. EVALUATION OF COOLANT ACTIVITY

4.2.1. Background

An evaluation of coolant activity in operating plants is the first step to fuel reliability evaluation. Monitoring the activity of selected fission product isotopes in the off-gas system or the primary coolant provides useful information on fuel performance in operation. From the data produced, it is possible to identify the time during an operating cycle when fuel failures occurred, to estimate the approximate number and type of failures, and to predict the approximate exposure of failed fuel.

Qualitative and quantitative characterization of the state of defective fuel is usually based on three chemical families of nuclides which do not produce significant deposits on circuit walls and for which correlations between defective rod release and activity are easily established [4.1–4.3].

The radionuclides from these families which are easy to measure using gamma spectrometry and which can be used in diagnosing the fuels are as follows:

- Noble gases: ^{133}Xe , $^{133\text{m}}\text{Xe}$, ^{135}Xe , ^{138}Xe , $^{85\text{m}}\text{Kr}$, ^{87}Kr , ^{88}Kr ;
- Iodine: ^{131}I , ^{132}I , ^{133}I , ^{134}I , ^{135}I ;
- Caesium: ^{134}Cs , ^{136}Cs , ^{137}Cs .

As a result of their chemical passivity, the escape of fission noble gases is governed only by physical factors such as diffusion and defect size. In contrast, escape of the other fission products is influenced by both their chemical and physical behaviour, including: solubility, volatility, chemical affinity, etc. The iodines, in particular, can be easily ‘trapped’ in the gap between fuel and cladding, and will only be released into the primary coolant if water or steam enters the gap.

Other fission products, produced in large quantities in fuel — such as those belonging to the alkaline earth group (barium, strontium) or to the lanthanides group (lanthanum, cerium) — are not volatile and have very low diffusivity in uranium dioxide. Therefore, they are not released in appreciable quantities into the primary coolant during normal operating conditions. Nevertheless, in the case of large cladding defects, it may happen that water is

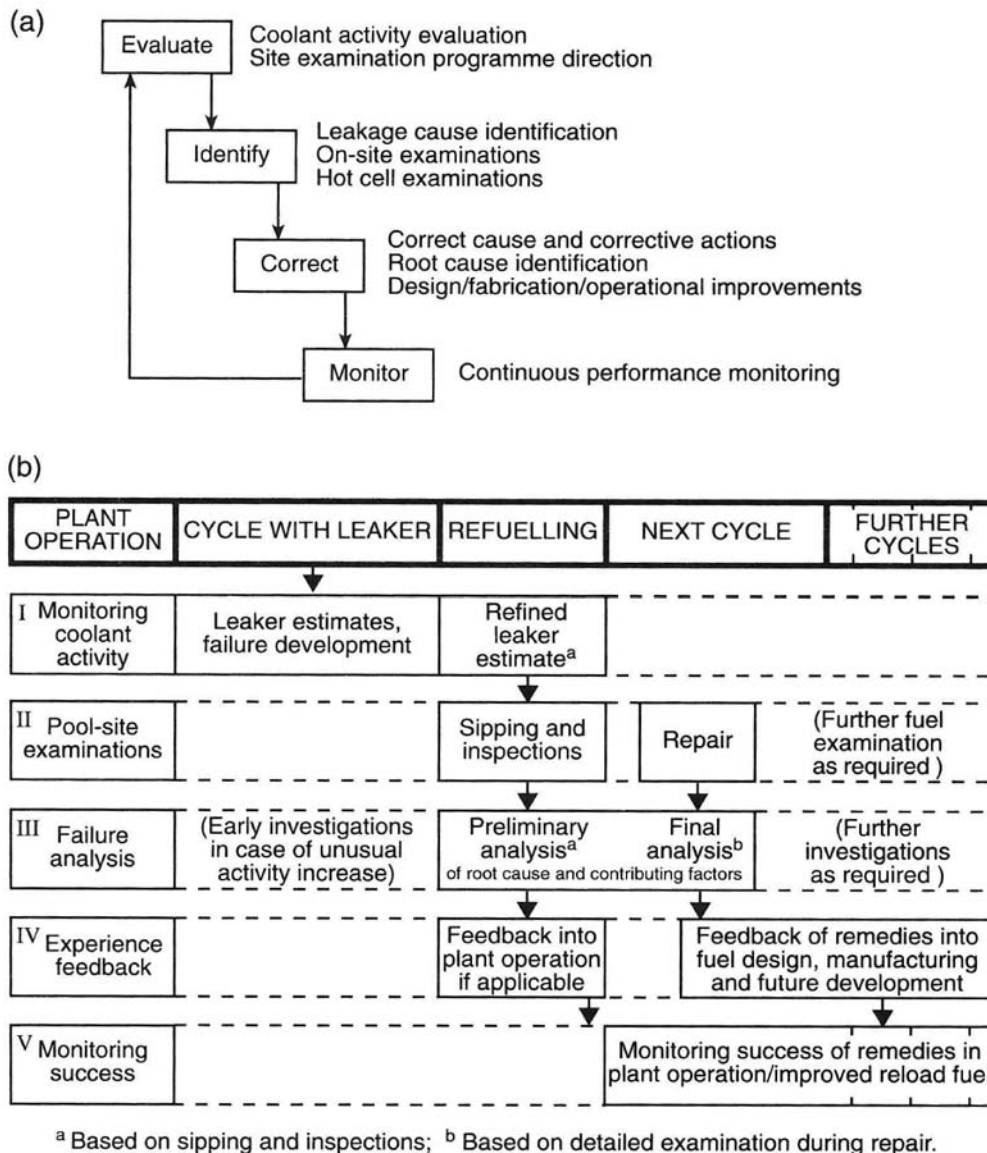


FIG. 4.1. Typical sequence for fuel failure evaluation (a) from Ref. [4.1], (b) from Ref. [4.2].

in direct contact with oxide fuel pellets, leading to fuel erosion. The fissile material is then dispersed into the primary coolant together with the contained fission products.

The actinide ^{239}Np , produced as a result of neutron capture in ^{238}U , is also a good indicator of degraded fuel conditions and significant fuel loss.

Gross activity, sum of the noble gases or dose equivalent iodine (D.E.I.) are also used to compare coolant or off-gas activity to technical specification limits. The purpose of these requirements is not to characterize fuel failures but to limit the amount of activity available for release in an accident event. Dose equivalent iodine is defined as the concentration of ^{131}I which alone would produce the same thyroid dose as all the iodines present. An example of the thyroid dose conversion used for this calculation is given in ICPR 30 [4.4].

Simultaneous to the measurement of fission product data, reactor power and the purification flow rate should be documented, as both have an effect on activity levels.

4.2.2. Leaker diagnostics

Estimating the occurrence of a failure

The onset of a fuel element failure is usually detected by monitoring the gamma activity level associated with specific radioactive fission products in the reactor coolant or off-gas.

Fission product data (in reactor coolant or off-gas) measured at regular intervals and trends throughout the operating cycle allow for a determination of failure times, and an assessment of further operational behaviour.

When a cladding breach occurs along a fuel rod during normal reactor operation, coolant can enter into the fuel to cladding gap and fission products (i.e., notably the volatile species of noble gas and iodine) will be released into the primary, causing a sudden increase in activity levels. With the entry of high pressure coolant through the defect, fuel may be oxidized, which can potentially enhance fission product release. Iodine release can also occur upon reactor shutdown, when the temperature in the fuel to cladding gap drops below saturation, permitting liquid water to dissolve the soluble iodine species in the gap and resulting in an 'iodine-spiking' phenomenon. Iodine rich water remaining in the gap can also be released upon subsequent startup, since the size of the gap is reduced due to fuel expansion.

Any significant change in fission product activity levels should be analysed to determine if it is due to a new fuel failure or an expansion of previously known fuel failures, or whether it can be attributed to other factors such as changes in reactor power, the letdown flow rate or the degassing procedure.

The best early failure indication is an increase in ^{133}Xe activity. Any significant increase above the tramp level or any permanent increase in the steady level should be viewed as a potential failure. The interpretation of corresponding changes in iodine activity is less reliable because small, tight defects cannot induce a measurable change in the steady state level of iodine concentrations. Alternatively, the absence of iodine spiking following a power transient is a reliable indicator of a defect free plant. Iodine spiking following a rapid power change indicates that the core contains one or more leaking fuel rods.

Moreover, the INPO/WANO fuel reliability indicator (FRI) provides an estimate of fuel reliability based on fission product activity present in reactor coolant. This indicator includes an approximate correction for tramp uranium activity and can be used to establish a threshold value below which a unit has a high probability of operating without defects.

Monthly indicator value is calculated using the following equation [4.5]:

$$\text{FRI} = [(A_{131})_N - k(A_{134})_N] * [(L_N / \text{LHGR}) * (100 / P_0)]^{1.5}$$

where:

$(A_{131})_N$ is the average steady state activity of ^{131}I in the coolant, normalized to a purification constant of $2 \text{ E-}5 \text{ s}^{-1}$;

k is the tramp correction coefficient (a constant of 0.0318 based on a tramp material composition of 30% uranium and 70% plutonium);

$(A_{134})_N$ is the average steady state activity of ^{134}I in the coolant normalized to a purification constant of $2 \text{ E-}5 \text{ s}^{-1}$;

L_N is the linear heat generation rate used for normalization (5.5kW/ft);

LHGR is the actual average linear heat generation rate at 100% power (kW/ft);

P_0 is average reactor power in percent at the time activities were measured.

Another useful tool for the analysis of coolant activity during steady state operation is a plot of the release to birth rate ratio (R/B) of measured gas and iodine isotopes against their decay constant. This plot is simply generated by measuring the coolant activity for each isotope and dividing this by its fission yield fraction to provide an effective R/B. Recoil is the dominant mechanism of fission product release from tramp uranium. When fission occurs in tramp materials, fission product nuclides are released directly into the coolant with no waiting time for decay. Thus, the R/B for fission product nuclides from tramp materials in the first approximation would be independent of nuclide half-life. Plants free of defect will exhibit horizontal slopes in R/B plots versus decay constants λ as shown in Fig. 4.2(a).

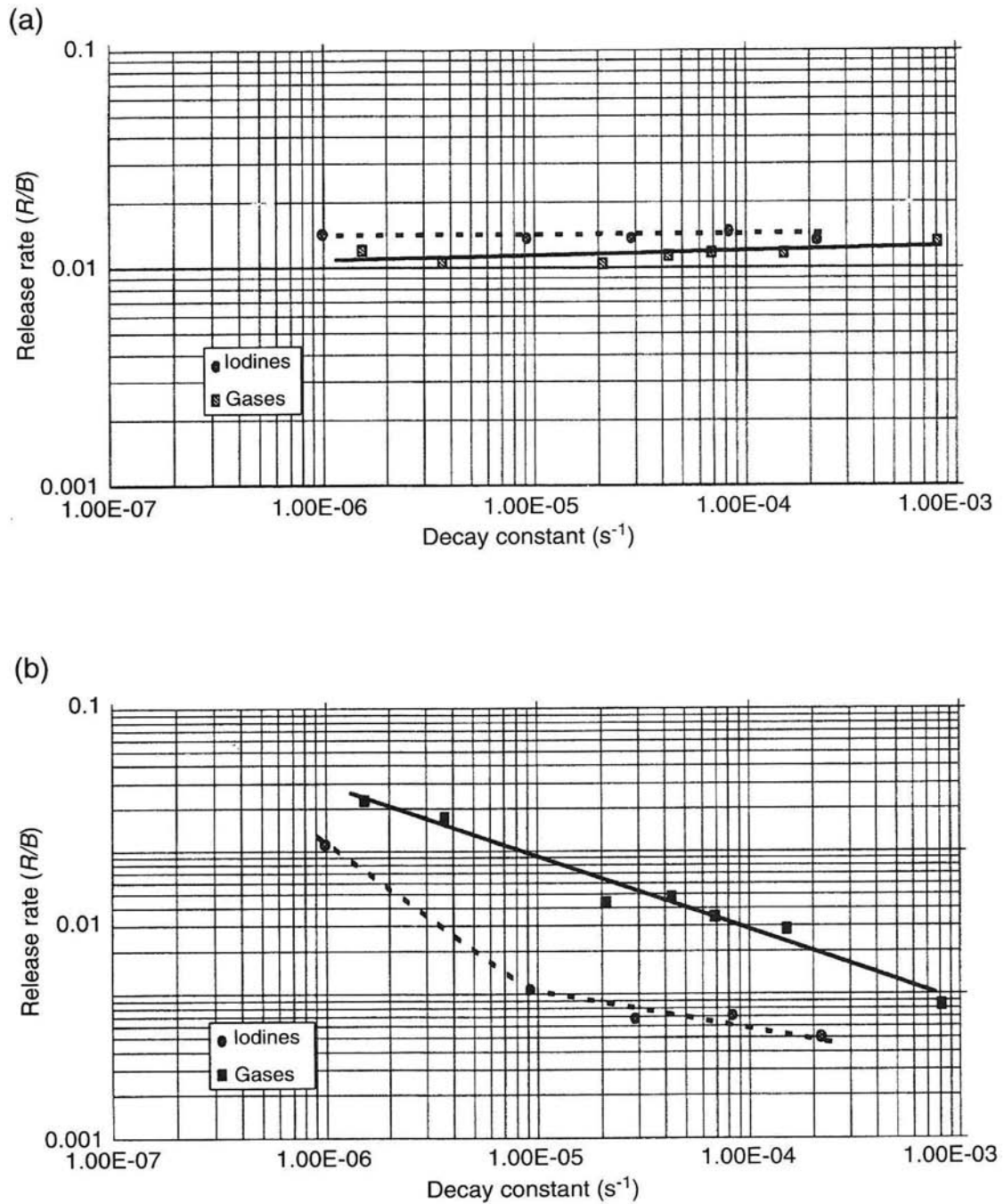


FIG. 4.2. Typical curves of R/B versus λ : (a) core with tramp uranium only; (b) core with defects.

Alternatively, as soon as a defect appears, the slopes in these plots become strongly negative, as shown in Fig. 4.2(b). The slopes of the plots are heavily dependent on the operating power of the failed rods and the defect sizes. Also, a comparison of the R/B of iodines and gases provides an indication of the portion of iodines 'trapped' in the fuel-cladding gap.

In addition to R/B plots, different isotopic activity ratios provide an indication of the presence of failure. The presence of failed rods in the core modifies the distribution of long lived versus short lived isotopes in the primary coolant. In the absence of fuel failure, short lived and long lived isotopes are released into the primary coolant from tramp uranium without delay. When a fuel defect is present, the release of isotopes into the primary coolant is delayed by their diffusion in the fuel pellets, the pellet-cladding gap then through the defect. This delay results in a more important release of long lived isotopes than short lived ones into the primary coolant. The most common

ratios utilized as indicators of fuel failure are: ^{133}Xe to ^{135}Xe , ^{133}Xe to ^{138}Xe or $^{85\text{m}}\text{Kr}$ to ^{87}Kr . A significant change in the value of such ratios is a clear indication of fuel failure.

Estimating the release of fission products from tramp uranium versus defective fuel: Fuel degradation monitoring

The ability to distinguish between releases from tramp uranium and defective fuel is essential in assessing the in-reactor performance of fuel. For example, an increase in the release of fission products from both tramp and defective fuel elements may indicate the appearance of a large open defect releasing fuel to the coolant. Similarly, a constant tramp release accompanied by an increase in the release of the longest lived soluble and gaseous fission products would indicate growth in the number of small defects.

Experience has shown that the yield corrected release of a short lived isotope such as ^{134}I or ^{138}Xe is a good indicator of tramp uranium buildup in the core. Conversely, the release of the longest lived isotopes is mainly due to fuel failure, and uranium contamination effects can be ignored for these isotopes if no large open failures are present in the core.

Thus, degradation or uranium release can be monitored by observing the trends of short lived fission product gamma activities, such as those for ^{134}I or ^{138}Xe . A slowly increasing trend at constant reactor power is an indication of uranium release and deposition within core boundaries, whereas a stepwise increase generally indicates a sudden increase in hole size.

As indicated in Ref. [4.6], the measurement of fission products such as ^{91}Sr and ^{92}Sr , or fuel activation product ^{239}Np can also provide a good indication of fuel dissemination into reactor coolant. However, ^{239}Np cannot be used for fuel degradation assessment in a BWR reactor applying hydrogen water chemistry. At reducing conditions, transuranic nuclides will form an insoluble complex and consequently, its concentration would be largely underestimated.

In addition, the presence of transuranic isotopes (actinides) in the primary coolant is an indication of the presence of fissile materials in the primary coolant due to erosion of fuel pellets through large defects. The most significant actinides are:

- Neptunium ^{239}Np ;
- Plutonium ^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Pu ;
- Americium ^{241}Am ;
- Curium ^{242}Cm , ^{243}Cm , ^{244}Cm .

In CANDU-6 reactors, a delayed neutron (DN) system measures delayed neutron activity in the coolant of each channel. This activity comes from short lived fission products ^{137}I and ^{87}Br , whose activity concentrations are also proportional to tramp uranium levels in the core. When the average DN signal begins to increase, uranium release to the coolant is occurring. When this happens, the suspect channel — identified by having the highest DN signal count rate — is placed at the top of the refuelling list [4.7].

Identification of the fuel failure mechanism

Continuous activity measurements can also provide early information on the fuel failure mechanism. Indeed, some of these failure mechanisms have somewhat typical activity release characteristics which can be utilized in estimating the types of failure present in the core. Some examples of the correlation between fission product escape and fuel rod defect type are provided in Ref. [4.8].

Table 4.1 of Ref. [4.9] lists the general activity characteristics for various fuel failure mechanisms. This table has been compiled from industry experience and is intended to demonstrate a typical situation; actual numbers may vary depending on individual conditions.

TABLE 4.1. TYPICAL CHARACTERISTICS OF VARIOUS FUEL FAILURE MECHANISMS [4.9]

BWR				PWR		
Possible cause	Time in cycle per rod ^a ratio	Activity characteristics slope ^b	Off-gas	R/B versus λ 131/I-133	I-131/rod ^c	Uncorrected
Debris	Typically 0–60 days (but could be any time)	Instantaneous increase following each event; usually followed by gradual iodine activity decrease; typically results in high tramp (I-134)	Low to medium	0.6–0.8	Average	0.5–0.7
Grid fretting	Any time	Multiple failure events; usually constant iodine activity following each event	Not experienced		Average	0.3–0.6
Baffle jetting	Any time (observed BOC to MOC)	Gradual increase in iodine activity, usually with high tramp (I-134)	Not applicable		Low	0.1–0.2
Handling damage	0–60 days	Similar to grid fretting	Not experienced		Average	0.3–0.6
Primary hydriding (moisture)	<100 days	Fast — usually after power manoeuvres; usually multiple events	Average to high	to high 0.5–0.7	Average	
End caps	Any time	Progressive increase	Low	0.6–0.8	Low >1.0	
Pellet-cladding interaction	Following power manoeuvres	Days after manoeuvre	Increasing within a few		Low to medium	0.7–0.9 Not experienced
Crud induced localized corrosion	Any time	Progressive increase in off-gas; number of failures is often high	Low to medium	0.7–0.9	Not applicable	
Cladding manufacturing flaw	Following power changes	Within a few days after power change, often followed by secondary	Low early, high later	0.5–0.7	Not experienced	
Secondary damage	Following power changes	Somewhat abrupt increases in off-gas	High	0.3–0.5	High 0.3–0.5	

^a Typical relative level of noble gases measured in off-gas system, per failed fuel rod. Low means sum of six gases less than 1000 mCi/s per leaking fuel rod; medium means 1000 to 5000 mCi/s per rod; high means over 5000 mCi/s per rod (1 Ci = 3.7×10^{10} Bq).

^b Refers to typical range of slopes for linear fit to log (release to birth) versus log (decay constant) for the six gases. Isotopic release should be corrected for pre-existing tramp fission gas sources.

^c Typical ^{131}I activity in coolant per failed fuel rod. Low means less than 3×10^{-3} mCi/g per leaking fuel rod; medium means 3×10^{-3} to 5×10^{-2} mCi/g per rod; high means over 5×10^{-2} mCi/g per rod (1 Ci = 3.7×10^{10} Bq).

4.2.3. Leaker estimates

To estimate and characterize the number of fuel defects, it is important to define the condition of fuel in the core, to provide information to mitigate degradation or additional failures (for example, through the use of power manoeuvring restrictions), and to plan for end of cycle refuelling outage activities and fuel inspection.

Rough estimates of the number of defect rods in the core can be obtained simply by dividing plant activity by average activity values, which will be generated by a leaking fuel rod in BWR off-gas or PWR coolant. Because the operating power of failed rods — as well as the nature and size of defects — vary significantly, this method only provides satisfactory results if there is an idea of the kind of defects present in the core.

For this reason, numerous methods have been developed to obtain more accuracy by analysing trends in the activities of individual fission products and using sophisticated models based on physical principles. Such methods for estimating the number of failed rods have been developed by EPRI, various fuel vendors, and some utilities, and have been widely documented [4.3, 4.10–4.22].

All approaches are based on the same fundamentals but differ in mode of application. They use analytical models that theoretically describe the nuclide activity release from failed rods. Activity release is divided into two parts: release rate from fuel and release rate from rod to coolant. The coefficients of the models are then determined either using theoretical calculations, or empirically from a large set of coolant activity data for which the number of leaking rods in each of the operating cycles is well characterized.

In these models, activity release is generally characterized by the release to birth ratio R/B, where R is the instantaneous release of nuclide (atom/s) into the primary coolant from all defective rods and B is the birth rate (atom/s) of any radioisotope in a rod operating at average power.

As reported in many studies in the references, the analytical model can then be expressed in terms of an R/B ratio of a given species of fission products released by x failed fuel rods and from tramp uranium deposited on fuel assemblies as:

$$\frac{R}{B} = x \cdot \frac{e}{1 + e} \cdot F_c + c \quad (4.1)$$

where:

- x is the number of failed fuel rods;
- e is the escape rate constant of fission products from fuel to cladding gap (s^{-1});
- λ is the radioactive decay constant (s^{-1});
- F_c is the fission product release ratio from the fuel (this depends on the diffusion coefficient in the fuel and the radioactive decay constant);
- c is the release from uranium surface contamination.

The R/B ratio can also be calculated from activity data collected from the primary coolant or in the off-gas system.

For PWRs:

$$\left(\frac{R}{B} \right)_{meas} = \frac{1 + b}{1} \cdot \frac{mA_m}{FY} \quad (4.2)$$

where:

- β is the purification rate constant (s^{-1});
- m is the mass of primary coolant;
- F is the average fission rate per defective fuel (fission/s);
- Y is the fission product yield;
- A_m is the measured activity concentration per unit of mass.

For BWRs:

$$\left(\frac{R}{B}\right)_{meas} = \frac{\lambda + \beta}{\lambda} \times \frac{mA_g}{FY}, \left(\frac{R}{B}\right)_{meas} = \frac{\lambda + \beta}{\lambda} \times \frac{mA_g}{FY} \quad (4.3)$$

where:

A_g is the measured off-gas activity in disintegrations (s^{-1}).

Equation (4.1) is fitted to measured R/B ratios for a given fission product species. The condition of the core (ϵ, x) can then be determined by fitting parameters to the model.

Fuel failure predictions require coolant sample data from steady state operations. The most accurate estimates of the number of leaking fuel rods in the core, therefore, are based on activity measurements obtained after a few weeks of core operation at constant power following a failure event to allow activities to stabilize.

The limitation in these empirical rod failure models is related to their inability to model large variations in rod power, burnup and defect sizes. They provide an acceptable prediction of the number of failed rods when the failed rods have powers close to the core average and when the defects are small to moderate in size. The accuracy of leaker estimates is then within a factor of two. But when failed rod powers are much higher or lower than average, and/or when defect sizes are large, the empirical models can over-predict or under-predict the number of failures. In these cases, Ref. [4.3] proposes that the average power of defective fuel rods be determined by measuring the ratio of release rates of a given noble gas isotope under steady state conditions for two different power levels (e.g. at 100% and 50% of full reactor power). The ratios of a short lived isotope in relation to a long lived isotope ($^{138}\text{Xe}/^{133}\text{Xe}$ in BWRs or $^{131}\text{I}/^{133}\text{I}$ in PWRs) are also used to indicate the degree of fuel degradation and estimate fuel defect size.

For BWRs, the assessment is mainly performed using off-gas activities. For PWR plants, the assessment has historically been performed by following iodine activities, but some authors claim that only noble gases can be used to estimate the number of defects with reasonable precision.

Experience shows that the BWR off-gas based model and PWR noble gas based models provide a better prediction of fuel rod failure than the PWR iodine based model. Owing to their chemical inertness and ease of release, noble gases are increasingly used to evaluate fuel reliability for PWR analysis.

The best known codes in use are:

- The DIADEME code, developed within the framework of R&D cooperation between the French Atomic Energy Commission (CEA), Electricité de France (EDF) and AREVA [4.17, 4.18, 4.21]. Its diagnosis is based on the analysis of steady state R/B ratios of noble gas and iodine in the primary coolant. In the event of a ^{133}Xe coolant activity peak during a power transient, revealing gas retention in the gap, defect number evaluation may be more accurate with a DIADEME calculation of the total iodine release during the transient, assuming that all the iodine trapped in the gap is released;
- The CADE code, developed by Westinghouse for its PWR reactors [4.10], is based on iodine activities and ^{133}Xe activity;
- The CHIRON code, developed by the Electrical Research Institute (EPRI) for estimating the number of defective fuel rods in PWRs and BWRs [4.12]. This code uses a combination of seven noble gas nuclides and five iodine nuclides — separately and combined — in order to provide an assessment of the number of defects;
- The MERLIN application, developed by Electricité de France (EDF) in the early 1990s [4.19]. The primary objective of this application is to programme daily measurements for nuclear power plants, according to chemistry and radiochemistry specifications, and to collect all measurement results. It constitutes the local and national EDF chemistry and radiochemistry database. A module of this application also permits fuel failure evaluation. This part of the programme uses the seven noble gas nuclides and the five iodine nuclides to provide an assessment of the number of existing defects. During the 1990s a satisfactory agreement between predictions and the inspection results was obtained. However, in 2001, the number of failures through fretting observed in a 1300 MWe nuclear power plant was underestimated by a large margin. Since

then, new empirical correlations have been introduced to fit MERLIN parameters to the number of fuel defects in accordance with more recent experience [4.20];

- For CANDUs, a steady state model has recently been developed into the Defective Element/Tramp Estimate/Characterization Tool (Visual_DETECT) software package. This prototype code numerically fits the fission product release model to coolant activity data using a Marquardt–Levenberg non-linear least squares fitting routine, where the fitting parameters of the model provide for a characterization of the fuel failure(s) and in-core tramp uranium contamination based on well-characterized data derived from in-reactor loop experiments at Chalk River Laboratories (CRL). The analysis procedure specifically provides an estimate of the number of failures, average defect size, and amount of tramp uranium [4.22]. Some other techniques for estimating the number of defective elements in CANDUs are provided in Refs [4.15, 4.16];
- Many of the Russian Federation's power plants also use methods of assessing defective fuel rod parameters in operation. The methods utilized are based on the monitoring of primary circuit coolant activity and application of a design code to interpret data [4.23–4.25]. After assessing the level of reactor core integrity, optimal solutions are taken relevant to the scope and time of implementing control cladding integrity (CIC) during a reactor shutdown, taking into account the unit radiation condition and economic resources of the NPP.

In practice, it is important to realise that all of these models attempt to extract considerable information from small amounts of input data and must be used cautiously and by specialists. One equation must be solved with multiple unknowns. Indeed, in the case of multiple failures, coolant activity is a composite of all activity release from large and small fuel failures in the core, located at different positions on the failed rods and which are often operating at quite different power levels and burnups. The mechanisms which govern the release of fission product into the fuel, then into the gap between the pellet and cladding, are not sufficiently known to develop physical models of release and require the empirical fitting of models. Also, to aid fuel failure evaluation, it is important to use and analyse activity release trends over a cycle to try and identify changes in the number and type of fuel failures in the core.

4.2.4. Burnup estimates: The $^{134}\text{Cs}/^{137}\text{Cs}$ activity ratio

The burnup level of failed rods (and thus sometimes the location of failed fuel assemblies) can be estimated through the $^{134}\text{Cs}/^{137}\text{Cs}$ activity ratio. These two isotopes have different production mechanisms and half-lives, but their concentration within any fuel assembly can be estimated based on time, burnup and enrichment. The half-life of ^{137}Cs is 30.17 years, when fuel is being used up to ~ 4–6 years in a reactor. This is why ^{137}Cs concentration is approximately proportionate to assembly burnup. Meanwhile, ^{134}Cs has a shorter half-life of 2.062 years and is mainly produced from the neutron capture of ^{133}Cs (stable). The production of ^{134}Cs is, therefore, a second order rate reaction with respect to burnup. Measured $^{134}\text{Cs}/^{137}\text{Cs}$ activity ratios are compared to burnup dependent values recalculated using codes such as ORIGEN or APOLLO. In estimating the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio, corrections must be carefully applied to account for any time spent out of the reactor, or for the effects of different enrichments [4.26]. Figure 4.3 shows a typical curve used for such estimates.

A refined method to determine the batch location of failed fuel is proposed in Ref. [4. 27]. This method is based on using the ratio of $^{136}\text{Cs}/^{137}\text{Cs}$ in addition to the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio. The equilibrium activity of ^{136}Cs short life is directly related to rod power, and the $^{136}\text{Cs}/^{137}\text{Cs}$ ratio is more a function of power than burnup. Thus, by using the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio to assess failed fuel rod burnup and the $^{136}\text{Cs}/^{137}\text{Cs}$ ratio to assess failed fuel rod power, it is easier to determine the batch containing the fuel rod failure.

Only a small fraction of the fission product caesium in a defective rod is released into the coolant, while the reactor operates at constant power. For this reason, the best caesium ratio to use in such estimates is the peak value, collected after a significant power reduction during spiking. Samples should be allowed to decay prior to counting to avoid interference from short lived isotopes. The method provides good results when one failed fuel assembly or several failed rods with the same burnup are present. However, burnup estimates may be wrong when multiple failures are present simultaneously in low and high exposure fuel because the caesium ratio is then dominated by exposure to the failed high burnup rods.

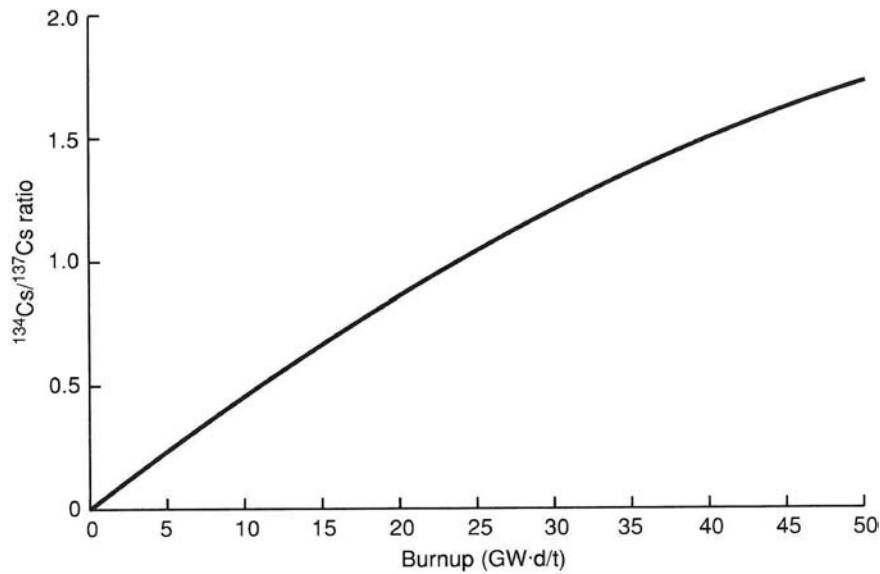


FIG. 4.3. Typical $^{134}\text{Cs}/^{137}\text{Cs}$ activity ratio versus burnup.

4.2.5. UO_2/MOX fuel discrimination: The $^{85\text{m}}\text{Kr}/^{135}\text{Xe}$ activity ratio

For reactors using mixed oxide fuel assembly (MOX), it can be useful to identify early the type of failed rod (MOX or UO_2 fuel rod), especially when activity levels indicate a possible fissile matter release in the primary coolant. Because fission product yields of ^{235}U and ^{239}Pu are similar for xenons and different for kryptons, it is possible to discriminate UO_2 from MOX rod failure through the $^{85\text{m}}\text{Kr}$ to ^{135}Xe activity ratio in the primary coolant. Theoretical calculations performed at CEA [4.28] show that this ratio is greater than 12 for MOX and smaller than 9 for UO_2 fuel regarding burnup up to 40 $\text{GW}\cdot\text{d/t}$ U. However, at higher burnup, use of this ratio can be more difficult, since in UO_2 fuel most fission is also generated in the plutonium that has build up at the pellet periphery. Nevertheless, experience feedback has proven the activity ratio to be useful, as it has permitted identification without ambiguity regarding MOX fuel failure during reactor operation through routine gamma activity measurements, for instance in EDF reactors [4.28] or in Bezneau 1 [4.29].

4.3. LOCALIZATION OF FUEL FAILURES

4.3.1. Sipping procedures

Sipping is the most common technique used to locate fuel failures in both PWRs and BWRs. Identification of fuel rod failure is based on the detection of fission product activity released through defects during sipping. The more common radioisotopes measured are xenon and krypton, and caesium or iodine in water samples.

Various versions of sipping have been used to detect leaking fuel assemblies, depending on:

- The details of configuration and system;
- The physical phenomenon used to promote fission product release, i.e. application of vacuum, heat, elevation (Δp).

These different techniques are known as vacuum sipping, wet sipping and in-mast or telescope sipping, depending on the physical phenomenon and configuration used.

Wet sipping uses the increase in internal pressure of failed rods due to heating of water to expel fission products. Vacuum sipping uses the decrease in external pressure created by a pumping station to expel fission products.

Following these techniques, a fuel assembly is placed either in a sealed container in the spent fuel pool or below a sipping hood for BWR assemblies which have a flow channel. In this last case, sipping can be performed in the spent fuel pool while assemblies are still in the core. The sipping hood is placed over several fuel assemblies and filled with air to isolate the top of each assembly. The air's presence restricts coolant flow to the assemblies and causes a temperature increase which expels fission products. Each assembly is tested individually. In PWRs and WWERs, wet sipping is performed in a cell located in the spent fuel pool. In-core sipping equipment has also been developed for the WWER-440 Paks power plant (Hungary), and the Loviisa plant (Finland).

For sophisticated installations, the assembly is placed in an enclosed canister and activity concentrations in sampling water and cover gas circuits are monitored on-line. A gamma detector is used, or in the case of gas circuits, a beta-type scintillation detector. Water samples are collected for further evaluation and precise determinations of activity concentrations are made with a Ge detector.

Descriptions of typical sipping equipment are given in Refs. [4.30–4.32]. An example is shown in Fig. 4.4.

Each method has its advantages and disadvantages. However, vacuum sipping — due to its accuracy — was the most frequently used technique in BWRs.

In-cell sipping is the most reliable testing method (more than 99% sure). However, flushing of the equipment is extremely important to avoid cross-contamination between assemblies, and the method is still time consuming.

In-mast or telescope sipping

In-mast sipping has been widely developed and used in French PWRs, as shown in Fig. 4.5 [4.33]. The system, installed on the refuelling machine in the reactor building, is designed to identify irradiated leaking fuel assemblies during core unloading or shuffling operations.

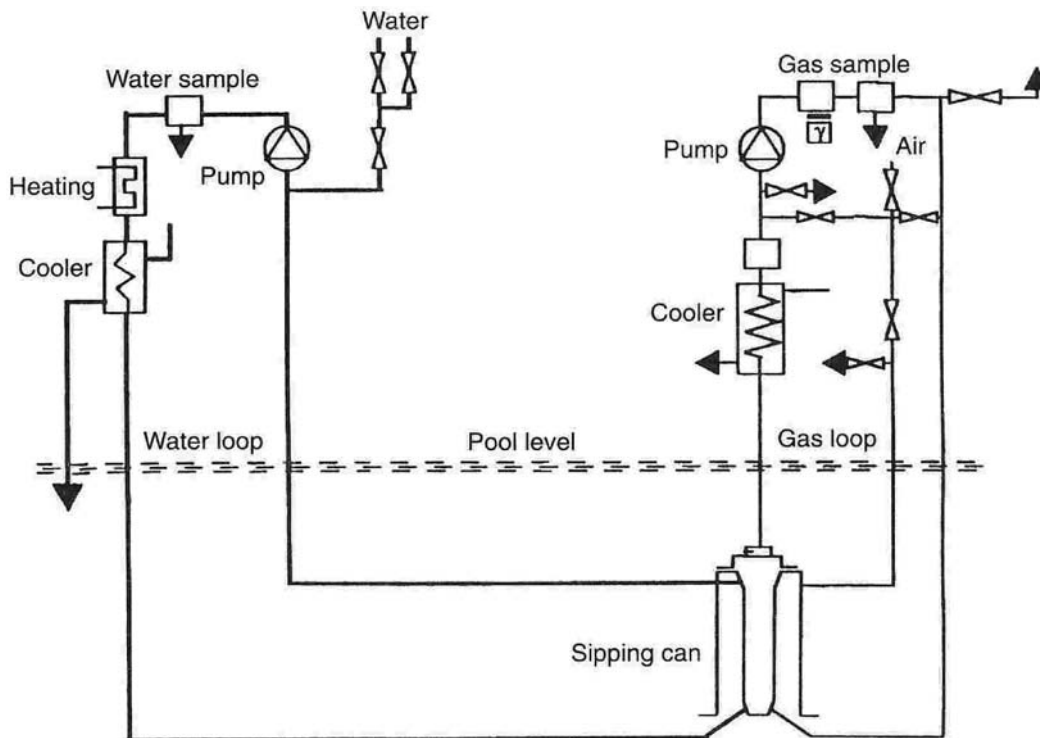


FIG. 4.4. Sipping equipment for a fuel storage pool [4.32].

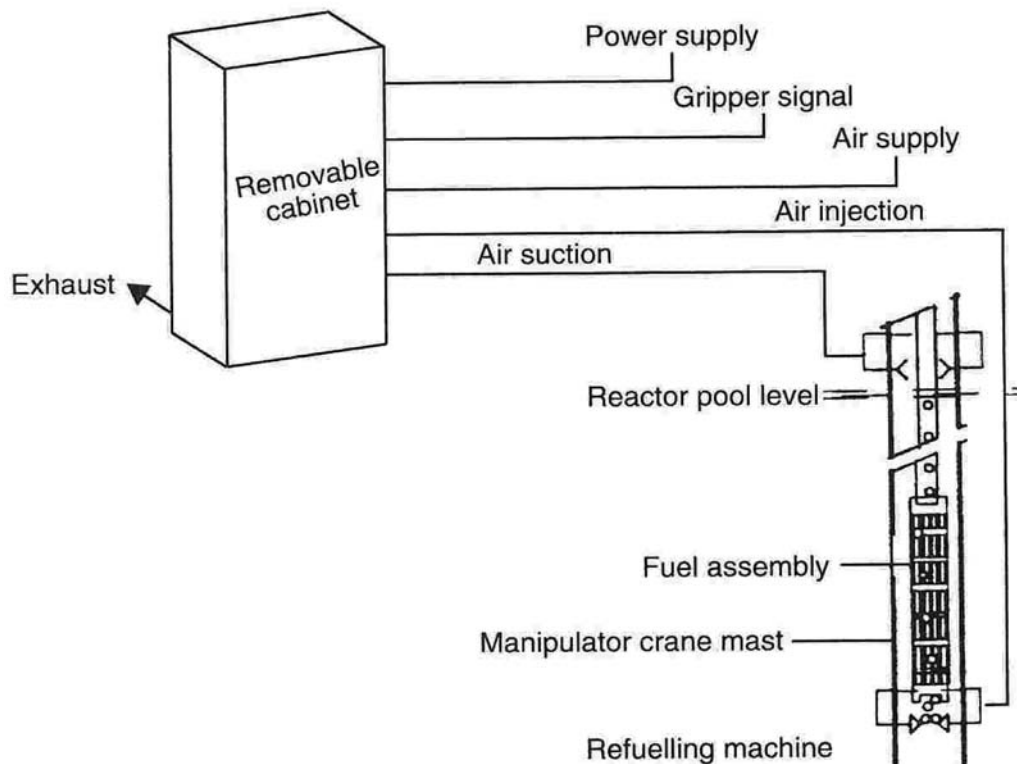


FIG. 4.5. In-mast sipping [4.28].

The on-line system uses a gas sipping method which provides inherently greater sensitivity than conventional water sipping, and is unaffected by reactor pool water contamination. When the core is unloaded or shuffled, each assembly is raised from the core to an upper position inside the machine mast, and the differential pressure caused by the change in elevation promotes the release of fission products from the defective rods. An air stream is continuously injected under the assembly and entrains the gaseous products. This steam is collected above the pool level and routed to a gamma activity measurement unit, where activity is permanently recorded. In order to provide the best signal/background ratio, counting is performed during the gamma peak of the ^{133}Xe isotope for two minutes with the fuel assembly in the upper position within the mast before further movement of the refuelling machine.

From French operating experience, in-mast sipping provides a highly reliable and easy to operate technique for identification of leaking fuel assemblies. Feedback based on experience shows that a failed rod which exhibited a low release rate of fission product activity during the operating cycle, or rods with significant damage, are clearly detected using this technique. Feedback has also shown that more than half of the under calls made were due to poor operation of this device. To remedy this problem, a functional requalification of the device is now being undertaken by the sampling injection of active gas into the sky of the mast.

Similar devices have been developed to detect defective fuel leaks in BWRs [4.34] and now in PWRs. During transportation, water is sucked from the fuel assembly into a hose mounted along or inside the telescope mast. The water is led to degassing equipment placed on the fuel handling machine or on the service room floor. Fission gas is separated from the water and measured on-line with a beta sensitive detector. Soluble fission products in the water can be measured either on-line or in a plant laboratory.

The main advantage with the last two solutions is that sipping can be performed parallel to fuel handling during shuffling or unloading of fuel without significant loss of time or outage schedule impact. Due to its satisfactory reliability, in-mast sipping is now the main technique used to identify failed fuel in both BWRs and PWRs.

4.3.2. Locating and identifying failed fuel in CANDUs

As a result of the CANDU design, it is relatively easy to detect defective fuel during operation. Two systems have been developed for locating fuel defects in the core: the delayed neutron (DN) and feeder scanning (FS) systems.

The first system is used when the reactor is at-power. It locates defective elements by scanning coolant activity immediately downstream of the fuel channels [4.35]. On each side of the reactor, sampling lines carry coolant away from the outlet end of each fuel channel to a common room within the reactor building (see Fig. 4.6). The presence of delayed neutron emitting fission products (^{137}I and ^{87}Br) in sampling lines is detected in this room using BF3 detectors. If neutron activity in a coolant sampling line is higher than normal, the corresponding fuel channel is suspected of containing a defective element. The DN system has sufficient sensitivity to locate fuel defects with very small holes. The Bruce and CANDU-6 reactors employ the DN system.

The second system is used when a reactor is shut down. It locates defective elements by scanning activity on the inside surfaces of the outlet feeders connecting the fuel channels to a common outlet header [4.36]. The presence of gamma emitting fission products is detected by Geiger-Müller detectors which move within guide tubes that transverse the outlet feeders. If gamma activity in a specific feeder is higher than normal, then the corresponding fuel channel is suspected of containing a defective element. The FS system only locates defects that have deteriorated to the point of releasing uranium and fission products which deposit immediately downstream of the defect. The Darlington reactors employ the FS system.

The Pickering reactors have no failed fuel location system. Normally, a fuel defect cannot be located in the core, but it can be detected during discharge from the core.

Fuel removal confirmation

After a suspect channel has been refuelled in a CANDU, the defective fuel element is confirmed to have been discharged by various methods, again depending on the station:

- Inspecting the discharged bundles in the bay which, of course, corresponds to LWR practice (see Section 4.3);
- Monitoring gamma activity near the spent fuel handling system when bundles are en route from the reactor to the fuel bay (wet or dry sipping) and/or when they are residing in the fuel bay (wet sipping);
- Monitoring delayed neutron activity of the coolant at the outlet end of the channel during refuelling.

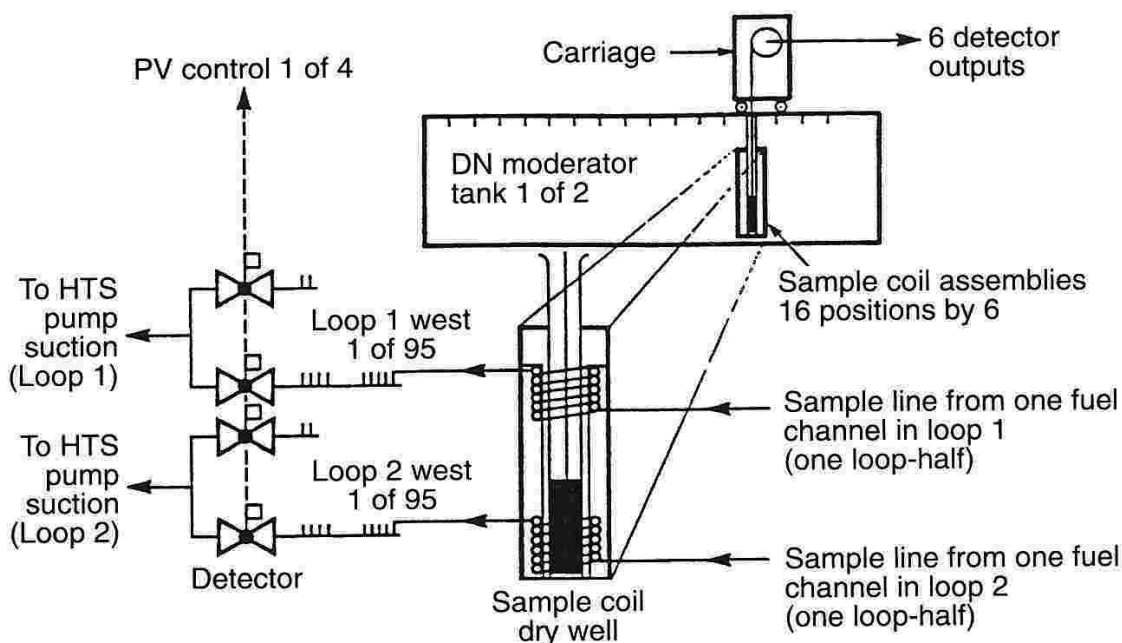


FIG. 4.6. Schematic diagram of a scanning room for the delayed neutron system in a CANDU-6 reactor.

The first method provides direct confirmation that a defective bundle has been removed. Portable underwater TV cameras are used at multiunit stations operated by Ontario Hydro, whereas periscopes are used at single unit CANDU-6 stations [4.36]. Photographs of the defective elements confirm that a defective bundle has been removed.

The second and third methods provide indirect confirmation that a defective bundle has been removed by monitoring gamma activity near the spent fuel handling system or in fuel bays. At Bruce, dry sipping techniques [4.37] are used to monitor airborne gamma activity in spent fuel transfer mechanisms where bundles are transferred from the heavy water to the light water environment of the fuel bay. A higher than normal signal that lingers after bundles have been transferred usually indicates the presence of a defective element. Two other techniques have been developed at CANDU-6 stations to confirm that defects have been discharged from the core. One technique depends on the radiation levels of fission products in the heavy water which is inside the fuelling machine. Before discharge of irradiated fuel to a bay, fuelling machine heavy water is transferred to a nearby drain tank. The presence of a defect is indicated when gamma fields near the tank trigger an area alarm gamma monitor. Another technique developed in the inspection bay at Point Lepreau is based on 'wet sipping', or measuring the gamma activity of water samples near recently discharged bundles. Again, a defect is present if gamma activity is unusually high.

Monitoring delayed neutron activity at the outlet end of a fuel channel during refuelling also provides some confirmation that a defect is being discharged from the core [4.38]. At CANDU-6 sites, special refuelling procedures are sometimes used which involve slow displacement of the fuel column while monitoring the DN signal of the channel. When the signal drops to below 'pre-defect' levels, the defect has been pushed outside the core boundary.

4.3.3. Locating and identifying failed fuel in WWERs

Two different systems are currently used during scheduled outages to control cladding integrity in Russian Federation NPPs; a 'sipping' system and a DAD (defective assembly defection). The 'sipping' system, used at Russian Federation plants [4.25], is based on registration of FPG activity releases through a defect in a leaker cladding within a tested fuel assembly placed in a reloading machine bar. FPGs are released as a result of a decrease in coolant external pressure when a tested fuel assembly is raised to the top transportation position. The resultant drop between internal and external pressures leads to gas and fission gas products being dissolved from a leaker into surrounding coolant.

The DAD system comprises a case into which a fuel assembly to be tested is placed, a system of pumping a coolant (pure water or a boric acid solution containing water) and of changing pressure inside the case. At the expense of a pressure rise in the case, water enters a defective fuel rod and compresses gas volumes inside that failed fuel rod. After a pressure drop, compressed gas volumes push out an excess volume of coolant with fission gas products dissolved in it. Spectrometric measurements of the samples taken register nuclides that have left a leaker.

4.3.4. Ultrasonic testing

Ultrasonic testing (UT) is a technique in the industry well known for locating failed assemblies and rods in PWRs [4.39]. The concept was originally developed by Brown Boveri and company in the early 1970s. Failed fuel rod detection systems can be based on different techniques; the pitch and catch, the through transmission or the pulse-echo.

All of these systems inspect fuel rods either circumferentially or radially for the presence of water in the pellet-cladding gap. The loss of energy at the inside diameter when water is present inside a rod is sufficient to provide a measurable difference in the amount of energy transmitted through the rod or reflected from the incident surface and arriving at the receiver. An array of probes mounted on flexible blades are inserted in the spacing between the rows of fuel rods in a fuel bundle. These blades are moved by a remotely controlled automatic manipulator that is positioned under water on the spent fuel storage racks.

Until recently, UT was the most common technique used in the United States of America to identify leaking assemblies and rods for PWRs, but its use has decreased in favour of more reliable in-mast sipping to identify failed assemblies. Though this technique has demonstrated satisfactory accuracy and reliability (currently estimated at

80% to 90%), it has its limitations, which have led to occasional under calls and over calls. To detect failed rods, the presence of water at the probe/transducer location is necessary. If water is not present at the location being tested, the rod will not be identified as a leaking rod. Furthermore, if water is not present around the entire rod circumference, a degraded signal may result which is difficult to interpret. To improve system reliability, several scans at different levels may be useful. Other effects, such as heavy crud on a rod surface, thick oxide, or pellet-clad bonding can also attenuate the UT signal and lead to misinterpretations. The effects of crud and pellet-cladding contact become more of a concern at higher burnup levels.

Nevertheless UT remains very useful for failed rod identification in failed fuel assemblies before fuel failure cause evaluation or repair.

4.3.5. Flux tilting (BWRs)

Flux tilting is used to locate fuel failure in BWRs during operation. If the number of failed rods is believed to be small, the operational procedures of flux tilting are used with some success to identify suspect regions or fuel cells prior to the end of a cycle, thus reducing the number of assemblies that need to be sipped.

The procedure involves creating a local power change through sequential insertion of control blades while monitoring changes in off-gas activity. If a failed rod is influenced by a power change, there will be an increase in the release rate of fission products from the defect. Increased fission gas release can be recorded in the off-gas system by using existing current measuring systems, but accuracy can be significantly improved by on-line measuring with Ge detectors for nuclide specific off-gas analyses.

The most common practice, described in Ref. [4.40], consists of fully inserting single or multiple control blades at reduced power (<65% of rated power) to exclude extra-secondary damage to failed fuel by utilizing severe local power changes.

The typical procedure consists of:

- Reducing core power;
- Fully inserting single or multiple control blades;
- Leaving the blade(s) inserted for a sufficiently long time to observe a change in off-gas activity;
- Extracting the control blade(s) to their original position;
- Waiting for a sufficiently long time to observe a return to unperturbed activity levels;
- Repeating the sequence with the next control blade(s).

The authors of Ref. [4.41] recommend carrying out flux tilting investigations at as high a reactor power level as possible, with moderate control rod movement, and only 10% control rod insertion. This makes use of the fact that fission product release is higher at high fuel temperatures, thus making flux tilting as sensitive as possible and, at the same time, reducing the transient necessary to produce a significant signal.

Data provided by these procedures can reduce the scope of sipping campaigns, but this approach is subject to the risk of missing some failures. Despite this limitation, as a result of increasing pressure to reduce plant operational and maintenance costs, methods such as these are becoming more widespread in use. Some years ago, flux tilting was widely used to locate leaking Zr liners or other sensitive fuels as early as possible; severe degradation could be prevented by shadowing leaks with fully inserted control blades.

4.4. FUEL EXAMINATION

4.4.1. Poolside inspection techniques

Poolside inspection techniques are widely documented in Ref. [4.42]. Visual examination of a whole assembly (directly or after removing the channel, if applicable) is, of course, the least expensive technique and can provide useful information on the behaviour of a fuel assembly and its structural parts. If failed fuel rods are on the periphery, it is sometimes successful in identifying failure mechanism. However, this is the exception rather than the rule, thus other techniques have to be used to provide more information.

Disassembly of the assembly

In general, root cause determination of a fuel rod leak necessitates rod removal and individual fuel rod examination. For that purpose, the most valuable information comes from fuel repair campaigns.

All recent BWR and PWR fuel designs have a removable top nozzle. They can be disassembled under water by removing the upper tie plate and channel for BWR assemblies. This flexibility provides great advantages in inspecting and repairing failed fuel assemblies.

All fuel vendors have developed fuel assembly repair stations. The typical repair of a fuel assembly follows the steps below:

- Installation of the fuel assembly into an assembly container or fuel elevator;
- Lifting of the elevator to its operating position (when about 3 m water is above the assembly);
- Removal of the end fitting assembly;
- Extraction of identified failed fuel rods;
- Insertion of replacement rods into the free position;
- Reassembly.

If there is damage to the structure of the assembly, the sound fuel rods can be transferred to a replacement cage. In this case, the upper end fitting of the damaged assembly can be removed and all rods transferred individually to the new skeleton.

Visual inspection

Visual inspection of extracted fuel rods is the best method of identifying a failure mechanism. Common mechanisms such as debris induced failure, grid-rod fretting and some manufacturing defects can be quickly identified by visual examination.

Visual examination is performed with a high definition underwater camera or a periscope, which allows for the examination of particular features in more detail.

Eddy current testing (ECT)

Eddy current testing (ECT) techniques are used to determine the overall condition of an individual fuel rod. ECTs detect partial or through-wall defects in fuel rod cladding and permit identification of the axial position of defects. Generally, the technique used is based on an encircling coil. Examination is performed during upward or downward rod motion through a cylindrical measuring coil. Two components (phase and amplitude) of the EC signal are recorded during this rod motion. The probe is calibrated using a calibration rod with standard defects of well defined size.

One advantage of EC is that it can be used both for reliable identification of clad defects and also to identify small defects where water intrusion has not yet occurred. However, in some cases, the presence of heavy crud on fuel rods may affect ECT results, making defect identification more difficult or resulting in over-calls.

Other detailed measurements

Various other non-destructive techniques can provide supplementary information to identify a failure mechanism if visual observations are not sufficient. The choice of which to use depends on the suspected failure mechanism. Some techniques that may be employed are:

- Oxide thickness measurements using eddy current lift-off signals;
- Dimensional measurements such as assembly or rod irradiation growth, rod bow, assembly bow and twist;
- Measurement of cladding wear at the grid level using a visual or UT device;
- Measurement of fuel diameter using profilometry to investigate excessive cladding creep down;
- Measurement of rod withdrawal force;
- Crud scraping (crud samples are taken from the fuel rod and analysed to determine their constituent elements).

4.4.2. Hot cell examinations

For a small fraction of fuel identified as failed, the cause of failure cannot be determined by poolside inspection techniques, and hot cell examinations are desirable. Hot cell examinations are also useful to more accurately determine the mechanism of failure and to take appropriate counter-measures. For example, it is important to determine whether the cause of a PCI failure is non-classic (i.e., due to missing pellet surface) or classic (i.e., iodine assisted cracking). In the first case, examination results can provide very direct feedback to the fabrication process. The fuel supplier can re-evaluate current pellet integrity specifications, rod handling specifications, and rod loading techniques. In the second case, the fuel supplier can re-evaluate uncertainties in core design and surveillance methods, accuracy of modelling control element effects on rod local power, and susceptibility of cladding material to stress corrosion cracking.

While hot cell examinations can provide detailed insight into the root causes of failures, such examinations are expensive and require fuel to be shipped from the plant site to a laboratory. Examinations away from the reactor site are especially considered to be valuable when there are multiple fuel defects of unknown cause. Depending on the suspected cause of failure, it is prudent to ship a corresponding intact rod to the hot cells in order to have a reference sample.

The facilities used in hot cells for post-irradiation examination (PIE) of failed rods are widely documented in Refs. [4.43–4.46].

Non-destructive tests such as visual inspection, X ray or neutron radiography, profilometry, ECT, axial gamma scanning and pressure testing of the ends of defective fuel rods may be used to locate and identify the cause of failure, as well as the extent of cladding and fuel damage. Sometimes for LWR and more frequently for CANDU fuel, puncture and gas collection of intact fuel elements neighbouring defective ones are carried out. Higher than expected pressures may indicate high power operation, degraded heat transfer, or internal contamination.

Neutron radiography of intact or defective fuel elements to search for regions of hydride is particularly useful for CANDU fuel. Owing to different neutron absorption properties, it is possible to distinguish between zirconium hydride and deuteride in regions of secondary damage on fuel elements. Deuterided regions are expected as secondary damage in association with coolant ingress, whereas hydrided regions may be indicative of excess hydrogen gas inside the fuel element stemming from manufacturing.

The most important part of hot cell work is the destructive tests — such as metallography, ceramography, hydrogen analysis, fractography and X ray microanalysis of fuel and cladding — which may be employed to clarify mechanisms of fuel failure. Microscopic examination of a specimen may provide some information about oxide thickness on cladding surfaces and along crack surfaces, incipient cracking within cladding, graphite retention on inside cladding surfaces, hydrogen levels in the structural components, isotherms within pellets, etc. The results of these microscopic examinations may also provide hints regarding progression of a defect. For example, oxide formation along a fracture surface confirms that a crack developed under hot conditions in a core rather than during handling. Examples of hot cell examinations on failed fuel rods are described in Refs. [4.47, 4.48].

4.5. FUEL FAILURE ANALYSIS

4.5.1. Background

It is now relevant to consider in more detail a further search for root causes and contributing factors in relation to fuel failures. Failure analysis has some similarities to criminology. Few failure categories can be easily clarified by fuel examination, i.e. mainly external mechanical damage. In most other cases, the identification of the failure mechanism may require substantial efforts beyond fuel examination, and a full clarification of root causes and all contributing factors can take months or years of criminal type work. Good examples of this are some corrosion type failures and grid–rod fretting, as will be discussed in Section 5.

Several attempts have been made to develop decision trees in the search for root causes. This may be helpful in a limited number of cases, but for general applications it is of little use and can be sometimes misleading. The best approach in complex situations is to list all conceivable root causes and to produce evidence for and against them until the cause has been identified and attributed to a specific area such as manufacturing and handling, plant operation and maintenance, or fuel and plant design features. Determining contributing factors from all these areas

is essential to define effective counter measures and to optimize future fuel designs. In some situations, it may be necessary to perform supplementary laboratory tests for final verification.

Some general aspects for performing analysis in different areas will be discussed in the following sections.

4.5.2. Manufacturing and handling

For all failures which are not obviously caused by external or operational effects alone, it is mandatory to look for possible manufacturing or handling influences. The first step is to identify the specific field (e.g. clad/pellet/rod/spacer fabrication, assembling, handling, transportation, and repair) where potential deficiencies could lead or contribute to a given type of failure, and to examine related quality documentation. Normally, results are disappointing, since routine documentation must not contain deviations which could lead to fuel failure; thus the chance of finding the root cause in this way is extremely small. Nevertheless, this routine check must be done in case something has been overlooked.

Usually a much deeper analysis is required including, for example, the examination of available documents and notebooks at the manufacturing plant, the evaluation of statistics on quality data and rejected materials, and interviews with the people involved in respective manufacturing and quality control steps. Occasionally, the root cause is found. More frequently, the exact cause of a suspect manufacturing failure cannot be finally clarified. However, the chance of finding and eliminating a specific root cause for a given failure is not the only benefit of this tedious work. Another advantage is that during such analyses, it is possible to find previously unnoticed weaknesses in manufacturing or quality control procedures, and to take adequate counter measures to prevent potential deficiencies in later manufacturing. This latter effect has significantly contributed to lowering failure rates from manufacturing causes. A good example is the search for the root cause of mid-grid fretting in earlier B&W fuel with Inconel grids [4.4].

Sometimes, an even more complex, unknown situation arises if the root cause of a problem is a deviation from specified quality or there is an inadequate specification. In such cases, specifications themselves have to be thoroughly analysed for potential weaknesses. Another example is the identification of ‘manufacturing flaws’ in GE cladding, which could equally influence primary and secondary failures [4.50].

4.5.3. Operation and design

The most important operating information for failed fuel is the power/burnup history, and the related influence of ‘control history’ if applicable, particularly in BWRs and CANDUs. Power/burnup histories are not only relevant to PCI failures, for which reconstructions of local power histories have been traditionally performed. For many types of corrosion problems, power histories are equally important, together with thermal hydraulic operating conditions and potential effects of water chemistry and crud depositions. Assembly bowing is strongly influenced by power/burnup and the respective gradients throughout an assembly cross-section (see Section 5.10). Evaluation of respective data should be a standard part of analysing fuel failures taking note of correlations to operation.

Beyond power histories, exact core positions — as well as the design and operating data of neighbouring assemblies — can be relevant for failure analysis. Trivial examples include some problems with assembly bow, where the gap (or interference) between neighbouring assemblies, i.e. the difference of neighbouring bow vectors, determines the consequences. Fretting problems can be influenced by cross-flow and thus be related to the design features of neighbouring assemblies. Peripheral core positions require particular attention in PWRs since (besides baffle jetting failures) an increasing number of fretting problems have been observed in recent years in these positions.

Fuel failures sometimes bring into question design adequacy. A classic example is PCI, for which early operations revealed that failures could occur in normal operation and manoeuvring guidelines were required. Since specific design and operation features can be part of the root cause of, or at least contribute to, many failure mechanisms, checking adequacy should also be a routine part of fuel failure analysis in relevant categories.

REFERENCES TO SECTION 4

- [4.1] WILSON, H.W., MILLER, R.S., KUNISHI, H., “Westinghouse fuel performance experience”, Proc. Tech. Comm. Mtg on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management, Dimitrovgrad, 1992, IAEA-TECDOC-709, IAEA, Vienna (1993) 133.
- [4.2] VON JAN, R., “SIEMENS KWU experience with LWR fuel: Failure evaluation, mechanisms and remedies”, *ibid.*, p. 50.
- [4.3] LEUTHROT, C., BRISSAUD, A., HARRER, A., “Correlation between fission product activity in PWR primary water and characteristics of defect in fuel cladding”, *ibid.*, p. 67.
- [4.4] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Limits for Intakes of Radionuclides by Workers, Part 1, ICRP Publication No. 30, Pergamon Press, Oxford (1979).
- [4.5] INSTITUTE OF NUCLEAR POWER OPERATIONS, Detailed Description of Overall Performance Indicators, INPO (1993) 96–003.
- [4.6] INGEMANSSON, T., RUDLING, P., LUNDGREN, K., “Assessment of Fuel Washout in LWRs — New Methodologies”, Proc. Int. Mtg on LWR Fuel Performance, Orlando, 2004, ANS (2004) 833–844.
- [4.7] MANZER, A.M., “Transport mechanisms of uranium released to the coolant from fuel defects”, Proc. 1st Int. Conf. on CANDU Fuel, Chalk River, 1986, CNS, Toronto (1986).
- [4.8] SCHUSTER, E., VON JAN, R., FISCHER, G., “Evaluation of fuel performance from coolant activity data”, Proc. Int. Top. Mtg LWR Fuel Performance, Avignon, 1991, Vol. 1, SFEN, Paris (1991) 285.
- [4.9] ELECTRIC POWER RESEARCH INSTITUTE, Fuel Integrity Reliability Improvement Guidelines Rep. Tr-1000659, EPRI, Charlotte, NC (1992).
- [4.10] BURMAN, D.L., et al., “Development of a coolant activity evaluation model & related application experience”, Proc. Int. Top. Mtg on LWR Fuel Performance, Avignon, 1991, Vol. 1, SFEN, Paris (1991) 363.
- [4.11] DUMONT, A., “FRAGEMA fuel reliability: From detection of fuel failures to the feedback on design and fabrication”, Proc. Tech. Comm. Mtg on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management, Dimitrovgrad, 1992, IAEA-TECDOC-709, IAEA, Vienna (1993) 46.
- [4.12] BEYER, C.E., “An analytical model for estimating the number and size of defected fuel rods in an operating reactor”, Proc. Int. Top. Mtg on LWR Fuel Performance, Avignon, 1991, Vol. 2, SFEN, Paris (1991) 437.
- [4.13] LEWIS, B.J., et al., “An expert system for the monitoring of failed fuel in Candu reactors”, *ibid.*, p. 385.
- [4.14] STORA, G.P., CHENEBAULT, P., “Defect fuel rod evaluation in EDF PWR reactors”, Proc. ANS–ENS Top. Mtg on Reactor Safety Aspects of Fuel Behaviour, Sun Valley, ID, 1981, Vol. 1, CENG (1981) 119.
- [4.15] MANZER, A.M., SANCTON, R.W., MACICI, N., “Canadian CANDU-6 perspective on fuel integrity performance indicators”, Proc. Int. Mtg on Nuclear Power Plant Operation, Chicago, IL, 1987, ANS, LaGrange Park, IL (1987).
- [4.16] MANZER, A.M., “In-core assessment of defective fuel in CANDU-600 reactors”, Proc. Spec. Mtg on Post-Irradiation Examination and Experience, Tokyo, 1984, IAEA-IWGFPT/22, IAEA, Vienna (1985) 263.
- [4.17] GENIN, J.B., et al., “DIADEME: A computer code to assess in operation defective fuel characteristics and primary circuit contamination”, Int. Conf. on Water Chemistry of Nuclear Reactor Systems, Avignon, 2002, FNES.
- [4.18] PARRAT, D., et al., “Failed rod diagnosis and primary circuit contamination level determination, thanks to the DIADEME code”, IAEA-TECDOC-1345, IAEA, Vienna (2003).
- [4.19] TIGERAS, A., et al., “MERLIN: Modelling fuel defects at EDF power plants”, Int. Conf. on Water Chemistry of Nuclear Reactor Systems, San Francisco, 2004, EPRI (2004).
- [4.20] TIGERAS, A., et al., “Improvement of fuel failure assessment based on radiochemical parameters (MERLIN code) taking in account the thermal mechanical fuel rod calculations (CYRANO3 code)”, Int. Conf. on Water Chemistry of Nuclear Reactor Systems, San Francisco, 2004.
- [4.21] PARRAT, D., et al., “Failed rod diagnosis and primary circuit contamination level determination, thanks to the DIADEME code”, IAEA-TECDOC-1345, IAEA, Vienna (2003).
- [4.22] LEWIS, B.J. GREEN, R.J., CHE, C., “A prototype expert system for the monitoring of defected nuclear fuel elements in Canada deuterium uranium reactors”, Nucl. Technol. **98** (1992) 307–321.
- [4.23] SLAVYAGIN, P., LUSANOVA, L., MIGLO, V., “Regulation of the fission product activity in the primary coolant and assessment of defective fuel rod characteristics in steady-state WWER-type reactor operation”, Proc. IAEA Int. Tech. Mtg on Fuel Failure in Water Reactors: Causes and Mitigation, Bratislava, 2002 (2002) 326–337.
- [4.24] HÓZER, Z., “Simulation of leaking fuel rods”, Proc. 6th Int. Conf. on WWER Fuel Performance “Modelling and Experimental Support”, Paper 3B.9, Albena, 2005.
- [4.25] SLAVJAGIN, P., “Optimization of the parameters of cladding integrity control using working rod fuel discharge machine in WWER-1000 reactors”, Rep. Bulgarian–Russian Practical Seminar on Operation and Implementation of WWER Fuel of New Generation, Slnchev Brjag, Bulgaria, 2–6 October 2006 (in Russian).
- [4.26] ELECTRIC POWER RESEARCH INSTITUTE, Fuel Integrity Monitoring and Failure Evaluation Handbook, Revision 1 Tr-1003407 (2003).

- [4.27] SAYLES, C.W., CHANG, R.Y., "The use of three cesium isotopes for the batch identification of a single fuel failure in a pressurized water reactor", Int. Topical Mtg on Light Water Reactor Fuel Performance, Portland, 1997.
- [4.28] PARRAT, D., LEUTHROT, C., HARRER, A., DANGOULEME, D., "Behaviour of a defective Mox fuel rod in a PWR", CT IAEA Recycling of Plutonium and Uranium in Water Reactor Fuel, Newby Bridge, Windermere, Cumbria, 1995.
- [4.29] BAY, H., et al., "Debris induced fuel failures on U an MOX fuel in Beznau", Proc. Tech. Comm. Mtg on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management, Dimitrovgrad, 1992, IAEA-TECDOC-709, IAEA, Vienna (1993) 133.
- [4.30] KNAAB, H., KNECHT, K., JUNG, W., "New pool-side inspection and repair techniques based on computer support", Proc. Tech. Comm. Mtg on Poolside Inspection, Repair and Reconstitution of LWR Fuel Elements, Lyon, 1991, IAEA-TECDOC-692, IAEA, Vienna (1993) 26.
- [4.31] BEUNECHE, H., et al., "Overview of fuel sipping in French power plants", Proc. Int. Top. Mtg on LWR Fuel Performance, Williamsburg, 1988, ANS, LaGrange Park, IL (1988) 304.
- [4.32] SIEMENS AG, Siemens Fuel Assemblies — References '94, Siemens AG Power Generation Group, Erlangen (1994).
- [4.33] BORDY, M., DANGOULEME, D., PARRAT, D., PENOZA, R.W., "On-line mast sipping in PWRs: Overview and recent results", Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, 1994, ANS, LaGrange Park, IL (1994) 542.
- [4.34] INGEMANSSON, T., DELERYD, R., LUNDHOLM, L., HALLSTADIUS, L., "Fuel failure management in BWRs", Jahrestagung Kerntechnik 93, Köln, 1993.
- [4.35] MANZER, A.M., SANCTON, R.W., "Detection of defective fuel in an operating CANDU-600 MW(e) reactor", Proc. Conf. on Fission Product Behaviour and Source Term Research, Snowbird, 1984, ANS, LaGrange Park, IL (1984).
- [4.36] LIPSETT, J.J., STEWART, W.B., Failed fuel location in CANDU-PHW reactors using a feeder scanning technique, IEEE Trans. Nucl. Sci. NS-23 1 (1976).
- [4.37] MacDONALD, R.D., FLOYD, M.R., LEWIS, B.J., MANZER, A.M., TRUANT, P.T., Detecting, Locating and Identifying Failed Fuel in Canadian Power Reactors, Rep. AECL-9714, AEC, Chalk River (1990).
- [4.38] MANZER, A.M., "Defective fuel detection in CANDU 600s", Proc. 6th Annual CNS Conf. Ottawa, 1985, CNS, Ottawa (1985).
- [4.39] MUSTAN, A., State of the art ultrasonic detection of failed fuel, Trans. Am. Nucl. Soc. 61 (1990) 47.
- [4.40] POTTS, G.A., "Failed fuel management", TopFuel 95 (Proc. KTG-ENS Topical Mtg, Würzburg, 1995) Vol. 2, INFORUM Verlags- und Verwaltungsgesellschaft mbH, Bonn (1995) 121.
- [4.41] INGEMANSSON, T., et al., "An advanced method for flux tilting", Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, FL, 1994, ANS, LaGrange Park, IL (1994) 459.
- [4.42] INTERNATIONAL ATOMIC ENERGY AGENCY, Poolside Inspection, Repair and Reconstitution of LWR Fuel Elements, IAEA-TECDOC-692, IAEA, Vienna (1993).
- [4.43] INTERNATIONAL ATOMIC ENERGY AGENCY, Post Irradiation Examination Techniques for Water Reactors, IAEA-IWGFPT/37, IAEA, Vienna (1991).
- [4.44] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidebook on Destructive Examination of Water Reactor Fuel, Technical Reports Series No. 385, IAEA, Vienna (1997).
- [4.45] INTERNATIONAL ATOMIC ENERGY AGENCY, Integrated Nuclear Fuel Cycle Information Systems, www-nfcis.iaea.org
- [4.46] INTERNATIONAL ATOMIC ENERGY AGENCY, Recent Development in Post-Irradiation Examination Techniques for Water Reactor Fuel, IAEA-TECDOC-822, IAEA, Vienna (1995).
- [4.47] BOURGOIN, J., VAN SWAN, L.F., BOURNAY, P., "Hot cell analysis of a fuel rod fractured during handling in the pool", Contribution of Materials Investigation to the Resolution of Problems Encountered in PWR Plants (Proc. 3rd Int. Symp., Fontevraud, 1994), Vol. 1, SFEN, Paris (1994) 161.
- [4.48] BOURGOIN, J., et al., "Enseignements tirés des resultats d'expertises de crayons de combustible non étanches", Proc. 2nd Int. Symp. on Contribution of Materials Investigation to the Resolution of Problems Encountered in PWR Plants, Fontevraud, 1990, SFEN, Paris (1990) 131.
- [4.49] WILLSE, J.T., COLEMAN, T.A., "Recent results from the B&W Fuel Company's fuel performance improvement program", Proc. Int. Top. Mtg on LWR Fuel Performance, Avignon, 1991, Vol. 1, SFEN, Paris (1991) 66.
- [4.50] POTTS, G.A., PROEBSTLE, R.A., "Recent GE BWR fuel experience", Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, FL, 1994, ANS, LaGrange Park, IL (1994) 87.

5. MECHANISMS AND ROOT CAUSES OF FUEL FAILURE

5.1. INTRODUCTION

In this section, fuel failures are classified according to mechanisms commonly used in literature [5.1, 5.2]. Ten mechanisms for fuel rod failures are identified, as shown in Table 5.1. No new mechanism of fuel failure in a reactor has been observed during the reporting period included in this review. However, during ramp tests, a new failure mechanism was recently observed in which a crack initiated at a massive hydride layer at the clad outer surface propagated through the entire cladding thickness; this failure mechanism could potentially limit high burnup operation in the future.

Grid to rod fretting

Grid to rod fretting has been the major mechanism of fuel failure for PWRs in the reporting period. The main root causes of failure have been identified as insufficient fuel rod support in the assembly due to improper design and/or fabrication, fuel rod vibration from fluid elastic instability caused by cross-flow in the assembly, and flow induced assembly and rod vibration.

Debris fretting

Debris fretting continues to be a common mechanism of fuel failure in all types of power reactors. Various types of debris circulating in the coolant, mainly machining chips from the core structure, protrude through the lower tie plate and either damage the lower part of the rod or are trapped further up in the grids and cause through wall fretting of the cladding.

Corrosion

In PWRs, corrosion is uniform on the cladding surface, and excessive uniform corrosion leading to cladding failure is very rare under normal operating conditions. Such failures are either due to abnormally high heat flux exceeding heat flux/burnup corrosion limits or to water chemistry problems leading to excessive crud deposits. In BWRs, crud induced localized corrosion (CILC) was one of the major causes of fuel failure in the 1980s. This has progressively decreased with the use of heat treated cladding more resistant to nodular corrosion. More recently, crud induced corrosion involving high nodular corrosion resistant cladding was observed on several BWRs. In these cases, the corrosion mechanism seems to be connected to unusually aggressive water chemistry conditions. In BWRs and WWERs, the high content of other impurities in the coolant, such as carbon or degraded water chemistry, also resulted in fuel failure.

Pellet-clad interaction

Pellet-clad interaction (PCI) has been a frequent cause of fuel failure in BWR and CANDU reactors, but has been rare with PWRs. PCI failure occurs as a result of a rapid power increase (power ramp), typically involving neighbouring control rod movement. PCI fuel failure has been reduced remarkably, mainly as a result of operating guidelines being followed and changes in fuel design (barrier cladding, CANLUB, 9×9 or 10×10 designs). Failures at-power ramp below the assumed PCI failure limit have been reported as a result of cladding defects.

Manufacturing defects

Manufacturing defects other than those leading to primary internal hydriding (which has historically been treated separately) have been identified. The major types of defects discussed in this review are end plug defects, and several types of end plug weld deficiencies.

TABLE 5.1. OBSERVED PRIMARY FAILURES AND THEIR ROOT CAUSES

Fuel rod failures	Root causes			Related areas			Affected plants		
	Manufac.	Operation	Design	BWR	PWR	WVER	CANDU		
— Grid-rod fretting	Large variety of root causes and contributing factors								
	x		x		x				
	— Insufficient rod support (design/manufac. related)								
		x	x		x				
		x	x		x	x			
— Debris fretting		x			x				
				x	x	x			
— Corrosion									
	x	x			x	x ^b			
	x	x		x					
	x	x		x					
	x	x		x ^b		x ^b			
— Manuf. defects (other than internal hydriding)									
— PCI/SCC									
— Baffle jetting									
— Hydriding									
— PCMI/DHC									
— Dryout									
— Clad collapse									

^a Isolated event.^b Earlier occurrences, no noticeable problems in the reporting period of this review (1994–2006).^c Potential cause of failure, none observed in operation.

Cross-flow/baffle jetting

Baffle jetting resulted in fuel failure in a limited number of PWRs before the reporting period. The root cause of failure was improper joining of the baffle plates surrounding the core. The failure became rare as a result of the use of various remedies, including a change in coolant flow behind the baffle plate, which reduces differential pressure across the plates.

Primary hydriding

Fuel failure through local internal hydriding is also known as sunburst failure because of its metallographic appearance. In the 1970s, hydrogen sources were mainly residual moisture or organic contamination in fuel pellets/rods. This cause of failure has been practically eliminated through improved manufacturing. Secondary hydriding takes place after the ingress of steam into the rods, and is described in Section 7.

Delayed hydride cracking (DHC)

Although this type of failure has not yet been reported in commercial power reactors, a section was devoted to delayed hydride cracking (DHC). Results running from ramp testing to failure showed that long through-wall cracks identified as DHC failures can be initiated at the outer surface of the cladding and propagate in an axial/radial direction, so-called outside-in failures. These failures seem to be primary failures in the sense that they occur in rods that are intact prior to ramp testing. This failure mechanism may potentially limit high burnup operation, especially in BWRs, since hydriding becomes more pronounced at high burnup.

Dryout

Dryout of fuel rods has been reported only once, in a BWR. The root cause of the failure was identified as excessive channel bow which caused large local power peaking in the fuel rods. It will not be discussed further in this review. For more details see Refs [5.3–5.5].

Cladding collapse

Cladding collapse due to densification of the fuel pellets forming axial gaps in the pellet column resulted in fuel failure in the early 1970s in many PWRs. This type of failure has been completely eliminated through the use of pellets with moderate densification and pre-pressurization of rods. Therefore, it will not be discussed further in this review. For details see, for example, Ref. [5.1].

5.2. GRID TO ROD FRETTING

5.2.1. General background

Grid to rod fretting continues to be the dominant failure mechanism in PWRs during the last decade. Grid to rod fretting failures are due to rod/assembly vibrations induced by turbulence and flow heterogeneities which are always present, particularly in the inlet range up through the bottom grid. Failure occurs if either turbulence is higher than anticipated in design, or fuel rod support in the spacer grid is not sufficient. Most grid to rod fretting failures occurred during fuel assembly in the planned final cycle of operation and, in some cases, when the assembly was located on the periphery of the core, adjacent to the core baffle plates. This phenomenon has been the original cause of several incidents with multiple failures, some of which are reported below.

The Angra 1 (PWR 2 loops, Brazil) fuel failure is an example of these incidents. After premature shutdown of Angra 1 in 1993 because of high activity in the reactor coolant system during cycle 4, inspection revealed that 17 out of 121 fuel assemblies failed due to grid to rod fretting [5.6]. The fuel assemblies had been manufactured by INB (Industrias Nucleares Brasileiras) according to Siemens design and technology transfer. The damage seemed to



FIG. 5.1. Typical rod wear through fretting in EDF PWR 1300 [5.8].

be caused by premature loss of spring force. Siemens assumed the losses were present before the fuel was in the reactor, and probably took place when the fuel rods were inserted into the skeleton, possibly in combination with loads sustained during transport to the site. Alternatively, the utility believes that the root cause was inadequate spring design. Siemens had developed a new spring design, the main characteristics of which were a new shape and a higher initial force for the new reload batch. In spite of the higher grid spring force, new fretting failures appeared on the fuel assemblies two cycles later [5.7]. The new spring design, with a new shape and a higher initial force as main characteristics, did not correct the problem, because flow-induced vibration is fundamentally related to mixing vane design (shape and orientation), which had not been changed. Alternatively, the higher spring force changed vibration characteristics (amplitudes and frequencies of the different mode), leading to the strong grid to rod fretting interaction in cycle 6.

At the beginning of the 2000s, several multiple failures from fretting were observed on some four loop, EDF 1300 MW(e) reactors using fourteen foot F/A design. In particular, in Cattenom unit 3 at the end of cycle 8, 28 F/As were affected by fretting failure at the bottom grid level. The wear marks observed at other grid levels were not significant. Only very slight or non-significant marks were found on the sound rods. All the leaking F/As were localized in an intermediate ring in the core, and nearly half of presumed failed rods were peripheral.

Since it was the first time that a large fretting problem had been experienced with AREVA fuel, all potential fretting origins were analysed, and a large investigation programme was initiated which involved examining conditions of reactor operation, manufacturing conditions, and on-site checks of failed fuel assemblies, with detailed examination of lower grids and cells after some fuel assembly reversals (visual inspection of springs and dimples, cell size and drag force measurements). AREVA and EDF performed calculations of flow distribution at the bottom of the reactor vessel and assemblies (axial and cross-flow), and undertook analysis of fuel rod vibration behaviour using appropriate models. Finally, hydraulic tests in the HERMES loop test facility were conducted. These tests included a parametric test to attempt to reproduce phenomenon which had occurred in Cattenom unit 3, and a long term wear test. A combination of cross-flow and axial flow conditions were used in these tests and grid cells were pre-sized to simulate beginning of life and end of life conditions.

The investigations concluded that the grid-rod fretting phenomenon originated due to important cross-flow in the lower part of the fuel assemblies, associated with low holding force in the first grid due to spring relaxation. The low holding force in the first grid due to spring relaxation was associated with increased cycle length in these reactors (passing through a 12 month cycle to an 18 month cycle). The risk was only identified for the 1300 MW plants, which have higher cross-flow at the bottom grid level than 900 MW(e) plants.

Later, similar fretting failures were observed in Nogent unit 2 (22 failed fuel assemblies), Cattenom unit 4 (7 failed fuel assemblies) and some other 1300 MW(e) units (1 or 2 failed FAs each time).

Grid to rod fretting at mid-grid locations has been observed in several plants containing Westinghouse Vantage 5H fuel assemblies with low pressure drop Zircaloy grids (LPD). Grid to rod fretting at mid-grid locations in V5H fuel assemblies was initially observed at Salem 2 and Beaver Valley 1 and later in another plant. Numerous fuel rods were found to have developed fretting wear and perforation [5.9]. At Beaver Valley 1, 11 fuel assemblies

with 35 failed fuel rods were found, and at Salem-2, there were eight fuel assemblies with 13 fuel rods [5.10]. Other plants with the same fuel design and similar duty did not exhibit such fretting. Testing performed at the Westinghouse Columbia, South-Carolina Fabrication Plant demonstrated that fuel assemblies with LPD grids were susceptible to self-excited fuel assembly vibration at specific flow conditions. Thus the root cause of fretting was attributed to this type of fuel assembly vibration which led to selected fuel rods fretting at mid-grid. Testing determined that rotation of alternate LPD grids eliminated fuel assembly vibration and, therefore, the potential for grid to rod fretting defects from this cause. The goal of this grid rotation was to balance the forces and movements generated by coolant flow through the grids. By balancing the forces and moments imparted to mixing vane grids, assembly vibration could be reduced or eliminated. Nonetheless, Westinghouse continued to experience leaking rods in fuel assemblies with rotated middle grid design. This remedy also caused DNB problems in a few plants [5.11], highlighting the difficulty of design optimization, considering all influential aspects.

Fluid elastic instability from cross-flow was the original cause of grid to rod fretting failures at Wolf Creek during two successive refuelling shutdowns. The failures occurred in the same Westinghouse batch fuel (standard assemblies) which had completed two and three cycles of operations. Inspection of the failed fuel rods concluded the failure was due to grid to rod fretting wear, and that primary defect sites were located at the bottom grid [5.12]. Although the most severe wear appeared to occur at spring and dimple locations, some non-leaking rods also exhibited minor fretting. Profilometry of the rods, fiberscope inspections of grid cells, measurement of spring forces, and detailed visual examinations of leaking assembly rods were conducted. It was confirmed that grid retention forces were in the expected range and no damage was observed in the bottom grid. The various examinations conclusively confirmed that grid to rod fretting was the principal failure mechanism, but did not reveal a definite cause [5.13].

Westinghouse performed core flow evaluations, and computer codes were used to obtain core flow characteristics and analyse fuel rod stability. The results of core flow evaluation indicated that fuel rods become unstable through a combination of high axial flows and high cross-flow, and that those fuel rods indicated decreasing stability with decreasing spring force. These fuel failures have been attributed to high cross-flow caused, in part, by mixed fuel designs which induced fuel rod vibration with fretting wear at the lower grids. The mixture of standard and VANTAGE 5H fuel (with debris filter bottom nozzles) resulted in axial mismatches between the bottom nozzles and grid spacers of the two fuel types. The failed assemblies were all standard fuel types which had been subjected to high cross-flow [5.13].

5.2.2. Failure mechanism

Nuclear fuel assemblies are exposed to severe thermal, mechanical and radiation conditions during operation. Global core and local fuel assembly flow fields typically result in fuel rod vibration. Under certain conditions, this vibration, when coupled with other factors, can result in excessive cladding fretting wear. This phenomenon is of significant concern for nuclear fuel designers, especially in light of a growing need for higher burnup, longer cycle lengths, and operational safety margins. Understanding fretting wear margins, the probability of a fuel assembly being at risk of excessive wear, and the factors affecting fretting wear resistance are important in order to better guide design, testing, and operational flexibility. Thus, an important effort has been made in this area during the last several years and is described below.

Fretting wear is governed by a set of complex physical phenomena, which do not remain constant during nuclear fuel operation in reactor cores. Based on various field data for grid to rod fretting wear, we believe that the amount of grid to rod fretting wear depends on:

- The extent of flow induced vibration caused by fuel design and/or plant specific operating conditions;
- Grid to rod support conditions;
- The initial grid to rod contact area;
- Grid materials.

A larger flow induced vibration, larger grid to rod gap, smaller initial grid to rod contact area and softer cladding tube contribute to a higher probability of grid to rod fretting wear.

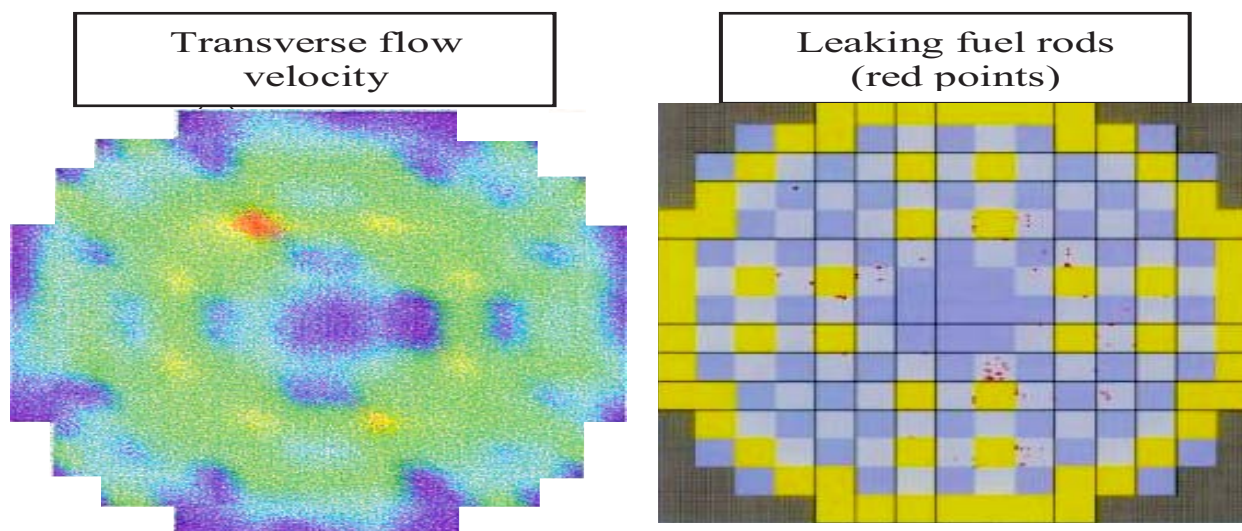


FIG. 5.2. Comparison between cross-flow computation (a) and leaking rod locations due to grid to rod fretting (b) on EDF 1300 MWe core [5.14].

Unexpected excessive flow induced vibration may occur at certain locations with severe flow conditions in the reactor core. Calculations of flow distribution at the bottom of the reactor vessel and within assemblies (axial and cross-flow), performed in the case of Cattenom 3, showed that higher cross-flows (red and yellow areas in Fig. 5.2) were distributed in an intermediate ring around the core. This correlated strongly with failed fuel assembly locations [5.14].

Grid design may sometimes lead to self-excited fuel assembly vibration and/or self-excited spacer grid strap vibration. A good example is the VANTAGE 5H fuel assembly with low pressure drop zircaloy grids, which were susceptible to self-excited fuel vibration at specific flow conditions.

Also, some events demonstrate that failure risk from fluid–elastic instability can increase in mixed cores when flow mismatches in neighbouring assemblies lead to added cross-flow and turbulence. Many existing spacer designs seem to be insufficiently robust to resist turbulence caused by added cross-flow. During cycle 9 of D.C. Cook unit 2, four Siemens fuel assemblies failed in the burnup range 45–50 GWd/t. Poolside examinations revealed that the cause of failure was grid to rod fretting [5.15]. The four failed Siemens assemblies were all surrounded by one cycle Westinghouse Vantage 5 assemblies and the core baffle. The fretting failures were located at spacers close to the intermediate flow mixers (IFMs) in the Vantage 5 assemblies.

Finally, in some reactors, more severe flow conditions can exist, mainly at fuel assemblies located on the core periphery.

The second, more important parameter is grid to rod gap support condition, which is determined by initial elastic spring deflection, cladding creep-down, irradiation induced spring force relaxation and fuel assembly location in the reactor core. Knowledge of fuel rod support in the grid cell is an important parameter for comprehension of global behaviour in relation to local behaviour of a rod (fretting wear). Generally, rods are supported by a spring dimples system. Under irradiation, grid spring force is decreasing and a rod to grid gap opening may appear, impacting the vibration of rods. An example of grid to rod support condition evolution calculated with the model presented in the reference [5.16] is illustrated in Fig. 5.3. This model takes into account the evolution of cladding diameter, grid growth, and spring and dimple creep under irradiation. As a result, the model provides the evolution of fuel rod support force (or corresponding cell size) versus burnup. The evolution calculated is in good agreement with operating feedback results.

It is obvious that larger initial elastic spring deflection, less cladding creep-down and lower spring force relaxation will produce a smaller grid to rod gap.

Also, assuming the same previous operating history, fuel assemblies located at the core periphery may generate a larger grid to rod gap than those inside the core because the former show relatively lower fuel rod temperatures and thus smaller fuel rod diameter than the latter.

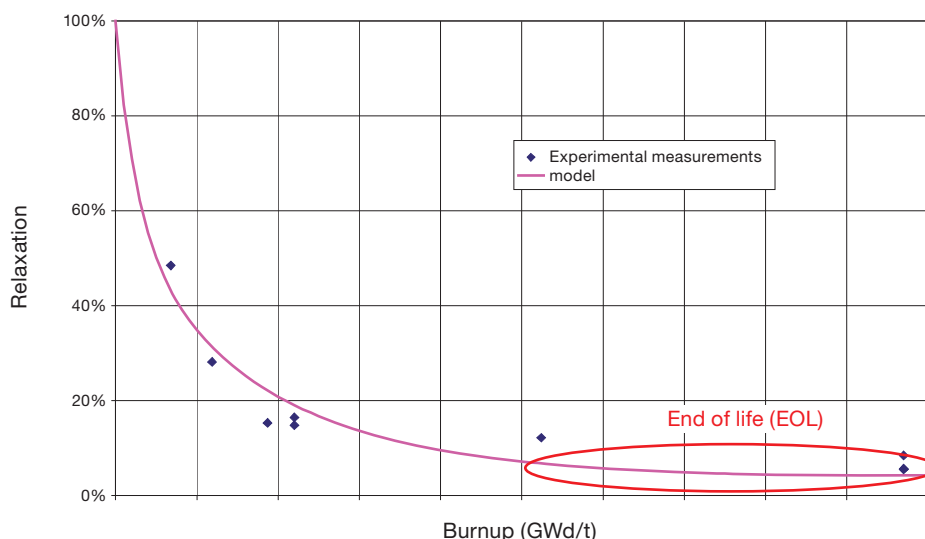


FIG. 5.3. Evolution of fuel rod support condition versus burnup [5.16].

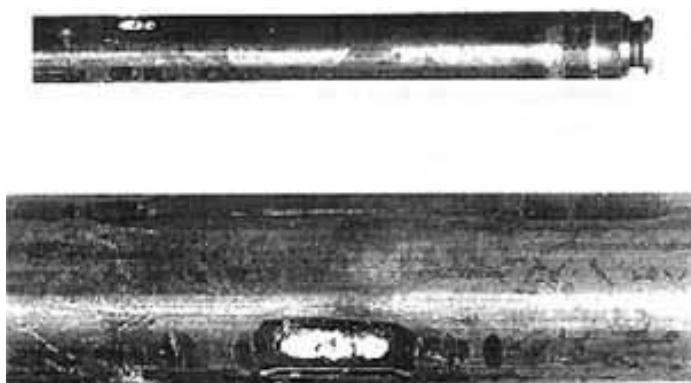


FIG. 5.4. Fretting defect induced by grid to cladding interaction [5.18].

A larger grid to rod contact area generates less grid to rod contact stress and thus reduces the fretting wear rate when the same vibration induced force is present for different failed fuel rods. Therefore, it is recommended that spacer grid spring and dimple design with a larger grid to rod contact area be developed to eliminate fretting wear.

Some other specific causes of fretting failure are worth mentioning. An unexpected root cause of grid to rod fretting due to spacer breakage in the bottom grid occurred in 1994–1995 in two German plants [5.17]. The cause of failure was traced back to a combination of high stress and high stress corrosion cracking susceptibility in spacer springs, caused by improperly heated Inconel springs and the use of these spacers at the lowermost spacer position, which is below the active length in the affected plant. This problem was solved by using proven Inconel spacers outside the active region.

Other cases of fretting failure due to spacer fabrication were also related. Fretting defects were observed in two Framema fuel assemblies [5.18]. A defect is shown in Fig. 5.4 [5.18]. A very extensive programme of investigation was performed on the fuel assemblies containing the two affected rods, yielding detailed grid cell dimension information. The results of dimension measurements on the bottom grid cells show very little difference between cells in which fuel rods were securely held and the two cells where fretting corrosion took place. In both fuel assemblies, the defect occurrence is estimated to take place either at the end of the second irradiation cycle or during the third cycle. The conditions for such a defect occurrence might plausibly be a combination of: (a) specific transverse flow at the bottom end grid; and (b) initial grid cell characteristics in excess of specification values, leading to excessive relaxation of the restraining force.

Grid to rod fretting failures also occur occasionally as a result of a damaged spacer, which can stem from interference between adjacent PWR fuel assemblies during fuel shuffling.

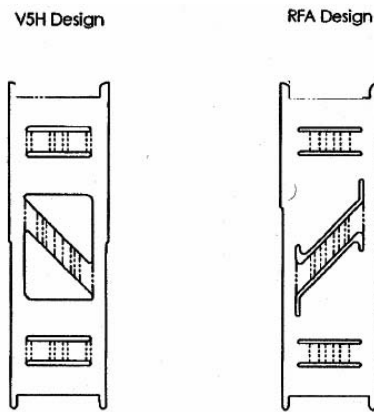


FIG. 5.5. Comparison of V5H and RFA design [5.20].



FIG. 5.6. Protective grid [5.8].

5.2.3. Corrective actions

Modifications to grid structure and spring designs were implemented to increase Westinghouse fuel resistance to grid to rod fretting [5.19]. The 17×17 RFA grid design provides both a grid to rod fretting margin and a DNB margin. The features alleviating grid to rod fretting include an increase in contact surface area between fuel rod and grid supports relative to the prior design, and a modification in the porosity (or opening) of the grid strap itself, achieved without impacting pressure drop. Changes in the mixing vane pattern have eliminated unbalanced lateral forces and have reduced susceptibility to fuel assembly vibration. In addition to the above changes, the mixing vane design itself was modified. These grid changes also provided a significant critical heat flux margin.

To provide more protection against elastic excitation of fuel rods caused by cross-flows present at the inlet of fuel assemblies, Westinghouse introduced a protective grid, which supports fuel rods at the bottom [5.12]. The protective grip holds the bottom end of a fuel rod and allows the bottom structural grid to be moved up, thus reducing the span between the two structural grids at the bottom of the fuel assembly. This changes the vibration characteristics of the fuel rod sufficiently to eliminate the flow induced mechanism.

As a remedy for grid to rod fretting in 1300 MWe EDF nuclear power plants, AREVA proposed a modification to the 14 ft fuel design and added a reinforcement grid to the bottom of fuel assemblies. This modification induces better embedding of the rod. Significant reduction of wear risk was confirmed by an in loop test (HERMES) at CEA [5.21]. The new fuel design is shown in Fig. 5.7.

To reduce grid to rod fretting, AREVA also proposed use of HTP design. The flow channels in HTP spacers also serve as springs and constrain lateral motion of the fuel rod throughout the operation of the fuel assembly. The flow channels maintain a long contact length of eight positions between each spacer cell and the fuel rod. This prevents fretting wear even after full irradiation induced stress relaxation of the spacer has taken place. This capability has been demonstrated through different tests and in-reactor experience feedback.



FIG. 5.7. AREVA Bi-Grid on AFA-3GLr design [5.8].

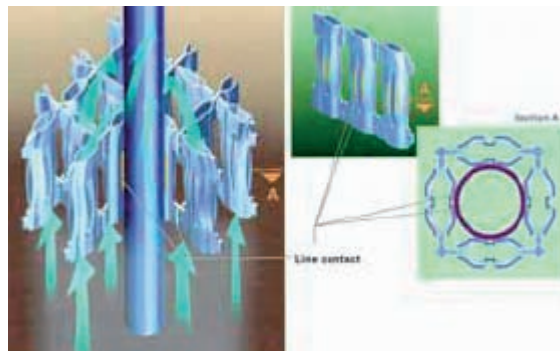


FIG. 5.8. HTP grid [5.22].



FIG. 5.9. Robust Fuelguard bottom nozzle [5.22].

The blade array geometry of the robust Fuelguard bottom nozzle also allows for excellent homogenization of flow distribution at the bottom nozzle outlet and reduces flow vibratory excitation on the bottom end of the fuel rods (see Figs 5.9 and 5.10).

Mitsubishi has developed an improved grid called the 'I type grid', which improves the performance of fuel rod support [5.23]. Its springs characteristics feature good stability of spring force even after an excessive displacement due to rod vibration or decreased deflection due to creep-down of the outer diameter.

Some plants with fuel rod wear in specific baffle locations have implemented mitigating actions, such as the use of stainless steel rods in fuel assemblies known to be in susceptible core locations, or fuel spring clips on peripheral fuel assemblies as an added barrier to baffle jetting. Initial feedback indicates that operating experience with these new grid designs is promising.

Fuel assembly design optimization guidelines against grid to rod fretting are presented in Ref [5.135].

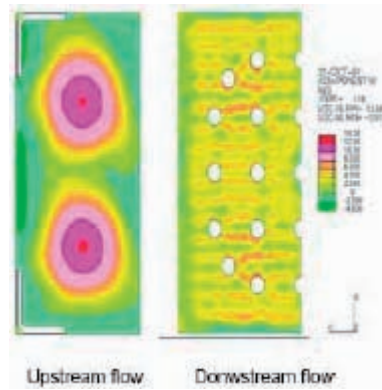


FIG. 5.10. Downstream flow homogenization [5.22].

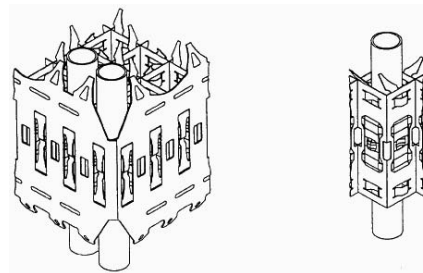


FIG. 5.11. I type grid [5.23].

5.3. FRETTING DUE TO DEBRIS

5.3.1. General background

Fuel failure due to fretting by debris occurred in all types of reactors. It is the second cause of failure over the past decade in PWRs, and the dominant failure mechanism in BWRs. Latest statistics show that the number of confirmed debris fretting failures has increased by a factor of two to four for European and United States of America fuel suppliers in spite of the implementation of debris filtering tie plate features [5.24].

A review on fretting failure due to debris and preventive measures has been published by A. Strasser [5.25]. The cause of failure is entrapment of metallic debris in fuel assemblies, which can lead to rapid fretting and penetration of the cladding wall. The debris are introduced into the reactor coolant system mainly during repair or maintenance operations, such as replacement of steam generators, removal of thermal shields in PWRs, repair and replacement of spray headers in BWRs, and repair and replacement of reactor coolant pumps in both PWRs and BWRs. In CANDUs, debris found within primary circuits came mainly from construction during the building of new reactors. Other sources of debris can appear after startup when reactors are shutdown for routine maintenance and inspection of primary circuit components. In one case, a bunch of turnings was caught at the end plate of a fuel bundle [5.26].

Fuel failures due to debris fretting have been also observed in WWER reactors. One case of fuel failure caused by debris fretting was reported at Zaporozhe 1 (1991) [5.27]. The cause of debris generation was thermal barrier damage.

Loose metallic debris can circulate in the reactor system and become trapped in the fuel assembly. Typically, debris is trapped between fuel rods and a spacer, usually the bottom one, and less frequently in spacers higher up in the assembly. The debris particle vibrates under influence of the coolant flow and causes fretting erosion of the cladding [5.28, 5.29]. The typical appearance of debris defects is shown in Fig. 5.12.

Unlike other defect types, fretting defects are not power related. Defects can appear anywhere in the assemblies or bundles, depending on the size of the debris.

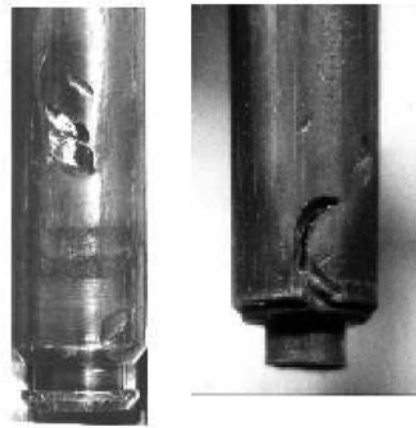


FIG. 5.12. Example of debris fretting failures at bottom fuel rod.

5.3.2. Debris characteristics

According to Strasser [5.25], the observed debris have typically included turnings and shavings such as those resulting from metal working operations performed during primary system maintenance. In a few cases, large objects such as tools, screws, bolts, nuts, metal clips, electrical connectors, pieces of lock wire and other wires, shavings from defective pumps or valves, parts of gaskets and saw blades have been found in damaged fuel assemblies.

Westinghouse evaluated a large number of TV tapes showing debris induced damage from the early to the late 1980s [5.29]. The wear scars were characterized on the basis of size (small, medium, large), shape (small hole, gouge, gross, crescent), and whether there were multiple or single scars. Three trends were seen in this study:

- (a) 'Small hole' wear scars were not seen in the early 1980s, but became prevalent in the late 1980s;
- (b) Single wear scars became dominant in the late 1980s. In the early 1980s, multiple scars prevailed;
- (c) More than 95% of debris scars occurred below the bottom grid.

These observations imply that the debris causing scars in the late 1980s, after efforts to minimize debris, were smaller in size than debris in the early 1980s. As debris became smaller, another trend was observed. The majority of debris induced fretting defects were found to occur early in the operating life of a fuel. This suggested that fuel rods have properties which make them more resistant to debris damage as burnup increases. One possible reason for this is an oxide film which forms on the cladding surface as a result of corrosion during irradiation and which is much harder than the base alloy.

Since the introduction of anti-debris filters, only long and thin or small debris are able to infiltrate; these now constitute the main risk of fretting caused by debris. Another consequence is the tendency to more frequently find defects in spacers higher up in the assembly as shown in Fig. 5.13.

A statistical analysis of debris fretting failures in Westinghouse BWR 10 × 10 [5.30] fuel also shows that debris fretting failures occur preferably:

- In the high spacer position where flow velocity is higher;
- In younger high power fuels with higher flow velocity, and less wear resistant oxide coating than older fuels.

5.3.3. Corrective actions

Debris filtering nozzles or tie plates

In the past, fuel vendors have developed fuel designs with debris resistant features. However, the effectiveness of these devices is not complete and advanced designs are under development.



FIG. 5.13. Example of debris induced fretting failures at spacers higher up in the assembly.

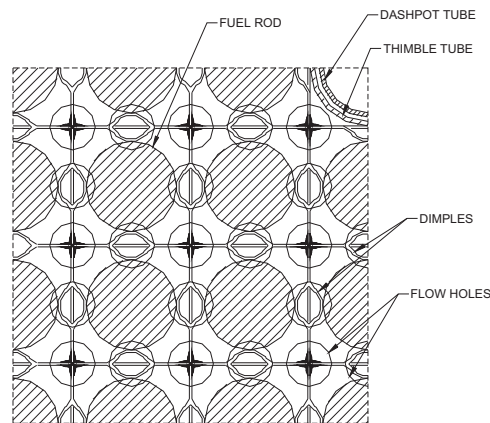


FIG. 5.14. Protective bottom grid and bottom nozzle interface [5.12].

For PWR fuel assemblies, Westinghouse introduced the debris filter bottom nozzle (DFBN) [5.29] in 1988. Compared to the previous bottom nozzle, the DFBN has reduced diameter flow holes through which coolant enters the fuel assembly. The DFBN, although very effective in reducing the occurrence of debris induced wear, did not totally eliminate it. To increase efficiency of the filtering device, an additional mitigation feature was connected to the bottom protective grid [5.12]. Located on top of the fuel assembly bottom nozzle, the grid straps of the protective grid are positioned so that they intersect the flow holes in the DFBN. The grid straps reduce the flow area by a factor of two or four, following the position of the flow holes. To take into account experience, which has shown that the majority of debris entering fuel is trapped in or below the bottom grid, fuel incorporating the protective grid additionally incorporates a longer fuel bottom end plug. If debris is caught under or in the protective grid, which is what usually occurs, any resulting wear will be located on the solid plug and will not breach the cladding.

For PWR fuel assemblies, AREVA has developed the TRAPPERTM bottom nozzle, equipped with an anti-debris filter (see Fig. 5.15) of AFA-3G fuel design and also proposed the debris resistant Robust Fuelguard (see Fig. 5.16) which provides the greatest protection to the fuel assembly against debris. It employs parallel rows of curved blades arranged in such a way that there is no line of sight path through the grid. This design provides a highly effective debris filter while retaining minimal flow resistance. Larger particles are stopped due to the close spacing of the blades, and long linear particles are trapped because of the curved path through the blades. Thus, the passage of debris into the fuel rod region of the fuel assembly, which might cause fretting failures, is reliably prevented. A similar design has also been developed for BWRs.

Debris is the leading cause of fuel failure in modern BWR fuel. Industry has observed, and testing seems to support, that 10×10 designs are more susceptible to debris failures than the 8×8 and 9×9 designs previously used, in spite of the implementation of protective debris filters. The 10×10 design has tighter flow gaps at the

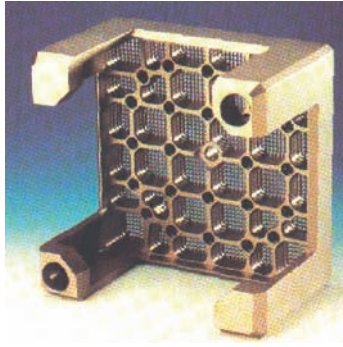


FIG. 5.15. Trapper bottom nozzle [5.22].



FIG. 5.16. Robust Fuelguard bottom nozzle [5.22].

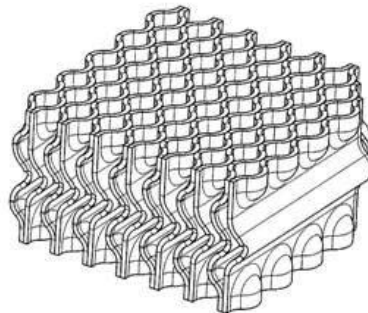


FIG. 5.17. TripleWave debris filter unit [5.32].

spacer interfaces, making debris capture more likely. Additionally, it has more spacers concentrated in the two phase region of the bundle where flow velocities are sufficient to cause fretting failure. GNF has approached the elimination of debris failures on several fronts [5.31]. It has developed test facilities in Japan and Canada to better understand and specifically design for debris protection. This has led to introduction of the DefenderTM debris filter. Testing revealed that any debris that can fit through a lower tie plate filter would eventually find its way into the bundle. So the DefenderTM was designed to filter out any size of debris identified in the past as having caused fretting failures. Testing also revealed that spacers could be designed to reduce the probability of debris capture. The GNF2 Advantage spacer is designed in such a way that any debris small enough to bypass the DefenderTM can be expected to pass through the spacers.

To mitigate debris induced fuel failure for 10×10 BWRs, Westinghouse has developed an advanced debris filter, the TripleWave debris filter, inserted in Lead Fuel Assemblies (LFAs) in 2001. The new design is aimed at catching long and thin debris, since studies have shown that this type of debris constitutes the largest fretting risk, while small objects like blasting grit will pass through. Trapping efficiency tests demonstrate that TripleWave reduces the risk of harmful debris entering the fuel assembly by one to two orders of magnitude, while assemblies equipped with TripleWave are compatible with total core pressure drop, and even with large amounts of debris trapped in the filters.

Other protective design features

Another feature incorporated in new Westinghouse designs is a ZrO_2 coating on the bottom portion of the fuel rod cladding to provide a harder surface, thus adding wear resistance during the first cycle of operation [5.12].

A further example of combined remedies has been applied to the Mitsubishi 17×17 fuel (PWR) [5.23]. Besides an improved bottom nozzle equipped with a debris filter, a particular feature of this design is a shift of the weld position inside the bottom grid through a combination of lengthening the bottom end plug and lowering the bottom grid. The dimple of the debris filter is located at the bottom end plug, so that debris will be trapped by the debris filter, and fretting will occur at the bottom end plug away from active fuel, preventing fuel leakage. Moreover, a skirt is installed in the bottom nozzle. The skirt is effective in preventing debris from passing through from the bottom nozzle to the gap between fuel assemblies.

Some Swedish BWRs have also installed centrifugal filter devices on feedwater lines to collect and trap debris.

5.4. CORROSION

5.4.1. General background

Uniform corrosion of fuel cladding is one of the main limitations in the use of Zircaloy 4 in PWRs. With increasing burnup and 'fuel duty', significant reductions in margin to cladding corrosion limits for Zircaloy 4 have been observed. The development of advanced cladding, such as M5 for AREVA or ZIRLO for Westinghouse, was necessary to achieve the burnup objectives of many plants operating under more demanding conditions and new 'fuel duty'.

The current generation of Zircaloy 2 fuel cladding possesses good resistance to nodular corrosion, which was the major corrosion mechanism in BWRs in the past (CICL).

Fuel failure through corrosion has been rare except for that caused by unfavourable water chemistry or excessive corrosion due to localized crud deposition. There have been no corrosion related failures in CANDUs or in WWERs, only infrequent cases of abnormal operation.

The majority of failures have occurred in BWRs as a result of unfavourable water chemistry conditions, through crud induced corrosion or a combination of unfavourable water chemistry and corrosion susceptibility of the cladding.

5.4.2. Excessive corrosion

It has been generally observed that the corrosion of standard Zircaloy-4 tends to accelerate approaching high burnup. Several mechanisms have been proposed to explain the accelerated corrosion. One is that the thermal conductivity of zirconium oxide is low and the accumulation of oxide has a feedback effect on oxidation of zircaloy via enhanced temperatures at the metal surface. Another is that the zirconium hydrides tend to precipitate at the outer surface of the cladding, and corrosion of hydrides is known to take place much faster than that of cladding material.

The technological performance limitation through waterside corrosion of PWR fuel has been demonstrated during irradiation of high power test assemblies in a Siemens 'hot plant' [5.33]. Several fuel rods had failed towards the end of their third irradiation cycle. Corrosion failures associated with local hydride concentrations were observed during poolside inspection of failed rods, as shown in Fig. 5.18. Under these conditions, the oxide layers away from the local defect were always on average larger than $140 \mu\text{m}$ in circumference. In an earlier high power experiment with substantially higher heat flux and smaller oxide thickness, local disturbances in uniform corrosion behaviour occurred; these were associated with loss of integrity of the oxide and degradation of its thermal conductivity, ultimately resulting in local perforations [5.34].

Siemens/KWU experience with thick oxide layers, including the above experiments, is shown in Fig. 5.19 on a map of oxide thickness and heat flow. The figure shows a tentative locus of constant corrosion rate (i.e. constant temperature at the oxide/metal interface) assuming a thermal conductivity of 1.5 W/mK for the oxide.

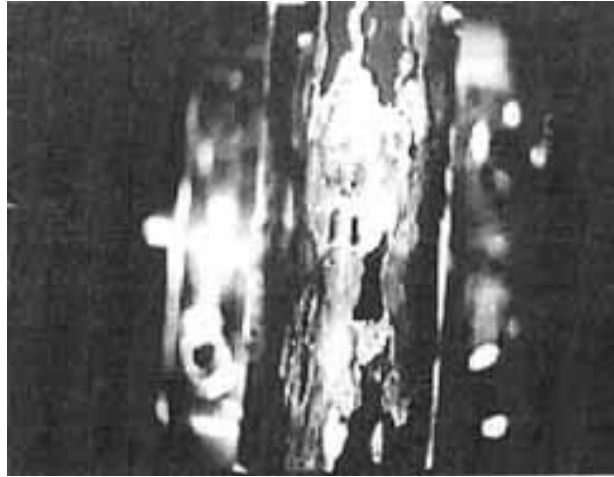


FIG. 5.18. Typical appearance of a fuel rod that has failed as a result of corrosion [5.33].

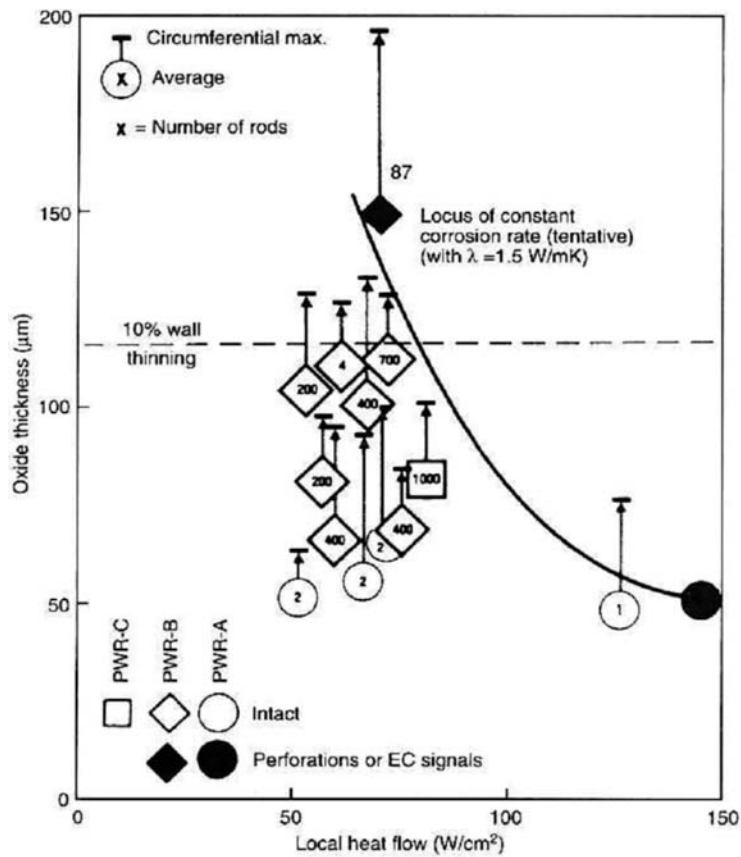


FIG. 5.19. Relation between corrosion defect, oxide thickness and local heat flow [5.34].

EdF [5.35] experienced one fuel failure due to excessive cladding corrosion. During shuffling of fuel assemblies at the end of cycle 9 at Blayais 1 in 1991, five failed assemblies were identified. All five assemblies were ultrasonically tested to determine which rods had failed and were visually inspected on four sides. Pellet-cladding contact reduced the possibility of water reaching the lower end of the rods and made interpretation of ultrasonic signals difficult, but there was one failed rod per each assembly. One assembly was chosen for oxide thickness measurements on peripheral rods. The average peak oxide thickness was 140 mm in the upper spans with a maximum peak oxide thickness of 197 mm. Oxide was observed to be spalling, and the values given reflect maximum remaining oxide thickness. These fuel assemblies were delivered by a foreign vendor, which used

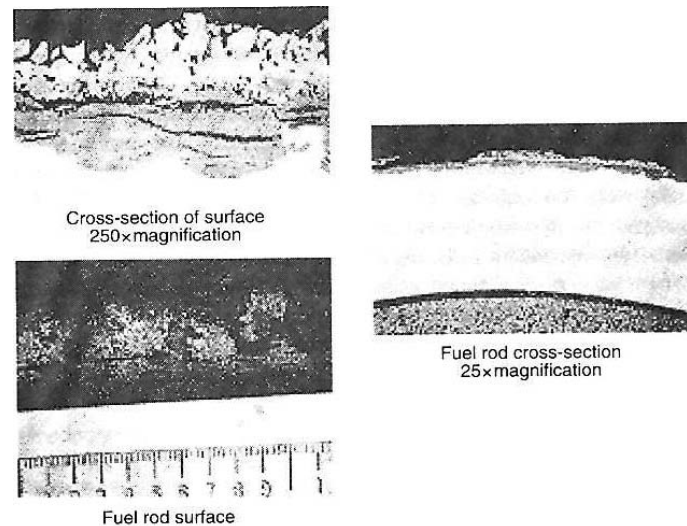


FIG. 5.20. Crud and oxide (formally nodular) on CILC damaged fuel rod [5.39].

cladding made under an old manufacturing process and which were irradiated for four cycles, i.e. one cycle beyond their design burnup. The cause of these defects was attributed to excessive cladding corrosion. No additional investigation was performed.

Fuel failure through corrosion was experienced in 1980 on six MOX rods in the BR3 as a result of excessive crud deposition [5.36]. This failure was due to primary water chemistry and a high surface heat flux level. It had no relation to the use of MOX fuel, and disappeared when a more restrictive permissible heat flux level as a core design criterion was adopted.

Fuel failure occurred in the WVER-440 at Novovoronezh 4 in 1986 as a result of a neutron field disturbance leading to increased power and probably partial coolant boiling [5.37]. The cladding temperature rose to 490°C, and a coolant activity increase was registered. PIE of failed fuel rods revealed intensive corrosion damage in the form of thick spalling oxide, some of which took the form of through-wall defects.

More recently, between 1997 and 2005, about 40 Zircaloy 4 corrosion induced failures occurred in Republic of Korea PWR fuel at high burnup [5.38].

5.4.3. Crud induced localized corrosion (CILC)

Failures by crud induced localized corrosion (CILC) were initially found in 1979 in some GE BWRs. The fuel failure was associated with localized fuel cladding corrosion, as shown in Fig. 5.20, where ‘scale type crud’ was formed [5.39], and is often accompanied by secondary hydride bulges and longitudinal splits [5.28]. In BWRs, two types of crud deposits (plant corrosion products) can be found. The most frequently observed crud is low density, loosely adherent crud with good heat transfer capability. The other is a rarely observed tightly adherent crud of high density, through which lamination can lead to low heat transfer capability. Scale type crud contains high concentrations of copper, up to and in excess of 50%, and heavy copper bearing crud was observed to be sandwiched between a layer of zirconium oxide at the cladding surface. The failures occurred at mid-life fuel exposure levels and were highly concentrated in burnable fuel rods containing urania–gadolinia pellets.

Detailed investigations, including fuel examinations, surveillance and extensive research, have led to a practical understanding of this failure mechanism. It was found that three special conditions of environment, duty and material susceptibility must occur simultaneously for CILC failure to occur [5.39].

With regard to the environment, CILC failure has been exclusively found in BWR plants with copper containing condenser tubes and a filter–demineralizer condensate cleanup system. All BWR plants without a significant copper source in the primary circuits exhibited immunity to CILC failure. The existence of significant amounts of copper in the coolant has been found to be detrimental in three ways. First, it promotes the formation of dense, adherent and insulating scale type crud. Second, copper bearing crud preferentially deposits between normal nodular oxides. Third, dense, copper bearing crud is deposited within normal laminations of the oxide formed on the zircaloy. In addition to the low thermal conductivity of copper bearing crud in laminar layers, its effect of

impeding the ingress of coolant leads to formation of a stagnant, highly insulating steam layer in laminations of the oxide.

With regard to duty, CILC failures have occurred almost, but not quite, exclusively in gadolinia containing fuel rods, which always operate at lower heat flux. Zircaloy has been found to be more susceptible to nodule nucleation and growth in gadolinia containing rods than in uranium rods with higher heat flux conditions. The underlying mechanism for this phenomenon is considered to be related to highly oxidizing, radiolytically produced species at the fuel rod surface. At low heat flux, the residence time for these species (e.g. H_2O_2 , HO , HO_2 , etc.) is sufficiently long in comparison to that of high heat flux uranium rods to promote the formation of nodular corrosion. However, it is reported [5.39] that no clear correlations have been observed between local linear heat generation rates (W/cm) and either CILC susceptibility or extent of CILC damage suffered; the best correlating parameter appears to be bundle power density.

With regard to materials, GE has developed out of pile methods to produce nodular corrosion, which is similar to that observed in BWRs with a large number of zircaloy materials samples from the same sources and processes which produced CILC failure. Tests confirmed variability in nodular corrosion resistance, and this was found to be inherent in tube shells having no effect from subsequent reduction and annealing processes. A small percentage of materials are susceptible to failure on gadolinia containing rods in environments which produce CILC failure. This percentage is in agreement with the observed failure frequency, thereby confirming the importance of material susceptibility in the CILC mechanism.

In Germany (Siemens/KWU) [5.40] and in Japan [5.41], no CILC failure has been reported.

Influence of transient water chemistry and cladding material on CILC failure

It has been noted that water chemistry in early fuel life may influence the corrosion of CILC type failure later in fuel life. EPRI [5.42] analysed the correlation between CILC fuel failure and specific off normal chemical in-leakage parameters over a ten year period at five EPRI member utilities that operate the bulk of US BWRs susceptible to CILC failure. Only certain parameters of early life chemical history (less than $3.2 \text{ MW}\cdot\text{d/kg U}$) of a given batch of fuel correlate predictably with peak oxidation rates and the incidence of subsequent localized corrosion failure. Early life conductivity transient severity and off-normal feedwater copper (above 0.4 ppb) in-leakage were found to be strong predictors of long term cladding oxidation rates and later life failure. It was noted that the 20 fuel batches consistently studied fell into two groups, one group more sensitive to the effects of chemical in-leakage than the other. Differences in cladding material condition could well account for this fact. From a mechanistic point of view, it is interesting to note that there was no correlation between late in life copper in the coolant and batch failure fractions. It may be that copper deposits observed in thick, cracked oxide are only concomitant with steam blanketing in the thick oxide; it may be the steam, not the copper, which raises the oxide-metal interface temperature and leads to through-wall failure. Early in life, when the zirconium oxide film is very thin, high copper in the feedwater, especially in combination with chemical transients which raise coolant conductivity, may cause the acceleration of oxidation necessary to form the thick, cracked oxides capable of failure.

To eliminate CILC failure, several actions have been taken and, since then, no CILC failures have been reported. Specific recommendations included: improved water chemistry monitoring, copper-concentration limited startup procedures, and more stringent target values for maximum allowable copper concentrations. A heat treatment process for cladding material has been developed that improves nodular corrosion resistance of Zircaloy 2 cladding. The standard Zircaloy-2 with variable corrosion resistance has been replaced by cladding material with special heat treated cladding [5.42]. Also, the number of CILC susceptible plants has decreased through hardware improvements, including copper condenser replacement and the installation of deep bed condensate treatment systems [5.42].

5.4.4. Crud induced corrosion

With regard to CILC, crud induced corrosion involves failures of heat treated cladding not susceptible to nodular corrosion. The corrosion characteristics are different and do not involve nodular corrosion, even if the root cause may be similar.

Recently, four US BWRs experienced fuel failure due to accelerated cladding corrosion; these involved different fuel suppliers. About 14 months after the startup of cycle 11 at River Bend, an off-gas activity increase was noted. During core off-load at EOC-11, six failed one cycle ATRIUM™10 fuel assemblies were identified, with burnups in the range of 14.6 to 19.0 GWd/MTU [5.43]. Examinations performed during the refuelling outage and after startup of cycle 12 indicated that Span 2 (the axial location between the second and third spacers from assembly bottom) failed in both, and one cycle assemblies which did not fail had an unusually thick tenacious crud accumulation on the peripheral rods. An example of heavy crud in lower Span 2 before and after brushing is given in Fig. 5.21. It was observed that the thickest tenacious crud was very specific to a particular region of the River Bend core and axial location on new fuel rods. The failed assemblies were located in core locations which formed a ring, positioned seven assemblies from the core centre. Visual examinations after rod brushing/cleaning showed that the peripheral rods had accumulated thick tenacious crud mainly on the side of the rod facing outward toward the fuel channel, while interior rods were generally unaffected by the thick crud. With this asymmetric crud deposition and oxide formation, excessive rod bow toward the fuel channel was experienced by higher power rods (failed and unfailed), and some of these rods may have bowed enough to touch the inside surface of the fuel channel at or near the location of failure. Several crud flakes were successfully acquired and evaluated and these indicated high levels of copper and zinc. The cause of failure in River Bend rods during cycle 11 was determined to be accelerated oxidation of the cladding in Span 2, resulting from unusually heavy deposits of insulating tenacious crud. Detailed reviews pointed out the River Bend water chemistry was within EPRI guidelines and no specific events of concern were noted. However, it was noted that River Bend was unique relative to other units due to its combination of high copper, zinc, and iron. The most probable cause of insulating tenacious crud was that copper and zinc were available in sufficient quantity to plug either normal wick boiling paths within the crud or any delamination within the crud or clad oxide, resulting in diminished heat transfer in local areas of the cladding surface.

The majority of other failures were localized in Browns Ferry Unit 2 where, during cycle 12, which operated from April 2001 to March 2003, 63 GE13 fuel assemblies in their second cycles of operation had fuel rods that failed from accelerated corrosion [5.44]. The entire second cycle reload, first loaded in May 1999, exhibited accelerated corrosion peaking towards the top of the bundle. The first and third cycle bundles operating in the same cycle, as well as previously discharged fuel in the fuel pool, were normal, well inside GNF's experience base. Despite a significant effort, including detailed reviews of reactor water chemistry, fuel manufacture and fuel operation, no root cause of the fuel corrosion failure at Browns Ferry has been determined. It is likely that a combination of conditions existed which resulted in the fuel failure. The combined failures at Units 2 and 3, and their timing, seem to indicate an unusually aggressive water chemistry condition, perhaps related, during or immediately after one or both of the affected reloads' first cycle of operation. Cladding material which was lower in alloy content and higher in duty was most affected by this environment. A hot cell exam to provide further insight is in progress.

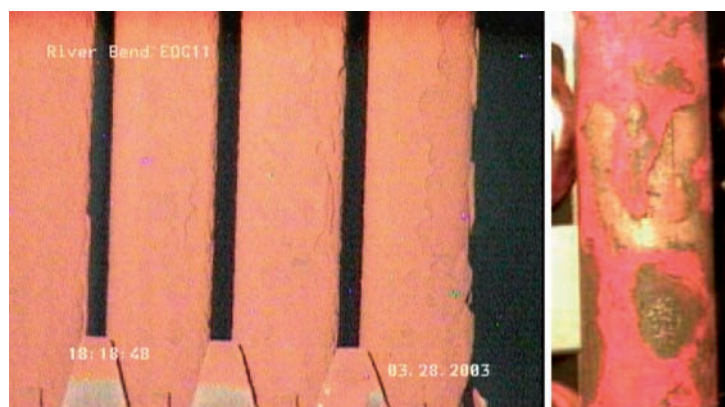


FIG. 5.21. Example of heavy crud in lower Span 2 before and after brushing [5.43].



FIG. 5.22. Failure location noted from bubbles escaping at a defect in a peripheral rod [5.43].

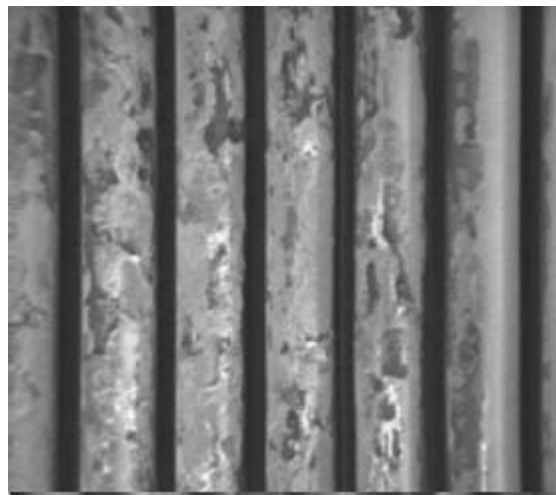


FIG. 5.23. Distinctive crud pattern (DCP) in TMI fuel assemblies [5.46].

Enhanced clad corrosion due to crud deposit was also reported in a few US PWR plants between 1995 and 2000. In particular, during cycle 10 in 1995, TMI-1 experienced nine failed rods owing to unusual buildup of corrosion products in the upper spans on the outer surface of peripheral fuel rods [5.45]. This crud buildup and the following fuel failures occurred in the first cycle with highly enriched fuel of 4.75% (in a 24 month cycle). The distinctive crud pattern (DCP) of the failed rod was characterized by a mottled appearance, specifically a dark, nearly black surface with jagged patches of white showing through and occurring around 2.8 metres from the bottom of the fuel rod as shown in Fig. 5.23.

Poolside and hot cell investigations to delineate the cause of fuel failures were performed [5.46]. Results indicated greater corrosion and hydrogen pickup than predicted in the upper spans, regions of cladding recrystallization in areas of unusual crud pattern as shown in Fig. 5.24, and hydrogen migration away from those areas. These observations indicated that higher than expected temperatures occurred in DCP areas. Cladding oxidation and hydrogen pickup were also higher than predicted at the elevation of DCP regions on rods, consistent with high cladding temperatures.

Localized cladding penetration due to crud induced localized corrosion was identified as the failure mechanism, see Figs 5.25 and 5.26. Evaluations attributed the problem to a combination of high temperature due to higher power and some flux tilt with fresh assemblies, and a low pH value because of the applied high level of boron poison used to transition from 18 month to 24 month cycles [5.46, 5.47].

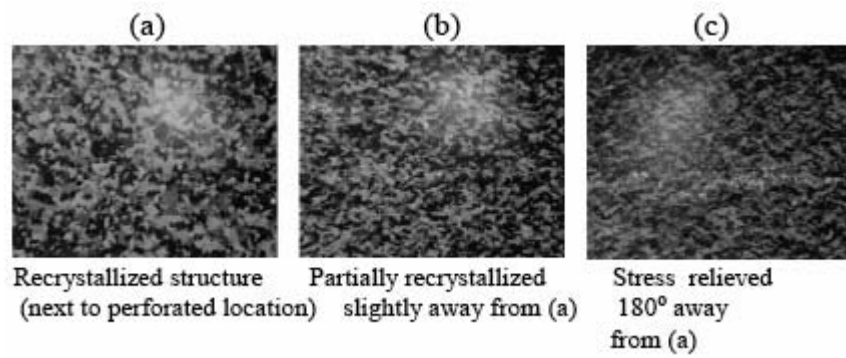


FIG. 5.24. Grain structure of the zircaloy cladding [5.46].

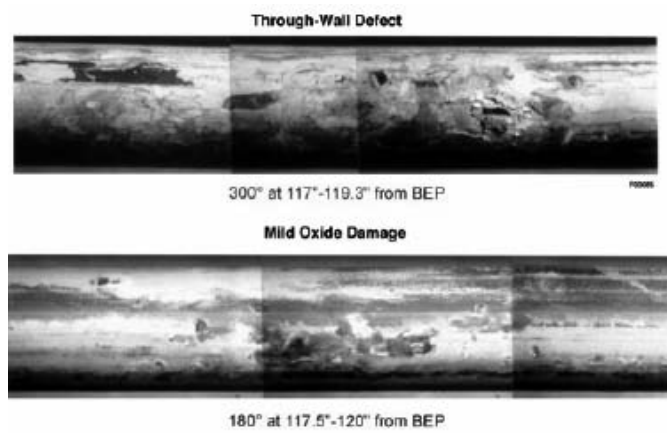


FIG. 5.25. TMI 1, Through-wall defect on a rod O11 [5.46].

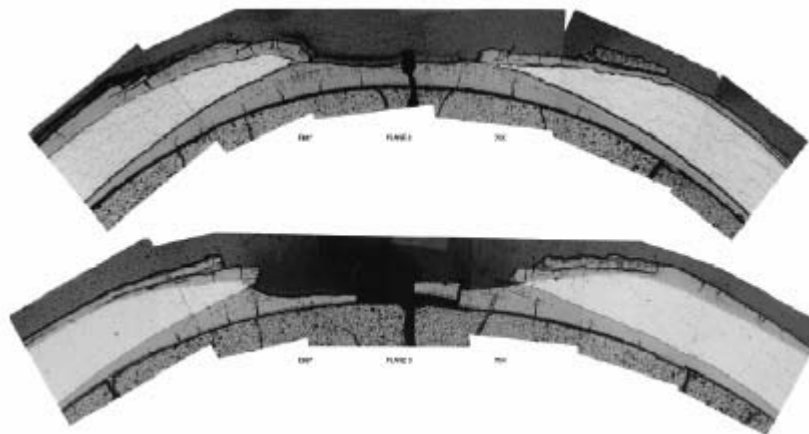


FIG. 5.26. TMI 1, Cladding corrosion on a rod O11 at an elevation of 118.5 inches [5.46].

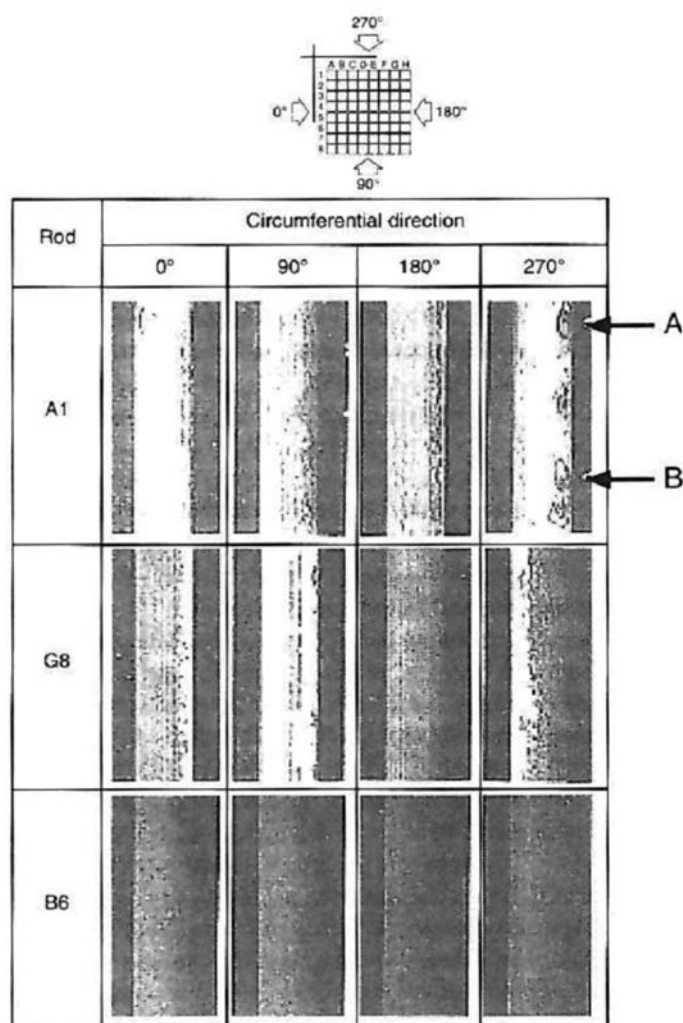


FIG. 5.27. Typical appearances of failed Hamaoka 1 rods 'A' and 'B' showing primary defect locations [5.41].

5.4.5. Corrosion failure caused by cladding susceptibility and irregular water chemistry

Five fuel assemblies were identified as failed at Hamaoka 1 in cycle 11 in 1990 [5.41]. Crud and the outer surface of the oxide layer had been spalled off in many of the assemblies. The typical appearance of the fuel rods is shown in Fig. 5.27. Although the appearance of the rods was similar to that existing with CILC, the probability of this mechanism was quite small because copper concentration in the coolant was 1–2 orders of magnitude lower than in cases described in the previous section. An extensive investigation was performed, including surveys of reactor operations, water chemistry, cladding material and post-irradiation examinations [5.41, 5.48].

Coolant water chemistry records from cycle 1 to cycle 11 were reviewed. Electric conductivity of the coolant at the startup of cycles 9 and 11 was higher than in other cycles, but low enough to be within regulation limits. Sulphate ions (SO_4^{2-}) were also detected at the startup of cycles 9 and 11. The increase in electrical conductivity was considered to originate from chemical species which were formed by the decomposition of organic materials due to heat and radiation. During reactor operation, average sodium ion content was slightly higher than in other Japanese BWRs. The increase in Na^+ ions could be explained by the decomposition of cation resin in the condenser demineralizer. These Na^+ ions entered the core during normal operation. During reactor shutdown of these cycles, Na^+ and SO_4^{2-} were detected. These ions were thought to be dissolved from Na_2SO_4 , which adhered to the cladding tube surface.

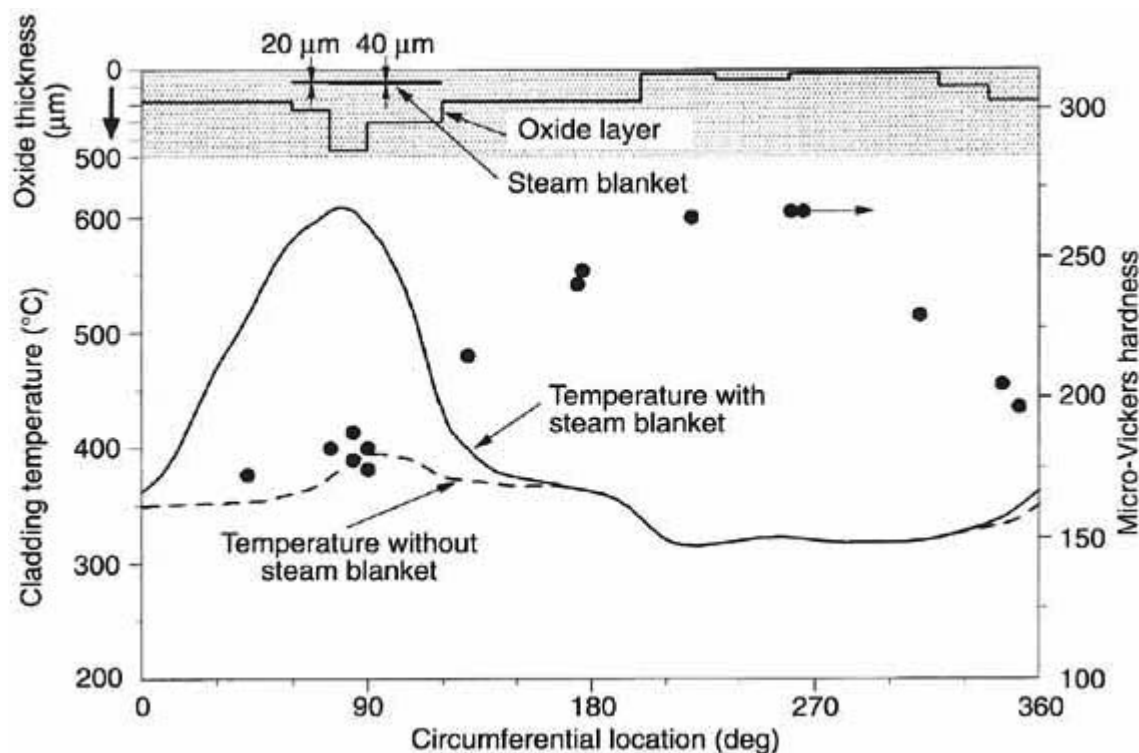


FIG. 5.28. Results of model calculation of the cladding temperature, taking into account oxide effects, with and without the steam blanket [5.41]. (I — hardness)

Nodular corrosion of BWR cladding is known to decrease with a decreasing annealing parameter. This susceptibility was calculated for tubes used in Hamaoka 1, and found to be relatively large for tubes used in cycles 8 and 9. A high temperature corrosion test was performed on archive tubing samples. The corrosion weight gain of the tubing during the high temperature corrosion test for these materials was relatively large and widely distributed.

A detailed PIE of the failed fuel bundle was conducted to explore the cause of failure. Most fuel rods located at the bundle periphery, including failed ones, showed an abnormally thick white oxide with spalling, in contrast to the normal nodular corrosion existing on other rods. Primary defects were characterized by formation of a thick oxide layer and subsequent through-wall penetration from localized oxidation. Metallographies of cladding with thick oxide revealed circumferential oxide cracks leading to gaps. The accumulated gap width ranged up to 60 μm and was considered to act like an insulating steam blanket.

The failure mechanism was considered to be as follows: chemical species such as SO_4^{2-} and other anions were present in the coolant when the reactor was started up. Na^+ and organic compounds entered the core during operation. These ions and organic compounds accelerated the beginning of corrosion on susceptible claddings. Once corrosion had started, Na_2SO_4 and similar compounds condensed on the cladding surface and may have enhanced the corrosion rate.

As stated above, gaps were observed in the thick oxide. When these gaps formed a steam blanket, the cladding temperature may have been enhanced. Figure 5.28 shows a circumferential cladding temperature profile calculated with a model that utilizes measured oxide thickness, together with measured radial cladding hardness distribution. Without the steam blanket effect, maximum cladding temperature was around 400°C, which was not enough to cause recovery of irradiation hardening. However, on the assumption of a steam blanket thickness of 40 mm, maximum temperature increased to around 600°C, which was consistent with the observed hardness recovery. The zircaloy corrosion rate was high enough at such high temperatures to penetrate the cladding wall.

5.4.6. Corrosion due to carbon containing crud

In Kola 2 (WWER-440) practically all fuel assemblies were overheated as a result of a decrease in coolant flow rate (20–30%) in 1990. In-core inspection revealed five leaking assemblies, and PIE of one of the assemblies

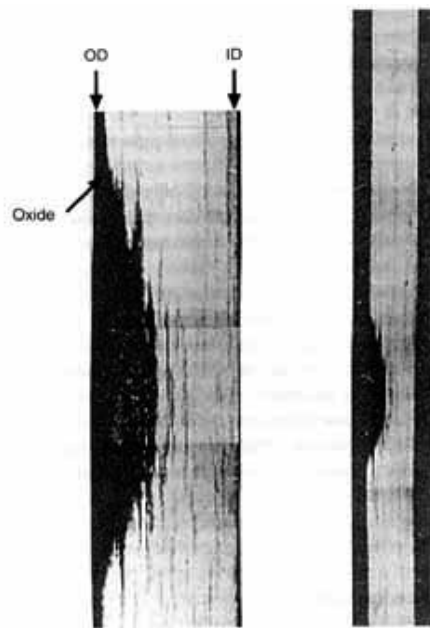


FIG. 5.29. Area of localized hydriding at axial position associated with a gap in the fuel column [5.42].

showed heavy deposits on the fuel rod surfaces, filling the gaps between the rods as well as the gaps between rods and spacer grids. The maximum amount of deposit — with a thickness of up to 1 mm — was on peripheral fuel rods in the zone of spacer grids 1 and 2 from the bottom of the fuel assembly, where maximum defects were observed. Chemical analyses of the deposit showed that carbon and iron were the main elements, leading to the suggestion that the cause of failure was deposition of organic compounds from the coolant into zones with high flow turbulence (spacer grids). The origin of the organic elements was not identified [5.27].

Reactor coolant pump vibration and thermal barrier damage took place in the South Ukraine plant in 1983, with detection of Np in the coolant. Later it was found that several fuel rods were covered with a deposit containing much carbon. Under the deposit, pittings — resulting from nodular corrosion — were discovered. The pitting depth was 50% of the fuel rod cladding thickness. The carbon was considered to originate from the graphite inserts of one of the reactor coolant pumps. The vibration of the reactor coolant pump and thermal barrier damage took place prior to fuel damage, though no fuel failure occurred [5.49].

The ingress of organic compound into the primary circuit of the Kalinin 1 plant was registered in 1991, and the leaking of two fuel assemblies was later observed [5.27].

5.4.7. Local concentration of corrosion hydrogen (suspect)

In a hot cell examination programme, jointly sponsored by EPRI and a utility owners group, ultrasonic examination of fuel rods indicated failed rods to be predominantly located near the central instrument tube, suggesting a power dependent failure mechanism [5.42]. When eddy current examination of neighbouring sound rods showed indications of incipient defects, these rods were sent to a hot cell laboratory for further examination. Hot cell examination of the sound rods revealed highly localized oxidation and hydriding at axial positions, corresponding to the regions of highest cladding temperature. Extensive hydriding was concentrated in areas associated with gaps in the fuel column as shown in Fig. 5.29. Such areas act as a sink for hydrogen. The source of hydrogen is believed to be cladding corrosion. These results indicate that local concentration of corrosion hydrogen is a potential cause of failure at high burnup.

5.4.8. Exceptional crud after plant decontamination, WWER-440

An isolated event of exceptional crud buildup was observed in 1995 in Loviisa-2 (WWER-440) in the initial cycle after plant decontamination [5.50, 5.51]. Anomalies in the coolant outlet temperature were noticed during

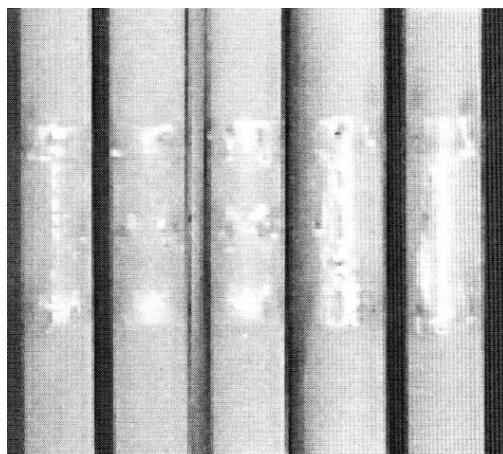


FIG. 5.30. Accelerated corrosion under an Inconel spacer grid [5.54].

operation. Later fuel inspection revealed high crud deposits in many assemblies and some fuel rod failures in seven assemblies from fretting due to increased local flow resistance and turbulence. The crud was attributed to insufficient cleaning of the circuit, which left crud from the decontamination process in stagnant zones.

Hydraulic anomaly, e.g. decreasing flow in the primary loop, rise of the pressure drop in the core, crud detection on FA heads, and observation of crud particles fallen from FAs during refuelling movements, was observed in 2001–2003 at Paks-1, -2 and -3, but not in Paks-4 [5.52]. Worsening of hydraulic parameters was most severe for two year FAs, and not so severe for freshly loaded FAs. Finally, the flow through two year FAs at Paks-3 after reloading in December 2002 reached the design limit; Unit 3 was stopped and the whole core was changed. After detailed analysis/experiments, this phenomenon was explained, as had been earlier observed at Loviisa-2, by crud corrosion deposits on fuel assemblies after a large number of steam generator decontamination procedures. The correlation between rise of the pressure drop in the reactor and the number of steam generator decontamination procedures during one preventive maintenance was studied and confirmed [5.52] with recommendations on its mitigation.

5.4.9. Shadow corrosion (BWRs)

Enhanced local corrosion of zirconium base alloys near spacer grids or structural parts of Inconel and stainless steel have been commonly observed not only in BWRs but also in SGHWRs, and have been investigated in the past. The visual appearance often resembles a shadow of the other component. Therefore, the phenomenon is referred to as ‘shadow corrosion’. Normally, shadow corrosion is of no practical significance. Typically, an oxide thickness of 3–4 mils has been observed on fuel rods at Inconel spring contact locations without there being any significant impact on fuel reliability.

However, in 1997, several fuel failures due to accelerated corrosion under Inconel spacer grids were observed in Leibstadt, a BWR in Switzerland [5.53]. Unexpected extensive corrosion of the rod cladding under grid was found during routine inspection on four and five cycle ABB fuel assemblies. Figure 5.30 illustrates the general corrosion enhancement under the Inconel spacers [5.54].

The mechanisms of shadow corrosion under an Inconel spacer grid and their variation with geometry, material and water chemistry are not fully understood. Still, recent investigations differ considerably in their conclusions. One proposed explanation is irradiation induced electrochemical corrosion. The experimental evidence in connection with shadow corrosion is, with few exceptions, remarkably consistent with a galvanic corrosion mechanism. However, shadow corrosion develops in-core, even when zircaloy and another metal are galvanically insulated, and does not develop out of core, even when zircaloy and another metal are galvanically coupled. Thus, the effect of irradiation seems necessary, in addition to the galvanic corrosion mechanism, to explain most experimental facts and observations in connection with shadow corrosion [5.55].

However, these factors were by far not sufficient to explain the observed effect in the Leibstadt reactor. Further investigation, including modifications to the water chemistry, led to the conclusion that the main causes were an unusually low Fe/Zn ratio in the coolant due to pipe contamination, and zinc addition [5.56].

5.5. PELLET–CLADDING INTERACTION

5.5.1. General background

More rarely observed in PWRs, PCI failure remained the second cause of failure in BWRs during the reporting period. Extensive investigations of fuel failure due to PCI have been performed with related ramp testing and laboratory work. A large number of papers on this subject were published in the late 1970s and at the beginning of the 1980s. Reviews have been published by Garzarolli [5.1], El-Adham [5.2] and Cox [5.57].

PCI failures may occur during severe local power ramps after prolonged low power operation. Many earlier and some of the most recent PCI failures in BWRs have been related to control blade manoeuvres resulting in significant changes in local fuel rod power levels. This correlation is further evidenced by the observed location of PCI failures in failed assemblies. PCI fuel rod failures typically occur in the first or second row of the bundle closest to the control blade. The changes in fuel rod power due to control blade movement are greatest for those fuel rods.

The outer appearance of a non-degraded PCI defect is characterized by fine axial cracks, often associated with ‘X’ marks on the cladding outer surface. When the defect is small, it appears to be the size of a pinhole. Metallographic examination reveals tiny brittle cracks starting from the inner surface, mostly at pellet–pellet interfaces and opposite radial pellet cracks, but sometimes opposite other pellet imperfections, such as missing chips or chips wedged into the pellet–cladding gap. Detailed examination of these cracks with SEM showed transcrystalline facets (cleavage fractures) linked by flutings (ductile fractures) as well as intercrystalline fractures.

For practical purposes, PCI failure is characterized by the following five operational factors associated with a power ramp:

- Burnup accumulated prior to the ramp;
- Maximum rod power during the ramp;
- Ramp height, i.e. power increment beyond the pre-irradiated power level;
- Average power ramp rate;
- Dwell time at high power.

For the defect to occur in an LWR, all five parameters have to be in a critical range simultaneously.

Ramp testing of light water reactor fuel rods in a test reactor has been a valuable experimental technique to investigate and qualify fuel performance under severe power variations. In particular, the PCI effect, which is a potential mechanism for cladding rupture during normal reactor operation, is a cause of concern when new fuel rod designs are introduced in reactor. Figure 5.31 provides an illustration of the power sequence during a test. To highlight burnup dependence on PCI failure, correlations between maximum rod power and burnup are most commonly used. Figs 5.32 [5.58] show typical PCI failure thresholds obtained by test reactor ramp experiments for PWR rods. It seems that the PCI failure threshold does not continuously decrease at high burnup. No failure caused by a power ramp test has been reported on PWR fuel at burnup higher than 50 GW·d/t U.

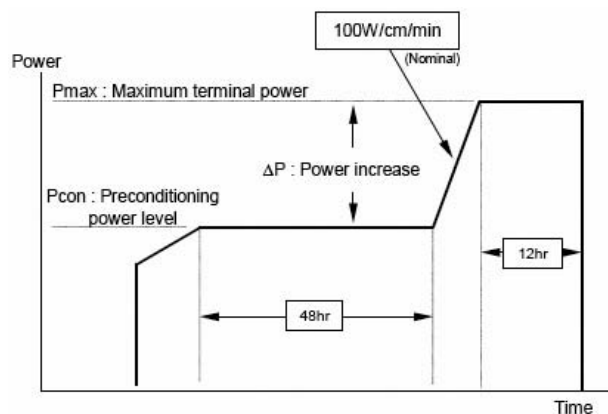


FIG. 5.31. Power sequence of the power ramp test [5.58].

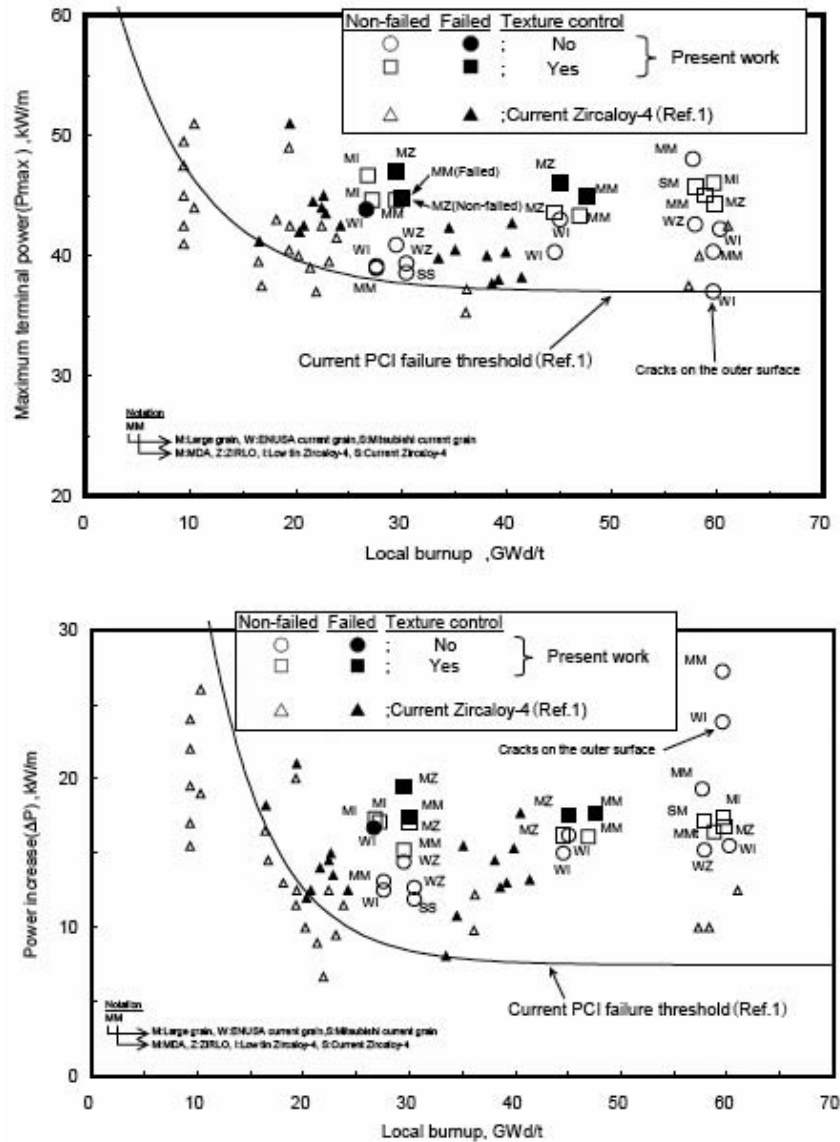


FIG. 5.32. Typical threshold for PCI failure using power ramping tests, PWR fuel rods [5.58].

For a PCI defect to occur in CANDU fuel, a recent reassessment [5.59] of extensive power ramp data has established that propensity to PCI is much better measured by pre-ramp power than by maximum rod power during the ramp. The recent reassessment also shows that there is no correlation with dwell time in the range for which in-reactor CANDU data are currently available. Further, rather than expressing propensity for defects using two independent curves for rod power and for power increase as functions of burnup, the reassessment suggests it is far superior to use a single curve to connect the three variables — power increase, burnup, and pre-ramp power. This is done through ‘damage parameter’ and ‘equivalent power increase’; see Ref. [5.59] for details. The above evolutions in formulations reduce scatter in the data by about a factor of three, from about 26 kW/m to about 7.6 kW/m [5.60] in a database that contains about 700 ramps near defect thresholds in power reactors, including some 126 defective bundles. They also render the PCI formulation more consistent with data for kinetics of fission gas release [5.60] and with out reactor mechanistic tests [5.61]. Figs 5.33 and 5.34 show current empirically derived failure thresholds for the CANDU reactor after making the above adjustments in formulations.

The above formulation has been found to be very effective in assessing PCI in current fuel designs. Nevertheless, its capabilities of addressing some types of design evolutions — such as optimizations of pellet shape, pellet density, radial and axial clearances, weld shape, etc. — are limited. To enable such design evolutions and optimizations, a more mechanistically based model/correlation has also been developed, called the INTEGRITY suite of codes [5.62]. This model uses mechanistic parameters, such as stresses, strains, fission product

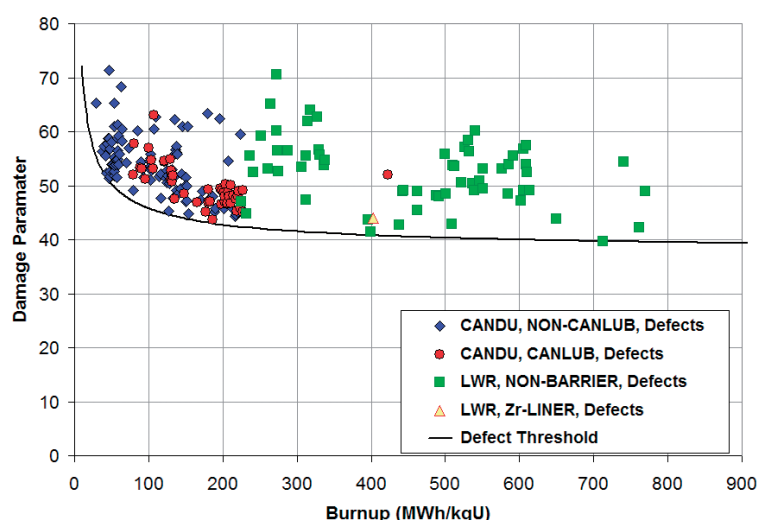


FIG. 5.33. Power ramp defects: dependence on pre-ramp power.

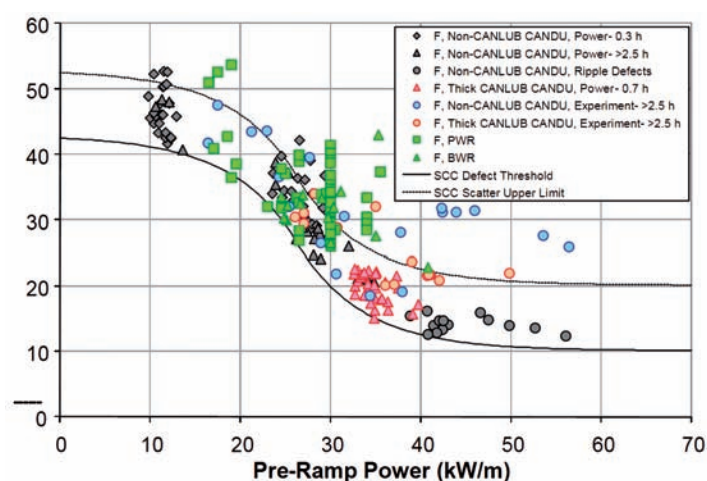


FIG. 5.34. Threshold for PCI failure of CANDU fuel element dependence on burnup.

concentration, etc., and information can thus be more confidently extrapolated to guide design evolutions and/or variabilities in operational and manufacturing parameters. Other design differences have also been studied using this approach, such as the effect of coolant pressure, smaller or larger fuel element diameters, different pellet densities, etc. Further details are available in Ref. [5.62].

5.5.2. Failure mechanism

PCI fuel failures result from the combined effects of fuel pellet expansion — especially at pellet radial cracks and pellet–pellet axial interfaces, leading to stresses in the cladding — and the presence of an aggressive fission product environment. Since the characteristics of PCI failures are quite similar to those of laboratory zircaloy stress–corrosion cracking (SCC) tests, including metallography and scanning electron microscopy (SEM), the PCI failure mechanism can be considered to be fission product induced.

Generally, SCC is a slow fracture process which occurs under a tensile stress lower than the fracture stress faced by the original material in the corrosive environment. For the occurrence of SCC, three fundamental conditions have to be satisfied simultaneously: a sufficient level and duration of stress in the material, susceptibility of the material to SCC, and an aggressive environment. Stress and environment are directly correlated to the five operational factors discussed in the previous section.

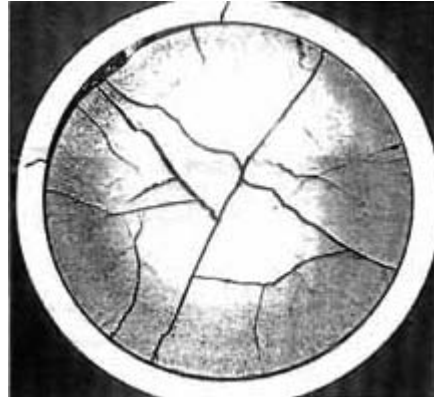


FIG. 5.35. PCI failure due to a wedge shaped pellet chip lodging between pellet and cladding [5.66].

Cladding stresses and strains

A large number of laboratory experiments have been performed on the tensile stress needed to cause SCC. The stress depends on iodine concentration, holding time, material and temperature [5.63].

PCI defects tend to occur at locally highly strained/stressed regions such as circumferential cladding ridging, at pellet chips, or at cladding defects above a certain depth. In-pile measurements of ridge formations and their correlation with PCI failure have been conducted in the Halden reactor, and the data have been useful for fuel code verifications [5.64]. PCI failures due to missing or wedged pellet chips in the fuel rod were reported in both PWRs and BWRs. In the early days of Salem 1, fuel failure due to pellet lockup and excessive rod growth was reported. Rod growth was due to PCMI/ratcheting and irradiation induced elongation. Internal tube stress and fission product iodine created crack initiation via SCC [5.65]. In addition, PCI defects in CANDUs have also occurred at the notch formed at the re-entrant corner near the weld between the sheath and the end cap at the end of the pellet stack.

Hot cell examination of a failed BWR zirconium barrier rod (8×8 fuel) from one manufacturer indicated that the primary cause of failure had been PCI, and was attributed to power ramps by control rod movements in a power range where no failures are expected from ramp testing. Similar failures were observed in a few other plants, leading to about 20 barrier rod failures due to PCI. Since there are over two million barrier rods which operate successfully without PCI failures, barrier rods are considered effective in preventing PCI failures. In the above hot cell examination case, a missing pellet chip was identified as the cause of PCI failure. Manufacturing changes have been implemented to reduce pellet chipping.

In Gundremmingen B, 9×9 fuel rod failure was found to be a result of the presence of fuel fragments in the gap between pellet and cladding, as shown in Fig. 5.35 [5.66]. The metallographic cross-section indicated that the chips are wedge shaped, and originating from the pellet end face, which slid into the gap during operation. In contrast to the usual finding of one crack, this had led to two local brittle fractures in the cladding at an angle of about 80° . It was pointed out that the fuel had been fabricated before current standard practices were applied, which include improved pellet quality, introduction of new end face geometry, improved loading and handling procedures, and extended quality control.

More recently, at the LaSalle and Quad Cities reactors, several fuel rod failures were attributed to the primary failure mechanism of stress corrosion cracking from pellet clad interaction (PCI), exacerbated by the presence of missing pellet surface. The resultant stress intensity concentration factor associated with the missing pellet surface condition, coupled with large power changes and power recovery ramps during/following sequence exchanges in the presence of relatively high concentrations of fission product inventory at the inner surface of the cladding, created the combination of conditions necessary to produce these failures. Indeed, during manoeuvring of the control blades, the location of failures was subject to power ramping, although the magnitude of the ramp was well within expected limits required to prevent PCI failure. Therefore, the missing pellet surface appears to have increased local stress levels and contributed to the failure [5.67].

PCI failures were also recently observed at some PWRs in the United States of America. These failures appeared during axial shape power control rod (ASPR) pull or during restart after refuelling. They were due to pellet chips in the fuel-clad gap associated with significant changes in the local fuel rod power level.

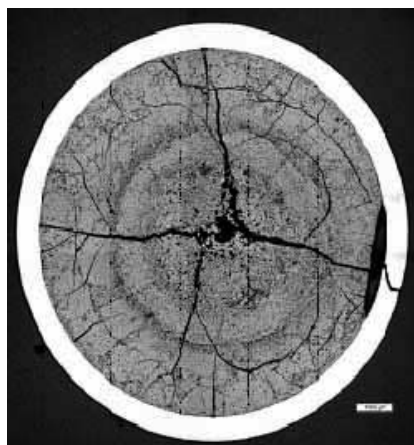


FIG. 5.36. PCI failure due to missing pellet surface [5.67].

Cladding SCC susceptibility

All commercially used zirconium alloys are susceptible to PCI failures and iodine induced SCC. In laboratory SCC tests, unalloyed zirconium is the most resistant to cracking and requires significant ‘plastic strain’ to initiate cracks. Unalloyed zirconium is thus used as a barrier layer in the cladding as a remedy for PCI, and the reduction in ductility of irradiated barrier material has been reported to be small and to saturate with burnup [5.57]. For zirconium alloys, susceptibility to SCC in laboratory tests increases with increasing yield strength, probably because higher yield strength permits higher stress intensities to be sustained at incipient cracks. However, this is not confirmed by ramp tests with rods of significant burnup, which indicates that the effect disappears during irradiation. The influence of irradiation on the SCC susceptibility of zircaloy was reported earlier to be significant after exposure to a fast neutron fluence of about 10^{24} n/m², which corresponds to about 1 MW·d/kg U [5.68].

The influence of zircaloy texture is considered significant, first because the SCC crack propagates on the pseudo-cleavage of the basal plane and second, because plastic deformation takes place in the form of prismatic slip on a plane perpendicular to the basal plane. Consequently, when pseudo-cleavage is promoted, prismatic slip is difficult to activate and does not reduce the force transferred to the basal plane. Laboratory test results on the influence of the zircaloy texture have been reported by several researchers [5.69, 5.70] and show that it is favourable when the basal pole is in the radial direction.

Environment

Iodine has been found to be the most probable fission product causing SCC. Some metal iodides such as those of Fe, Al, Zr and Te are known to cause SCC of zircaloy just like iodine itself [5.71–5.73]. Among them, gaseous ZrI₄ is found to be the most corrosive agent [5.73]. SEM examination has revealed that the morphology of fracture surfaces caused by iodide induced SCC is similar to that caused by iodine induced SCC. Furthermore, EPMA analysis of fracture surfaces suggests that, in the case of iodide induced SCC, ZrI₄ forms as a result of a reaction between the metal iodides and Zr might be a cause of SCC rather than the iodides themselves. CsI was found to induce SCC on zircaloy in the laboratory and under gamma ray irradiation. It took longer to cause failure than with the same batch of specimens in iodine vapour. Under irradiation, sufficient iodine partial pressure is estimated to be available from the radiation dissociation of CsI, and this assumption has been verified by experiments. Figure 5.37 illustrates that iodine partial pressure significantly increases under irradiation (compare ‘rad’ versus ‘non-rad’). The estimated iodine partial pressure is above the experimental data as shown by filled square marks [5.74]. Chemical species other than iodine, particularly Cs and Cd, were found to be capable of causing SCC, and the mixture of the two was more potent than either alone. Furthermore, the fractography of Cs/Cd-induced failure was very similar to that of iodine-induced cracking [5.73].

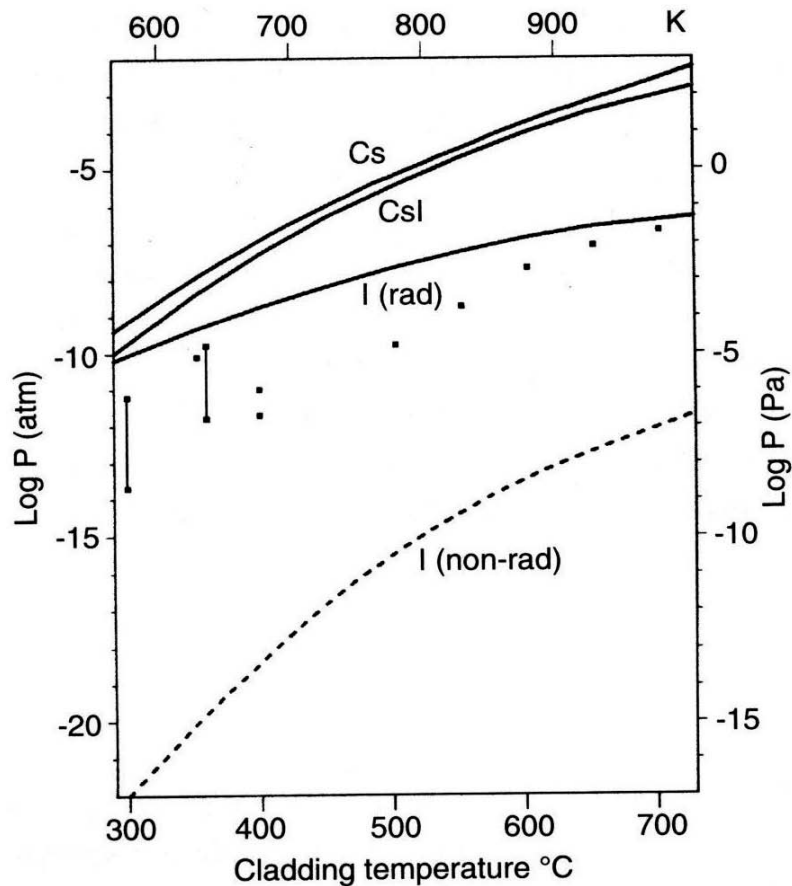


FIG. 5.37. Effect of irradiation on the iodine partial pressure and critical value for SCC [5.74]. Filled squares represent experimental critical values for SCC of zircaloy.

Time dependence

SCC is a time dependent phenomenon. In most cases, the time to reach PCI failure after power ramp is not known in power reactors. In test reactors, where parameters related to PCI failure are well controlled, PCI failure can be traced more systematically. The INTER-RAMP Project, which works with irradiated BWR segmented rods, highlighted very clear correlations between power increase and time to PCI failure [5.75]. The time needed to cause PCI failure ranged from some minutes to one day. To further investigate PCI crack initiation and propagation during power ramps, very rapid power ramps with short holding times above the PCI failure threshold were performed in the TRANS-RAMP I/II Project [5.76]. Clear power-time correlations for crack initiation on the inside surface of zircaloy cladding; crack through-wall propagation and out leakage of fission products to the water coolant were obtained. Figure 5.38 shows results for BWR fuel segments (TRANS-RAMP I). From this figure, it can be deduced that crack initiation occurs rapidly within approximately 10 seconds, and through-wall penetration occurs within about one minute. The PCI failure boundary coincided with the conventional failure threshold at long hold times [5.76].

5.5.3. PCI failure during abnormal transients

Occurrences of PCI failure are usually related to normal operating conditions in BWRs and PWRs. The duration time is short enough to avoid PCI failure in abnormal or off normal transients which result in high power, causing reactor scram. It has been reported that fuel failure has occurred as a result of PCMI (pellet cladding mechanical interaction) in reactivity initiated transient tests [5.77]. This type of failure, however, is related to accident conditions and is outside the scope of this review.

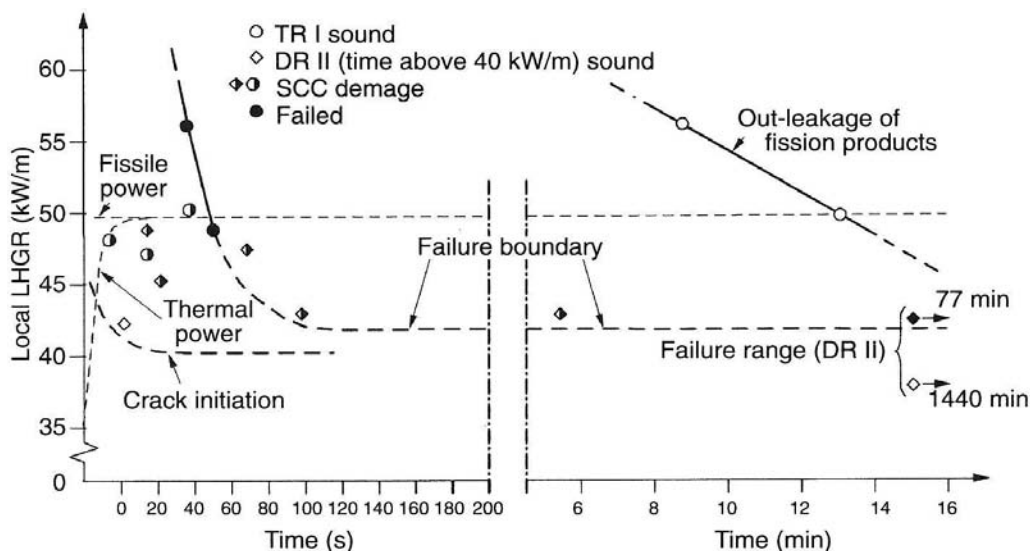


FIG. 5.38. TRANS-RAMP I PCI failure progression, BWR fuel [5.76].

In CANDU and WWER reactors, PCI failures were reported to have occurred during abnormal transients and are therefore described below.

In November 1988, a reactor trip occurred at Pickering Unit 1. During trip recovery, all adjusters were withdrawn from the core and reactor power rose to 87% for 40 minutes. This was outside the range of normal operation, as reactor power is normally limited to 65% in this type of situation. As a result of the transient, about 200 fuel bundles in 40 central channels sustained large power ramps. Thirty-six defective bundles contained about 290 failed outer fuel elements [5.78]. The mechanism of this failure was PCI as described in the previous section.

5.5.4. Other potential defect causes

In the 1980s, some fuel channels at Bruce NGS A were unavailable for refuelling, which allowed several bundles to achieve burnups of about 450 MW·h/kg U, well in excess of the normal range. Four bundles for which burnups ranged between 583 and 772 MW·h/kg U were found to be defective [5.79]. The cause of these failures is uncertain, but they did have SCC defect features in the sheath and end caps. There were no power ramps due to refuelling that would have led to SCC. The intact elements from the same bundle outer ring as that belonging to the defective elements were found to have very high gas release, in the order of 23%–26% [5.80]. These high releases may have contributed to fuel failure in two ways. First, the high concentration of fission products within the pellet to sheath gap may have given rise to corrosive conditions conducive to SCC. Second, the high internal gas pressure, which may have exceeded coolant pressure, may have made the sheath ‘lift off’, causing a reduction in heat transfer and elevated fuel temperatures, again leading to stress levels in the sheath conducive to SCC [5.81].

In December 1983 and early 1984, Bruce A experienced one of the largest defect excursions in its history [5.82]. At least 43 fuel bundles containing 140 defective fuel elements were found to have defected. Most of the defective bundles (37 out of 43) had been made by the same manufacturer and had been irradiated in Unit 3 at Bruce A. The cracks were generally circular, and occurred very close to the end cap weld. In some cases, the cracking resulted in complete separation of the end cap from the element.

The above failures occurred during power ramps, but the location of failure was not in the usual place along circumferential ridges. Instead, the cracks occurred at the junction of the clad and the end cap. PIE of failed fuel elements found evidence of hard pellet/end cap axial contact, which was most unusual at the ratings experienced by these fuels. Companion unfailed fuel elements had trivial fission gas release consistent with the very low burnups of such fuel. This suggests there was no likelihood of gas pressure driven defects.

A review of the manufacturing processes indicated that the affected batch of fuel may have contained, simultaneously a statistically unlikely combination of low axial and diametral clearances, high pellet density, and large geometric stress concentration at the sheath/end cap junction [5.83].

On-power diametral expansion, in-reactor sheath creep, and reduced in-reactor pellet densification then resulted in unusually low on-power clearances (axial and diametral). Because of this, during a subsequent power ramp, pellet expansion caused excessive strain and stress at the sheath/end cap junction. This, in turn, caused environmentally assisted cracking at the sheath/end cap junction.

5.5.5. Corrective actions

To prevent PCI failure, it is necessary to remove at least one of the fundamental conditions causing SCC (stress, susceptibility, environment). There are two principal types of remedies. One is to improve plant operational strategies to limit power transients, and the other is to improve the design of fuel rods. The first remedy is described in Section 8. The second remedy, design improvement, consists of several approaches:

- In BWRs changing the lattice configuration to have more fuel rods per assembly (from the 9×9 to the 10×10 , for example), contributes to lower average linear heat generation rates (LHGRs) which is desirable, particularly in high power density BWRs, to reduce the potential risk of PCI failure;
- Another approach is development of Zr liner cladding at the inner cladding surface. Pure zirconium (of either crystal bar or sponge origin) has been found to provide excellent resistance to PCI and, following extensive irradiation and ramp testing, was introduced in commercial BWR fuel in the early 1980s. The barrier is soft and serves to reduce local stress and hence provides cladding resistance to SCC. Zirconium barrier fuel has been generally successful in preventing PCI failures over many years. However, several BWR barrier fuel failure incidents have resulted in high off-gas, plant contamination and even forced outages; this has become a serious problem for utilities and fuel vendors. The incidents were characterized by primary failures caused mostly by debris fretting, manufacturing defects or PCI, followed by power changes resulting in severe fuel degradation. This degradation was due to secondary damage sustained during post-primary defect operations. A pure zirconium barrier appears statistically to be more susceptible than standard non-barrier cladding to severe secondary degradation following a primary rod failure. To remedy secondary degradation, fuel suppliers developed new liners which combine pellet clad interaction (PCI) resistance and better corrosion resistance on the inside, plus a pure zirconium liner to significantly reduce the risk of high post defect activity release. These ‘duplex liners’ consist of a very thin Zry surface layer and thick pure Zr layer, an iron enhanced zirconium liner cladding (0.2 to 0.5 wt.% Fe) developed by SIEMENS [5.84] or a Sn alloyed Zr liner developed by Westinghouse and AB Sandvik Steel [5.85]. The introduction of these modified barrier concepts has seriously reduced secondary damage, which will be described in more detail in Section 7.
- An additional mitigation method currently under development to improve PCI resistance involves the use of softer UO_2 fuel pellets doped with small amounts of chemical additives for both PWR and BWR applications. Westinghouse has developed ADOPT (Advanced Doped Pellet Technology) UO_2 fuel, containing chromium and aluminum oxides in order to improve performance by reducing fission product gas release through enlarging the mean grain size, and also by making fuel ‘softer’ under pellet clad mechanical interaction, through a higher fuel creep rate. The additives facilitate densification and diffusion during sintering, which results in a higher density and an enlarged grain size. The ADOPT pellets have been proven to provide the rod with valuable internal properties, such as reduced FGR and increased PCI margins [5.86]. With the same goal, AREVA NP has been investigating the ability of large grain doped UO_2 microstructures [5.87] since the early 1990s. There are high expectations for the optimized chromium doped UO_2 fuel. Power ramp tests performed on fuel rods from the CONCERTO programme, launched to verify the behaviour of different doped fuels in PWR conditions, proved that doped fuel exhibits significantly higher performance than UO_2 reference fuel (Fig. 5.39). Currently, emphasis is being placed on expanding that fuel qualification with the aim of producing a full set of consistent and valuable licensing data to respond to nuclear utilities’ requests.

In CANDU fuel, CANLUB coatings with either graphite or siloxine are used. The effectiveness of the coatings can be attributed to both lubrication and adsorption of iodine (or modification of the Zr/I system chemistry).

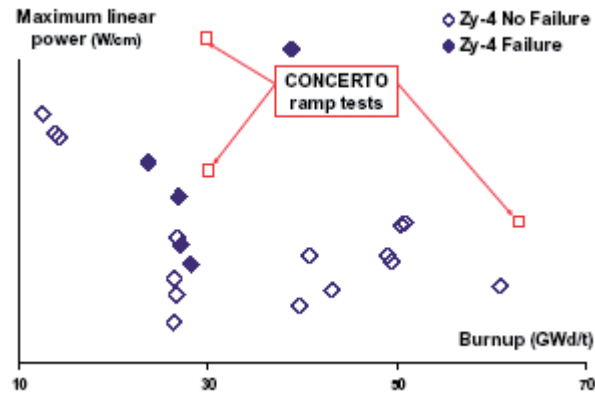


FIG. 5.39. Ramp test results for PWR Zy-4 and CONCERTO rods [5.87].

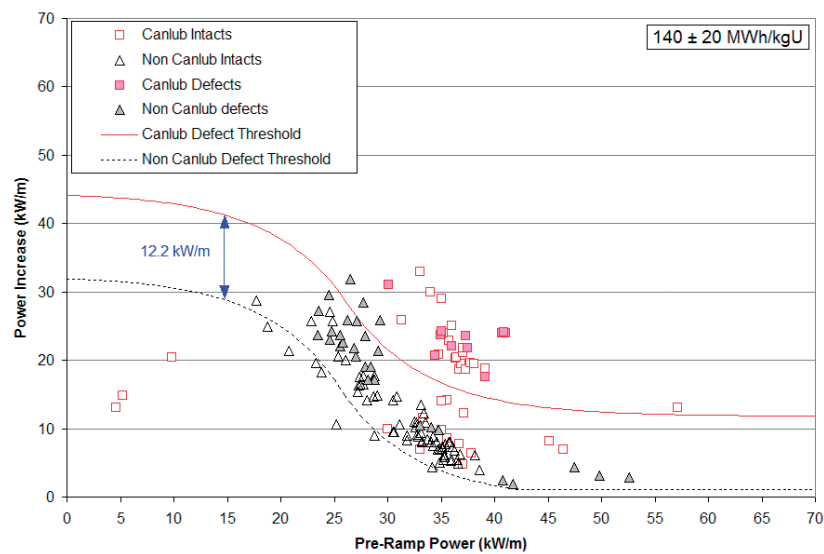


FIG. 5.40. Improvement of CANDU fuel elements PCI threshold due mainly to CANLUB.

A pioneering work on failure thresholds was published in 1977 [5.88] on CANDU reactor fuel. This has since been modified using statistical data, including CANLUB coated fuel in current reactor types. Current failure thresholds are shown in Fig. 5.40, along with non-CANLUB thresholds. All normal operations at CANDU stations are carried out in such a way that the power conditions of all bundles in the core do not exceed these thresholds.

Manufacturing process changes have been implemented addressing the overall quality of fuel pellet surface conditions, the effectiveness of the pellet visual inspection process and inspection acceptance standards and, finally, in the technology employed to load pellets into the fuel rod, with a primary focus on improving surface conditions and reducing pellet chips [5.89].

5.6. MANUFACTURING DEFECTS

5.6.1. General background

The number of failures resulting from manufacturing defects has decreased over the years owing to improvements in manufacturing processes. However, such failures still occur and include mainly end plug and welding defects as well as, in some cases, tubing reduction flaws. An isolated event of PWR fuel failure by Ni

contamination of the cladding surface due to improper loading machine operation during assembly has been described. In addition, fuel failure supported by insufficient free volume has taken place in CANDU fuel.

Incomplete welds and leaking end plugs have a common characteristic: the primary defect is a small hole near the end of the fuel rod or element. Because of the initial size of the hole, both types tend to limit the amount of coolant ingress and fission product release to the coolant. This means that these defects can remain undetected in the core until the primary hole size increases or secondary damage develops. The incubation period for secondary hydriding damage depends on the initial primary hole size, fuel temperature and other parameters as described in Section 7.

There has been a current tendency for fuel vendors to use automated process control to eliminate operator dependent faults in fabrication. Inspection methods have also improved. For these reasons, fuel failure due to manufacturing errors are at a low level.

5.6.2. Leaking end plugs

The fabrication of zirconium alloy end plugs can result in defects providing a leakage path for gases through the end plug. Chloride stringers present in some zirconium products sometimes form one such defect, which can be aligned longitudinally along the axis of the bar stock in the fabrication process [5.90]. Cold swaging of a barstock has also resulted in internal voids at the end of bars by deforming the outer surfaces more than the centre and eventually leaving central voids. Failures of this type have been rare. Clearly, the remedy is careful fabrication process control and inspection. Ultrasonic inspection of finished barstock is necessary.

Westinghouse reported that several rods identified as leaking may have problems as the result of an extrusion defect of end plug material [5.91]. An extrusion tail is introduced in end plug bar material during the manufacturing process and is subsequently removed as a specific step in the process. In a few isolated cases, the tail was not entirely removed. Most fuel rods believed to be leaking as a result of this mechanism have been identified through correlation with fabrication records rather than through direct visual confirmation. Two rods were confirmed to be leaking due to this mechanism during detailed destructive examinations in a hot cell. Several types of corrective action have been taken to eliminate this problem. The extrusion bar is cropped further away from the end to ensure that the actual end of the defect is removed. The end of the extrusion bar is then inspected metallographically. Ultrasonic testing of the final bar material has also become increasingly sensitive and provides a final check that the defect has been removed.

5.6.3. Welding defects

Contamination of the weld

The tungsten inert gas welding process for a zircaloy end plug is conducted either in a vacuum or in an inert gas atmosphere. Poor control of the gas atmosphere or leaky vacuum chambers have led to nitrogen contamination of the welds, and the resulting corrosion was sufficient in some cases to breach weld integrity. Clearly, the remedy is better atmosphere control and monitoring using impurity sensors [5.90].

Recently, several welding defects were observed on M5 cladding supplied by AREVA. According to information in [5.92], these failures were due to accidental pollution of the end plug welding during manufacturing. In order to reinforce a return to good reliability of M5 rods, AREVA put a 'cleanliness plan' in place in its factories and decided to replace laser and tungsten inert gas (TIG) welding processes with the upset shape welding (USW) process, which is less sensitive to pollution.

Incomplete welds

Fuel failures due to incomplete end cap welds are extremely rare, and no particular incident has been reported regarding LWRs in recent years. An example of an incomplete end plug weld is given in Fig. 5.41. This defective product was inadvertently not rejected during controls [5.18].

For CANDUs, two events have been recorded. During the first year in service of Units 1 and 2 at Bruce NGS-A (1977–1978), defective fuel bundles were found [5.81]. Post-irradiation examinations showed the primary defects to be fabrication flaws. Of the defective elements examined, 80% were seen to have defects due to

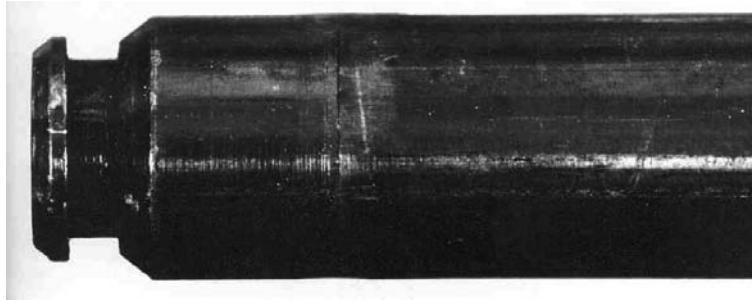


FIG. 5.41. Incomplete end plug weld.

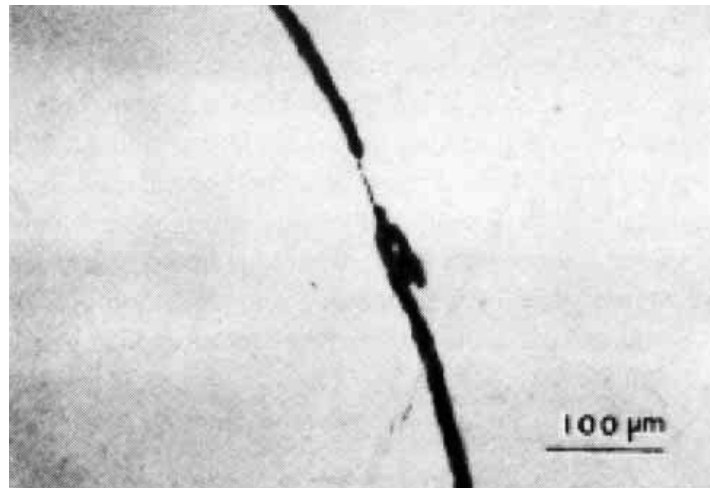


FIG. 5.42. Typical leak path due to incomplete cladding at the end cap weld [5.93].

incomplete sheath to end cap closure welds. The majority of flaws were very small oxide filled stringers in the weld area which allowed ingress of the primary heat transfer fluid into the fuel element. Figure 5.42 shows a typical leak path.

A preferential distribution of defects was observed in 1981–1982 at the Douglas Point CANDU reactor during a defect excursion. At the time, the core contained many bundles with elements having incomplete closure welds. The affected bundles were randomly distributed in the core, but the defects were preferentially located at high power positions towards the outlet end of the channel. The number of failures due to incomplete welds has diminished as a result of quality control programmes.

Grain boundary separation

In 1992, a weld defect type not previously recognized was experienced and identified by GE [5.94]. The weld defect type is known as grain boundary separation and is characterized by a microscopic crack-like defect initiated at the weld inner surface and extending radially along prior beta grain boundaries. This defect results from the application of tensile stresses in the weld during the cool down phase of weld operation. Figure 5.43 shows a representative grain boundary separation defect [5.94]. Corrective actions taken by GE include: (1) modification of weld station hardware and processes to minimize the extent and effect of applied tensile stresses, and (2) development and implementation of extended ultrasonic capability for 100% inspection of end plug welds for grain boundary separation.

Undercut in end cap welding (potential cause of failure)

BNFC [5.95] investigated the root cause of fuel failure and reported that approximately 2.8 per 100 000 fuel rods failed because of an undetermined failure mechanism, possibly a manufacturing defect. A potential cause of this failure mechanism was discovered: X ray examinations have identified an undercut in the cladding that occurs

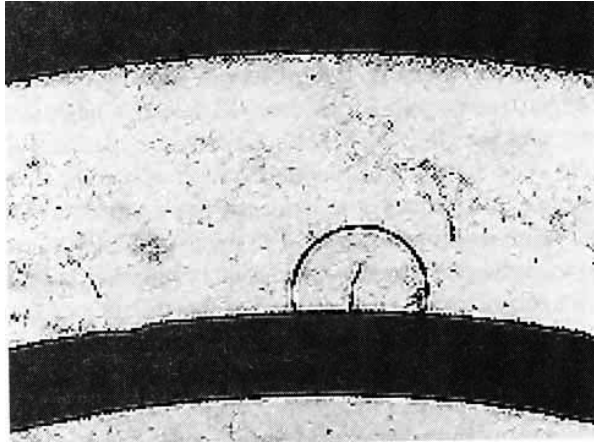


FIG. 5.43. Typical grain boundary separation (GBS) defect [5.94].

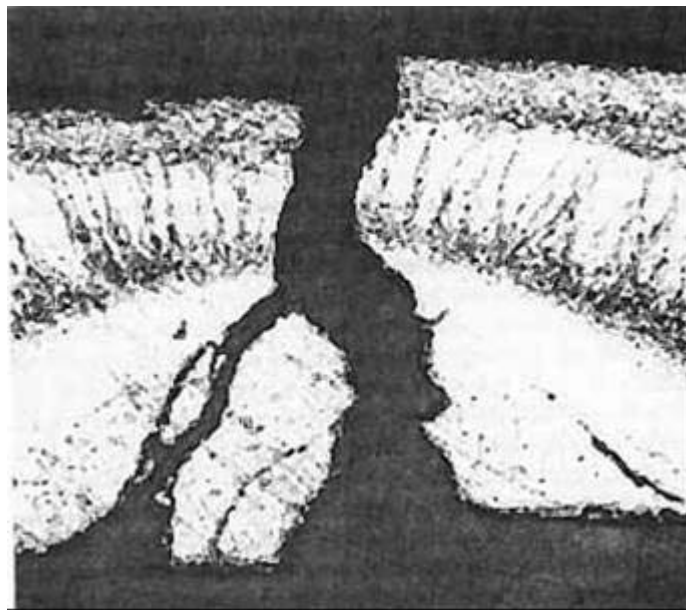


FIG. 5.44. External heavy local hydriding and crack due to Ni contamination of the cladding surface [5.96].

very infrequently during end cap welding. This anomaly was not detectable using ultrasonic inspection. BWFC now X rays every end cap weld to ensure that all weld defects are eliminated. These defects, in most cases, are too small to result in fuel rod failure, although the occurrence of undercuts large enough to lead to failure have been discovered at a frequency similar to the rate at which fuel rods have been failing from an unknown mechanism.

5.6.4. External hydriding due to Ni contamination of the cladding surface

Two fuel assemblies failed at Ohi 1 in 1989 as a result of external local hydriding in the bottom area of the fuel rods. An increase in outer rod diameter from external hydriding occurred in peripheral rods on one side. Only one rod in each assembly was leaking.

The root cause of failure was identified to be misloading of fuel rods because of a deformed lower finger guide in the fuel rod assembling machine. The deformed finger guide had led to a smaller than normal contact area between the rod and dimple or spring of the grid and to abnormally high loads, such that Ni from grid material coating deposited on the fuel rods, and residual strain was left on the cladding surface. Under these conditions, hydrogen from the coolant entered the cladding at the locations of Ni deposition and led to heavy local hydriding, as shown in Fig. 5.44 [5.96]. The problem was resolved through design improvements and better control of the bundle assembling process [5.96].

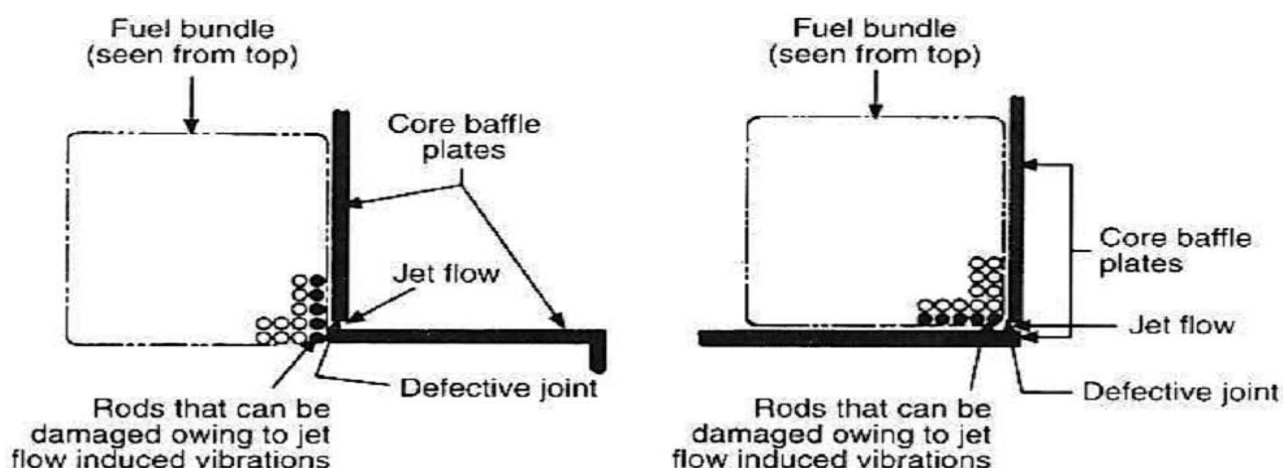


FIG. 5.45. Cross-section view of fuel bundles and baffle plates showing how gaps may open in the baffle plate joints [5.99].

5.6.5. Cladding flaws

An increase in manufacturing defects affecting GE BWR fuel was observed at the end of the 1980s [5.97]. The cause was identified as an inability of standard tubing flaw inspection systems to reliably detect certain tubing reduction flaw configurations. Local undetected flaws can increase cladding stress during power transients and result in fuel failure. Such a failure mechanism is similar to PCI failure. To eliminate tube reducer generated flaws as a cause of in-reactor fuel failure, GE initiated corrective actions, including improvements to the ultrasonic (UT) tubing flaw inspection system and introduction of a supplemental, alternate technology (Eddy Current) tubing flow inspection system.

5.7. CROSS-FLOW/BAFFLE JETTING

5.7.1. Fuel failure due to baffle jetting

Some causes of rod fretting are plant specific and therefore require plant specific corrective actions. Within this category, there is a specific mechanism of baffle jetting which can cause rod fretting and other severe damage to fuel due to cross-flow jets from defective baffle joints. Fuel failure from baffle jetting did occur in PWRs which use the down flow baffle barrel design. In this design, a small portion of total primary coolant flow is diverted into the region surrounding the reactor core, opposite the direction of flow through the core.

In these designs, there is a bypass flow path from the reactor inlet plenum which is directed downwards in the vessel between the outside of the core baffle and the inside of the core barrel [5.98]. Some of this flow passes through gaps between adjacent baffle plate joints owing to the differential pressure existing between the ascending flow of coolant through the core and the descending flow providing internal cooling for the baffles.

Figure 5.45 [5.99] shows two ways in which the joint between two core baffle plates can open and allow a high pressure jet of water to impinge on some of the fuel rods. Such gaps are caused by stress in the baffle plate assembly during reactor operation, and may occur at any point on the joint. When such a gap occurs, coolant will rush through it because of the pressure difference across the juncture. The direction of the jet will be transverse to the fuel rods and the resulting flow will be turbulent. It will be composed of localized velocity and pressure fluctuations occurring over a wide range of frequencies. The transverse flow can produce vortices which are shed in an alternating fashion. The alternating shedding of vortices creates a fluctuation force on the fuel rod cladding. The unsteady force has components both in the direction of flow and perpendicular to this direction, i.e. drag force and lift force. When the frequency of vortex shedding equals the natural frequency of the fuel rod, or half that frequency, or any multiple of it, the structure goes into resonance and fails. This phenomenon induces severe rod damage.

The defect generally appeared to be a long axial fretting ‘flat’ worn along the centre line of the rod. This resulted in a large axial split in the rod, often with loss of fuel in the split area. Rupture of some rods also occurred and fuel pellets were observed to have fallen off failed rods.

5.7.2. Corrective measures

Several corrective measures have been applied since initial observation of the baffle jetting problem. In the short term, corrective action included performing mechanical peening on the baffle joints to reduce the gap size between segments and thus reduce the flow which can cause rod vibration. However, subsequent incidents of jetting type fuel rod failures suggest that the peening operation was not always successful.

Another corrective action has been the modification of peripheral fuel assemblies. For assemblies that would reside in baffle joint locations, one modification consists of replacing the fuel rods in front of baffle joints and adjacent fuel rods with solid stainless steel rods, which are more resistant to vibration. In addition, in some cases a partial grid was installed in each mid-span of assemblies at a location next to the joint.

The Advanced Nuclear Fuels Corporation (ANF) developed special anti-fretting clips to prevent rod vibration and damage resulting from baffle plate leakage cross-flow. The clips can be attached to peripheral rods of assemblies in sites known to be susceptible to baffle jetting, and removed prior to movement of the fuel [5.100, 5.101].

The ultimate solution was modification of the reactor vessel internal assembly to reduce potential rod failure possibly related to baffle joint jetting. The modification work reverses the flow of coolant water to eliminate differential pressure. The conversion of down flow design to up flow design is achieved by plugging existing core barrel holes and adding appropriate new holes in the top former baffle plate to allow water to continue flowing upwards.

These corrective actions have successfully reduced the number of plants and fuel assemblies affected by this failure mechanism. No evidence of fuel failure as a result of baffle jetting has been observed in recent years.

Similarly, some Babcock and Wilcox (B&W) designed plants have experienced grid to rod fretting failure in the position of a baffle plate LOCA hole [5.102] in the past and again more recently. These plants have a unique baffle design that includes loss of coolant accident (LOCA) flow holes and slots that result in a pronounced cross-flow environment near the periphery of the core. The LOCA holes, which are 3.5 cm in diameter, are designed to equalize pressure across the baffle plate during a LOCA event. During normal operation, the flow rate through LOCA holes was thought to be minimal because the up flow through the annulus between the core barrel and the baffle plate was controlled to minimize pressure drop across the baffle plate.

5.8. PRIMARY HYDRIDING

5.8.1. General background

Fuel failure caused by local hydriding of the cladding due mainly to moisture or organic residues in fuel rods frequently occurred in the 1960s and 1970s in both BWRs and PWRs. Local hydriding may also occur when a fuel rod fails and steam enters the rod, producing a hydrogen source which is exposed to the cladding. This is distinguished as secondary hydriding and is described in Section 7.

Although primary hydriding is a kind of manufacturing defect, it is usually treated separately, since the cause of failure has been extensively investigated and it has features common to secondary hydriding. Effective measures have been taken to prevent primary hydriding, including careful drying of pellets and cladding, and the use of getter alloy in some BWR rods. The pellet density of PWR fuel was increased to 95% theoretical density (TD) after densification/collapse failures occurred in the early 1970s. This change resulted in a reduction of the open porosity of pellets and helped decrease the adsorption of moisture. Thus, the occurrence of fuel failure due to primary hydriding has become rare; however, it still occurs today, mainly as a result of occasional quality control errors in the fabrication processes.

The root cause of fuel failures at Chinsan 1 (BWR) was deduced to be primary hydriding caused by improper drying of cladding tubes [5.103]. In this case, human error (failure to turn on the dryer after tube loading) was behind the problem. Some failures in first cycle FRAMATOME BWR fuel rods attributed to fabrication

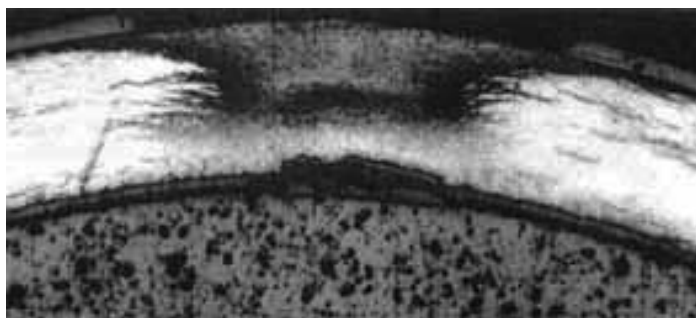


FIG. 5.46. Typical sunburst failure.

deficiencies are reported in Ref. [5.104]. Poolside examinations revealed primary hydriding due to hydrogen containing material within the rods as the most likely cause of failure. Measures were taken in fuel fabrication to prevent or at least reduce such failures.

Another typical leak of this kind occurred at the end of 1994 in a French reactor. After two weeks of operation, high fission product activity was measured and the reactor was shutdown. Ten fuel rods in two fresh assemblies were leaking. All the failed rods had blisters just above the bottom end plug. The information gathered from hot cell examinations and fabrication investigation demonstrated that the cause of the failure was pollution of some pellets by hydrogenous compounds. Measures have been taken in the fabrication plant to prevent repeated occurrences of pollution [5.17].

Fuel defects from primary hydriding in Point Lepreau (CANDU) in 1991 and 1992 were due to the insufficient curing of CANLUB graphite coatings. At least 20 defective bundles, mostly from high power positions, were discharged from the core. The amount of gas had exceeded the technical specification of 1 mg. The defective elements failed at linear powers exceeding 50 kW/m [5.105]. After the CANLUB curing procedure was corrected, there were no more fuel defects.

Wolsong Unit 1, a CANDU 6 reactor designed by AECL and located in the Republic of Korea, also experienced a number of fuel failures between September 1995 and August 1996 [5.106]. In order to identify the root cause of the failures, a review of fuel manufacturing processes was jointly performed by KAERI and AECL, and some fuel elements were selected for destructive examinations. These investigations concluded that the root cause of failure was — similar to the experience at Point Lepreau — hydrogen content within the fuel element resulting from insufficient baking of CANLUB graphite coatings. After the manufacturing process was improved, the total amount of hydrogen within the fuel element remained below 0.6 mg and no new failure has occurred.

5.8.2. Failure mechanisms

Published work on failure mechanisms has been reviewed by Garzarolli [5.1], Pickman [5.107] and El-Adham [5.2]. Besides moisture, contamination with organic material is also a source of hydrogen because of its decomposition by radiolysis. Oxygen reacts with pellet and cladding, and releases hydrogen within the cladding. As long as hydrogen pickup is uniform, no significant consequences arise. However, if hydrogen pickup is locally enhanced, massive hydriding forms. As the density of hydride is much lower than that of zircaloy, the cladding swells locally and a stress field is formed in the cladding. Hydride precipitates perpendicular to the stress, and these precipitates appear in the shape of a sunburst (example in Fig. 5.46). This massive localized hydriding leads to hydride blisters, and eventually to local breakthrough of the cladding owing to the brittle nature of the hydrides. The breakthrough is accelerated by hydrogen migration down the temperature gradient [5.107].

The main source of hydrogen in fuel rods is residual moisture in pellets. The water is mainly adsorbed in the open pores during wet grinding at the time of fabrication. The amount of residual moisture in the loaded pellets depends on the drying and handling procedures, and the amount and shape of the open porosity [5.1]. Desorption of water by drying is very effective. Drying of pellets and cladding tubes together with fuel rod internals is standard procedure in fuel fabrication. The standard pellet density nowadays is 95% TD, and in some BWRs, pellet density has been increased up to 96.5% TD in order to reduce fission gas release at high burnup [5.108]. High density pellets have less open porosity, which helps to reduce residual moisture.

Concerning hydrogen pickup in the cladding, a substantial number of out of pile tests have been performed on zircaloy specimens in H_2/H_2O atmospheres. Results indicate the existence of critical ratios of hydrogen to water vapour pressure, above which massive hydriding occurs. Table 5.2 shows critical values obtained by several researchers [5.109, pp. 107–109]. A similar critical level of hydrogen absorption by zircaloy 4 cladding under simulated LOCA conditions was observed in Ref. [5.110].

Massive hydriding can start when the concentration of H_2O oxidant needed to continuously repair the protective oxide film falls below critical values. Since depassivation of the surface becomes inhomogeneous under conditions close to this critical value, locally enhanced hydrogen pickup will frequently occur.

Two mechanisms are postulated to explain how zirconium oxide films permit hydrogen penetration [5.111]. First, mechanical defects such as microcracks, pores, intermetallic particle sites, as well as subgrain boundary and dislocation networks are proposed as short circuit paths in the oxide corrosion film through which hydrogen can come into contact with the zirconium substrate. The second mechanism relates hydrogen solubility and diffusivity through the oxide film to the presence of substoichiometry, which permits the more rapid movement of hydrogen through oxygen vacancies. Through this mechanism, very thin films or those formed under reducing conditions are substoichiometric and, hence, hydrogen permeable. Actually, both mechanisms operate simultaneously. Occasionally, local oxide patches with accelerated hydriding have been observed opposite radial pellet cracks. It was postulated that these high hydrogen/oxide patches are the result of local attacks on the normally protective oxide film from aggressive fission products such as caesium and iodine.

It is common practice to limit equivalent moisture content in pellets to 10 ppm or to 2 mg H_2O per cubic centimetre free cold volume for LWR fuel [5.1]. These limits are similar to the 1 mg H_2 limit in a CANDU fuel element [5.83]. LWR limits, especially the 2 mg H_2O limit, are primarily based on Joon's analyses [5.112]. Joon reviewed reported moisture values and in-pile hydride failures from HBWR and other sources and tried to correlate failure limits to various moisture concentration parameters. His analyses showed that the parameter 'mg H_2O/cm^3 cold void' provided the best approximation to the failure threshold, and that failures were observed only above 2 mg H_2O/cm^3 cold void. The acceptable moisture level of the UO_2 pellet indicated in the Standard Review Plan (SRP) of the United States Nuclear Regulatory Commission is 20 ppm, which is equivalent to 2 ppm of hydrogen as stated by the American Society for Testing and Materials [5.2, 5.113, 5.114].

5.9. DELAYED HYDRIDE CRACKING (DHC)

5.9.1. Introduction

Recently, some results from ramp testing of high burnup BWR fuel rod segments showed that long through-wall cracks identified as DHC failures can initiate at the outer surface of the cladding and propagate in an axial/radial direction; so-called outside-in failures [5.115–5.118]. These seem to be primary failures in the sense that they occur in rods that are intact prior to ramp testing. None of this type of failure has been reported in commercial power reactors.

TABLE 5.2. REVIEW OF THE CRITICAL PARTIAL PRESSURE RATIO OF HYDROGEN TO VAPOUR FOR CATASTROPHIC HYDRIDING [5.109]

Reference	Material	Temperature (°C)	Critical β (p_{H_2}/p_{H_2O})
Une	zircaloy-2	300	10^5 , $p_{H_2} = 1$ atm
Une	zircaloy-2	400	10^2 , $p_{H_2} = 1$ atm
Shannon	zircaloy-2	400	10^2 , $p_{H_2} = 1.3 \times 10^{-2}$ atm
Gibby	zircaloy-2	320	10^5 , $p_{H_2} = 1$ atm
Boyle and Kisiel	zircaloy-2	343	10^6 – 10^8 , $p_{H_2} = 6.5 \times 10^{-2}$ –1 atm



FIG. 5.47. Photo of crack (dark region) which started from the outside after the power ramp of a high burnup rod, (20X).

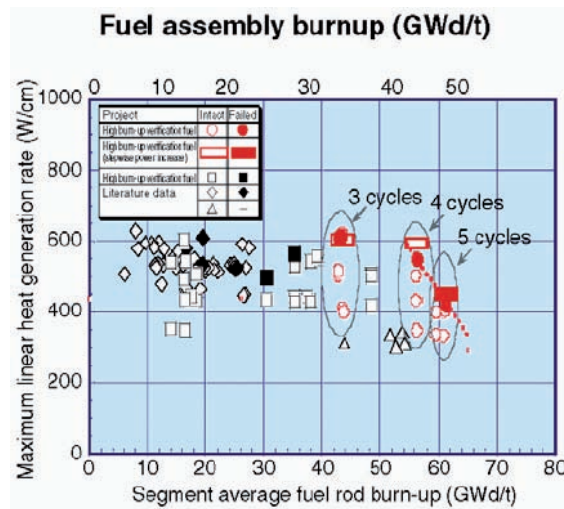


FIG. 5.48. Power ramp test results.

Hypothesis that the process was initiated by cracking of radial hydrides formed at the cladding outer rim, followed by crack propagation through step by step cracking of hydrides around a crack tip has been deduced from metallographic and fractographic examinations of failed rods. [5.119]. Alternatively, there have been no recent reports of fuel failure caused by a power ramp test on PWR fuel rods at burnups higher than 50 GW·d/t [5.120].

5.9.2. Observations

During ramping of two segments of an 8×8 BWR rod in the Halden reactor, an axial crack extending the whole length of the rod was observed on one segment. The rod had been irradiated in Ringhals 1 from 1980–1992 to a peak local burnup of approximately 68 MWd/kg, at a very low linear heat rating during the final cycle (approximately 4–5 kW/m). The rod was then examined in Studsvik [5.121–5.124]. The axial cracks propagated both inward (radially) and in both longitudinal directions from the initiation sites. Figure 5.47 shows how a thumbnail shaped crack started from the outside of the cladding. It was observed that several cracks had started along the same scratch, raising stress along the cladding. In SEM microscopy, the fracture surface had the same appearance as that of axial splits caused in BWR reactors after secondary failures, as shown in Section 7.3.

Similar observations were made during power ramp tests carried out in JMTR using the Boiling Water Capsule under simulated BWR temperature and pressure conditions, on 25 segment rods with burnup ranging from 43 to 61 GWd/t, irradiated in a Japanese commercial BWR. Ramp test results showed a trend to decrease abruptly in failure power level over burnup of 55 GWd/t, as shown in the Fig. 5.48 summary of the published data [5.117–5.122].

Five segment rods failed during power ramp tests. The failed segment rods can be seen in Fig. 5.49. One segment rod, irradiated for three cycles (43 GWd/t), failed a single step ramp test after nine minutes at a ramp terminal power (RTP) of 614 W/cm with a pinhole due to PCI/SCC as shown in Fig. 5.49(a). One segment rod, irradiated for four cycles (56 GWd/t), failed a single step ramp test after 149 minutes at 551 W/cm with an outer side axial crack. Out of nine segments irradiated for five cycles (61 GW·d/t), two failed a single step ramp test after 100 and 68 minutes at 421 and 428 W/cm respectively, and another failed a stair ramp test at 446 W/cm; all three had outer side axial cracks.

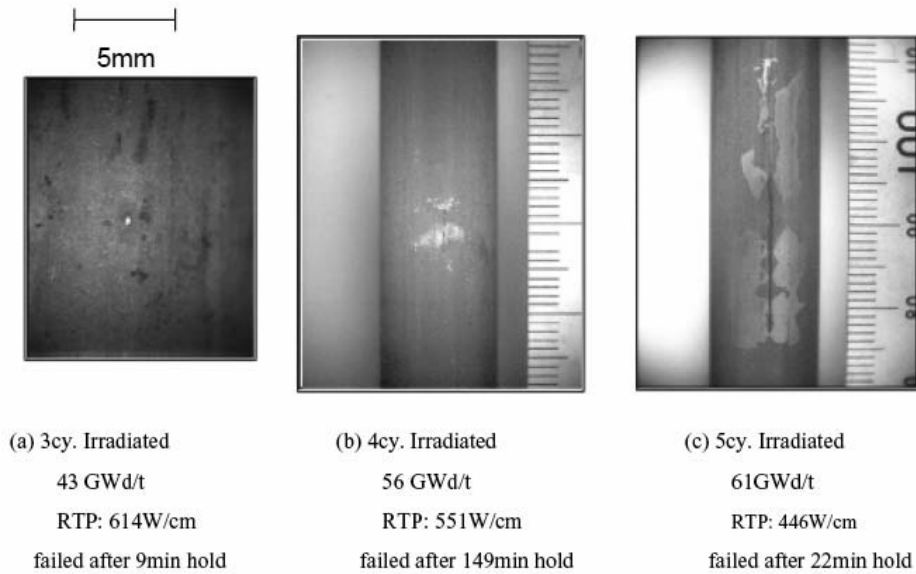


FIG. 5.49. Visual appearance of failed segment rods.

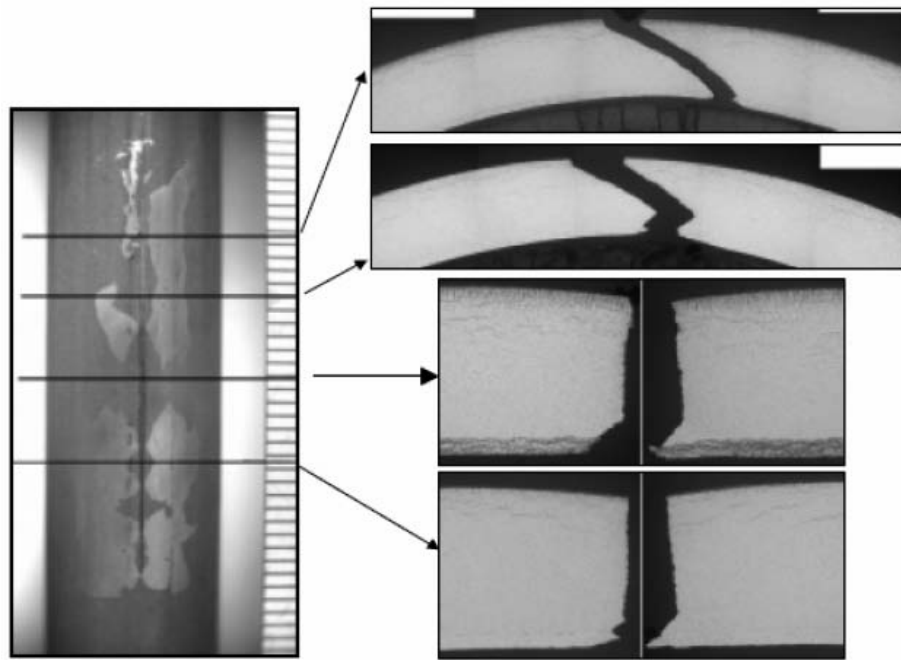


FIG. 5.50. Visual appearance and cross-sectional metallography of a failed segment rod (irradiated five cycles, RTP: 446 W/cm, failed after 22 min hold).

The following characteristics of failed segment rods have been observed through detailed PIEs before and after ramp tests. Figure 5.50 shows cross-sectional micrographs at four elevations of the segment rod through wall crack irradiated for five cycles. The direction of the crack near the centre of the axial crack is perpendicular to the outer surface of the cladding tube and the failure mode is strongly brittle. At the upper part of the crack, there is a small brittle portion in the outer rim of the cladding tube and a large ductile area in the middle part [5.120]. Radial hydrides are observed at the outer rim of the cladding tubes, having a length of about 70 micrometres. These radial hydrides are observed on all rods failing ramp tests after four and five cycle irradiations; the length of the hydrides depends on ramp terminal power (RTP). No radial hydrides were found in the rods before ramp tests [5.120].

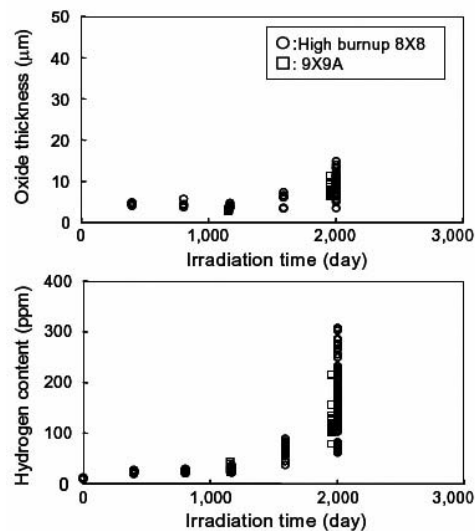


FIG. 5.51. Oxide thickness and hydrogen content of BWR fuel claddings as a function of irradiation time.

The splits were visually similar to the cracks observed with fuel degradation in BWRs, as shown in Section 7.3, except for their length; the split length on the failed segment rods was short and ranged from 4–40 mm depending on ramp test conditions. In any case, the visual appearance of the failed segment rods after four and five exposure cycles suggested that the segment rods failed, not due to the usual PCI/SCC, but because of a different mechanism.

5.9.3. Failure mechanisms

An important research programme was initiated to understand the failure mechanisms. Within the SCIP (Studsvik cladding integrity programme) organized by Studsvik as part of an OECD/NEA project in 2005, a testing technique was developed in order to study DHC as a *primary* failure starting from the cladding outer surface of an initially intact rod. To evaluate the technique, mechanical tests have been performed on medium to high burnup BWR and PWR rods (40–75 MW·d/kgU) with dense hydride formation at the outer surface. The first results of the modified ring tensile testing are in good agreement with previous results from ramp testing and thereby indicate that the testing technique can be used to predict failure mechanisms induced during ramp testing. The objectives of the studies were: (1) to simulate incipient crack formation in a dense hydride rim at the outer surface of irradiated cladding; (2) to study under what conditions and by which mechanism(s) such incipient cracks may propagate into the cladding and cause through-wall failure, and; (3) to compare outside-in cracks formed during mechanical testing with those formed during ramp testing.

The Japan Nuclear Energy Safety Organization (JNES) has also been carrying out an important research programme since 2003 to clarify the axial crack mechanism, taking each of the processes in the hypothesized mechanism into consideration [5.125].

The main conclusions of these programmes can be summarized as follows:

Hydrogen pick-up

To understand hydrogen pick-up behaviours, which have been reported to accelerate at high burnup [5.117, 5.126, 5.127] as illustrated in Fig. 5.51, microstructures of the cladding outer surface oxide layer were examined using a transmission electron microscope (TEM). The most noticeable result of TEM observations on cladding oxide films was seen in the density of the layer rather than in crystal structures. Post-transition oxide film of unirradiated cladding showed a large number of microscopic pores in the vicinity of the oxide–metal interface and small cracks in the middle in contrast to a dense oxide layer in a pre-transition oxide film. Those on a high burnup cladding tube also exhibited increased porosity similar to the post-transition oxide of unirradiated material. Formation of a porous oxide film after transition may lead to enhancement of both cladding oxidation and hydrogen pick-up fraction resulting in accelerated hydrogen pick-up.



FIG. 5.52. Incipient crack formation through an EDC crack propagation test with no holding period.

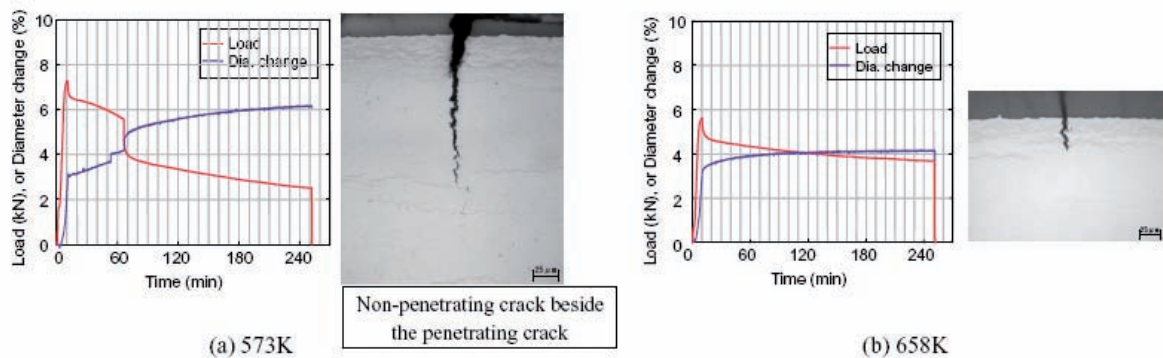


FIG. 5.53. Typical cracks after an EDC crack propagation test with a holding time of four hours.

Hydride redistribution

A hydride redistribution test was conducted under a linear heat flux rate of 300W/cm and with a circumferential stress of 200 MPa for 4h. It showed that hydrogen initially concentrated in the inner Zr liner layer diffused to the outer zircaloy matrix during the test, but no significant hydride reorientation was observed up to 300 W/cm of LHR. Tests with an LHR greater than 400 W/cm are necessary to simulate behaviour during ramp testing.

Temperature dependency of DHC

The crack propagation behaviour of a PWR cladding tube was examined using an expanding mandrel test (EDC test) at temperatures of 573 K and 658 K, with consideration for the differences in BWR and PWR conditions. In order to confirm incipient crack formation, a test specimen was expanded up to about 2% circumferential strain and subjected to metallographic observation. Results are shown in Fig. 5.52. Several incipient cracks were seen in the region of concentrated hydrides at the cladding outer rim on both specimens tested at 573 K and 658 K.

Propagation behaviour of such incipient cracks was examined by holding the expanding device (i.e. keeping the mandrel push rod location at a specified point) for four hours after applying the loading — up to about 2% of circumferential strain. Results are shown in Fig. 5.53. A load curve for the expanding device (push rod) vs. time at 573 K showed a stepwise decrease after about 60 minutes during the holding time, suggesting cracking of a specimen. Metallography after holding for four hours revealed a penetrating crack and several non-penetrating cracks which were deeper than the incipient one. Alternatively, no abnormal change in the load–time curve was seen in the test at 658 K. Crack depth observed by metallography after four hours holding at 658 K was comparable to that for incipient cracks, showing no indication of propagation.

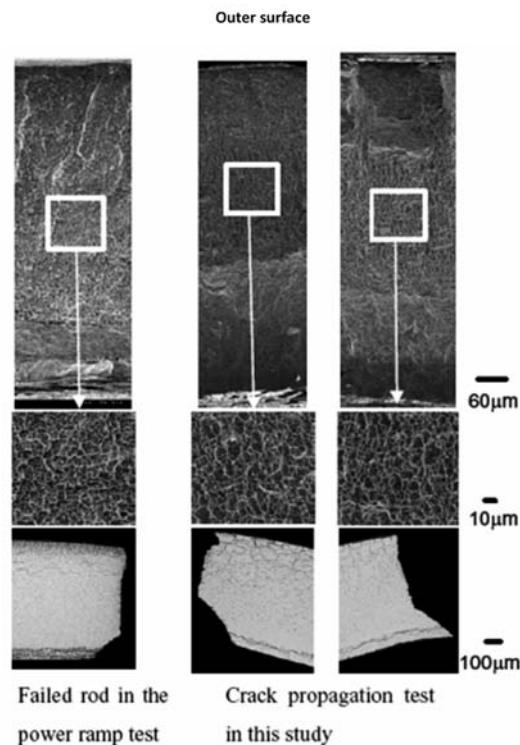


FIG. 5.54. Fractographs of specimens fractured in a brittle manner by crack propagation test.

Test results at 573 K suggested delayed crack propagation after the formation of incipient cracks. Non-propagation at 658 K could be attributed to low stress intensity due to enhanced ductility at the crack tip compared to conditions at 573 K. The above results may describe power ramp test results for PWR fuel rods which exhibited incipient cracks but showed no indication of propagation [5.120].

A mechanical test on irradiated PWR fuel cladding showed incipient crack formation even with very little strain to a limited depth at a highly concentrated hydride region at the cladding outer rim. A crack propagation test using the expanding mandrel method caused a cladding fracture at 573 K with a time delay of about 60 minutes but no indication of crack growth at 658 K, demonstrating the effect of temperature on susceptibility to hydride induced crack propagation. The results support the idea that there is an upper temperature limit above which DHC will not occur in zirconium cladding.

Crack propagation behaviour

A crack propagation test was carried out through a stepwise increase in internal pressurization with holding intervals of one hour. Some specimens exhibited a similar fracture surface to that observed on fuel cladding failed during ramp testing: a brittle surface at the outer rim followed by a mixture of brittle and ductile surfaces in the middle. [5.119]

Figure 5.54 compares fracture surfaces in the middle portion of the thickness between cladding specimens fractured in stepwise pressurization and power ramp tests. Fractographic observations suggest a crack initiation and propagation processes. The brittle surface at the outer rim may be caused by a fracture of radial hydride precipitated at the outer rim. Hydride precipitates appear in a network of circumferential and radial orientations around the cladding outer region. Although the stresses which initiated outer surface cracking were comparable to hoop strength measured in the internal pressurizing burst test, they might potentially be lowered by more concentrated radial hydride precipitates at the outer rim. A mixture of ductile and brittle surfaces can be formed during crack propagation through repetition of small hydride precipitation at the crack tip and its fracture. Since the crack propagation tests were conducted in the atmospheric environment, the hydrogen required to form hydrides at the crack tip should be supplied by stress induced diffusion from circumferences of the crack tip. The roughly estimated stress intensity factor and crack growth length were seen to be equivalent to those for delayed hydride cracking (DHC) [5.128, 5.129].

Since the classical DHC model requires a stress concentrator to initiate the fracture mechanism [5.130, 5.126], there has been speculation as to where the DHC failures start in these rods. Destructive examinations of failed rods suggest that outside-in DHC failures initiate from radial hydrides [5.119] or dense hydride rims formed at the outer surface. If incipient cracks form in either radial hydrides or dense hydride rims, they can serve as stress concentrators, to which hydrogen in solid solution will diffuse, resulting in a local increase in hydrogen concentration in front of the crack tip — required to initiate DHC. Since the stress field around the crack tip will depend on crack geometry, it is important to induce cracks with geometry similar to that formed in a reactor when simulating in-pile DHC failures.

Outside-in cracking model

When optical microscopic and fractographic observations are comprehensively taken into account, a possible crack initiation and propagation process could schematically be summarized as in Fig. 5.55. The temperature gradient and stress produced by power ramp tests causes radial hydrides to concentrate at the outside cladding surface. Then hydride fracture due to strain (or stress) by PCMI triggers outside-in cracking. Since hydride platelets fracture mechanically at the initiation process, the morphology of the fracture surface is most brittle. In the crack propagation process, hydrides precipitate at the crack front and then grow to a certain length to crack at the tip. By repeating the same process, the crack advances and finally, when the residual wall thickness becomes thin enough to be torn mechanically, the cladding shears with a ductile process.

Open issues [5.119]

Although hydrides in the cladding tube are responsible for crack initiation, an evaluation of hydrogen content distribution in the cladding tube suggests it would be difficult for hydrogen in the tube to be responsible for hydride precipitation at the crack tip in most of the propagation region. A hydrogen content profile along the tube wall during a ramp test was analysed using PIE results. The analysis revealed the hydrogen content was too low for the hydrogen in cladding to precipitate in most of the propagation region at a high temperature, even when hydrogen solubility — affected by stress concentration at the crack tip — is taken into account. Thus, another mechanism is required to explain hydride precipitation at the crack front. One possible idea is the absorption of hydrogen at the fracture surface, especially at the crack front surface. When radial hydrides fracture, coolant goes into the crack and the diffusing steam begins to react with fracture surfaces to create hydrogen.

When the gas mixture becomes sufficiently enriched in hydrogen, absorption will occur through breaking down of the protective oxide on fracture surfaces. A simple calculation reveals that only a few nanometers of oxide thickness is enough to change the steam completely into hydrogen gas, because the crack width is very narrow, about 10 μm or less. This indicates that the crack would be filled with hydrogen in a comparatively short time after steam invasion. Since the steam changes to the same number of moles of hydrogen gas, the crack would be filled with almost pure hydrogen of about 7 MPa, the coolant pressure of the ramp test. After the steam changes to hydrogen, since the crack is very narrow and deep, hardly any new coolant steam enters the crack space to degrade the hydrogen purity. Thus, once a crack is filled with hydrogen, it propagates through a hydrogen gas cracking (HGC) mechanism [5.131–5.135]. When some amount of hydrogen is absorbed through the fracture surface and new steam flows into the crack, since the crack tip is located at the bottom, hardly any new steam reaches the bottom and an almost pure hydrogen atmosphere is maintained there. Furthermore, the hydride fracture at the crack front exposes a new metal surface at the front, which promotes hydrogen absorption there, leading to hydride precipitation and again, hydride cracking. Through repetition of hydrogen absorption, precipitation and cracking, the crack advances further towards the inside and finally, when only a thin residual wall remains, it penetrates the cladding in the ductile fracture.

In conclusion, careful examination of segment rods which failed ramp tests after four and five cycles of base irradiation show that hydride cracking is the responsible failure mechanism. For crack initiation, hydrogen behaviour caused by power ramp testing is essential. The temperature gradient along the cladding wall and stress from a ramp test precipitate radial hydrides on the tube outside.

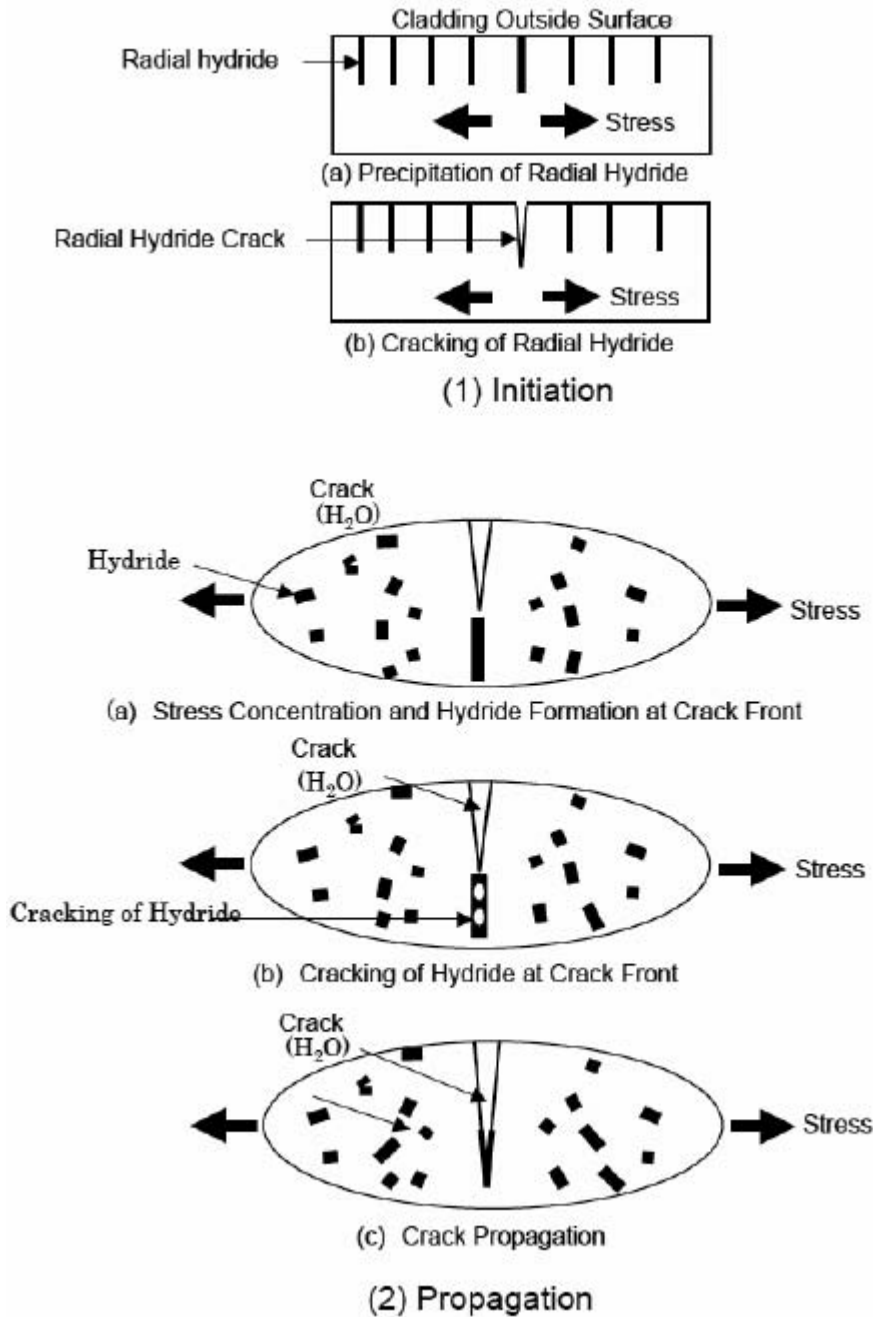


FIG. 5.55. Schematic diagram of crack initiation and propagation.

One more issue is hydride fracture due to hoop strain or stress by PCMI, namely, the initiation of outside-in cracking. Since hydrogen originally included in the cladding plays a main role, the mechanism is somewhat similar to that of secondary degradation failure, or delayed hydride cracking (DHC). However, in the crack propagation process, it has been deduced that, although hydride cracking at the crack front is fundamental to the process, hydrogen absorbed at the crack tip might be a mechanism of outside-in cracking observed in power ramp tests, which would be different from that of the usual secondary degradation mechanism in which hydrogen in cladding tubes is responsible for crack propagation.

Ramp test results suggest that hydride orientation plays a critical role in fuel rod behaviour as well as hydrogen content. In order to assess fuel rod behaviour at high burnup, better understanding of the dissolution, diffusion and local precipitation of hydrogen in cladding tubes in operation is required.

REFERENCES TO SECTION 5

- [5.1] GARZAROLLI, F., VON JAN, R., STEHLE, H., The main causes of fuel element failure in water-cooled power reactors, *At. Energy Rev.* **17 1** (1979) 31–128.
- [5.2] EL-ADHAM, K., Fuel failure mechanisms in operating US plants from 1981 to 1986, *J. Nucl. Safety* **29 4** (1988) 487.
- [5.3] *Nucleonics Week*, **30 42** (1989) 1–11.
- [5.4] Nuclear Power Experience, Vol. BWR-2, I. Fuel (1990) 24–25.
- [5.5] JONSSON, A., SUNDSTROM, U., HALLSTRADIUS, L., “In-reactor mechanical performance of BWR fuel channels”, *Proc. Int. Top. Mtg on LWR Fuel Performance*, Avignon, 1991, Vol. 1, SFEN, Paris (1991) 184.
- [5.6] CHAPOT, J., FREIRE, J., Tracing fuel failures at Angra 1, *J. Nucl. Eng. Int. Sept.* (1994) 32.
- [5.7] CHAPOT, J.L.C., SUANO, R., de COUTO, N., “Fuel failures at ANGRA 1 : Cause and mitigation”, IAEA-TECDOC-1345, IAEA, Vienna (2003).
- [5.8] PROVOST, J.L., DEBES, M., “EDF PWR Fuel operating experience and high burnup performances”, *TopFuel 2006*, Salamanca, 2006 ENS, Brussels (2006).
- [5.9] Nuclear Power Experience, Vol. PWR-2, I. Fuel (1994) 79.
- [5.10] Nuclear Fuel, Vol. 5 McGraw-Hill, New York (1993) 3.
- [5.11] Nuclear Fuel, McGraw-Hill, New York (1994) 3.
- [5.12] WILSON, H.W. et al., “Westinghouse fuel performance in today’s aggressive plant operating environment”, *Int. Topical Mtg on Light Water Reactor Fuel Performance*, Portland, 1997, ANS, Le Grange, IL (1997).
- [5.13] Nuclear Power Experience, Vol. PWR-2, I. Fuel (1993) 75, 79.
- [5.14] BAILLON, N., “Grid to rod fretting wear in EDF PWR — From operating problems to new design qualification method”, IAEA-TECDOC-1454, IAEA, Vienna (2005).
- [5.15] WOODS, K.N., KLINGER, W., “Siemens fuel performance overview”, *Proc. Int. Topical Mtg on LWR Fuel Performance*, Portland, 1997.
- [5.16] BILLEREY, A., “Evolution of the fuel rod support under irradiation and its impacts on the mechanical behaviour of the fuel assemblies”, IAEA-TECDOC-1454, IAEA, Vienna (2005).
- [5.17] KLINGER, W., PETIT, C., WILLSE, J., “Experience and reliability of FRAMATOME ANP’S PWR and BWR fuel”, IAEA-TECDOC-1345, IAEA, Vienna (2003).
- [5.18] BLANC, C., BOURNAY, P., DANGOULEME, D., “FRAGEMA fuel reliability”, *Proc. Int. Top. Mtg on LWR Fuel Performance*, Avignon, 1991, Vol. 1, SFEN, Paris (1991) 353.
- [5.19] KNOTT, R. P., KETERSON, R. L., HALLSTADIUS, L. G., YOUNG, M. Y., “Advanced PWR fuel designs for high duty operation”, *Int. Conf. TopFuel 2003*, Würzburg, 2003.
- [5.20] SCOTT, D., et al., “Post-irradiation examination of the lead westinghouse robust fuel assemblies after three cycles of operation in the Wolf Creek generating station”, *ibid.*
- [5.21] VALLORY, J., “Methodology of PWR fuel rod vibration and fretting evaluation in HERMES Facilities”, IAEA-TECDOC-1454, IAEA, Vienna (2005).
- [5.22] GOTTUSO, D., CANAT, J.N., MOLLARD, P., “A family of upgraded fuel assemblies for PWR”, *Int. Conf. TopFuel 2006*, Salamanca, 2006.
- [5.23] NAKAJIMA, I., TESHIMA, H., YAMADA, M., “Improvement and innovation of Mitsubishi PWR Fuel”, *Int. Conf., TopFuel 2005*, Kyoto, 2005.
- [5.24] KENNARD, M.W., “Nuclear fuel performance, trends, remedies and challenges”, *Int. Conf. TopFuel 2006*, Salamanca, 2006.
- [5.25] STRASSER, A., GINGOLD, J., “Evaluation of debris fretting failures and preventive methods”, *Proc. Tech. Comm. Mtg on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management*, Dimitrovgrad, 1992, IAEA-TECDOC-709, IAEA, Vienna (1993) 133.
- [5.26] CARTER, J., MANZER, A.M., “Overview of defect mechanisms in CANDU fuel”, *ibid.*, p. 121.
- [5.27] VELUKHANOV, V.P., et al., “Features of operating a VVER reactor core containing leaking fuel rods”, *ibid.*, p. 152.
- [5.28] BAILY, W.E., et al., “Recent GE BWR fuel experience”, *Proc. Int. Top. Mtg on LWR Fuel Performance*, Avignon, 1991, Vol. 1, SFEN, Paris (1991) 26.
- [5.29] WILSON, H.W., MILLER, H., KUNISHI, H., “Westinghouse fuel performance experience”, *Proc. Tech. Comm. Mtg on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management*, Dimitrovgrad, 1992, IAEA-TECDOC-709, IAEA, Vienna (1993) 138.
- [5.30] RYTTERSSON, K., HELMERSSON, S., WRIGHT, J., HALLSTADIUS, L., “Westinghouse BWR fuel reliability — Recent experience and analyses”, *Int. Conf. on LWR Fuel Performance*, TopFuel 2007, San Francisco, 2007, Paper 1035.
- [5.31] SCHARDT, J., “GNF 10X10 Fuel pursuing performance and reliability”, *Int. Conf. TopFuel 2006*, Salamanca, 2006.
- [5.32] SODERLUND, A., et al., “Mechanical design of the triplewave debris filter”, IAEA-TECDOC-1345, IAEA, Vienna (2003).
- [5.33] HOLZER, R., KNAAB, H., “Recent fuel performance experience and implementation of improved products”, *Proc. Int. Top. Mtg on LWR Fuel Performance*, Williamsburg, 1988, ANS, LaGrange Park, IL (1988) 69.
- [5.34] GARZAROLLI, F., et al., “Progress in understanding PWR fuel rod water side corrosion”, *Proc. Top. Mtg on Light Water Reactor Fuel Performance*, Orlando, 1985, Vol. 1, ANS, LaGrange Park, IL (1985) 3.
- [5.35] DANGOULEME, D., private communication.

- [5.36] DERAMAIX, P., BAIRIOT, H., BILLAUX, M., LIPPENS, M., "MIMAS fuel performance in commercial reactors", Proc. Int. Top. Mtg on LWR Fuel Performance, Avignon, 1991, Vol. 1, SFEN, Paris (1991) 94.
- [5.37] DUBROVIN, K.P., et al., "Results of post-irradiation examination of fuel assemblies in the vicinity of the control rod assemblies of WWER-440 reactor", Proc. 3rd Conf. on Radiation Metallurgy, Dimitrovgrad, 1992, SRIAR, Dimitrovgrad (1994) 89 (in Russian).
- [5.38] SOHN, D.-S., "Status and development: Plan in Korea", 2007 Int. Conf. on LWR Fuel Performance Meeting, TopFuel, San Francisco, 2007.
- [5.39] MARLOWE, M.O., et al., "Nuclear fuel cladding localized corrosion", Proc. Top. Mtg on Light Water Reactor Fuel Performance, Orlando, 1985, Vol. 1, ANS, LaGrange Park, IL (1985) 3.
- [5.40] LILL, G., KNAAB, H., URBAN, P., "Advanced BWR fuel designs: Incentives, implementation, performance verification", *ibid*, p. 74.
- [5.41] OGATA, K., MITSUOKA, H., YAMANAKA, T., NAKAMURA, T., "Post-irradiation examination on the failed rod in the Hamaoka atomic power station unit 1", Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, 1994, ANS, LaGrange Park, IL (1994) 510.
- [5.42] YANG, R.L., OZER, O., KLEMPFER, O.H., "Fuel performance evaluation for EPRI program planning", Proc. Int. Top. Mtg on LWR Fuel Performance, Avignon, 1991, Vol. 1, SFEN, Paris (1991).
- [5.43] RUZAUSKAS, E.J., SMITH, D.L., "Fuel failures during cycle 11 at River Bend", Int. Mtg on LWR Fuel Performance, Orlando, 2004.
- [5.44] KEYS, T.A., LEMONS, J. F., OTTENFELD, C., "Fuel corrosion failures in the Browns Ferry Nuclear Plant", Int. Mtg on LWR Fuel Performance, Orlando, 2004.
- [5.45] Nuclear Fuel, October 9 1995, McGraw-Hill, New York, NY (1995) 5.
- [5.46] TROPASSO, R., et al., "Crud-induced cladding corrosion failures in TMI-1 Cycle 10", Int. Mtg on LWR Fuel Performance, Orlando, 2004.
- [5.47] YANG, R., CHENG, B., DESHON, J., EDSINGER, K., OZER, O., "Fuel R & D to improve fuel reliability", Int. Conf. TopFuel, Kyoto, 2005.
- [5.48] HYASHI, Y., MATSUMOTO, T., KUZUSHIMA, M., MURAMOTO, T., OKUBO, T., "Fuel failure at Hamaoka Unit 1", Proc. Tech. Comm. Mtg on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management, Dimitrovgrad, 1992, IAEA-TECDOC-709, IAEA, Vienna (1993) 128.
- [5.49] SMIRNOV, A.V., et al., "WWER-1000 and WWER-440 fuel operation experience", Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, 1994, ANS, LaGrange Park, (1994) 31.
- [5.50] BIBILASHVILI, Yu., et al., "Operation experience of WWER fuel, including analysis of abnormal condition", Proc. 2nd Int. Sem. on WWER Reactor Fuel Performance, Modelling and Experimental Status, Sandanski, 1997, BAS, INRNE, Sofia (1997) 11.
- [5.51] TERÄSVIRTA, R., HANSSON-LYYRA, L., "Exceptional crud build-up in Loviisa-2 fuel bundles", OECD Specialists' Meeting.
- [5.52] NEMES, I., "Hydraulic anomaly and its management on Paks NPP WWER-440 units in 2002-2003", Proc 4th Int. Conf on WWER Reactor Fuel Performance, Modelling and Experimental Support", Albena, 2003, BAS, INRNE, Sofia (2004) 77.
- [5.53] Nucleonics Week, July 17 (1997) 6.
- [5.54] ZWICKY, H., et al., "Enhanced spacer shadow corrosion on SVER fuel assemblies in the Leibstadt nuclear power plant", ANS Fuel Topical Mtg, Salt Lake City, 2000.
- [5.55] ULLBERG, M., LYSELL, G., NYSTRAND, A. C., "Shadow corrosion mechanism of zircaloy", SKI report 2004:28 (2004).
- [5.56] Nucleonics Week, September 3 (1998) 9.
- [5.57] COX, B.J., Pellet-clad interaction (PCI) failures of zirconium alloy fuel cladding: A review, J. Nucl. Mater. **172** (1990) 249.
- [5.58] TSUKUDA, Y., et al., "Performance of Advanced Fuel Materials for High Burnup", Int. Conf. TopFuel, Würzburg, 2003.
- [5.59] TAYAL, M., CHASSIE, G.G., "Extrapolating power-ramp performance criteria for current and advanced CANDU fuels", 21st Annual Conference, Toronto, 2000, CNS,.
- [5.60] CAMPBELL, F.R., DESHAIES, R., NOTLEY, M.J.F. "Transient fission gas release within UO₂ fuel elements following power increases", Atomic Energy of Canada Limited Report, AECL-4912 (1974).
- [5.61] PEEHS, M., STEHLE, H., STEINBERG, E., "Out-of-pile testing of iodine stress corrosion cracking in zircaloy tubing in relation to the pellet-cladding interaction phenomenon", 4th Conf. on Zirconium in Nuclear Industry, ASTM STP 681 (1979) 244-260.
- [5.62] TAYAL, M., et al., "INTEGRITY: A semi-mechanistic model for stress corrosion cracking of fuel", IAEA Tech. Comm. Mtg on Water Reactor Fuel Element Modelling at High Burnup and Its Experimental Support, Windermere, 1994.
- [5.63] ROBERTS, J.T.A., et al., "A stress corrosion cracking model for pellet-cladding interaction failures in light-water reactor fuel rods", ASTM-SPT-681, American Society for Testing and Materials, Philadelphia, PA (1979).
- [5.64] ICHIKAWA, M., et al., "Studies of LWR fuel performance under power ramping and power cycling utilizing in-pile measurement and fuel modelling", Proc. Int. Top. Mtg on LWR Fuel Performance, Williamsburg, 1988, ANS, LaGrange Park, IL (1988) 253.
- [5.65] NUCLEAR POWER EXPERIENCE, Vol. PWR-2, I, Stoller, Boulder, CO (1982).
- [5.66] SIEMENS, Siemens Nuclear Fuel Report 3 (1993) 35.

- [5.67] POWERS C., et al., "Hot cell examination results of non-classical PCI failures at La Salle", Int. Conf. TopFuel, Kyoto, 2005.
- [5.68] HARDY, D.G., BAIN, A.S., Proc. 3rd Int. Conf. on Zr in Nuclear Industry, Quebec City, 1976, ASTM-STP 633, ASTM, Philadelphia, PA (1977) 98.
- [5.69] SCHUSTER, I., LEMAIGNAN, C., Influence of texture on iodine-induced stress corrosion cracking of Zircaloy-4 cladding tubes, J. Nucl. Mater. **189** (1992) 157.
- [5.70] CHARLES, M., LEMAIGNAN, C., Fuel performance under normal PWR conditions: A review of relevant experimental results and models, J. Nucl. Mater. **188** (1992) 96.
- [5.71] CUBICCIOTTI, D., HOWARD, S.M., JONES, R.L., The formation of iodine-induced stress corrosion cracks in Zircalloys, J. Nucl. Mater. **78** (1978) 2.
- [5.72] HOFFMAN, P., et al., Chemical interactions between simulated fission products and Zircaloy-4 at elevated temperatures including influence on ductility and time to failure, J. Nucl. Mater. **102** (1981) 117.
- [5.73] SHIMADA, S., MATUURA, T., NAGAI, M., Stress corrosion cracking of Zircaloy-2 by metal iodides, J. Nucl. Sci. Technol. **20**, 7 (1983) 593.
- [5.74] KENOSHA, K., AMEX, M., Evaluation of iodine pressure in oxide fuel pins under irradiation, J. Nucl. Sci. Technol. **29** (1992) 1.
- [5.75] THOMAS, G., The Studsvik inter ramp project: An international power ramp experimental programme, J. Nucl. Mater. **87** (1979) 215.
- [5.76] MOGARD, H., et al., "International TRANS-RAMP I fuel project", Proc. Tech. Comm. Mtg on Fuel Rod Internal Chemistry and Fission Products Behaviour, Karlsruhe, 1985, IAEA-IWGFPT/25, IAEA, Vienna (1986) 157.
- [5.77] FUJISHIRO, T., "NSRR experiments to study the effects of burnup on the fuel behaviour under reactivity initiated accident conditions", paper presented at 22nd WRSIM, Bethesda, 1994.
- [5.78] FLOYD, M.R., CHENIER, R.J., LEACH, D.A., ELDER, R.R., "An overview of the examination of fuel as follow-up to the 1988 November overpower transient in Pickering NGS-A Unit 1", Proc. 3rd Int. Conf. on CANDU Fuel, Chalk River, 1992, Vol. 2, CNS, Toronto (1986) 21.
- [5.79] MANZER, A.M., CHOW, H.C., "Fuel management flexibility with single ended refuelling in CANDU-3", 11th Annual Conf., Toronto, CNS, 1990.
- [5.80] FLOYD, M.R., NOVAK, J., TRUANT, P.T., "Fission gas release in fuel performing to extended burnups in Ontario Hydro nuclear generating stations", Proc. Tech. Comm. Mtg on Fission Gas Release and Fuel Rod Chemistry Related to Extended Burnup, Karlsruhe, 1985, IAEA-IWGFPT/25, IAEA, Vienna (1986) 53.
- [5.81] MANZER, A.M., DENNIER, D., HU, R., YOUNG, E.G., "CANDU 6 reactor shim operation: Fuel performance guidelines", Proc. 12th Annual Conf. Canadian Nuclear Society, Saskatoon, 1991, CNS, Toronto (1991).
- [5.82] JUDAH, J., "Defected fuel location by dry sipping: Experience at Bruce NGS-A during the 1984 fuel defect excursion", Proc. Int. Conf. on CANDU Fuel, Chalk River, 1986, CNS, Toronto (1986) 470.
- [5.83] SEJNOHA, R., MANZER, A.M., SURETTE, B.A. "Hydrogen in CANDU fuel elements", Proc. Int. Conf. on CANDU Fuel, Pembroke, 1995, CNS, Toronto (1996).
- [5.84] SEIBOLD, A., et al., "Fe-enhanced ZR liner cladding", Int. Topical Mtg on Light Water Reactor Fuel Performance, Portland, OR, 1997.
- [5.85] SIHVER, L., et al., "Recent ABB BWR failure experience", ibid.
- [5.86] ARBORELIUS, J., et al., "Advanced doped UO₂ pellets in LWR applications", TopFuel Kyoto, 2005.
- [5.87] DELAFOY, C., DEWES, P., "AREVA NP New UO₂ fuel development and qualification for LWRs applications", TopFuel 2006, Salamanca, 2006.
- [5.88] PENN, W.J., LO, R.K., WOOD, J.C., CANDU fuel-power performance criteria, J. Nucl. Technol. **34** (1977) 249.
- [5.89] BORELL, S., WIDEGREN, H., "Improved fuel performance by enhanced UO₂ pellet fabrication and inspection methods", TopFuel 2006, Salamanca, 2006.
- [5.90] STRASSER, A., SUNDERLAND, D., "A review of recent LWR fuel failures", Proc. Tech. Comm. Mtg on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management, Dimitrovgrad, 1992, IAEA-TECDOC-709, IAEA, Vienna (1993) 17.
- [5.91] MILLER, S., et al., "Westinghouse fuel operation at high burnup and with advanced features", Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, FL, 1994.
- [5.92] AUTORITÉ DE SÛRETÉ NUCLÉAIRE, "Décision no. 2007-DC-0066 de l'Autorité de sûreté nucléaire du 19 juillet 2007 relative à la mise en oeuvre de la gestion du combustible dite", ALCADÉ dans les Réacteurs des Centrales Nucléaires de Chooz B et Civaux, ASN.
- [5.93] TRUANT, P.T., HASTINGS, I.J., CANDU Reactor Experience: Fuel Performance, Rep. AECL-8774, Atomic Energy of Canada Ltd, Chalk River (1985).
- [5.94] POTTS, G.A., PROEBSTLE, R.A., "Recent GE BWR fuel experience", Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, FL, 1994, ANS, LaGrange Park, IL (1994) 87.
- [5.95] WILLSE, J.T., MITCHELL, B., WILLIAMS, P.F., "Recent results from the B&W fuel company's fuel performance improvement program", ibid., p. 45.
- [5.96] MITSUBISHI HEAVY INDUSTRIES, personal communication.

- [5.97] POTTS, G.A., PROEBSTLE, R.A., “Recent GE BWR fuel experience”, Int. Topical Mtg on Light Water Reactor Fuel Performance, West Palm Beach, FL, 1994.
- [5.98] Nuclear Power Experience, Vol. PWR-1, I. Fuel (1980).
- [5.99] REPARAZ, A., ABRAHAM, J.P., A new solution to the baffle jetting problem, Power Eng. Sept. (1984) 46.
- [5.100] SOFER, G.A., HANSEN, L.E., VAN SWAM, L.F.P., PATTERSON, J.F., “Performance of Advanced Nuclear Fuels Corporation fuel in light water reactors”, Proc. Int. Top. Mtg on LWR Fuel Performance, Williamsburg, 1988, ANS, LaGrange Park, IL (1988) 41.
- [5.101] SOFER, G.A., VAN SWAM, L.F.P., EARHOS, C.A., “Performance of EXXON Nuclear Fuel Company fuel in light water reactors”, Proc. Top. Mtg on Light Water Reactor Fuel Performance, Orlando, FL, 1985, Vol. 1, ANS, LaGrange Park, IL (1985) 1.
- [5.102] WILLSE, J.T., COLEMAN, T.A., “Recent results from the B&W fuel company’s fuel performance improvement program”, Proc. Int. Top. Mtg on LWR Fuel Performance, Avignon, 1991, Vol 1, SFEN, Paris (1991) 66.
- [5.103] YANG, T., et al., “Experience on fuel inspection and root cause analysis during outage”, Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, FL, 1994, ANS, LaGrange Park, IL (1994) 531.
- [5.104] KLINGER, W., PETIT, C., WILLSE, J., “Experience and reliability of FRAMATOME ANP’S PWR and BWR fuel”, IAEA-TECDOC-1345, Vienna, IAEA, 2003.
- [5.105] MANZER, A.M., et al., “Fuel defect investigation at Point Lepreau”, Proc 3rd Int. Conf. on CANDU Fuel, Chalk River, 1992, Vol. 2, CNS, Toronto (1986) 30.
- [5.106] PARK, H.T., et al., “Fuel defect root cause investigation at Wolsong-1”, Proc. 5th Int. Conf on CANDU Fuel, Toronto, 1997, CNS, Vol. 1 (1997) 219.
- [5.107] PICKMAN, D.O., Interactions between fuel pins and assembly components, J. Nucl. Eng. Des. **33** (1975) 141.
- [5.108] OGUMA, M., MAKI, H., MATSUMOTO, T., YAMAMOTO, Y., NOMURA, M., Review of BWR fuel performance in Japan, J. Nucl. Energy **31** 1 (1992) 25.
- [5.109] UNE, K., Kinetics of reaction of zirconium alloy with hydrogen, J. Less-Common Met. **57** (1978) 93.
- [5.110] FURURA, T., et al., J. Nucl. Mater. **105** (1982) 119.
- [5.111] CLAYTON, J.C., Internal Hydriding in Irradiated Defected Zircaloy Fuel Rods: A Review, Rep. WAPD-TM-1604, Bettis Atomic Power Lab., Pittsburgh, PA (1987).
- [5.112] JOON, J., Primary hydride failure of Zircaloy-clad fuel rods, Trans. Am. Nucl. Soc. **15** 1 (1972) 186.
- [5.113] NUCLEAR REGULATORY COMMISSION, Fuel System Design, 4.2, Rev. 2, Standard Review Plan for the Review of Safety Analysis Report for Nuclear Power Plants, LWR Edition, Rep. NUREG-0800, USNRC, Washington, DC (1981).
- [5.114] AMERICAN SOCIETY FOR TESTING AND MATERIALS, ASTM 776-76, Philadelphia, PA (1976).
- [5.115] ALVAREZ-HOLSTON, A-M., LYSELL, G., GRIGORIEV, V., “Studies of hydrogen assisted failures initiating at the cladding outer surface of high burnup fuel using a modified ring tensile technique,” Proc. of the 2007 Int. LWR Fuel Performance Mtg, San Francisco, 2007, Paper 1080.
- [5.116] SCHRIRE, D., LYSELL, G., CARLSSON, M., JOSEFSSON, B., “Testing fuel rod integrity at Studsvik”, KTG Conf. on Verification of Fuel Performance through Hot Cell Examination and Experiments, Karlsruhe, 2000.
- [5.117] SAKURAI, H., et al., “Irradiation characteristics of high burnup BWR fuels,” Proc. Int. Topical Mtg on LWR Fuel Performance, Park City, 2000.
- [5.118] HAYASHI, H., et al., “Outside-in failure of high burnup BWR segment rods caused by power ramp tests”, Proc. ENS Top Fuel 2003, Wurzburg, Germany, 2003.
- [5.119] SHIMADA, S., ETOH, E., HAYASHI, H. TUKUTA, Y., “A metallographic and fractographic study of outside-in cracking caused by power ramp tests”, J. Nucl. Mater. **327** (2004) 97–113.
- [5.120] NUCLEAR POWER ENGINEERING CORPORATION, The Annual Report of FY2001 on Irradiation Tests of High Burnup Fuels-Integrated Evaluation on Irradiation Behavior of PWR High Burnup Fuel-, Nuclear Power Engineering Corporation, Tokyo, Japan (2002) (in Japanese).
- [5.121] HAYASHI, H., et al., “Irradiation characteristics of BWR Step II lead use assemblies”, Proc. 1997 ANS Int. Mtg. on LWR Fuel Performance, Portland (1997) 296.
- [5.122] OHARA, H., et al., “Fuel behaviour during power ramp tests”, Proc. 1994 Int. Topical Mtg. on LWR Fuel Performance, West Palm Beach, FL, (1994).
- [5.123] LYSELL, G., GRIGORIEV, V., EFSING, P., “Axial splits in failed BWR fuel rods”, Int. Topical Mtg on Light Water Reactor Fuel Performance, 2000, oral session.
- [5.124] JONSSON, T., “Hot cell examination of a defect rod from IFA-597”, Studsvik Nuclear AB, Published by the Swedish Nuclear Power Inspectorate in SKI Report 99:4.
- [5.125] OGATA, K., et al., “Progress in the research programs to elucidate axial cracking fuel failure at high burnup”, Proc. of the 2007 Int. LWR Fuel Performance Mtg, San Francisco, 2007, Paper 1024.
- [5.126] HIRANO, Y., et al., “Irradiation characteristics of BWR high burnup 9X9 lead use assemblies”, Proc. 2005 Water Reactor Fuel Performance Mtg, Kyoto, 2005.
- [5.127] OGATA, K., et al., “Study on the hydrogen assisted cracking of zircaloy fuel cladding at high burnup”, 15th Int. Symp. on Zirconium in the Nuclear Industry, Sunriver, OR, 2007.

- [5.128] TAKAGI, I., et al., “A simple model for hydrogen re-distribution in Zirconium-lined fuel, claddings”, J. Nucl. Sci. Technol., **39** **1** (2002).
- [5.129] GRIGORIEV, V., JAKOBSSON, R., “Delayed hydrogen cracking velocity and J-integral measurements on irradiated BWR cladding”, ASTM STP 1467 (2005).
- [5.130] DUTTON, R., NUTTALL, K., PULS, M.P., SIMPSON, L.A., “Mechanism of Hydrogen-Induced Delayed Hydride Cracking in Hydride Forming Materials”, Metall. Trans. A, **8A** (1977) 1553–1562.
- [5.131] MYERS, S.M., et al., “Hydrogen interactions with defects in crystalline solids”, Rev. Mod. Phys. **64**, No.2 April (1992) 575–576.
- [5.132] COLEMAN, C. E., COX, B., ASTM STP 824 (1984) 675.
- [5.133] ROWE, R. G., Scr. Mater. 38 (1998) 1495.
- [5.134] ROWE, R. G., Scr. Mater. 40 (1998) 249.
- [5.135] KIM, K-T., SUH, J-M., Impact of nuclear fuel assembly design on grid to rod fretting wear”, J. Nucl. Sci. Technol. **46**, No. 2 (2009)149–157.

6. FUEL STRUCTURAL DAMAGE AND OTHER FUEL ASSEMBLY ISSUES

6.1. INTRODUCTION

Besides traditional rod failures, new fuel assembly-related issues have appeared; fuel assembly bow and its consequences on incomplete control rod insertion (IRI), axial offset anomaly (AOA), and crud deposits on fuel. PWR fuel handling damage is also often related to assembly bow with the consequence of spacer damage during loading or offloading. As well, a variety of incidents involving fretting wear have been reported. Experience with these other observed, unexpected fuel issues implies that they can seriously affect plant operations, and it is clear that these concerns currently have an importance similar to that of fuel rod failures, at least for LWR fuel.

A list of fuel assembly related issues other than cladding breach are shown in Table 6.1 and they are described in more detail in this section.

6.2. ASSEMBLY BOW

6.2.1. PWRs

General background

In the middle of the 1990s, incomplete rod cluster control assembly (RCCA) insertions (IRI) during emergency shutdown or during drop time tests had been reported by several reactor operators and fuel vendors in the world. The first events of this type occurred in the Ringhals 4 nuclear power plant (Sweden) in 1994, when one rod control did not fully insert during a reactor trip [6.1]. During a subsequent rod drop test, four RCCAs stuck in the dashpot region.

An extensive investigation programme was initiated to identify causes of the sticking problem. Root cause investigations showed that IRI was experienced due to excessive deformation of fuel assemblies. Measurement of the axial shape of the FAs experiencing sticking showed that they were bowed in an 'S' shape with significant bending amplitude (up to 20 mm). This bowing, if it is great, increases friction between the control rod and the guide thimble, and can induce the breaking of RCCAs and even induce their sticking in the lower part of the fuel during their drop.

Afterwards, similar IRI events were observed in South Texas (United States of America), Almaraz (Spain), Wolf-Creek (United States of America), Doel and Tihange (Belgium), Nogent, Paluel and several other reactors in France [6.2–6.6].

Each time, a correlation was observed between drop time, drag force measured during reshuffling of RCCAs in the reactor vessel or in the spent fuel pool, and fuel assembly deformation, while there was not a pronounced correlation with burnup. Core deformations consist of an 'S' shape deformation and a 'C' shape deformation. An example of such deformations is shown in Fig. 6.1.

These RCCA anomalies can have a safety impact if RCCAs do not drop promptly to stop a nuclear reaction or do not insert sufficiently to provide a minimum required shutdown margin for those transients, which could add positive reactivity in the core. Another safety impact brought on by assembly bow is the creation of water gaps between fuel assemblies in a core, which lead to a redistribution of local power in comparison with nominal conditions. It is necessary to verify that perturbation of power distribution caused by assembly bow does not exceed the limitations existing in reference safety analysis. Due to their deformation, such fuel assemblies are also more difficult to handle during outages and the risk of damage increases. Occasional misalignment of an assembly to a nozzle with respect to the upper internal alignment pins may lead to assembly interaction and damage, as was observed at Byron and Nogent.

TABLE 6.1. FUEL STRUCTURAL DAMAGE AND OTHER FUEL ISSUES

Fuel issues	Root causes	Related areas			Affected plants			
		Manufact.	Operation	Design	BWR	PWR	WWR	CANDU/PHWR
— Assembly bow	High hold down forces		x	x		x		
	Creep and cross-flow						x	
— Channel bow	Irradiation growth	x	x	x	x			
	Creep mechanism							
	Shadow corrosion							
— Handling damages	Handling/hydridding brittleness		x		x	x		x ^b
— Axial offset anomaly	Crud deposits		x			x		
— Others miscellaneous damage								
— Hold down spring screws fracture		SCC	x	x	x		x	
— Resonance vibration	High cycle/low amplitude fatigue			x				x ^{a,b}
— Lower end cap fretting	Flow induced vibration		x		x ^{a,b}			
— Hydridding of Zr guide thimbles	Excessive hydrogen take-up	x	x			x ^b		
— Deformation of upper grid rim	Axial growth			x			x ^b	
— Displacement of SGs	Not reliable design			x			x	

^a Isolated event.^b Earlier occurrences, no noticeable problems in the reporting period of this review (1994–2006).

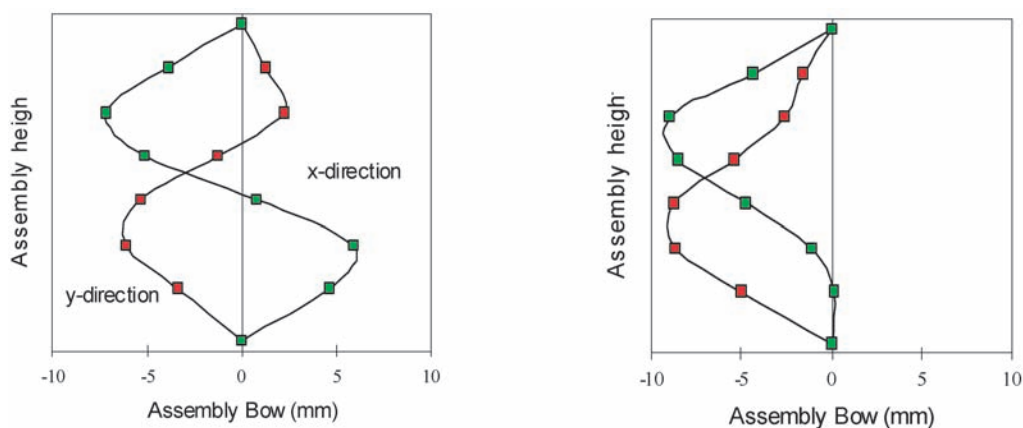


FIG. 6.1. Ringhals 3 — example of S-shaped and C-shaped assembly bows [6.5].

While waiting for a resolution to this problem, some preventive and monitoring actions were imposed by safety authorities to limit the risk of incomplete RCCA insertion and to guarantee safety during reactor operation. Among these were the limitation on fuel assembly burnup in the RCCA position, the optimization of loading pattern based on drag force measurement and drop times performed at the end of the previous cycle to select FAs in RCCA positions or survey RCCA drops following reactor trip, and the implementation of additional drop tests during and at the end of a cycle. A drop time test performed at the end of a cycle is an accurate means to detect severe core bowing early.

Bow mechanism

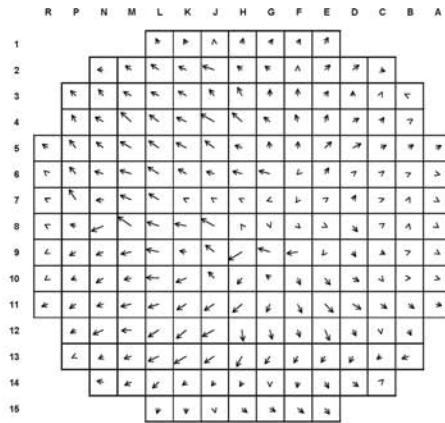
The causes and mechanisms of assembly bow are extremely complex and still not completely understood. Investigations and modelling have resulted in allotting these assembly deformations to two, probably dependent, phenomena:

- Overall assembly creep;
- Guide thimble creep. Measurement insertion guide efforts thimble by guide thimble confirmed the existence of such local deformations (bowing and misalignment of dashpot) on the lower stage level of the guide thimble. They also showed that the effort of insertion was not only related to total deformation of the assembly but can vary notably per guide thimble compared to others for the same assembly.

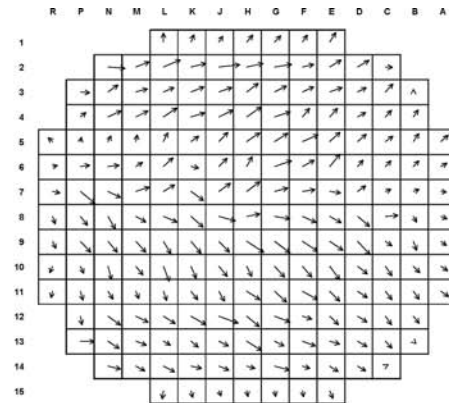
Irradiation induced creep deformation is a normal phenomenon function of several parameters and variables, including:

- Axial compression force applied on the FA;
- Guide thimble tube and rod growth;
- Initial deformation;
- Burnup;
- Fuel lateral stiffness;
- Creep properties of the material;
- Lateral hydraulic forces applied to FA;
- Interaction with neighbouring bowed assemblies.

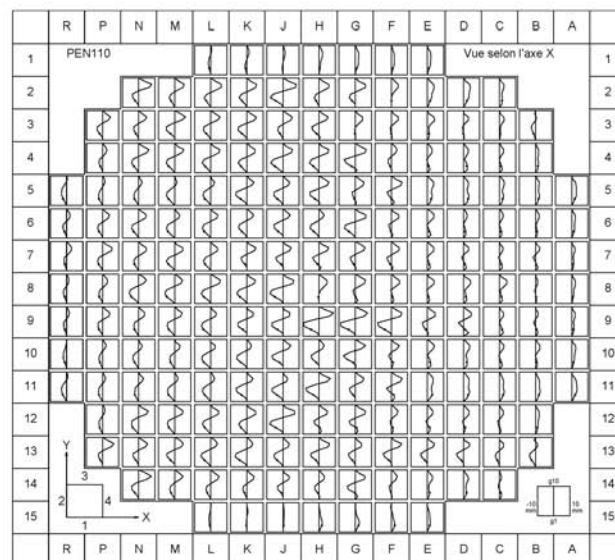
In the case of Wolf Creek, the main cause of fuel assembly bow was reported to be abnormal axial growth of the fuel assembly. Incomplete rod insertion (IRI) was experienced in non-IFM V5H fuel at exposures of greater than 50 MWd/kgU. The operation of these assemblies resulted in accelerated growth of the skeleton, which caused excessive bowing of the guide thimble tubes. The high growth was a combination of growth due to normal



Grid level 3 (a)



Grid level 7 (b)



Axial shape (c)

FIG. 6.2. Typical assembly bow in EDF 1300 MW(e) reactors [6.3].

saturation, oxide accumulation and accelerated irradiation growth [6.7]. However, in the other IRI events, a fuel assembly length increase was observed in the design limits.

An analysis performed at Ringhals concluded that the bowing in this reactor had been caused by a large creep deformation driven by excessive compressive forces of the hold down spring on the FAs, with a decrease in lateral stiffness due to a change from AFA to AFA2G design being an increasing factor.

A specific device was developed by EDF for FA bow measurement based on the use of an ultrasonic signal to measure the distance between fixed transducers and FA hang-up grids to the bridge in the spent fuel pit. During core unloading, each fuel assembly is stopped in front of the control process where FA bow is recorded in two directions. This recording device allows for on-line measurement of all core FAs during offload. A large database has been acquired using this device. These measurements performed on different reactors show differences in FA behaviour between 4-loop reactors and 3-loop reactors. Figure 6.2, extracted from Ref. [6.3], provides a typical FA bow at grid levels three and seven for a 4-loop reactor.

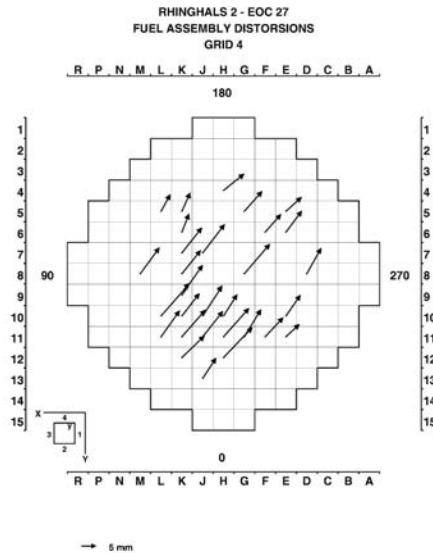


FIG. 6.3. Typical assembly bow in Ringhals reactors [6.5].

In all 4-loop reactors, deformations are always oriented radially in the lower part of the FA and follow two symmetrical directions in relation to a horizontal line in the upper part. As a consequence, ‘C’ shapes are observed in the east area of the core where a deformation has the same orientation over the entire height of an FA, and ‘S’ shapes are observed in the west area of the core, where deformations run in opposite directions in the lower and upper part of the FA. This explains why incomplete rod insertion problems are always found in the west area of the core in these reactors. This phenomenon is attributed to lateral hydraulic force distribution in the core, the distribution and amplitude of which may be reactor dependent.

A similar collective bow pattern was found on Ringhals reactors and EDF 900 MW(e) reactors [6.5] — Fig. 6.3.

A consequence of this reactor effect is that shape and bow amplitude observed at the end of a cycle are strongly dependent on FA location within a core. Burnup has only a mild effect on global deformation of an FA. The bow of one assembly does not evolve in the same direction throughout its life. It is a function of its successive position in the core. Thus, some utilities have developed fuel shuffling strategies that mitigate vulnerability to fuel assembly bow by changing the FA of a quadrant in a core at each cycle.

The effect of burnup is essentially seen through an increase in local deformation with time. Global deformation of assemblies can induce local deformation of the guide thimble, especially in the initial design of FAs with a constant guide thimble wall thickness combined with a reduction in the outer diameter in the dashpot region.

Several analytical tools were developed to model in-core assembly bowing. In its paper [6.8], AREVA presents an analytical approach based on a detailed single assembly. The influence of FA bow on drag forces and drop times was also analysed [6.9, 6.10]. These models are used to quantify the impact of design changes and operating conditions on bow evolution.

Corrective actions

To solve the problem of excessive fuel assembly deformation, different actions have been implemented. In the first change, undertaken in several reactors, the hold down springs of all irradiated fuel assemblies were plasticized to reduce compressive forces. The second action involved fuel suppliers proposing new features and fuel design such as:

- A change to the assembly hold down spring design to reduce and optimize the assembly hold-down forces;
- Development of a new guide tube with greater stiffness to upgrade the skeleton of the fuel assembly;
- Grid width increase to reduce inter-assembly lateral gaps by using low growth recrystallized zircaloy 4.

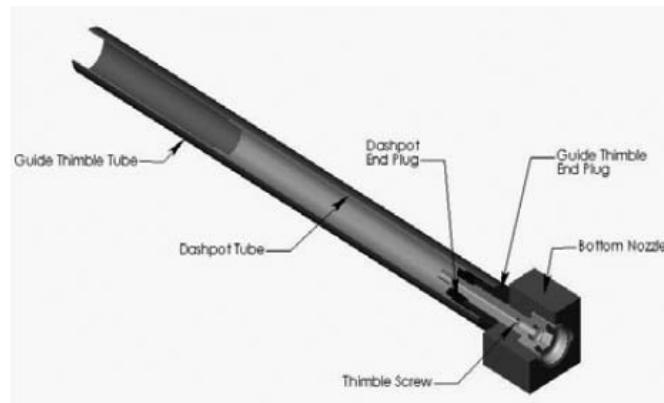


FIG. 6.4. RFA guide tube and dashpot configuration [6.12].

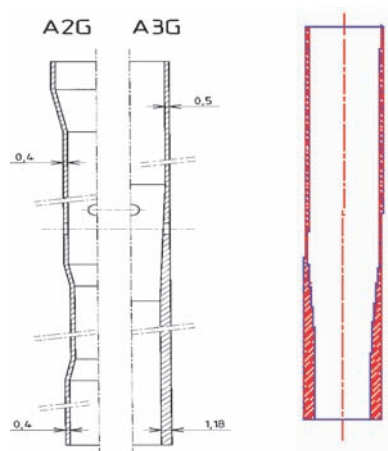


FIG. 6.5. Monobloc™ guide thimble [6.13].

Suppliers have also proposed introducing advanced cladding and guide thimble materials with a low growth rate and higher creep resistance to improve the dimensional stability of assemblies; M5 for AREVA, and ZIRLO for Westinghouse.

For Westinghouse, the enhanced structural capability of the RFA design stemmed from incorporation of a thicker guide thimble (12 and 14 foot designs) and incorporation of the tube in tube thimble design for 14 foot fuel (Fig. 6.4). The tube in tube design includes a full length, uniform structural tube, thereby eliminating the discontinuity that resulted from the swagged dashpot region of earlier designs. The dashpot effect is provided by a smaller inner tube at the bottom of the guide thimble. These features provide a structural margin for localized guide thimble distortion and overall fuel assembly bow. These improvements provide a margin to minimize susceptibility to incomplete RCCA insertion (IRI) and fuel handling incidents [6.11].

The use of ZIRLO material in structural components has also contributed to a structural margin because of its greater dimensional stability due to lower irradiation growth.

AREVA has achieved greater resistance to assembly bow with the introduction of the Monobloc™ guide thimble design [6.13]. The outside diameter of these enlarged, thick guide thimbles is constant all along the tube. In contrast to other solutions in the industry, the Monobloc™ guide thimble is made from a single piece, and the reinforced dashpot does not have any welds or insert sleeves (Fig. 6.5). The first fuel assemblies featuring Monobloc™ guide thimbles made of zircaloy-4 were utilized in 1998 in Belgium and Sweden.

All of these design changes have eliminated IRI or drop without recoil. Registered RCCA drop times have returned to normal values. For example, Figure 6.6 gives the evolution of an RCCA drop time anomaly indicator versus time in EDF reactors [6.13].

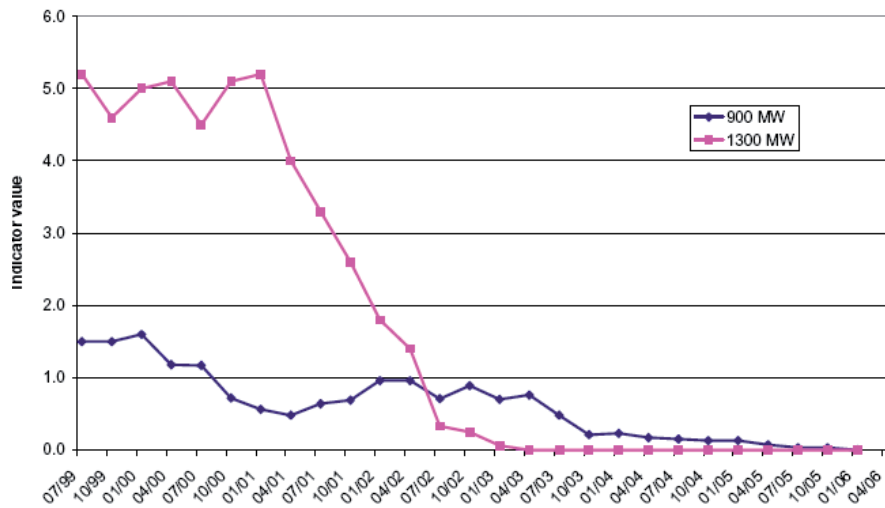


FIG. 6.6. RCCA drop time abnormality indicator versus time [6.13].

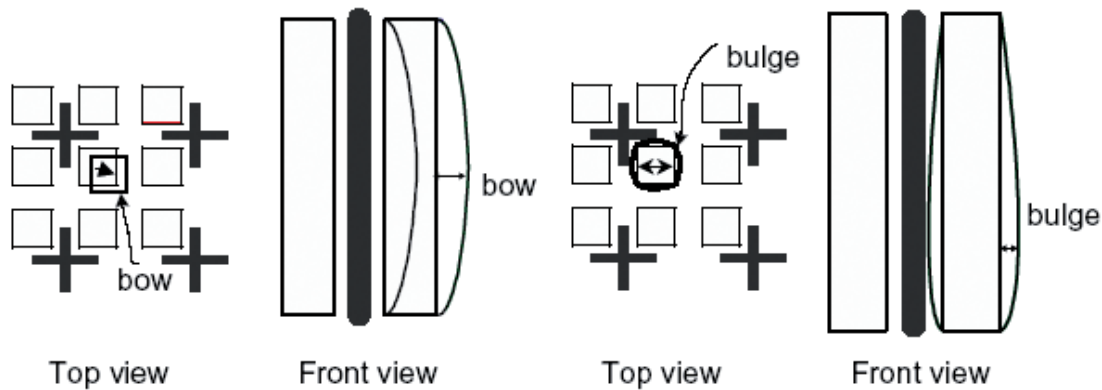


FIG. 6.7. Channel deformation mechanisms in BWRs [6.16].

Concerning assembly bowing, a decrease in average core value was generally observed with use of the new generation of FAs: AREVA AFA-3G and Westinghouse RFA. However, return to a normal situation will be gradual and will require several cycles, especially for 14 foot FAs, confirmed by recent fuel assembly bow measurements [6.14].

6.2.2. BWRs — Channel bow

BWR fuel elements are surrounded by zircaloy channels that provide guidance to control blades, separate regions of coolant with differing conditions and provide structural strength and stiffness to assemblies. They are designed to be used for more than one fuel assembly lifetime. Fuel channels undergo deformations due to various material and mechanical interactions when residing in an operating BWR.

Deformations such as bulge, bow and twist interfere with control blades and excessive moderation, leading to localized power increase due to asymmetry of the water gap.

The mechanisms of BWR channel deformation are described in Refs [6.15, 6.16], see Fig. 6.7. Channels operate at a positive internal pressure that causes outward elastic deformation of the walls. The combination of neutron irradiation and elevated operating temperature tends to convert the elastic deformation into permanent deformation via a creep mechanism. This deformation mechanism is called bulge.

Channel bow is due to a neutron flux gradient during irradiation and depends on core location history and exposure of the channel. As a result of neutron flux gradients across fuel cells, the fluence accumulated by a channel has an uneven planar distribution. Since irradiation growth depends on both material properties and fluence, channels bow to accommodate the resulting uneven growth. Small variations in heat treatment, material composition or texture tend to accentuate differential growth and bow.

Additionally, recent experience has revealed a new phenomenon called ‘shadow corrosion induced’ channel bow [6.17]. Several BWR plants have reported control blade interference issues, which is evident from increased blade settling times. Interfering channels were found to bow towards the control blade. A bow pattern appeared to indicate a new channel bow mechanism different from stress relaxation or differential growth due to fluence gradient.

To explain this phenomenon, coupons from several channels with varying degrees of bow and obtained at four elevations from opposing channel sides were brought to GE Vallecitos Nuclear Center for PIE. Channels selected for PIE had exposures in the range of 36–48 GWd/MTU and covered a wide range of measured bow. PIE performed on these coupons included visual examination, metallography, and hydrogen concentration measurements.

Hot cell examination results show that shadow corrosion of the channel outer surface can occur when a control blade is inserted next to the fuel channel. When controlled early in life, the resulting shadow corrosion can result in increased absorbed hydrogen induced growth of the channel wall closest to the control blade, which leads to channel bowing towards the control blade late in fuel bundle life, in addition to fluence gradient induced bow. Estimated channel bow due to hydrogen differential and fluence gradient correlates well with actual measured bow.

Fuel management can partly account for these effects and help to prevent excessive bow or unacceptable interference with control blades. In several plants, a specific ‘channel management procedure’ has been developed and applied. An example of such a methodology is described in Ref. [6.16].

Only one event of fuel failure due to excessive channel bow has been observed; this was already discussed in Section 5. Channel bow has been reduced through improved channel manufacturing methods such as thermal sizing and matching channel halves. In spite of improvements, channel bow continues to affect fuel performance. Utilities have spent significant resources to determine if their fuel designs are affected by channel bow and, if confirmed, to implement remedies (including channel replacement) and perform ongoing blade operability tests.

6.2.3. WWERs

Disturbances in the control and protection system due to assembly bow have been observed since 1992–1993 in some WWER-1000 plants transferred to three year operation mode in the Russian Federation [6.18], the Ukraine [6.19] and then in Bulgaria [6.20]. In these plants, the specified four second drop time for absorber rods was exceeded and, in some cases, absorber rods were sticking in bottom zones. The first studies of this phenomenon were conducted in a reactor spent fuel storage pool and in the reactor core of Balakovo Unit 2, where there were found to be excessive friction forces regarding absorber rod insertion [6.18]. Two such faulty FAs from the Zaporozhe NPP Unit 1 were transported and investigated in the Hot Laboratory of RIAR [6.21] where ‘intricately shaped assembly bow’ with 18–20 mm (17–24 mm for GTs) maximum deflection was found. It was understood that FA bow may cause a local power increase by creating enlarged water gaps between adjacent assemblies and that it thus might create a safety concern because of the possibility of a local power increase. An examination found no tracks of excessive corrosion, FR and GT irradiation growth, or changes in their mechanical properties.

Originally, these issues were discussed internationally at several IAEA Consultants’ Meetings on Control Rod Insertion Reliability for WWER-1000 NPPs in 1995, then later at the OECD/NEA Specialist Meeting on Nuclear Fuel and Control Rods: Operating Experience, Design Evolution and Safety Aspects in 1996 and again, in 1998, at the International Workshop on PWR and WWER Fuel Assembly Bow, organized by the Hans Weidinger Consultancy Firm in cooperation with the IAEA [6.22].

An, intensive investigation programme was initiated, including operating condition analysis, hot cell examinations of FAs, out of pile tests and modelling. Delays were found to be caused by friction increase between guide tubes and control rods due to fuel assembly distortion caused by excessive hold down force, radiation induced creep of fuel rod claddings and guide tubes, and insufficient FA rigidity at high burnup. The mechanisms of fuel assembly bow in PWRs and WWERs were identical.

The bowing was either C-shaped or S-shaped, i.e. similar to that observed in PWRs. To closely track the situation, a programme of additional measurements for RCCA drop time/friction forces was developed and introduced at all WWER-1000 units. In parallel, in some difficult cases, some units, e.g. Zaporozhe-1 (cycles 7 and 8, 1994–95) and 3 (cycles 7–9, 1994–96) and South Ukraine-2 (cycle 7, 1994), were transferred to the operational mode with three loop coolant circulation and a power reduction to 67% of nominal power [6.23].

Corrective actions

The following counter measures regarding reactor internals, FA materials/design and RCCA materials/design were performed during scheduled outages/repairs in 1993–1997 [6.23, 6.24]:

- Adjustment of the position of the protective tube units through which FA compression is provided;
- Provision of a separate hold down for each guide tube through an individual spring load for the first FAs (beginning in 1996);
- Replacement of spring material and an increase in spring motion to reduce forces of compression;
- Drilling of RCCA driver bars to reduce hydrodynamic resistance when inserting RCCA into the core;
- Use of newly designed heavier RCCAs;
- Replacement of steel by Zr based alloys for the FA skeleton. The presence in FA structural materials of different thermal expansion coefficients (grid and guide tubes made of steel, a central tube and fuel claddings of zirconium) leads to a stressed/strained FA state during a reactor heat-up and placement into power operation. The first loading of GTs and SGs made of E-110 alloy started in 1995; GTs made of E-635 alloy and SGs made of E-110 alloy were first loaded in 1996.

Use of GTs and SGs made of Zr-based alloys allows for a reduction in axial load to the FAs in a reactor hot state by ~40%. As a result, there is no spring relaxation and FA caps in the reactor reach a stable position. Also, homogeneity of materials provides minimum relative displacement of FA components [6.25].

Special calculation codes were developed to assess thermomechanical interactions of elements inside the assembly (fuel rods, guide tubes and spacer grids), and total FA behaviour in the core [6.26–6.29]. As mentioned above, monitoring measures were developed and proposed for introduction at NPPs, including:

- On-site measurement of FA bow;
- Periodic monitoring of RCCA drop time/friction force at NPPs;
- Optimization of fuel loading patterns.

Implementation of these countermeasures allowed for the practical elimination of, or significant reduction in the number of RCCA drop time and incomplete rod insertions (IRIs). For example, before these modifications (1993–1995), the average RCCA drop time, the number of RCCAs trips exceeding four seconds and the number of IRIs was for 11 Ukrainian WWER-1000s (unit average over a three year time span) equal to respectively 3.4 seconds, 19 times, 2.5 IRIs. After the implementation of countermeasures (1995–1st quarter of 1998), these values were equal to 2.5 seconds, 1 time and 0.1 IRI [6.18]. After the above mentioned modifications, no disturbances were registered at Balakovo 2 and 3 [6.18].

As was reported by Russian Federation fuel specialists V.L. Molchanov, V. Troyanov and others at the International Workshop on PWR and WWER Fuel Assembly Bow (17–19 February 1998, Řež, Czech Republic, organized by the Hans Weidinger Consultancy Firm in cooperation with the IAEA), over the period 1993–1997, more than 20 individual procedures for safe operation (with modelling FA behaviour in the core) of different cores with real FA geometry at 100% power had been made for Russian Federation NPPs with WWER-1000s. The results of FA bow measurements provided sufficient statistical data to assess gaps between FAs and to predict the evolution of gaps during operation. Important progress was made in the completion of a ‘generalized model’ WWER-1000 reactor in 1998 and the issuing of the Safety Analysis Report in 1999 for all WWER-1000 cores [6.24]. The ‘generalized model’ included a model of thermomechanical behaviour for all FAs and a code verified by measurements of more than 3000 FAs by February 1998. It allowed for calculation of the thermomechanical behaviour of FAs (including the shape of all FAs in the core, water gaps and rod linear loads at 100% power) for every specific fuel reload.

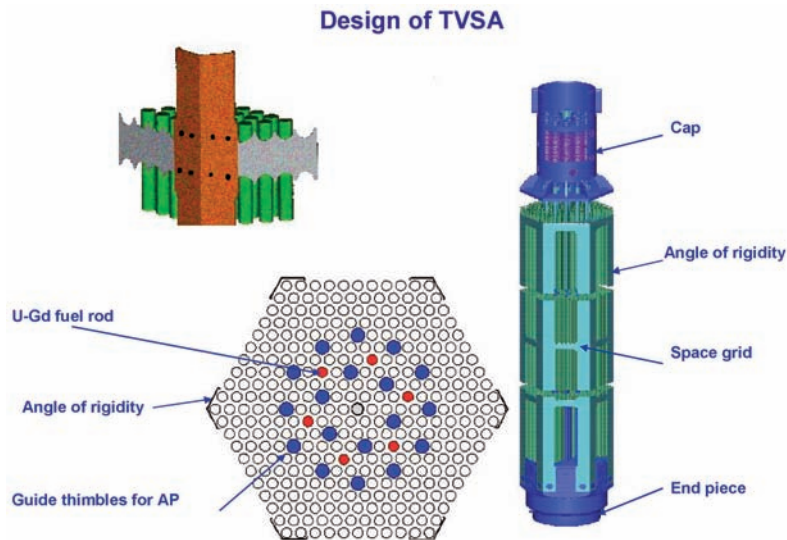


FIG. 6.8. TVSA–WWER-1000 FA designed by OKBM, Nizni Novgorod.

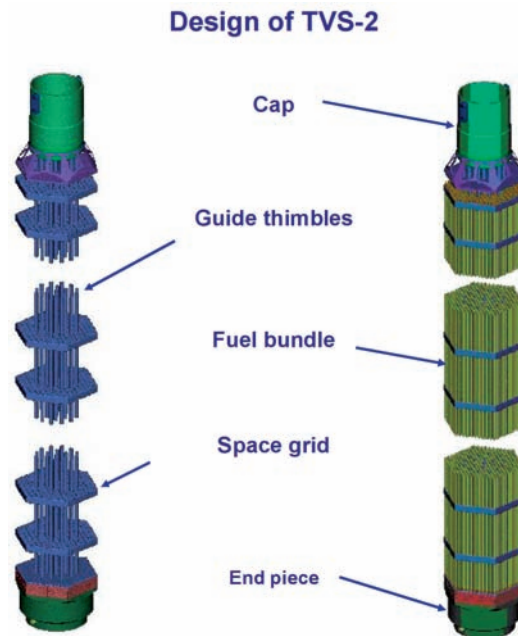


FIG. 6.9. TVS-2–WWER-1000 FA designed by OKB Hydropress, Podols.

An analysis of operational experience up to 1997–1998 showed that these measures successfully eliminated control rod disturbances for fuel operation conditions in those days. To cope with the FA bow–water gap phenomena at higher fuel duties (fuel residence time of 4–5 or more years), Russian Federation fuel design and research organizations developed advanced FA designs. Significant design modifications were introduced to increase individual FA rigidity, namely [6.30–6.32]:

- TVSA (or AFA), alternative FA, with a rigid Zr alloy skeleton formed by angles welded to SGs (Fig. 6.8);
- TVS-2, advanced FA, with a rigid Zr alloy skeleton formed by GTs welded to SGs (Fig. 6.9).

Both designs include a detachable head, anti-debris filter, removable FRs with collect fixing in the low anti-vibration SG, require U–Gd fuel, and are designed for no less than 4–5 years of operation. Mixing SGs are optional.

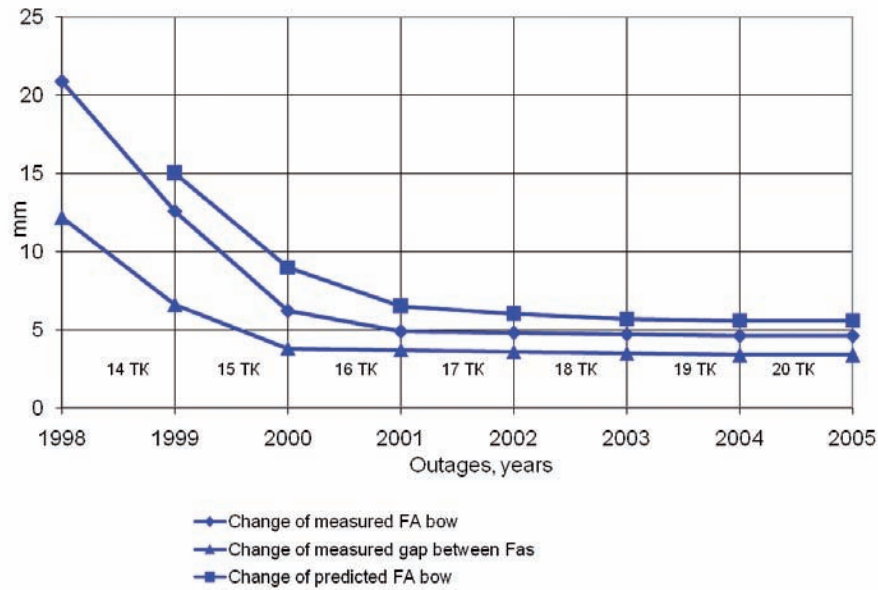


FIG. 6.10. Changes in FA bow and gaps between FAs at Kalinin-1.

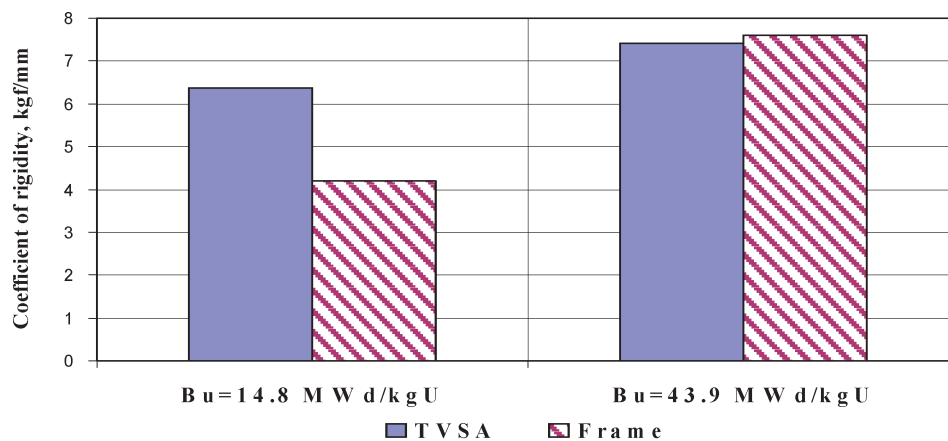


FIG. 6.11. Measured bending rigidity of FAs as a whole and skeletons for TVSA FAs.

The first batch of 12 TVSA FAs was installed in 1998 at Kalinin-1 (14th cycle) for five years of operation; their loading continued and the 18th cycle core consisted of 100% AFAs [6.33]. Results of in-core measurements of the FA's maximum curvature and gaps between FAs showed a high core bending geometrical stability, the gap decreased by ~four times and maximum bending and gap values were less than 5 mm after the 18th cycle [6.34]; see Fig. 6.10.

Four TVSAs, irradiated until burnup in the range of 14.8–55.3 MW·d/kgU, were transported and investigated in the RIAR Hot Lab [6.35]. A maximum bow of less than 7 mm at an elevation of ~2.5 m and a twisting angle of ~0.7 deg were measured for free standing FAs. Bending rigidity of the FAs as a whole and skeletons were measured (Fig 6.11). It can be seen that at the BOL, rod bundle contributes significantly to FA rigidity, while at the EOL; the major contribution is due to a strong skeleton. This might be explained by the fact that FR cladding diameter measurements indicated cladding creep down at high burnup.

TVS-2 FAs were first loaded at Balakovo-1 in 2003, Balakovo-2 and 3 in 2004 and Balakovo-4 in 2005 with full core operation using these FAs taking place within three years, e.g. Balakovo-1 in 2005 and Balakovo-2 and 3 in 2006. Measurement of FA bow (Fig. 6.12) showed that maximal FA bow diminished up to ~ 7 mm (with an average amplitude of less than 3 mm) with an increase in the share of TVS-2 FAs in the core, and the bow shape changed from an S to a C type [6.36].

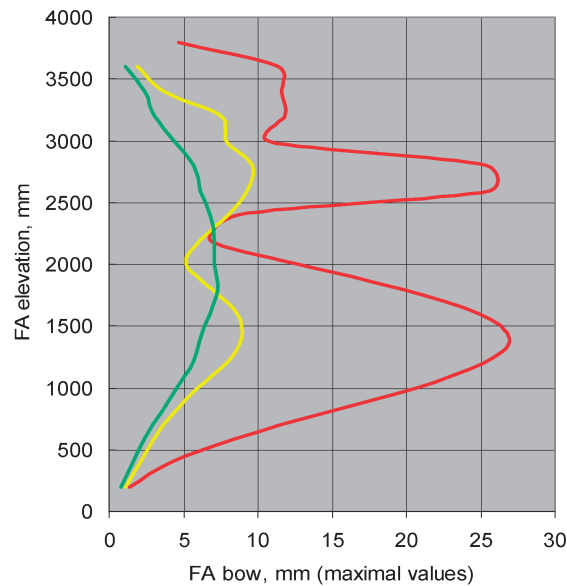


FIG. 6.12. Maximal FA bow measured at Kalinin 1 upon loading of TVS-2 FAs into the core: red line reload in 2003 when 54 TVS-2 FAs were loaded for the first time (no advanced FAs in the core during the measurement period); yellow line reload in 2004 with 54 more TVS-2 FAs loaded; green line reload in 2005 with the whole core consisting of TVS-2 FAs (108 advanced FAs in the core during the measurement period).

As a result of the implementation of FAs of advanced designs, periodic scram checks and bow amplitudes have been cancelled since 2006 [6.37].

Finally, during startup of a new unit at Rostov NPP, special attention was paid to the geometry of fitting surfaces of reactor internals to provide a better free FA vertical position.

Following these modifications, measurements showed a significant decrease in fuel assembly bow and RCCA displacement force in fuel assemblies. Designs with a rigid skeleton have demonstrated permissible use of fuel assemblies with high geometrical stability for up to six fuel cycles [6.35, 6.38].

All of the above mentioned information relates to FA bow phenomenon observed in WWER-1000 fuel supplied by Russian Federation fuel vendor Corporation TVEL. Temelin NPP in the Czech Republic, with its two WWER-1000 units, has been supplied from the beginning by Westinghouse. The basic FA design used is Vantage-6 with Zry-4 GTs and mid-SGs (bottom and top SGs are from Inconel). After the 1st cycle, no RCCA drop time increase was registered for either unit, but FA irradiation growth was at the upper boundary and FA bow corresponded to higher burnup [6.39]. During the 2nd cycle, 10 IRI events were observed at Unit 2 and FA bow was beyond the database. Periodic RCCA drop tests started. During the 3rd and 4th cycles, more fresh FAs were positioned under RCCAs and periodic RCCA drop time tests continued. Scram tests showed a constant increase in drop time during both cycles, in spite of operation in IRI mitigation mode. A sudden outage was executed in the course of the 4th cycle.

To cope with such an unfavourable IRI situation, the following FA design modifications were implemented:

- 1st step (loaded in 2006): optimization of GT dashpot (tube in tube design), and long thimble end plugs;
- 2nd step (loaded in 2007): ZIRLO GTs, Inconel vaneless SGs, double GT/SG sleeves.

Effectiveness of these countermeasures may be seen by the end of 2008 or beginning of 2009, i.e. after submission of this document for publication.

6.2.4. CANDUs

The short 0.5 m length fuel elements of a horizontally oriented CANDU fuel bundle tend to sag at mid-plane by less than 0.25 mm, depending on the clearance between appendages of neighbouring elements and between bearing pads and the pressure tube surface. In a reactor, the fuel elements of an irradiated bundle tend to sag and then creep in the direction of gravity. The maximum bow, or permanent sag, measured at the mid-plane of an

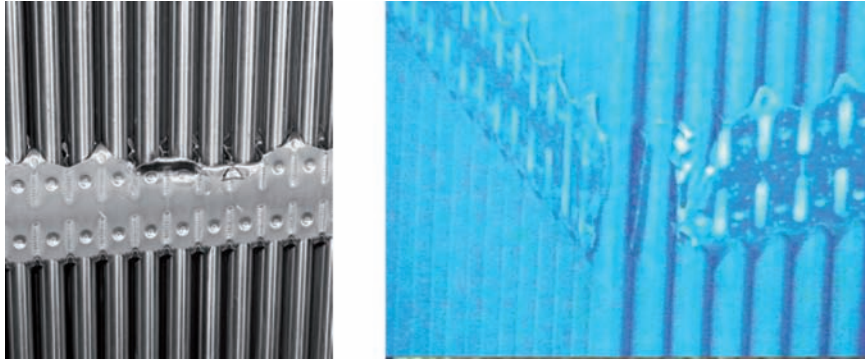


FIG. 6.13. Examples of spacer grid damage [6.14].

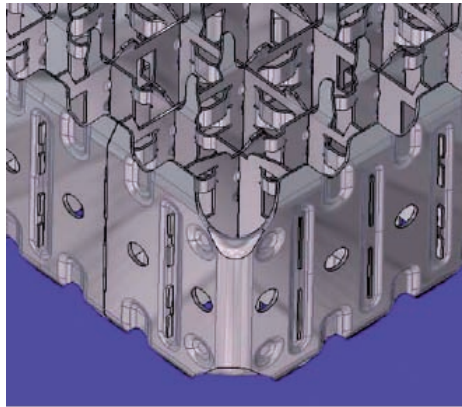


FIG. 6.14. Improved AFA-3G grid [6.13].

irradiated power reactor bundle is less than 1 mm [6.40]. Maximum sag generally occurs at the top elements. There have been no reported cases of element bow causing fuel bundle damage in CANDU power reactors.

6.3. MECHANICAL DAMAGE DURING HANDLING

Over the last decade, mechanical damage which takes place during handling has been reported by several utilities. Interference between PWR fuel assemblies during loading and unloading operations has resulted in hang-ups between spacer grids that damaged the grids. In rare cases, such grid damages can lead to rod failure through fretting wear of the fuel rods, which are no longer supported. These difficulties have led to increases in handling phase time, and consequently, outage duration. They can also generate debris in the core vessel, which must be removed during an outage. The causes of interference have included spacers prone to hang-up, bowed fuel assemblies, diminution of the gap between fuel assemblies due to the use of larger spacers to reduce assembly bow in the core, and handling practices.

Statistical data show an increase in FA damage in EDF plants over the past three years, especially for 1300 MWe units. Some examples of such grid damage are illustrated in Fig. 6.13. The cause of this increase can be explained by the mechanical behaviour of the FA structure under irradiation, and particularly significant bow and unexpected grid growth (recrystallized zircaloy-4), in addition to some inappropriate handling machine settings and operating modes [6.14].

A new version of the existing AFA 3G spacer grid, with implementation of design adjustments to further increase fuel assembly in-reactor straightness and to reduce the risk of hang-ups with neighbouring fuel assemblies during loading/unloading operations is being designed by AREVA (Fig. 6.14). The main modifications to the grid have to do with the external strap (vane shape and strap welding geometry).

In CANDU reactors, fuel bundles have occasionally been damaged during refuelling. Two examples are briefly described below [6.41].

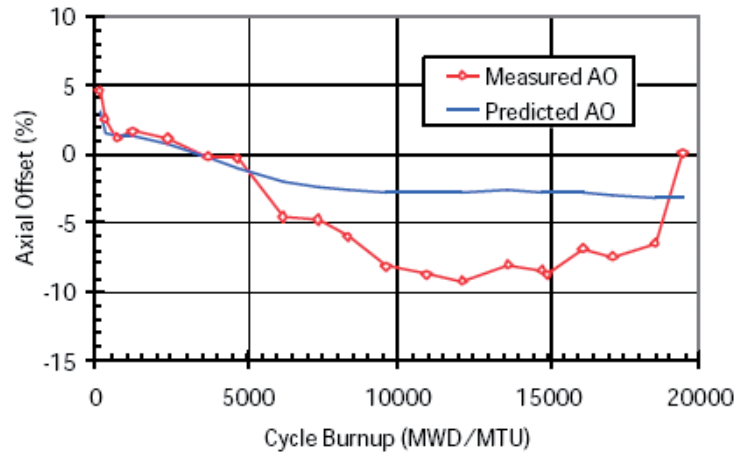


FIG. 6.15. Axial offset behaviour for a crud induced power shift (CIPS) core [6.11].

At Bruce NGS A, the fuelling direction is against the flow, which requires pairs of bundles to be transported to and from the fuel channel in 'fuel carriers'. In 1979, a fuel bundle escaped or was 'washed-out' from the carrier at the upstream end of channel P13 in Unit 1. This bundle was subsequently crushed when an empty carrier was returned to the channel to remove the next pair of bundles. The problem was corrected through a minor change to the fuelling sequence.

Owing to an interruption in refuelling operations at Pickering NGS A, bundles resided for several days in the cross-flow region near liner holes of the inlet end fitting of the fuel channel. Under these conditions, the end plates broke, causing a bundle to partially disassemble in the core. Normal residence time in the cross-flow is about two minutes.

On the basis of experience with fuel residing in the cross-flow region of the liners in CANDU-6 reactors, a limit of about one day has been placed on fuel residence time. If fuel resides in a cross-flow for a longer period, it is recommended that it be discharged from the core.

6.4. CRUD AND AXIAL OFFSET ANOMALIES (AOA)

Without existing at the origin of a fuel failure, a relatively new fuel issue is crud deposition on PWR fuel assemblies operating with high duty cores and significant levels of sub-cooled nucleate boiling [6.11].

In these plants, sub-cooled boiling occurs on the upper spans of high power fuel assemblies. Nickel and iron corrosion products, which circulate in coolant, deposit as 'crud' on the fuel cladding. In locations where sub-cooled boiling occurs, deposition is accelerated. Boric acid and LiOH also concentrate in these boiling porous crud deposits, accumulating boron compounds. The result of this boron buildup in the upper portions of some fuel assemblies is that core power distribution shifts unexpectedly toward the core inlet several months after the start of the cycle. This shift is referred to as a crud induced power shift (CIPS), or axial offset anomaly (AOA) (Fig. 6.15).

To explain the phenomenon, an alternative theory has been proposed by fuel vendor ENUSA [6.42]. It has been suggested that metallic cations, especially nickel in PWRs, precipitate as their hydroxides when local pH exceeds a precipitation threshold. In turn, it is proposed that boron co-precipitates as lithium metaborate. Furthermore, it is suggested that the AOA process is dependent on pH, temperature, flow conditions and concentrations of metallic ions. In particular, the steaming rate at the fuel cladding surface had a profound effect on core deposits and AOA.

Most AOA incidents in the United States of America have occurred shortly after a plant increased its power output, which increased sub-cooled nucleate boiling. Other factors can affect this susceptibility, since corrosion product transport and deposition have a relatively complex interaction. In particular, the release and dissolved fraction of corrosion products depend on coolant pH. Susceptibility can be reduced, for example, by reducing the release rate of corrosion products from steam generators (high lithium coolant programmes currently being

implemented are designed to reduce corrosion release). However, the maximum level of sub-cooled nucleate boiling in the core remains a good indicator of AOA risk.

In 2004, EPRI published the PWR Axial Offset Anomaly (AOA) Guidelines. This document provides utilities with guidance on how changes in various plant parameters, core design and chemistry can impact crud deposition and AOA. EPRI also developed the Boron-Induced Offset Anomaly (BOA) Code. The BOA code is an integrated thermal hydraulic–chemistry software package that predicts where crud will deposit, its thickness and the susceptibility of particular core designs to AOA [6.43].

Additionally, to mitigate the impact of crud deposition and AOA on plant operation, several utilities are using ultrasonic fuel cleaning technology to remove crud from reload fuel. During the spring of 2005, this new technology was applied at seven reactors in the United States of America and one in Spain. By the end of 2005, a total of 13 reactors worldwide will have used this technology [6.43].

6.5. OTHER MISCELLANEOUS DAMAGE

6.5.1. Hold-down spring screws fracture, PWRs

Over the last decade, a significant number of Westinghouse fuel assemblies have been affected by fracturing of the hold-down spring screw on the top nozzle. These failures have sometimes caused assembly handling difficulties. The cause of the failures was stress corrosion cracking.

In several cases, repair of fuel assemblies was required. The top affected nozzle in the Inconel 718 model was replaced by a new top nozzle equipped with screws less sensitive to stress corrosion cracking.

6.5.2. Resonance vibration, Darlington 2

In November 1990 [6.44], a routine fuelling operation for channel N12 of Darlington Unit 2 was aborted owing to difficulties encountered in the insertion of a pair of fuel bundles recycled from another channel. A follow-up investigation showed that the centre seven elements of the downstream bundle had broken loose, and had interfered with normal refuelling operations. These elements had been carried past the fuel latch by coolant flow through the channel, and had obstructed other bundles from being completely inserted into the channel. The bundle was extensively damaged during the attempted refuelling operation. The damage increased the activity levels of most fission products in the primary circuit. The increased activity levels were difficult to quantify at the time, since the GFP system was not fully commissioned.

Inspections of other outlet bundles in Darlington fuel bays revealed the presence of end plate cracks. PIE of the damaged fuel revealed that the end plate cracks were the result of high cycle/low amplitude fatigue. Subsequent investigations demonstrated that the five vane impellers of the primary circuit pumps introduced pressure pulsations which were acoustically amplified within certain channels. The pulsation frequency of 150 Hz coincided with the resonant frequency of the inner seven fuel elements of the 37 element bundle. With fuel column latch support, which is unique to the Darlington and Bruce reactors, the non-outer fuel elements are unrestrained and free to vibrate in an axial direction. Axial vibration at the resonant frequency led to end cap cracking.

To eliminate the acoustic amplification of pressure pulsations in the fuel channels and to decouple the axial resonant response of the fuel, five vane pump impellers were replaced with seven vane impellers. This change shifted the pressure pulsation frequency from 150 to 210 Hz, which eliminated the end plate cracking problem at the Darlington reactor.

6.5.3. Lower end cap fretting, Oskarshamn 1

During the May 1988 refuelling outage at Oskarshamn 1 reactor, foreign objects resembling ‘nails’ were found in the reactor vessel [6.45]. These objects were seated on the fuel inlet orifice plate located under each fuel element. Subsequent checks revealed that these objects were the remains of lower end cap shanks from spacer capture rods (SCRs) of Advanced Nuclear Fuels Corporation (ANF) supplied fuel. An inspection of all ANF fuel assemblies in the core revealed that of 338 irradiated assemblies from five different reloads, 127 had varying degrees of fretting damage to the SCR lower end cap. In 55 cases, the end cap had completely worn off.

The shape of the SCR end cap wear pattern led to the strong suspicion that the problem stemmed from flow induced vibrations. In 1986, the reactor underwent some modifications to improve fuel cycle economics. These modifications resulted in the maximum average core flow changing from 6950 to 7300 kg/s. Fuel assemblies loaded after 1983 had incorporated a longer SCR lower end cap with a 4 mm projection below the lower tie plate.

To evaluate the effect of changes to fuel design and reactor operating conditions on lower end cap fretting, a large number of tests were performed. The results of the testing programme and the evaluation showed that the root cause of the problem was excessive pressure fluctuation caused by lateral oscillations of a water jet at the inlet orifice directly below the lower tie plate of the fuel assembly. This jet was found to oscillate laterally, producing lateral cross-flows which, in turn, subjected fuel rods to oscillating lateral forces, causing the rods to vibrate. The magnitude of forces driving the vibrations was found to increase with flow rate.

In order to stabilize flow in the conical diffuse zone, a 10 hole orifice was developed. This orifice was tested and installed, as results demonstrated that rod vibration amplitude was small enough. Further steps in design were taken to prevent recurrence in any reactor.

6.5.4. Hydrogenation of Zry guide thimbles, Ringhals 2

In 1990, Zry guide thimbles in two fuel assemblies were damaged during handling in Ringhals 2. Hot cell examinations showed that the damage was due to material embrittlement in the cold state by excessive hydrogenation, i.e. external hydriding. More detailed investigations and laboratory tests revealed that the effect was caused by a combination of specific water chemistry parameters during initial startup after refuelling, and a particular susceptibility to hydrogen uptake of the fresh grit blasted inner surfaces of guide thimbles under these conditions. The main water chemistry factor was a combination of high initial nickel concentration plus the early addition of hydrogen. As coolant heats up, nickel can precipitate as a metal on the fresh surface and, probably assisted by specific impurities embedded in the surface, can act as a catalyst to intensify hydrogen uptake [6.46]. The plant startup procedures in place had been used before, and guide thimbles subject to the applied process of grit blasting had been used before in other plants. The incidental combination of these two proven practices created the problem. A manufacturing process improvement to reduce sensitivity of the fresh surface was introduced and has solved the problem.

6.5.5. Deformation of upper grid rim, WWER-440

Axial clearance for fuel rod irradiation growth in WWER assemblies is sufficient for elongations up to burnups beyond the design value. Nevertheless, in some cases, severe deformation of the upper grid and upper grid rim was observed in wrapped assemblies. The cause was found (and confirmed by testing) to be the jamming of growing fuel rods in the upper spacer grid, which was firmly fixed in the top nozzle. Design changes were made to avoid this effect [6.47].

6.5.6. Displacement of spacer grids in WWER-1000 FAs with Zr based alloy spacer grids and guide tubes

Advanced WWER-1000 FAs with Zr based alloy E-110 spacer grids (SGs) and guide tubes (GTs) were first loaded into reactors in 1995, and with E-110 SGs and Zr based alloy E-635 GTs starting in 1996. During inspections of FAs with SGs and GTs made of Zr based alloys operated during 2nd and 3rd cycles in Zaprozhe-3 and Rovno-3 reactors, vertical displacement of SGs was found. After two cycles (burnup ≥ 27 MWd/kgU), movement was observed for SGs from the 10th to the 12th positions (from the bottom, altogether 15 SGs). After three cycles (burnup ≥ 27 MWd/kgU), movement was observed for SGs from the 4th to the 13th positions, thus practically involving 2/3 of the FA volume [6.23] in the SG displacement process. This took place because the SGs were not structurally reliably attached to GTs (either mechanically or through welding). After design debugging, no SG movement was observed.

REFERENCES TO SECTION 6

- [6.1] JACOBSON, S., FRANCILLON, E. "Incomplete control rod insertion due to fuel element bow", Nuclear Fuel and Control rod: Operating Experience, Design Evolution and Safety Aspects, Madrid, 1996.
- [6.2] JADOT, J. et al., "Incomplete rod insertion at DOEL 4 and TIHANGE 3", International LWR Fuel Performance Meeting, TopFuel 1999.
- [6.3] BALEON, J.P, DANGOULEME, D., VRIGNAUD, E., "Traitement des anomalies de chute des grappes de commande rencontrées dans les réacteurs EDF", Fontevraud, 1998.
- [6.4] FRANCILLON, E., "Remedies to F.A. bowing and incomplete RCCA insertion in PWR", TOPFUEL 97, Manchester, 1997.
- [6.5] ANDERSON, T. "A decade of assembly bow management at Ringhals", Int. Mtg on LWR Fuel Performance, Orlando, FL, 2004.
- [6.6] KEE, E., "Experience with incomplete control rod insertion in fuel with burnup exceeding approximately 40 [6.7] GWD/MTU", 24th Water Reactor Safety Info. Mtg, Brookhaven, National Laboratory, 1996.
- [6.7] WILSON, H.W., et al., "Westinghouse fuel performance in today's aggressive plant operating environment", Int. Topical Mtg on Light Water Reactor Fuel Performance, Portland, 1997.
- [6.8] SALAUM, H., BALEON, J.P., FRANCILLON, E., "Analytical approach to PWR fuel assembly distortions", SMIRT 97, Lyon, 1997.
- [6.9] BOSSELUT, D., et al., "Insertion and drop of control rod in assembly: simulation and parametric analysis", Structural Behaviour of Fuel Assemblies for Water Cooled Reactors, IAEA-TECDOC-1454, IAEA, Vienna (2005).
- [6.10] AULLO, M., RAENSTEIN, W.D., "European Fuel Group experience on control rod insertion and grid to rod fretting", *ibid.*
- [6.11] KNOTT, R.P., et al., "Advance PWR fuel designs for high duty operation", TopFuel 2003, Würzburg, 2003.
- [6.12] CHAPLIN, D., et al., "EFG Fuel designs and experience in EDF reactors", TopFuel 2006, Salamanca, 2006.
- [6.13] GOTTUSO, D., CANAT, J.N., MOLLARD, P., "A family of upgraded fuel assemblies for PWR", TopFuel 2006, Salamanca, 2006.
- [6.14] PROVOST, J.L., DEBES, M., "EDF PWR Fuel operating experience and high burnup performances", *ibid.*
- [6.15] REPARAZ, G., et al. "Advanced BWR channels " Int. Topical Mtg on Light Water Reactor Fuel Performance, Portland, 1997.
- [6.16] KARVE, T., et al. "A methodology for calculating and a process for mitigate channel distortion and cell friction", TopFuel 2006, Salamanca, 2006.
- [6.17] MAHMOOD, R., et al. "Channel bow in boiling water reactors — hot cell examination results and correlation to measured bow", 2007 Int. LWR Fuel Performance Mtg, San Francisco, CA, 2007.
- [6.18] VASILCHENKO, I., DEMIN, E., "Operational indices of VVER-1000 fuel assemblies and their improvements", Proc. 1st Int. Conf. on VVER Reactor Fuel Performance, Modelling and Experimental Support", Varna, BAS, INRNE, Sofia (1995) 49.
- [6.19] AFANASYEV, A., "The summary of WWER-1000 fuel utilization in Ukraine", Proc. Int. Top. Mtg on LWR Fuel Performance, Portland, 1997, ANS (1997) 96.
- [6.20] BEKIREV, D., NIKOLOV, A., "Kozloduy NPP nuclear fuel cycle experience", Proc. IAEA TM on Structural Behaviour of Fuel Assemblies for Water Cooled Reactors, Cadarache, 2004, IAEA-TECDOC-1454, IAEA, Vienna (2005) 165.
- [6.21] BIBILASHVILI, Y., et al., "Operation experience of WWER fuel, including analysis of abnormal condition", Proc. 2nd Int. Sem. on WWER Reactor Fuel Performance, Modelling and Experimental Status, Sandanski, 1997, BAS, INRNE, Sofia (1997) 11.
- [6.22] VON JAN, R., WEIDINGER, H.G., "PWR and WWER fuel assembly bow", Workshop in Rez/Prague, 1998, Summary (Document HGW-02-98).
- [6.23] AFANASYEV, A., "Fuel cycle activity and the summary of VVER-1000 fuel utilization in Ukraine", Proc. TCM, 15th Plenary Mtg, of the Int. Working Group on Water Reactor Fuel Performance and Technology, Vienna, 1999, IWGFPT/49, IAEA, Vienna (1999).
- [6.24] VASILCHENKO, I.N., et. al., "Design measures for providing geometrical stability of WWER reactor cores", Proc. IAEA TM on Structural Behaviour of Fuel Assemblies for Water Cooled Reactors, Cadarache, 2004, IAEA-TECDOC-1454, IAEA, Vienna (2005) 169.
- [6.25] VASILCHENKO, I., DRAGUNOV, Y., MIKHALCHUK, A., "Results of trial operation of the WWER advanced fuel assemblies", Proc. 4th Int. Conf. on WWER Reactor Fuel Performance, Modelling and Experimental Support", Albena Resort, 2001, BAS, INRNE, Sofia (2002) 83.
- [6.26] TROYANOV, V., et. al., "Lessons learned from the computational simulation of thermo mechanical behaviour of the WWER-1000 reactor cores: FA development and its implantation into the Balakovo NPP Unit 1", Proc. 5th Int. Conf. on WWER Reactor Fuel Performance, Modelling and Experimental Support, Albena, 2003, BAS, INRNE, Sofia (2004) 200.
- [6.27] LIKHACHEV, Y., et. al., "Theoretical approach to the WWER core thermo mechanical modelling", Proc. 5th Int. Conf. on WWER Reactor Fuel Performance, Modelling and Experimental Support, Albena, 2003, BAS, INRNE, Sofia (2004) 209.
- [6.28] KISELEV, A.S., et. al., "Substantiation of thermal-mechanic stability and strength of TVSA-T type fuel assembly in four-year fuel cycle", Proc. 7th Int. Conf. on WWER Reactor Fuel Performance, Modelling and Experimental Support, Albena, 2007, BAS, INRNE, Sofia (2008).

- [6.29] TROYANOV, V.M., et. al., “Numerical and analytical investigation of WWER-1000 fuel assembly and reactor core thermal mechanics”, Proc. IAEA TM on Structural Behaviour of Fuel Assemblies for Water Cooled Reactors”, Cadarache, 2004, IAEA-TECDOC-1454, IAEA, Vienna (2005) 113.
- [6.30] ROZKOV, V., ENIN, A., BEZBORODOV, Y., PETROV, V., “Fuel improvement and WWER-1000 FA main operational results”, Proc. 5th Int. Conf. on WWER Reactor Fuel Performance, Modelling and Experimental Support, Albena, 2003, BAS, INRNE, Sofia (2004) 177.
- [6.31] SOLONIN, M., et al., “WWER fuel performance and material development for extended burnup in Russia”, Proc. 2nd Int. Seminar on WWER Reactor Fuel Performance, Modelling and Experimental Support, Sandanski, 1997, BAS, INRNE, Sofia (1998) 48.
- [6.32] BIBILASHVILI, Y., et al., “Development of alternate fuel assembly for WWER-1000 Reactor”, *ibid.*, p. 123.
- [6.33] MOLCHANOV, V., et. al., “The results of AFA Operation at Kalinin NPP and the trends of further perfection of the fuel based on AFA”, Proc 5th Int. Conf on WWER Reactor Fuel Performance, Modelling and Experimental Support, Albena, 2003, BAS, INRNE, Sofia (2004) 182.
- [6.34] LAVRENYUK, P., MOLCHANOV, V., TROYANOV, V., IONOV, V., “Nuclear fuel for VVER reactors. Current status and prospects”, Proc. 7th Int. Conf. on WWER Reactor Fuel Performance, Modelling and Experimental Support, Albena, 2007, BAS, INRNE, Sofia (2008).
- [6.35] MARKOV, D.V., et. al., “Basic results of post-irradiation examination of advanced VVER fuel”, *ibid.*
- [6.36] TYUKIN, V., “Operational experience of fuel assembly TVS-2 at Balakovo NPP”, *ibid.*
- [6.37] VASILCHENKO, I.N., et. al., “Choosing the governing solutions for FA of AES-2006”, *ibid.*
- [6.38] SMIRNOV, V.P., “VVER fuel results of post irradiation examination”, TopFuel, Kyoto, 2005.
- [6.39] MECIR, V., “Temelin NPP fuel experience”, Proc. 7th Int. Conf. on WWER Reactor Fuel Performance, Modelling and Experimental Support”, Albena, 2007, BAS, INRNE, Sofia (2008).
- [6.40] DENNIER, D., MANZER, A.M., RYZ, M.A., KOHN, E., “Element bow from new and irradiated CANDU fuel bundles”, 17th Annual Conf. of the Canadian Nuclear Society, Fredericton, 1996.
- [6.41] MANZER, A.M., personal communication.
- [6.42] TARIN, F., DONCEL, N. “ Experimental verification of pH and nickel influences on AOA ”, TopFuel 2003, Würzburg, 2003.
- [6.43] YANG, R., CHENG, B., DESHON, J., EDSINGER, K., OZER O., “Fuel R & D to improve fuel reliability ”, TopFuel, Kyoto, 2005.
- [6.44] JUDAH, J., “Overview of fuel inspections at the Darlington Nuclear Generating Station”, 3rd Int. Conf. on CANDU Fuel, Chalk River, 1992.
- [6.45] REPARAZ, A., NORDLOF, S., “An innovative solution to an end cap fretting problem in Oskarshamn Unit 1”, Proc. Int. Top. Mtg on LWR Fuel Performance, Avignon, 1991, Vol 1, SFEN, Paris (1991) 305.
- [6.46] SIEMENS, Siemens Nuclear Fuel Report, No. 2 March (1992) 42.
- [6.47] SMIRNOV, A.V., et al., “WWER-1000 and WWER-440 fuel operation experience”, Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, 1994, ANS, LaGrange Park, IL (1994) 31.

7. SECONDARY FUEL FAILURES

7.1. BACKGROUND

Degradation of failed fuel rods due to secondary hydriding, e.g. blisters, holes, cracks, failed or separated end plugs, and advanced stage rod fractures, has always been observed in Zry clad fuel and was extensively studied in the 1990s [7.1].

Significant secondary degradation has been a major issue for BWR fuel since the late 1980s, when several failures developed long axial splits. Failures in PWR fuel have generally been less severe, although there have been infrequent cases in which some axial cracking has been observed.

In BWRs, axial cracks of significant length had been observed after severe power ramps by control rods, but became very rare after implementation of PCIOMR. More frequent observations of axial cracks and splits were made after the introduction of barrier cladding with a pure Zr liner for BWRs, even under complete PCIOMR regulations. Reviews of the early 1990s experience showed an unusually high frequency of axial splits, in the range of 30–50% of failed rods, with unusual lengths of around 15 cm up to a range of 2 to 3 metres. Axial splits became a major concern due to extremely high activity release and partial fuel washout (10–100g), which forced several utilities into early shutdown just to remove a leaker [7.2]. Disputes have taken place regarding the extent to which this effect should be attributed to the Zr liner, the base material or the ‘fuel duty’ during operation.

Non-barrier BWR fuel from some — but not all — manufacturers was also affected by an increasing number of axial splits, although with much lower frequency and with typical lengths of a few decimetres. Again, it was not clear to what extent material variations and increased ‘fuel duty’ contributed to this effect.

Moreover, an increased frequency in circumferential fractures was found in some BWR and PWR fuel types and added to discussions.

No severe degradation of the type discussed here was reported by WWERs or CANDUs.

The discussion that follows in this section summarizes work that has led to our current understanding of severe fuel degradation mainly in BWRs; a limited amount of data and experience regarding the degradation of PWR fuel has been published.

7.2. OBSERVATIONS FROM EXPERIENCE

7.2.1. BWRs

In GE barrier fuel performance summaries [7.3, 7.4], the following damage distribution was given for 83 failed rods of 3.36 million rods in service:

- 25 axial splits >6 inch (30% average);
- 3 circumferential fractures (3.6% average);
- 49 minor damage;
- 6 uninspected.

GE states that circumferential fracture had been previously observed at a low frequency and is normal in the case of severe hydriding damage. Axial splits often correlate to control rod withdrawal, whereas a power increase is not required to initiate a circumferential fracture in a heavily hydrided cladding area.

In a combined paper with EPRI, Anatech and GE [7.5], hot cell examination results for seven rods were reported, highlighting three failed and three non-failed barrier rods, and one failed non-barrier rod. The failure distribution is shown in Fig. 7.1. Two of the failed barrier rods were degraded to axial splits with fuel washout. The

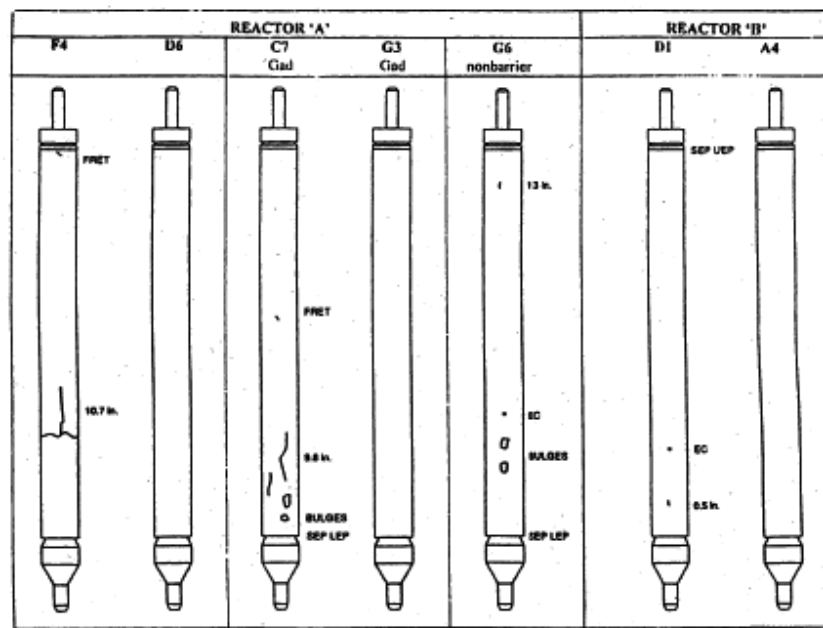


FIG. 7.1. GE failed fuel rods and symmetric sound rods examined in hot cells [7.5].

primary failure cause of the non-barrier rod and the non-degraded barrier rod was PCI. The other two rods had failed due to debris fretting.

All failed rods had suffered extensive secondary hydriding. The two degraded barrier rods showed evidence that most barrier oxidation had occurred after crack formation. The axial splits were initiated in a heavily hydrided region, but propagated in a brittle fashion through regions of moderately hydrided cladding. In the two non-degraded rods, the inner clad oxidation of the non-barrier rod was 30% to 40% heavier, but a comparison was difficult, since this rod operated in a failed condition significantly longer than the barrier rod. It was concluded that hydriding and hydrogen content in cladding assume a significant role, but the extent of cladding hydriding in itself is not sufficient to cause axial cracks. A severely hydrided rod, however, is degradable when there is a sufficient power rise. It was not possible to identify a simple direct correlation between one particular design feature and the propensity for degradation.

Since GE blamed cladding with small size precipitates for the splits, one of its remedies was 're-coarsening' of the cladding, using only a solution quench on the outer portion of the cladding to maintain small size precipitates for nodular corrosion resistance at the outside surface, while the iron content in the liner was increased for better steam corrosion resistance. Two fuel failures were observed with this coarsened cladding, with primary debris failures just below the top spacer. The first rod had no visible secondary damage, but had fractured at the lowest spacer level during handling. The second rod had two circumferential fractures 64 cm and 74 cm in elevation, but no splits were observed.

ABB made the following observations regarding the failures [7.6, 7.7]:

- (i) Seven axial splits >15 cm in liner and non-liner SVEA 64 fuel were observed, including five primary PCI failures. The largest split involved a sponge Zr liner rod. Splits were not observed in 10 x 10 fuels or in Zr-Sn liner fuel (a remedy for secondary degradation);
- (ii) Sixteen circumferential fractures were reported, all in the lower part of the rod with primary debris failures mainly in the upper part of the rod. There was no correlation to local or average maximum LHGR, and power suppression did not decrease degradation. Circumferential fractures were less severe than axial splits in terms of fuel washout;
- (iii) For five failures at high burnup >35 MWd/kg U, there was no or only minor degradation.

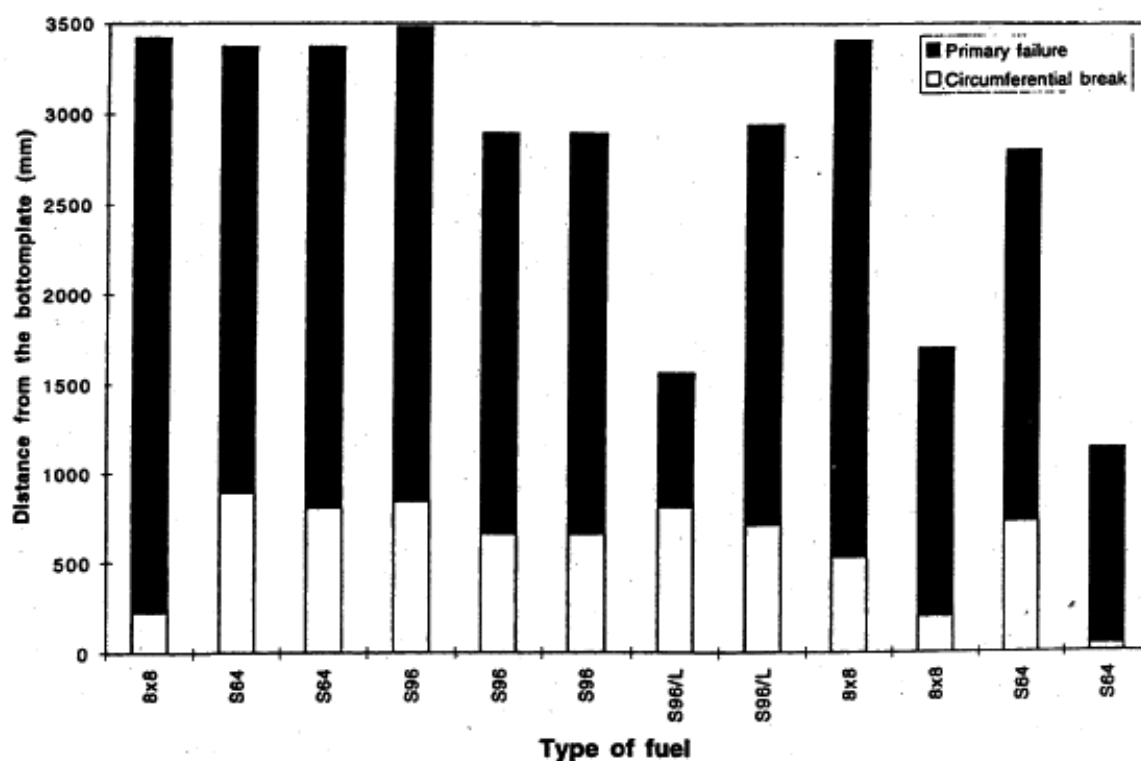


FIG. 7.2. Locations of primary failures and circumferential breaks on fuel rods for ABB SVEA fuel [7.7].

The frequency of both axial splits and circumferential fractures is judged to be high, which is supported by local TVO experience [7.8], with 14 failures in Olkiluoto-1, including one exception; a PCI failure in Siemens 9×9 fuel, and 11 failures in ABB SVEA fuel in Olkiluoto-2, mainly through debris fretting, but including three PCI failures in SVEA 64. Whereas the 9×9 fuel in Olkiluoto-1 did not show severe degradation, in Olkiluoto-2, the three PCI failures contained two rods with axial splits. The other eight failures (including two uninspected) revealed five rods with circumferential fractures.

A total of 18 failed rods were examined by ABB in hot cells [7.6, 7.7, 7.9]. Secondary hydrides and primary PCI cracks have functioned as incipience for crack propagation to axial splits. The very long split in the failed sponge Zr liner rod was attributed to large tensile hoop stresses in the cladding from rapid and extensive oxidation of the liner, thus promoting crack propagation. The circumferential fractures were a consequence of massive local pick-up of hydrogen and mainly occurred at low burnup, presumably due to the open pellet clad gap leading to good steam communication. The absence of severe degradation at high burnup was attributed to the closed pellet clad gap. Circumferential fractures occurred in the bottom part of the rod with primary fretting failures near the top, as is shown in Fig. 7.2.

The Krümmel plant (KKK) had a total of 10 failed barrier rods, as shown in Table 7.1 [7.10], with five degraded to axial splits >15 cm (ranging from 18 to 208 cm), two non-degraded, and four uninspected rods. Primary damage was believed to be caused by debris fretting, though this was not positively identified in PIE. Two failed rods with long axial splits were examined. One rod with a split had been in the reactor for nearly a year whilst the second had operated with an open crack for 29 days. The axial splits seen were considered to be secondary defects. The rods operated within PCIOMR, so PCI was not considered to be the root cause of the initial failures, but they did experience power ramps which may have influenced secondary cracking. One rod showed significant failure following a power manouever to reduce power in the other, pre-existing failed rod. [7.10].

From the KKK, the two rods with a spiral crack and a 95 cm axial crack (No. 2 and 3 in Table 7.1) were examined in hot cell, beside one sound rod [7.10]. It was concluded that oxidation of the barrier supported hydrogen production in the fuel rod, but that alone was not the root cause of the long axial split. It instead supported the theory of gap closure and tangential cladding stresses leading to strong radial orientation of hydrides observed in both fuel rods. Thus, a major cause of the axial splits was seen in the evenly distributed, as opposed to localized, hydrogen pick-up and the radial hydride orientation. Minor power ramps are then sufficient to start crack propagation.

TABLE 7.1. DEFECT BARRIER RODS AT KKK [7.10]

FA Number	Type	Burnup, MW·d/kg	Rod Position	Failure	Off-gas Activity
DA 039K	9 × 9-5	18.700	c1	180 mm split	moderate
JB 075K	9 × 9 QA	7.522	g1	250 mm spiral split	moderate to high
JB 039K	9 × 9 QA	10.640	k9	950 mm split	high
HA 144K	9 × 9 Q	27.422	f3	230 mm split	high
HA 057K	9 × 9 Q	36.000	g7	crack	moderate
HA 082K	9 × 9 Q	33.000	d7	40 mm split	moderate
KB 019K	9 × 9 QA	18.300	f9	2080 mm split	high
KB 077K	9 × 9 QA	17.610	—	not yet examined	moderate to high
HA 067K	9 × 9 Q	35.858	—	not yet examined	moderate to high
HA 087K	9 × 9 Q	41.431	—	not yet examined	moderate

In reports from all three hot cell programmes, it was emphasized that hydrogen produced through oxidation of the pellet surface contributes significantly to clad hydriding.

7.2.2. PWRs

In PWR fuel rods, the types of defect are essentially the same as those in BWRs except for severe axial cracks or splits which are less frequent than in BWRs. One of the rare cases of axial cracking is shown in Fig. 7.3. Typically, more secondary damage is located in the upper part of the rod in the region of high power and heat affected by the welding zone (upper end plug).

During the operation of Cattenom Unit 3 cycle 8 (15 Sept. 1999 to 27 Jan. 2001), a large number of fuel rods failed due to grid to rod fretting at the bottom grid level [7.11]. Coolant Xe-133 activity began to increase as early as October 1999 and continued to slowly increase in magnitude through June 2000. This coolant activity was consistent with the formation of primary defects caused by grid to rod fretting. From June through to the end of the cycle, coolant activity of short lived soluble fission products (I-134) and actinides increased considerably, indicating the development of secondary degradation. During on-site examinations, several large secondary defects such as circumferential breaks were observed on peripherals rods of leaking FAs. Two examples of circumferential break are shown in Fig. 7.4. The secondary break, perpendicular to the axis of the rod and on the edge of the broken area, indicated that the rod broke off at a pellet to pellet interface.

Since then, several similar defects have been observed on high burnup rods affected by fretting problems at lower grid levels in EDF reactors.

Similar secondary hydriding damage by circumferential cracks has also been observed in some United States of America PWRs over the last several years.

7.3. DEGRADATION CHARACTERISTICS

Primary failures

The sequence of events prior to secondary degradation is fairly well understood and documented in a number of publications [7.13–7.17]. The process begins with a primary defect that allows coolant to enter the fuel rod. Any type of primary breach can, in theory, lead to a secondary failure. Primary failure modes include grid to rod fretting, debris fretting, PCI cracks and manufacturing defects (e.g. tubing reduction flaws, weld porosity, etc.). With significant improvement in fuel manufacturing processes and widespread quality initiatives, manufacturing defects

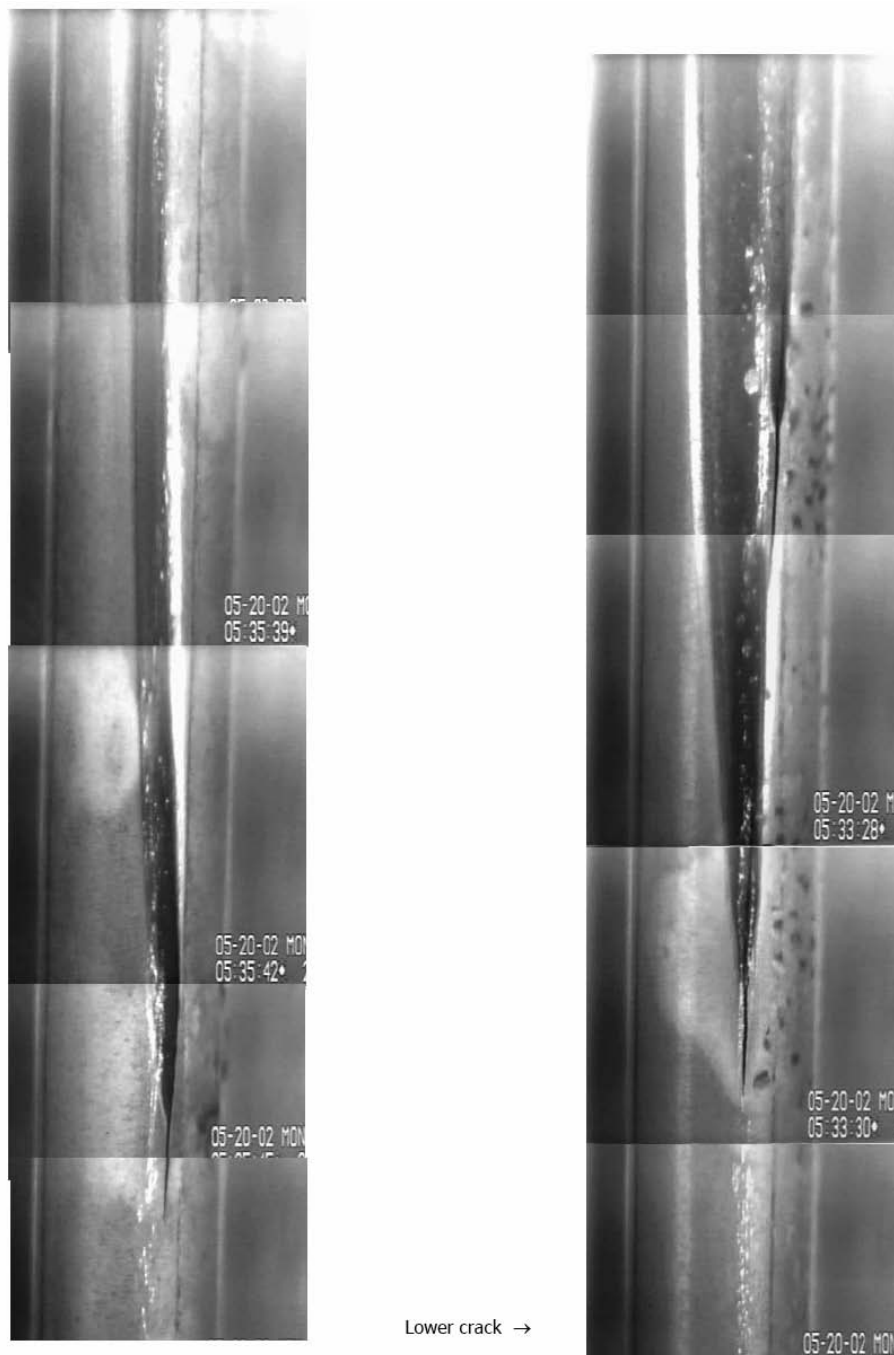


FIG. 7.3. Example of an axial split failure in a PWR fuel rod [7.12].

have been dramatically reduced in recent years. In addition, extensive application of barrier fuel (and operating restrictions on non-barrier fuel) has considerably reduced the frequency of PCI defects. Today, fretting is the most common initiator of primary failure. This is significant because, unlike other types of defects, frets are not sharp enough to initiate a longer crack. When degradation follows a fretting failure, it is always at a secondary site.

Steam starvation

When water enters a fuel rod through a primary defect, it flashes to steam and begins to react with the fuel and inside surface (IS) of the cladding. Steam continues to enter the rod until equilibrium with the system pressure is reached. Oxygen is stripped from the steam through the two simplified corrosion reactions below (a process known as steam starvation):

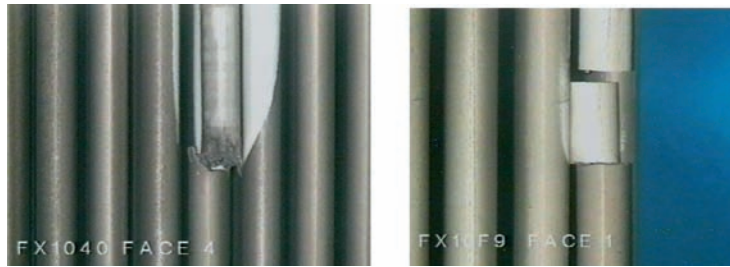


FIG. 7.4. Example of circumferential (left) or 'guillotine' (right) break.

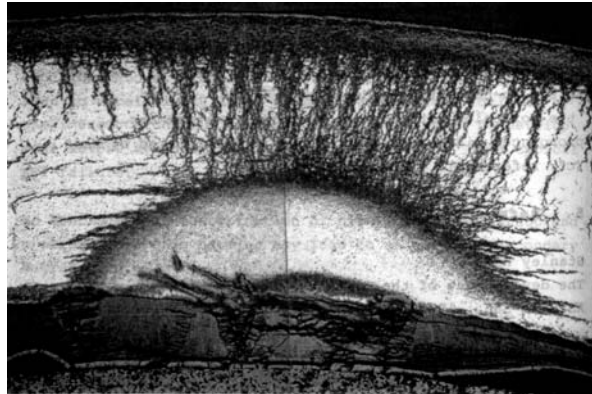
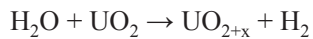
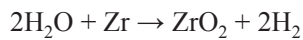


FIG. 7.5. Massively hydrided region with hydride sunburst [7.21].



Steam is also subject to radiolytic decomposition, generating additional hydrogen and hydrogen peroxide.

The gas mixture becomes continuously enriched in hydrogen, with a maximum concentration some distance from the primary defect.

Hydrogen absorption

When the gas mixture becomes sufficiently enriched in hydrogen, absorption occurs by breaking down the once protective oxide on the inside of the cladding. The exact hydrogen to steam ratio at which rapid absorption occurs (the 'critical ratio') has been found to depend on a number of variables, including material type, thickness and integrity of the oxide and test temperature and pressure [7.18–7.20]. In a series of tests at 350°C and 7 MPa (the BWR system pressure), Kim et. al. [7.20] found the critical ratio $\text{H}_2/\text{H}_2\text{O}$ for sponge zirconium to be between 1000 and 5000. Above this value, massive hydriding was inevitable, although it could be somewhat delayed with thicker pre-oxidized films.

Once conditions for absorption are reached, hydrogen is rapidly absorbed into the cladding. This gas phase absorption mode is much faster than the hydrogen diffusion rate in the cladding and blisters on the inside surface are formed. In the presence of a thermal gradient, the hydrogen is slowly transported to the outside surface (OS). Given enough time, all hydrogen above the solubility level will move to the OS and (potentially) form hydride blisters. At intermediate stages of the process, the hydrogen is distributed in a 'sunburst' pattern as shown in Fig. 7.5. The sunbursts are often found to align with cracks in the fuel pellet, which is likely a function of hydrogen (and fission gas) access to the cladding surface

Modes of secondary failure

Post-irradiation examinations of failed fuel rods have highlighted several stages in secondary degradation. The different types of secondary damages affecting failed rods are classified in the following way:

a. 'Sunburst'

This failure reflects localized hydriding radiating from the cladding. An illustration is given in Fig. 7.6.

b. 'Blister' or 'bulges'

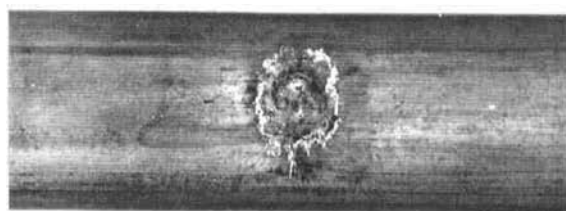
This is a local increase in the volume of cladding reflecting massive hydriding, typically with radial hydride precipitation under the 'blister'. An illustration of this type of defect is presented in Fig. 7.7.

c. Perforation or holes

The final stage in the evolution of the 'sunburst' or 'blister' is perforation or holes; fuel is visible. Some examples of perforations are presented in Fig. 7.8.

d. Small cracks

These can be of different types, longitudinal, transverse or circumferential and often leave a 'sunburst'. The different types of small cracks observed are presented in Fig. 7.9.



(a)

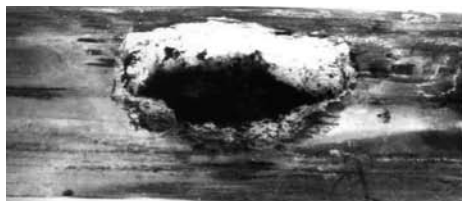


(b)

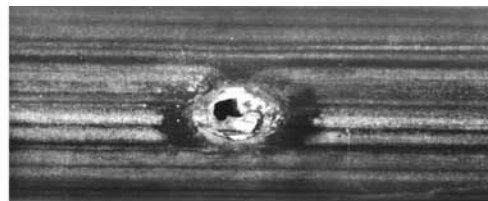
FIG. 7.6. Sunburst.



FIG. 7.7. Bulges.

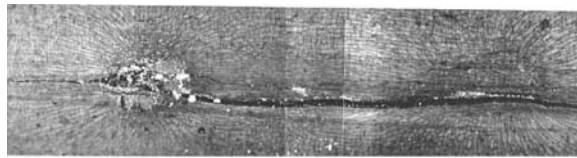


(a)



(b)

FIG. 7.8. Holes.



(a) Axial crack



(b) Transverse crack

FIG. 7.9. Small cracks.

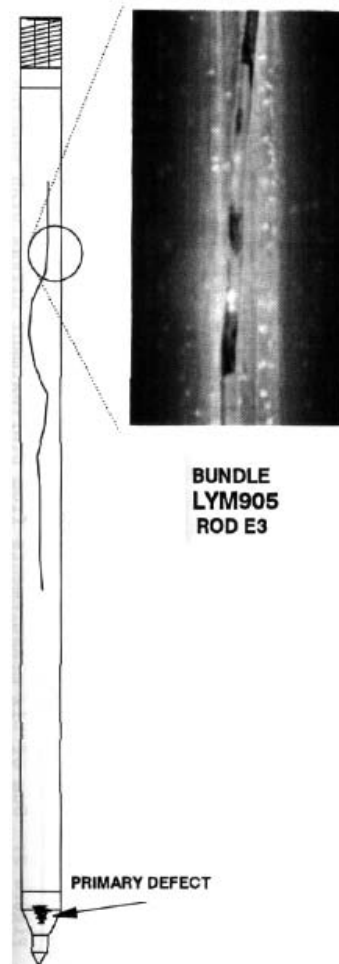


FIG. 7.10. Schematic and poolside visual of an axial split from a Hatch-2 reactor [7.22].

e. Severe degradation

Deterioration of a fuel rod beyond the previous stages can lead to two specific forms of severe degradation. One form is axial split, an example of which is shown in Fig. 7.10 in a BWR fuel rod. Axial splits can form either by extension of a primary defect (called a 'propagating primary') or by initiation and growth from a secondary hydride. The path chosen depends on the acuity of the defect, stress distribution along the rod and the local environment inside the fuel rod. Propagating primary splits are particularly interesting because the splits form (by definition) in an oxidizing environment away from any massive hydriding.

The particular significance of long axial cracks is that they tend to cause a high release of off-gas activity and can also result in large coolant contamination through greater fuel loss.



FIG. 7.11. Open long axial split from a PWR fuel rod [7.11].

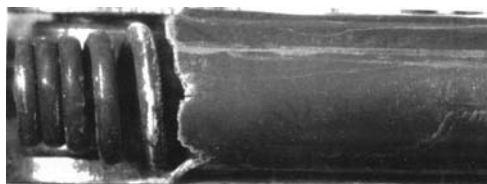


FIG. 7.12. Fracture at an upper end plug welding [7.11].

Although less frequent, a few cases of secondary axial splits were also reported in PWRs. An example of an open axial split observed on a PWR failed fuel rod is shown in Fig. 7.11. The hole, or cracked hydride blister, in the middle of the split suggests the crack propagated in both directions after nucleation at the location of the hydrides.

A second form of degradation is the circumferential break, in which cladding is massively hydrided around enough of the circumference to literally break into two sections.

For BWRs, these failures tend to occur in rods early in life, when the fuel-cladding gap is still open, and in the high power region of a rod [7.23]. Here the pellet-cladding gap is smallest and the hydrogen absorption rate is highest [7.14]. Hydride concentrations can be further localized in cladding at pellet-pellet interfaces that operate at slightly lower temperatures than the bulk of the cladding. The stress that produces the break is likely a combination of volumetric expansion (associated with the phase change to a zirconium hydride) and thermal expansion of the fuel column. The nominally open pellet-clad gap can be closed by oxidation, reduced conductivity and volume expansion of the hydrided region itself. Because of the susceptibility of zirconium to form hydrides in the presence of dry hydrogen gas, this type of failure is not easily mitigated.

For PWRs, secondary hydriding damage through a circumferential crack as shown in Fig. 7.4 has also been observed, especially in high burnup fuel rods.

In PWRs, circumferential breaks are frequently observed in the heat affected welding zone of the upper end plug. Figure 7.12 shows a fracture at the weld position of an upper end plug; no lower end plug fracture was found.

Since there is very little information published on circumferential breaks, the remainder of this review will focus on axial splits.

PIE characteristics

Over the years, a consistent description of degraded cladding has been developed from post-irradiation examination (PIE) at several hot cell facilities, including Studsvik [7.24, 7.25], GE Vallecitos [7.26, 7.27] and INER [7.28].

A low magnification SEM image of two fracture surfaces from a split in the Chinshan reactor is shown in Fig. 7.13.

The curved, periodic markings (wrongly called ‘striations’ by some) have been found on all splits reported to date and are a key identifying feature of the crack mechanism. More recent fractography by GE at Vallecitos on well preserved split fracture surfaces has indicated that curved features are always perpendicular to the crack front and are more accurately described as chevron patterns [7.27]. These findings have since been confirmed by an examination at Studsvik [7.25]. As shown in Fig. 7.14, chevron patterns (or half chevron patterns, when a crack leads close to the outer surface) define local crack advance through the varying micro-structural features encountered along the crack tip. A cross-section of the crack, shown schematically in Fig. 7-14c, shows that chevron patterns are roughly square channels with a brittle fracture along surfaces in the fracture plane and a ductile fracture along vertical surfaces connecting the channels [7.25, 7.27].

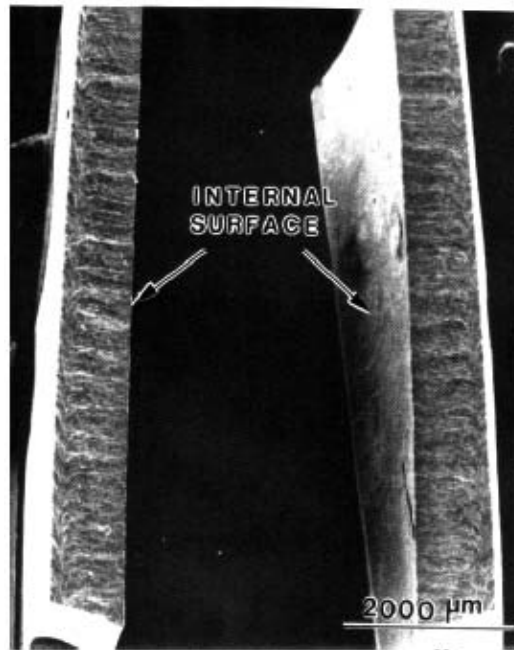


FIG. 7.13. SEM fractography of a split [7.28].

Overall, the fractures are quite brittle and radial through the cladding. As shown in Fig. 7.15, the only significant ductility is at the inner surface. In this case, the barrier exhibits a knife edge failure. More recent work by Studsvik has shown that non-barrier cladding also exhibits some ductility (measurable as wall thinning) at the inner surface [7.25].

A metallography specimen from a split in Oskarshamn 3, shown in Fig. 7.16, revealed hydrides concentrated at and aligned with the crack tip. This turned out to be a common observation in degraded cladding [7.24, 7.28], and suggested a DHC-like mechanism was operating. Unfortunately, it could not be proven that hydrides precipitated at the crack tip were involved in the fracture, since they could have just as easily precipitated during cool down in the stress field of the crack tip.

The key characteristics of axial splits based on present knowledge are summarized below [7.13, 7.29]:

- Axial splits have mainly been observed in BWR fuel rods;
- Crack initiation occurs at sharp flaws (massive hydrides, PCI defects, etc.);
- Distinctive fractography is characterized by a chevron pattern;
- Crack advance through a combination of axial and radial propagation with crack fronts lead to near the outside surface;
- Macroscopically, brittle crack propagation is indicated by radially oriented fracture surfaces and little measurable plastic deformation;
- Fracture surfaces are brittle in appearance since they consist of quasi-cleavage facets (or at least of featureless micro-areas, which are similar to quasi-cleavage facets) and at a microscopic scale show a perfect fit of the two opposing surfaces;
- Hydrogen (or hydrides) is/are involved in the fracture process;
- Brittle crack propagation in cladding often occurs well away from massively hydrided regions, with hydrogen as low as ~150 ppm;
- Sections at the tip of axial clad splits show precipitation of hydride in front of the crack tip;
- Ductile separation is observed in the zirconium liner (slight ductility also observed at the inner surface of non-barrier cladding).

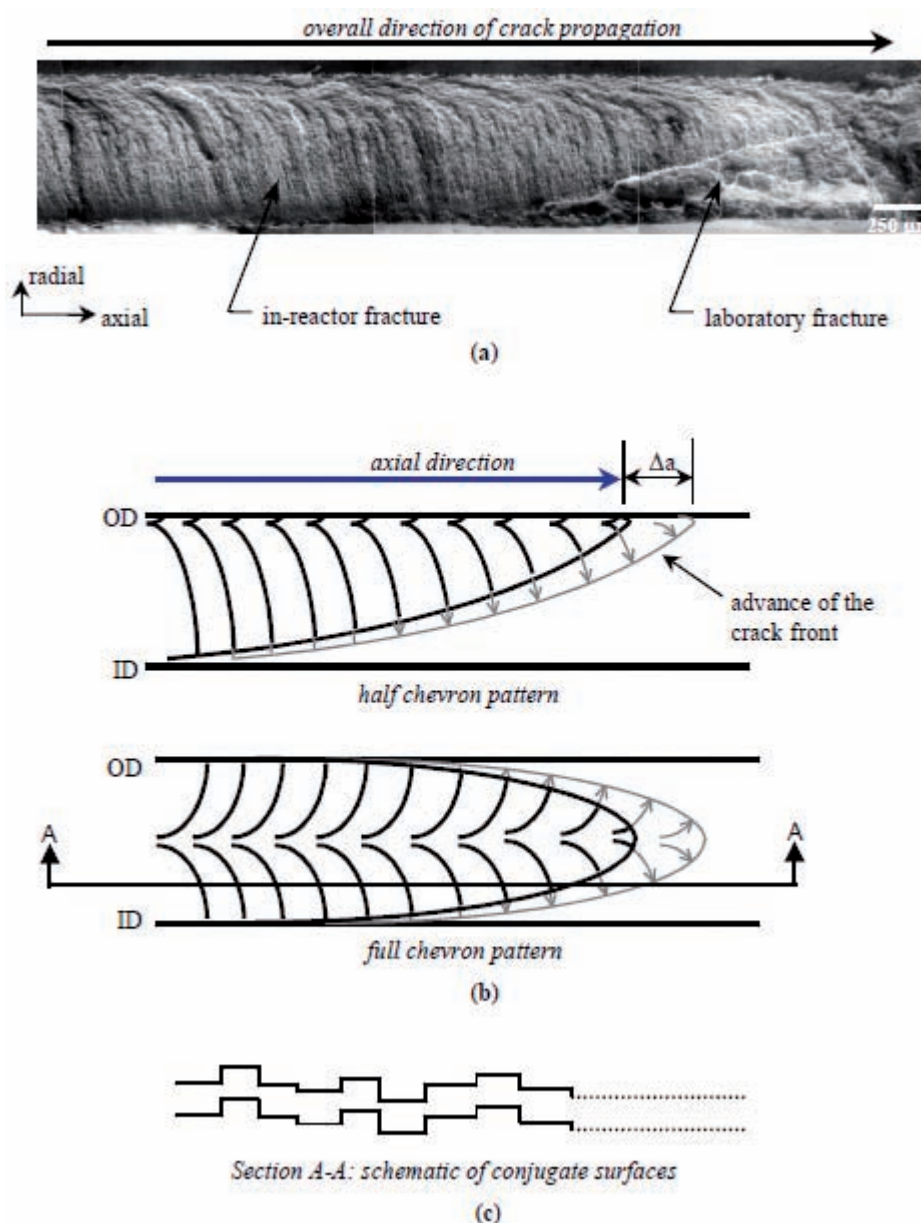


FIG. 7.14. (a) SEM fractography of a crack tip in better preserved splits with schematics of crack fracture patterns in (b) and (c) [7.25, 7.27].

7.4. MECHANISMS

7.4.1. AXIAL SPLIT

(1) Embrittlement of the cladding [7.13, 7.29]

A number of mechanisms have been proposed to explain axial splitting. A good review of environmental mechanisms in zirconium alloys was made by Cox [7.30]. Some failures could be simply explained by the massive hydriding associated with secondary failure.

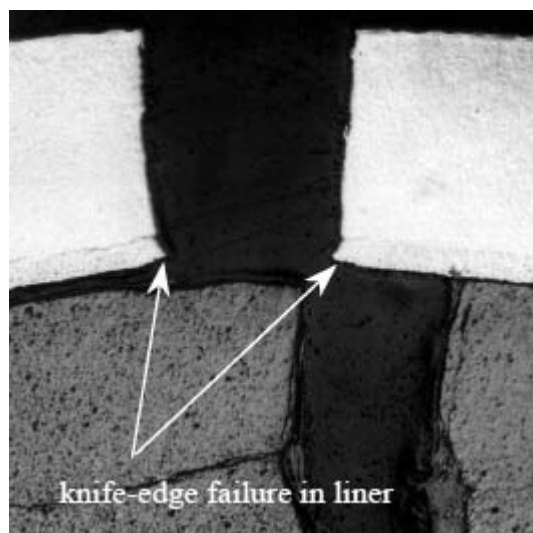


FIG. 7.15. Transverse metallography from a split revealed a brittle failure through the matrix and ductile in the liner [7.27].

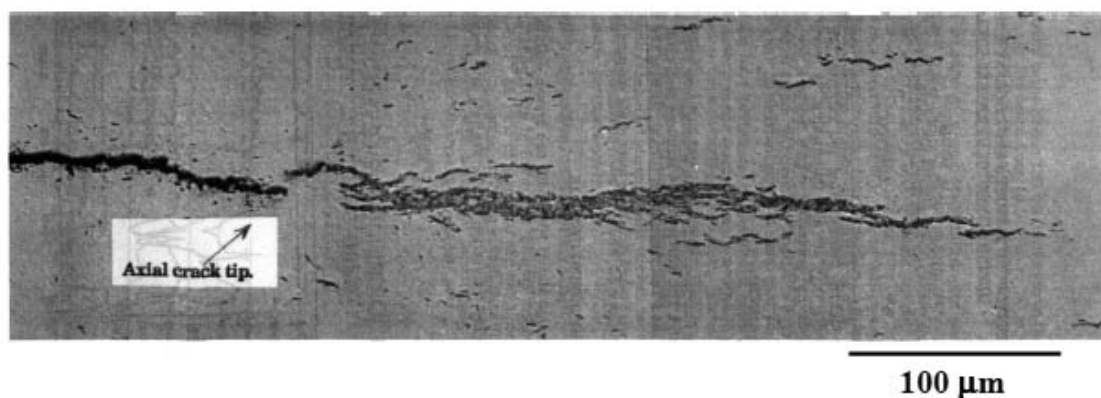


FIG. 7.16. Metallography at the crack tip reveals precipitated hydrides [7.21].

However, post-irradiation examinations (PIEs) have shown that the cracks often extended outside of the massively hydrided regions. A somewhat lower concentration of hydrides aligned with the fracture plane (i.e., radial hydrides) could explain brittle fracture outside of massively hydrided regions. However, this was not corroborated by PIE, which revealed splits running through regions with no observable radial hydrides. In fact, regions with 150 ppm total hydrogen had almost no precipitated hydrides present at the operating temperature.

One proposed mechanism is based on embrittlement of cladding material through irradiation damage and hydriding, and enhanced by secondary phase particle (SPP) dissolution [7.31, 7.32]. However, the experimental evaluation of resistance to axial crack propagation of cladding with different SPP characteristics [7.33] did not reveal any large embrittlement of the cladding with small SPP or hydrogen levels up to 1000 ppm. Recent fracture toughness tests, performed at Studsvik by means of the Pin-Loading Tension (PLT) technique, showed that the fracture toughness of cladding which had experienced axial splitting did not differ much from the toughness of unirradiated cold worked zircaloy cladding.

Based on these fracture toughness measurements, and since quasi-brittle axial splits can even run in a cladding with a hydrogen content as low as 150 wtpm [7.34, 7.35], one could conclude that the bulk properties of the cladding itself do not appear to be the reason for axial splitting.

(2) Delayed cracking of cladding [7.13]

Delayed hydride cracking

The DHC mechanism [7.36, 7.37] is generally seen to be a two step process. In the first step, hydrogen is attracted to a crack tip by a crack tip stress field. It then precipitates locally when the terminal solid solubility for precipitation (TSSp) is reached¹. When the precipitated hydrides reach a critical size, crack advance occurs. In the classic DHC picture, each crack advance is marked by blunting of the crack tip as the crack remains under load, while additional hydrogen accumulates and finally precipitates to further advance the crack. Since these markings indicate the location of the crack front at different times, they are properly referred to as striations [7.38].

The primary reason for skepticism regarding the role of DHC in degraded cladding failures is that the mechanism had not been demonstrated to occur in zircaloy-2 at operating temperatures (i.e., 288°C near the OS and 320°C or even higher at the IS). For example, a study by Mahmood and co-workers indicated that DHC could not be achieved above 250°C with 90–110 ppm total hydrogen (although results were subsequently published indicating that TSSp for irradiated materials could be higher) [7.39].

There were also questions about fractography when crack surface features were shown to be in chevron patterns rather than the striations generally associated with DHC.

A breakthrough paper in this area was presented in 1995 by Efsing and Pettersson [7.40]. They performed DHC-type tests on unirradiated, recrystallized and stress-relieved annealed (RXA and SRA) zircaloy-2 cladding. At 300°C, they achieved crack growth with many features of DHC, including striations, an incubation time before crack growth and crack velocity that would be expected by extrapolating from lower temperature DHC tests. An over temperature of 60°C was required and the pre-charged hydrogen was more than high enough to ensure hydrogen in a solution above TSSp. The authors went on to publish work on irradiated materials [7.41, 7.42] with similar results. The main factor in their success appears to be the large over temperature required to reach TSSp at higher test temperatures.

Primary criticism of the work centred on whether this actually represented ‘classical’ DHC, particularly since the hydrogen levels were so high (500–1000 ppm). However, the key point was that brittle fracture was achieved at BWR operating temperatures with crack speeds and fracture toughness levels consistent with expectations. Interestingly, the stronger materials (i.e., SRA and irradiated) tested at 300°C failed because of DHC, but did not exhibit striations.

Corrosion hydrogen cracking (CHC)

A breakthrough in degradation came from work at GE Vallecitos [7.27] on irradiated and unirradiated cladding using the split propagation laboratory investigation test (SPLIT). In these tests, a length of cladding with a central pre-crack was subjected to an internal hoop stress with a split mandrel while it underwent corrosion and hydrogen charging. In order to accelerate the corrosion reaction at the 300°C test temperature to in-reactor rates (on the inner surface (IS) of a failed fuel rod), lithium hydroxide was added to the water. The cracks that resulted were exact matches to the splits in degraded cladding in every detail. One such example is shown in Fig. 7.17.

More recent results from the SPLIT test demonstrated that, like DHC, crack advance occurs along hydrides precipitated at a crack tip. It was further shown that the required hydrogen comes from bulk corrosion [7.43] and not a more localized attack at the crack tip (as occurs with hydrogen gas cracking (HGC)). It was suggested that the main significant difference between CHC and DHC is the route taken to reach a condition in which hydride precipitation can occur (i.e., to reach TSSp).

DHC testing, at least above critical temperatures, has historically reached TSSp through an over temperature, while CHC achieves it directly with corrosion hydrogen. This is an important distinction, since fuel cladding does

¹ The exact conditions of precipitation are still somewhat controversial. One view is that the TSSp is unaffected by stress and precipitation occurs first at the crack tip due to a locally higher concentration of hydrogen in equilibrium with the stress field (i.e., TSSp is reached locally even though bulk hydrogen may be lower). A second view is that TSSp itself is influenced by stress (i.e., reduced at the crack tip).

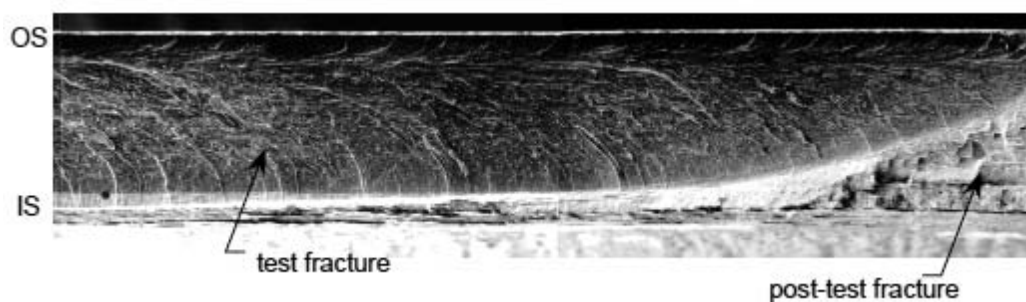


FIG. 7.17. Fractography from SPLIT test [7.27].

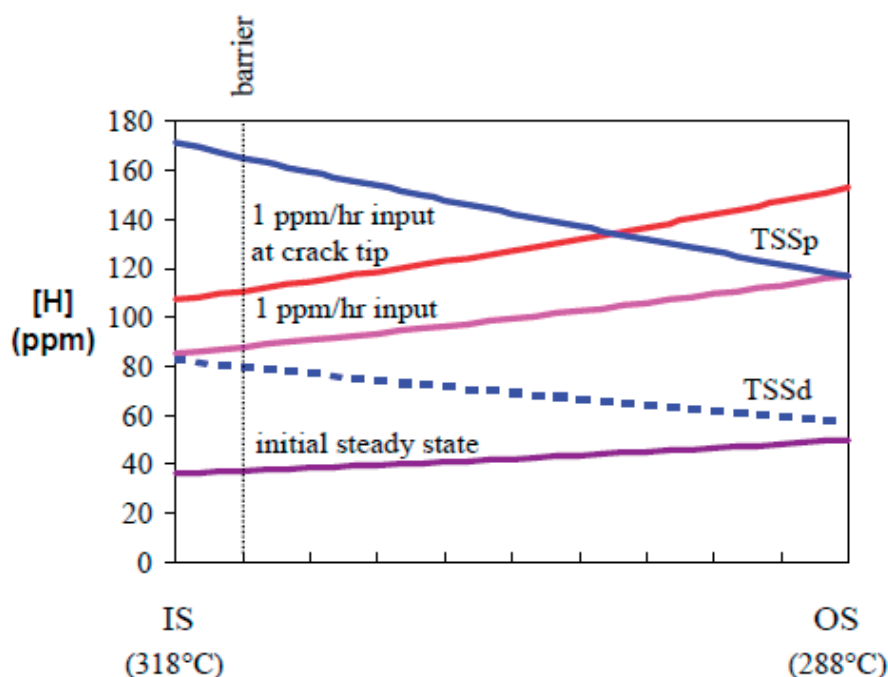


FIG. 7.18. Calculated hydrogen distribution [7.43] through the cladding wall for a temperature difference of 30°C across the cladding wall.

not undergo an over temperature. However, the bigger implication of an active source of corrosion hydrogen is that it counters the ability of the thermal gradient to drive the hydrogen in solution towards the outer wall and deplete the bulk of the matrix. This process is shown for a temperature difference of 30°C in Fig. 7.18. The calculation began from an initial steady state hydrogen level and proceeded with constant hydrogen input (i.e. from IS corrosion) until TSSp was reached at the outside surface. Under these conditions, hydrogen precipitation could only occur at the outside surface. With a crack tip present, hydrogen concentration is locally elevated and TSSp can be reached somewhat away from the outer surface (OS). However, these calculations only explain crack propagation through the outer 1/3 of the cladding wall. To achieve through wall propagation, an additional mechanism needs to be evoked. There appear to be two reasonable explanations. First, although the cracks are very tight, there could be some crack tip cooling; if the crack tip temperature is held close to the coolant temperature, precipitation can occur throughout the cladding wall and through wall propagation is easily explained. The second possibility is that once the hydrides nucleate at the outer surface with TSSp, hydride growth can occur with much less super-saturation of hydrogen. There is some evidence for this in literature [7.44], where it is written that to maintain growth, hydrogen concentration stays closer to the TSS line for dissolution (TSSd). The example in Fig. 7.18 shows concentration at the crack tip to be well above this line. In either case, the thermal gradient would be expected to accelerate crack growth by sweeping hydrogen towards the crack tip. A crack growth process relying purely on hydrogen in solution (in which hydrogen would have to diffuse to the crack tip against the thermal gradient) would be, by comparison, much slower.

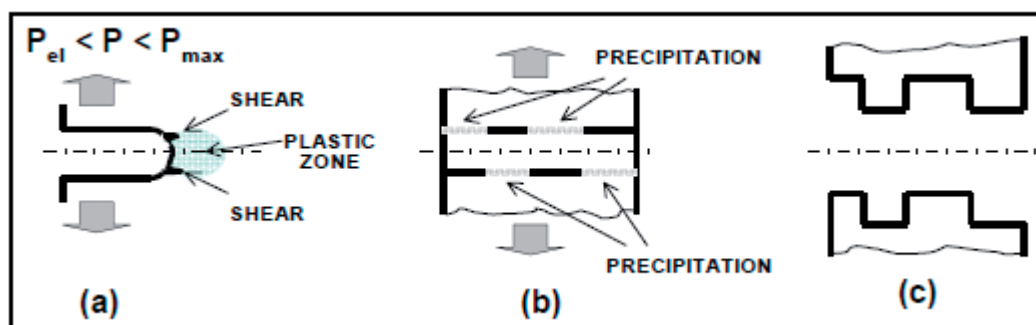


FIG. 7.19. Schematic of crack surface topography resulting from material separation in areas of shear distortion where material separation is facilitated due to hydride precipitation (according to the HALS mechanism).

Hydrogen assisted localized shear (HALS)

The HALS mechanism proposed by V. Grigoriev excludes precipitated hydrides from the cracking process [7.45]. In other words, as soon as hydride is precipitated, it follows matrix ductile behaviour without having an embrittling effect. The most deteriorating effect at the crack tip appears to be produced by precipitation itself, when an interface boundary between the matrix and precipitating hydride is created as a result of corporate movement of zirconium atoms created by the strength of the hydrogen–hydrogen interaction. The precipitation of hydride is accompanied by the creation of intensive shear stresses in the surrounding matrix. However, those stresses can cause material separation only when they are superimposed on shear stresses from external forces. Thus, according to the HALS mechanism, local plasticity at the tip of a growing crack is a required prerequisite for creating sufficient shear stresses at the crack tip. One could suppose that hydride precipitation under combined tension–shear stresses creates a weakened plane in the matrix, which is an interface boundary between the hydride and matrix. The following separation of the matrix along that plane might be compared to the lamination of a composite material and might be able to produce a quasi-brittle micro-structural appearance on the fracture surface.

In addition to the microscopic characteristics of the fracture surface, the HALS mechanism appears to be able to explain the topography of crack surfaces observed with axial splits. When a moderate load P is applied, a well-defined plastic zone is created at the notch or crack tip with two symmetrical areas of shear distortion (Fig. 7.19(a)). Precipitating hydrides facilitate material separation within the micro-areas above and below the plane of a macro-crack (Fig. 7.19(b)). Thus, a macro-crack is supposed to propagate as a self-balanced system of micro-cracks located above and below the plane of symmetry [7.46]. In such a case, cross-sectioning perpendicular to the macro-crack plane will show the relief, which is similar to that observed for actual in-reactor splits (Fig. 7.19(c)).

The HALS mechanism does not distinguish pre-existing hydrogen and hydrogen from ongoing corrosion. However, the provocation of hydrogen in relation to corrosion at the crack tip may be important, since shear distortion at the crack tip appears to be most intense near the crack surface, decreasing away from the crack tip.

Crack velocity

The question of crack velocity has been of interest primarily for two reasons. First, it can provide valuable insight into the crack propagation mechanism (although, now that the mechanism is understood, it should be possible to estimate crack velocity). In addition, it is useful for a utility in determining how quickly to react to fuel failure. Unfortunately, there are no direct observations of crack velocity. The best estimates from in-reactor data are lower boundaries based on the length of split and in-reactor time. Assuming that all cracks initiated in the centre of the final crack and grew at a constant velocity in both directions, estimates are in the range of 2.5×10^{-7} to 6.6×10^{-7} m/s (0.9 to 2.4 mm/hr) [7.28, 7.27]. At the highest rate, one could expect a 12 cm crack in one day.

Crack velocity estimates from what we understand of the mechanism are also in this range. Maximum DHC rates reported by Efsing and Pettersson [7.41] on irradiated cladding at 300°C are 9.5×10^{-7} m/s. They have also reported DHC crack growth rates on heavily cold worked unirradiated cladding as high as 5×10^{-6} m/s. For the SPLIT test, values of 2×10^{-7} to 3×10^{-7} m/s have been reported [7.27, 7.43]. However, since crack velocity is not

directly measured in the SPLIT test, these values are also averages of crack length and total time. Maximum crack speeds during the test were estimated to be two to three times higher (i.e., 6×10^{-7} to 9×10^{-7} m/s) [7.43].

Although lab results for DHC/CHC tests are in good agreement with lower bound estimates for in-reactor crack velocity, in-reactor rates could still be higher. One important factor is that the driving forces of stress field and thermal gradient can be superimposed to move hydrogen to the crack tip. Another factor is that the cladding stress state is probably more severe for fuel cladding than for either DHC or SPLIT tests.

Source of stress

In 1991, in an ANS paper by Jonsson and co-workers [7.21], it was concluded that the combination of high power, poor corrosion resistance of zirconium liners and (interestingly) low burnup were detrimental to secondary degradation resistance. A paper from the same conference by Davies and Potts [7.47] suggested that cladding, and especially fuel oxidation (decreasing fuel thermal conductivity and increasing fuel thermal expansion), play a key role along with rod power. By the 1994 ANS conference, experience based on degraded fuel characteristics had grown substantially. A paper by Schrire et. al. [7.24] stated that corrosion of a high purity zirconium liner increases cladding stress through (1) volume expansion of the oxidized liner and (2) thermally insulating the fuel column. They also measured fuel pellets from the split region and found an increase in diameter consistent with roughly half of the crack opening (the other half was from liner oxidation).

Industry consensus seems to be that all these factors are important to a greater or lesser extent, depending on failure specifics. A clear correlation has been shown between local rod power changes and the location of split initiation [7.48]. Indeed, some shorter cracks can be completely explained by local power changes. Given estimates for crack velocity and the fact that most cladding hoop stress relaxes through creep in one to two days, the maximum crack length expected from a single power change is in the order of 12–24 cm.

Splits that are significantly longer than 12–24 cm without power histories to explain the length (e.g. Oskarshamn-3, Hatch-2) require an additional mechanism of stress [7.21, 7.27]. Factors such as cladding–pellet gap closure due to cladding and fuel oxidation and reduced fuel conductivity of oxidized UO_2 , all contribute to the magnitude of stress on cladding during a power change. However, these factors do not directly influence the duration of elevated stress levels and this is key to explaining longer splits. The best explanation is that volume expansion of oxide through corrosion is faster than relaxation due to creep (as noted by Schrire and others above). The conceptual difficulties with this explanation were: (1) how the oxide could continue to grow when steam access was restricted by the filled cladding–pellet gap, and (2) how the oxide of high purity zirconium liners could support such stress when they were described as ‘fluffy’ by PIE. To check the effect of expanding oxide on cladding stress levels, simple corrosion tests were performed at Vallecitos.

In these tests, different types of cladding were loaded with tight fitting UO_2 pellets ($\sim 75 \mu\text{m}$ initial diametral gap) and corroded in 400°C steam. The specimens were recrystallized tubing with two low iron zirconium barriers (100 and 400 ppm) and one non-barrier. The results are shown in Fig. 7.20. The largest effect was observed for the 100 ppm Fe barrier specimen, which was visibly deformed by corrosion. In this specimen, the entire barrier was consumed in less than 300 hours. The effect on 400 ppm Fe barrier specimens was much smaller, but still significant on two of the three lots of tubing. The non-barrier specimen never did develop pellet–cladding contact and the pellets fell out after the test. Post-test metallography of the 100 ppm Fe specimen indicated the oxide was only 60% of theoretical density. This may well be the key to oxide imposed cladding stress (i.e. an oxide porous enough to allow continued transport of steam and dense enough to support cladding stress).

Figure 7.21 is a summary of liner oxide thickness in failed cladding as measured by PIE. The liner oxidation rates (or IS rates in the case of non-barrier cladding) were calculated from the average of all measurements published for a given rod divided by total time in a failed condition. (Note: it was not clear how to include the ABB Sn liner on a plot against iron levels, so it was arbitrarily placed at 400 ppm). While one might have anticipated considerable scatter (due to differences in the failure mode, locations of metallography specimens, etc.), the overall trend is what would be expected. The trend line is a linear regression fit to $1/\text{Fe}$ and has no physical significance. Unfortunately, except for the one point with the ABB Sn liner, no data were available on the current generation of alloyed liners.

It appears that much of the increase in frequency and extent of damage in fuel degradation can be attributed to rapid corrosion of the unalloyed zirconium liner. This is clearly the case for long axial splits. However, depending on the specific failure, other factors have had important roles. These factors include increased ‘fuel duty’

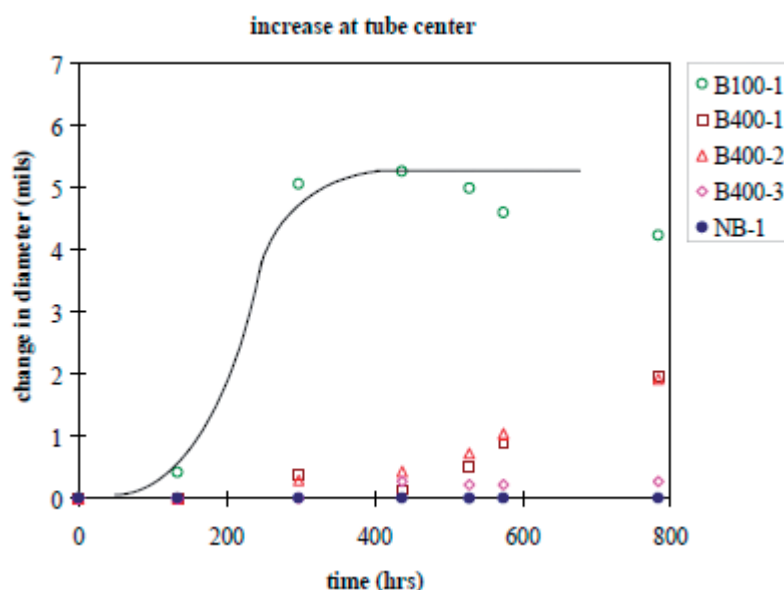


FIG. 7.20. Net increase in tubing diameter for barrier cladding with 100 ppm Fe ("B100"), 400 ppm Fe ("B400") and non-barrier (NB) cladding [7.49].

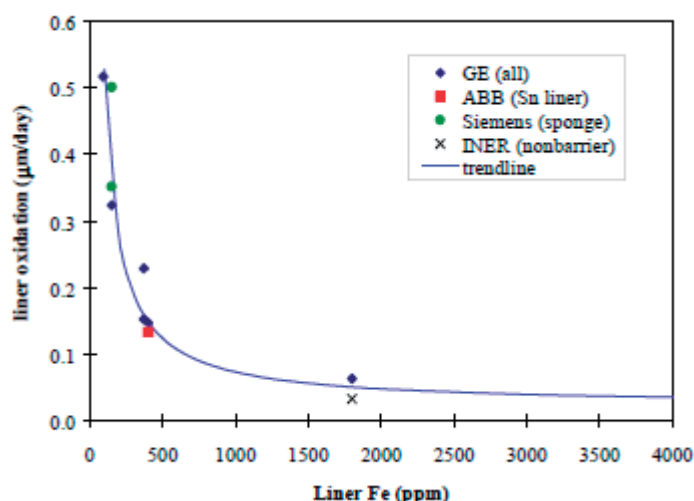


FIG. 7.21. Summary of liner oxide thickness data from PIE results published by ABB [7.50], GE [7.51–7.54], INER [7.28] and Siemens [7.55].

(no operating restrictions on full barrier fuel cores), higher cladding burnups (primarily pellet-cladding gap closure) and the increased frequency of debris fretting as a failure mechanism.

7.5. CIRCUMFERENTIAL FRACTURES

There is broad agreement in literature that circumferential fractures are the consequence of heavy local hydriding leading to cracks and eventually to rod fractures when pellet cladding contact has been established in an area [7.1].

GE discussed the mechanism in more detail [7.4]. Early life failures should be more susceptible to circumferential fracture due to the still open pellet cladding gap and good communication conditions along the rod, especially when the primary defect is far away from the peak power location, i.e. when the location of heavy secondary hydriding by oxygen starvation coincides with the peak power location where the pellet cladding gap

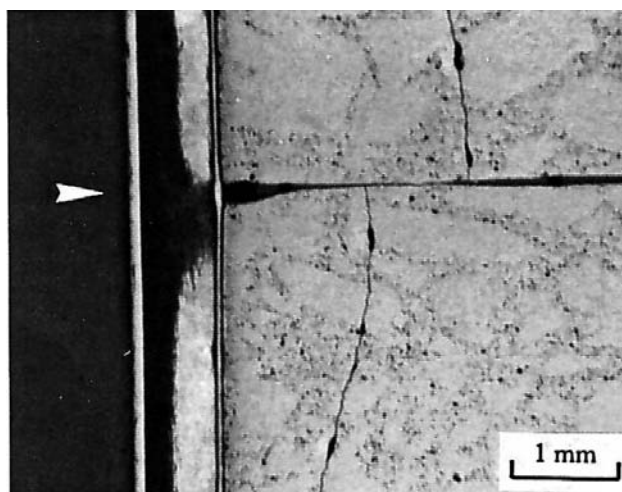


FIG. 7.22. Hydride repartition at a pellet to pellet interface [7.56].

closes earliest. This implies that a primary failure at the top more likely leads to fracture, since peak power is usually near the bottom. In fact, circumferential fractures are found in the bottom range, and GE has observed — with a primary failure near the bottom — heavily hydrided cladding in the top range of a rod without fracture, as no pellet clad has contact in this range.

This fully agrees with ABB observations and a similar description in [7.9], as reported circumferential cracks were found in the bottom area in conjunction with debris fretting at a higher axial level. According to Fig. 7.2, the average distance between primary and secondary failure was around two metres, which is quite compatible with oxygen starvation conditions. Of course, one cannot exclude an additional material effect on susceptibility to circumferential cracks, e.g. by the degree of hydrogen pickup and mode of hydriding. However, there is currently no direct evidence for a material effect.

Some years ago, EDF sent a broken fuel rod to hot cell for further examination in order to determine the cause of failure and to explain the fracture mechanism of the fuel rod [7.56]. The rod burnup was about 51 GW·d/t U. Metallographic investigations were performed on the axial sample and a radial cut was located close to the rupture.

It was shown that the high burnup rod had initially failed due to grid to rod fretting in the lower part of the rod. Examinations confirmed that secondary hydriding of the cladding was the cause of these fractures. Severe secondary hydriding of the cladding caused the formation of hydride rings at pellet to pellet interfaces, as shown in Fig. 7.22. At the pellet to pellet interface, cladding areas are typically cooler than in adjacent areas and hydride precipitates preferentially, thus forming hydride rings. The hydride causes severe embrittlement of the cladding, which when then submitted to a stress, has a tendency to break at a pellet to pellet interface.

REFERENCES TO SECTION 7

- [7.1] ONOUFRIEV, V., VON JAN, R., “Survey on fuel failures in water cooled power reactors in 1995–98 (CANDUs, BWRs, PWRs, and WWERs)”, IWGFPT/51, IAEA, Vienna (2000).
- [7.2] INTERNATIONAL ATOMIC ENERGY AGENCY, Review on Fuel Failures in Water Cooled Reactors, IAEA Technical Reports Series No. 388, IAEA, Vienna (1998).
- [7.3] POTTS, G.A., “Recent GE BWR fuel experience”, ANS 1997 Int. Top. Mtg on LWR Fuel Performance, Portland, 1997, ANS, Le Grange, IL, (1997) 261.
- [7.4] MONTES, J.A., POTTS, G.A., “Failed fuel degradation”, TopFuel '97 — BNES Conf. Manchester, 1997, p. 5.17.
- [7.5] YAGNIK, S.K., et al., “Assessment of BWR fuel degradation by post-irradiation examination and modelling in the defect-code”, ANS 1997 Int. Top. Mtg on LWR Fuel Performance, Portland, 1997, p.329.
- [7.6] VESTERLUND, G., HELMERSSON, S., “ABB fuel design and development”, OECD Specialists’ Meeting on Nuclear Fuel and Control Rods, Madrid, 1996, paper 4.A.1.
- [7.7] SIHVER, L., et al., “Recent ABB BWR failure experience”, ANS 1997 Int. Top. Mtg on LWR Fuel Performance, Portland, 1997, p. 356.

- [7.8] LUNABBA, R., MUTTILAINEN, E., "TVO's experience with fuel and control rods", OECD Specialists' Mtg on Nuclear Fuel and Control Rods, Madrid, 1996, paper 1.4.
- [7.9] SIHVER, L., et al., "Recent ABB BWR SVEA fuel failure experience", TopFuel '97 — BNES Conf., Manchester, 1997, p. 5.1.
- [7.10] HÜTTMANN, A., et al., "Post irradiation examination of failed KKK barrier fuel rods", ANS 1997 Int. Top. Mtg on LWR Fuel Performance, Portland, 1997, p. 350.
- [7.11] DANGOULEME, D., Personal communication.
- [7.12] KILLEEN, J., Personal communication.
- [7.13] EDSINGER, K., "A Review of Fuel Degradation in BWRs", Proc. of the Int. Top. Mtg on Light Water Reactor Fuel Performance, Park City, 2000, ANS, LaGrange Park, IL (2000).
- [7.14] LOCKE, D.H., The behaviour of defective reactor fuel, *Nuc Eng and Design* **21** (1972) 318–330.
- [7.15] DAVIES, J.H., "Secondary damage in LWR fuel following PCI detection — characteristics and mechanisms", IAEA Spec. Mtg on Behavior of Defected Zirconium Alloy Clad Ceramic Fuel in Water Cooled Reactors, IWGFPT-6, 1980, pp. 135-140., Chalk River, Ontario, Canada, September 17-20, 1979: Transactions of the ANS, 1982, pp. 269–270.
- [7.16] CLAYTON, J.H., "Internal hydriding in irradiated defected zircaloy fuel rods", 8th Int. Symp. on Zirconium in the Nuclear Industry, STP 1023, (SWAM, L.F.P., EUKEN, C.M., Eds), ASTM, Philadelphia (1989) 266–288.
- [7.17] OLANDER, D.R., et al., Chemical processes in defective LWR fuel rods, *J Nucl. Mater.* **248** (1977) 214–219.
- [7.18] ZIMA, G.E., "A Review of the Zircaloy-2 Hydriding Program", HW-66537 (1960).
- [7.19] UNE, K., LESS, J., *Common Metals* **57** (1978) 93.
- [7.20] KIM, Y.S., WANG, W.E., OLANDER, D.R., YAGNIK, S.K., High pressure hydriding of sponge-Zr in steam-hydrogen mixtures, *J Nucl. Mater.* **246** (1977) 43–52.
- [7.21] JONSSON, L., HALLSTATIUS, B., GRAPENGIESSER, M., LYSELL, G., "Failure of a barrier rod in Oskarshamn 3", Proc. Int. Top. Mtg on LWR Fuel Performance, ANS, Avignon, 1991, pp. 371–377.
- [7.22] TURNAGE, K.G., HUNT, B.E., BENNETT, J.D., GIBSON, E.B., "Fission product analysis and operational experience with leaking fuel rods at the Hatch Nuclear Unit", Proc. Int. Top. Mtg on LWR Fuel Performance, West Palm Beach, 1994, ANS, La Grange, IL (1994) 467–476.
- [7.23] VESTERLUND, G., HELMERSSON, S., "ABB BWR fuel design and development", 7th Annual Conf., Mexican Nuclear Society (1996) SP15–21.
- [7.24] SCHRIRE, G., LYSELL, G., FRENNING, G., RÖNNBERG, L., JONSSON, A., "Secondary defect behavior in ABB BWR fuel," Proc. Int. Top. Mtg on LWR Fuel Performance, West Palm Beach, 1994, ANS, La Grange, IL (1994) 398–409.
- [7.25] LYSELL, G., GRIGORIEV, V., PETTERSSON, K., "Characteristics of axial splits in failed BWR fuel rods", 9th Int. Conf. on Environmental Degradation of Materials in Nuclear Power Systems — Water Reactors, ANS/NACE/TMS, 1999.
- [7.26] DAVIES, J.H., ARMIJO, J.S., "Post-irradiation examination of failed KKL fuel rods", KTG Fuel Conference, Karlsruhe, 1993.
- [7.27] EDSINGER, K., DAVIES, J.H., ADAMSON, R.B., "Degraded fuel cladding fractography and fracture behaviour", 12th International Symposium on Zirconium in the Nuclear Industry, ASTM STP 1354 (SABOL, G.P., MOAN, G.D., eds) Philadelphia, PA, ASTM, (1999).
- [7.28] LIN, K.F., et al., "Investigation on the post-defect deterioration of non-barrier BWR failed rods", Proc. Int. Top. Mtg on LWR Fuel Performance, West Palm Beach, 1994, ANS, La Grange, IL (1994) 377–390.
- [7.29] LYSELL, G., GRIGORIEV, V., EFSING, P., "Axial splits in failed BWR fuel rods", Proc. of the Int. Top. Mtg on Light Water Reactor Fuel Performance, Park City, 2000, ANS, LaGrange Park, IL (2000).
- [7.30] COX, B., Environmentally-induced cracking of zirconium alloys — a review, *J Nucl. Mater.* **170** (1990) 1–23.
- [7.31] MONTGOMERY, R.O., RASHID, Y.R., OZER, O., "Evaluation of post defect fuel behaviour", Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, 1994, ANS, La Grange, IL (1994) 447.
- [7.32] WILLIAMS, C.D., et al., 11th ASTM Mtg on Zirconium in the Nuclear Industry, Garmisch, 1996, 676.
- [7.33] GRIGORIEV, V., JOSEFSSON, B., ROSBORG, B., "Fracture toughness of Zircaloy cladding tubes", 11th Int. Symp. on Zirconium in the Nuclear Industry, ASTM STP 2195 (1996) 431–447.
- [7.34] EDSINGER, K., DAVIES, J.H., ADAMSON, R. B., "Degraded fuel cladding fractography and fracture behaviour", ASTM meeting on Zirconium in the Nuclear Industry, Toronto, 1998.
- [7.35] LYSELL, G., GRIGORIEV, V., "Characteristics of axial splits in failed BWR fuel rods", 9th Int. Conf. on Environmental Degradation of Materials in Nuclear Power Systems — Water Reactors, TMS, Newport Beach, 1999.
- [7.36] DUTTON, R., PULS, M.P., "A theoretical model for hydrogen induced sub-critical crack growth", Effects of Hydrogen on Behaviour of Materials, AIME (1976) 516–525.
- [7.37] EADIE, R.L., SMITH, R.R., "Modelling delayed hydride cracking in Zirconium alloys", *Can. Metall. Q.* **27** (1988) 213–223.
- [7.38] NUTTALL, K., ROGOWSKI, A.J., Some fractographic aspects of hydrogen-induced delayed cracking in Zr-2.5 wt% Nb alloys, *J. Nucl. Mater.* **80** (1979) 279–290.
- [7.39] McMINN, E.C., DARBY, SCHOFIELD, J.S., "Terminal solubility of hydrogen in Zirconium alloys", 12th Int. Symp. on Zirconium in the Nuclear Industry, ASTM STP 1354, (SABOL, G.P., MOAN, G.D., Eds) Philadelphia, PA, ASTM (1999).
- [7.40] EFSING, P., PETTERSSON, K., "The influence of temperature and yield strength on delayed hydride cracking in hydrided Zircaloy-2", Zirconium in the Nuclear Industry, ASTM STP 1295 (BRADLEY, E.R., SABOL, G.P., Eds.) ASTM (1996) 394–404.

- [7.41] EFSING, P., PETTERSSON, K., “Delayed hydride cracking in irradiated Zircaloy cladding”, 12th Int. Symp. on Zirconium in the Nuclear Industry, ASTM STP 1354 (SABOL, G.P., MOAN, G.D., Eds.) Philadelphia, PA, ASTM (1999).
- [7.42] EFSING, P., “Delayed hydride cracking in irradiated Zircaloy”, PhD thesis, KTH, Stockholm (1998).
- [7.43] EDSINGER, K., VAIDYANATHAN, S., ADAMSON, R.B., “On the mechanism of axial splits in failed BWR fuel rods”, 9th Int. Conf. on Environmental Degradation of Materials in Nuclear Power Systems — Water Reactors, 1999, ANS/NACE/TMS.
- [7.44] SHI, S.-Q., SHEK, G.K., PULS, M.P., Hydrogen concentration limit and critical temperatures for delayed hydride cracking in Zirconium alloys, *J. Nucl. Mater.* **218** (1995) 189–201.
- [7.45] GRIGORIEV, V., JOSEFSSON, B., On the mechanism of Zircaloy cladding axial splits, *J. Nucl. Mater.* **257** (1998) 99–107.
- [7.46] GRIGORIEV, V., PETERSSON, K., “On the indications of shear fracture at the tip of an axial notch in Zircaloy cladding”, 9th Int. Conf. on Environmental Degradation of Materials in Nuclear Power Systems — Water Reactors, TMS, Newport Beach, 1999.
- [7.47] DAVIES, J.H., POTTS, G.A., “Post-defect behavior of barrier fuel”, Proc. Int. Top. Mtg on LWR Fuel Performance, Avignon, 1991, ANS, La Grange, IL, (1991) 272–284.
- [7.48] ARMIJO, J.S., “Performance of failed BWR fuel”, Proc. Int. Top. Mtg on LWR Fuel Performance, West Palm Beach, 1994, ANS, La Grange, IL (1994) 410–422.
- [7.49] EDSINGER, K., unpublished work, Vallecitos Nuclear Center, 1998.
- [7.50] LYSELL, G., SCHRIRE, D., SIHVER, L., “Disposition of oxygen and hydrogen arising from water ingress in a failed Sn-alloy Zirconium liner fuel rod”, HPR-349, 1998.
- [7.51] DAVIES, J.H., HOSHI, E.V., REAGER, R.D., ZIMMERMAN, D.L., WARNER, R.W., “Post-irradiation evaluation of BWR fuel from the Hatch-1 Reactor — Final Report”, EPRI TR-106036 (1995).
- [7.52] DAVIES, J.H., HOSHI, E.V., REAGER, R.D., ZIMMERMAN, D.L., WARNER, R.W., “Post-irradiation evaluation of BWR fuel from Hope Creek Reactor”, EPRI TR-106348 (1996).
- [7.53] EDSINGER, K.W., DAVIES, J.H., FARKAS, D.M., REAGER, R.D., HOSHI, E.V., internal GE report (1997).
- [7.54] SCHLEUNIGER, P., et al. internal GE/PSI report (1994).
- [7.55] HÜTTMANN, M., et al., “Post-irradiation examination of failed KKK-barrier fuel rods”, Proc., Int. Top. Mtg on LWR Fuel Performance, Portland, 1997, ANS, La Grange, IL (1997) 350–355.
- [7.56] BOURGOIN, J., VAN SWAM, L.F., BOURNAY, P., “Hot cell analysis of fuel rod fractured during handling in the pool”, Colloque de FONTREVAUD III.

8. FUEL FAILURE PREVENTION AND MANAGEMENT IN PLANT OPERATION

8.1. EXISTING OPERATING LIMITS AND RECOMMENDED PRACTICES

Fuel failures can affect plant operating activities. Paragraph 8.1 describes existing operating limits due to fuel failure, and recommendations for operating policies and practices taken to prevent fuel failures and to minimize their effects or propagation. Some of these are extracted from the work of EPRI to develop guidelines for improving fuel reliability and for operation with failed fuel [8.1].

8.1.1. Coolant activity

All plants must operate within their technical specifications for failed fuel as defined by the limit on coolant activity. These technical specifications on coolant activity are imposed by a regulatory body. Allowable coolant activity during normal operation can be different from plant to plant and from country to country. Typically, the 'dose equivalent iodine' shall be $\leq 0.2 \mu\text{Ci/g}$ (7.4 GBq/t) in United States of America BWRs [8.2] and $\leq 1.0 \mu\text{Ci/g}$ (37 GBq/t) in United States of America PWRs [8.3].

In France, more severe limits were imposed by French Safety Authorities after a multiple failure incident involving Cattenom 3 at cycle 8. For 1300 MW(e) nuclear power plants, new activity limits are set in terms of:

- Dose equivalent iodine ($< 20 \text{ GBq/t}$);
- The sum of noble gas with different limits as a function of burnup of failed rod estimates through caesium ratio during a power transient ($< 100 \text{ GBq/t}$ if $^{134}\text{Cs}/^{137}\text{Cs} > 1.4$ or $< 50 \text{ GBq/t}$ if $^{134}\text{Cs}/^{137}\text{Cs} < 1.4$);
- The increase in ^{134}I activity due to fuel failure after subtraction of ^{134}I activity due to uranium tramp ($< 1 \text{ GBq/t}$). This indicator is used to limit the dissemination of new fuel material in primary coolant.

For WWER reactors, the operational failure limit of fuel rods with defects of the gas leak type is 0.2%, and in the case of direct contact between fuel and coolant, it is 0.02%. Total iodine isotope activity of the primary circuit would then correspond to $1.5 \times 10^{-2} \text{ Ci/kg}$ for the WWER-1000. For the WWER-440, the specific activity of non-gaseous fission products at this point would reach $1 \times 10^{-4} \text{ Ci/kg}$. Such levels of coolant activity have never been reached or observed.

These limitations on the specific activity of reactor coolant or off-gas minimize personnel exposure and ensure that the resulting doses at the site boundary will not exceed an appropriately small fraction of dose guideline values following a reactor accident.

The iodine spiking phenomenon after plant transients has received particular attention in safety evaluations owing to its potential to increase the radiological consequences of postulated accidents. Therefore, a limit to the amount of iodine activity allowed in reactor coolant after some plant transients is specified in many LWRs. To control iodine spiking in some CANDUs, utilities have placed a limit on the number of defects in the core for operation at high power. Other CANDUs have reduced the shutdown limits for ^{131}I .

In addition, the majority of LWR plants are not authorized to startup with failed fuel. They must identify, locate and remove failed fuel before restarting. These activity limits on current licensing are well below the design limits of plant systems in which radwaste circuits are designed to accommodate 1% of fuel rod failures.

8.1.2. Guidelines for power changes and load following operation

For LWRs, one of the most important fuel performance aspects of power changes and load following operation is PCI in fuel rods. Indeed, power changes are directly accompanied by dimensional changes of the fuel and cladding in both directions.

The physical phenomenon at the origin of the risk of PCI failures results from differential thermal dilations between pellet and cladding. The pellet dilates and retracts much more quickly than the cladding during power variations. When power is lowered for a long time, the cladding comes to slowly restick through creep. This

mechanism, which takes some hours to a few days, is called deconditioning and leads to a new steady state condition at a lower power level. In the event of a quick increase in power, the pellet pushes violently on the cladding and the chips of pellet locally create constraints which can become excessive by initiating cracks through stress corrosion if such power ramps are not limited to acceptable levels. If the ramp rate is limited, the fuel rod may adapt to a higher linear heat generation rate (LHGR) level by creep and relaxation of the fuel and cladding. This mechanism, which takes some hours to a few days, is called conditioning and leads to a new steady state condition at a higher power level. The conditioning power level is defined as the equilibrium LHGR at which neither conditioning nor deconditioning of the fuel occurs (corresponding to equilibrium between pellet swelling and clad creep). The kinetics depends on rod design and power history.

Because of complex fuel rod behaviour, the PCI effect depends on pre-irradiation history and especially on conditioning power. Cyclic power variations below the pre-conditioning power level are safe against the risk of PCI. Only power ramps or local power overshoot beyond this level may lead to a risk of PCI failure. In all cases, some form of conditioning is required, and the allowable ramp rates are different for conditioned and unconditioned fuel. A more complete description of thermal mechanical fuel rod response to power changes is described, for example, in Ref. [8.4].

With the aim of protecting fuel against pellet-cladding interaction, most fuel vendors or utilities provide recommended power manoeuvring restrictions. These restrictions are generally in the form of prescribed preconditioning rules, limitations of operating time at reduced power to limit fuel deconditioning, ramp rates and power level holds to prevent excessive pellet-clad interaction for different operating conditions such as load following operation. Special attention is given at the first startup after handling of the fuel assemblies, where more severe power ramp rates are applied. For the majority of plants in the United States of America, power ramp rate restrictions (3 percent/hour) for unconditioned fuel begin at 40% power or less. Following recent PCI failures observed at some PWRs in the United States of America, more conservative ramp rates can now be used for these reactors. For EDF plants, power ramp restrictions begin at 50% power. Above 50% power, there are also limits on rod movement (3 steps per hour) in several plants.

The need for continuous PCI surveillance is much more pronounced in BWRs than in PWRs. One important feature of PWRs is the possibility to operate the core at constant load without deeply inserted control rods, since burnup compensation takes place through boron control. Fast load changes can be performed by temporarily inserting one or several groups of control rods, called D or grey banks, without noticeable local power perturbation. Moreover, the most modern plants are equipped with a sophisticated or even an automatic control system for power distribution using direct input from in-core detectors. They can then operate with a very low power spike and there are few restrictions on load following operation except for starting up after refuelling. Recent experience shows that only limited PCI failure has been observed in PWR fuel of modern design (see Section 5).

Moreover, in France, where more than 80% of electricity is produced by nuclear reactors and where plants have to keep up to grid demands and operate using frequency control, actions necessary to meet these needs such as

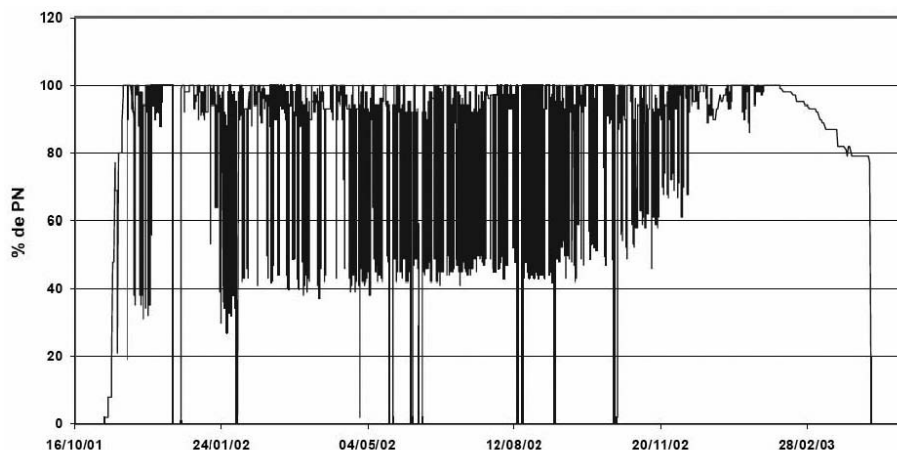


FIG. 8.1. Example of a typical power history during a cycle in EDF reactors [8.5].

daily load follow and extended reduced power operations have shown that the fuel failure rate is not affected by these operations or by the control rod movements associated with them [8.5].

BWRs are more subject to the PCI problem than PWRs. Indeed, most of the time they must operate with deeply inserted control blades used for burnup compensation. The movement of control blades induces more distortions in the power distribution of adjacent fuel rods and necessitates the use of operating rules to reduce the risk of PCI failures. These rules mainly imply limitations on the power increase rate during startup after refuelling or after control rod sequence exchange and on control rod withdrawal speed in a high power region when fuel is not preconditioned.

In 1972, GE developed and implemented its Pre-Conditioning Interim Operating Management Recommendations (PCIOMRs), to contain PCI risk in non-barrier fuel [8.6]. These recommendations essentially aim to provide cladding conditioning to avoid excessive local stress during power increases. The Siemens criteria for safe fuel operation with respect to PCI are based on the RSST approach (safe range, safe speed, safe steps and safe time), presented in Ref. [8.7]. ASEA-ATOM related operating restrictions are described in Ref. [8.8]. These rules are similar in nature but different in detail. The main difference resides in the fact that European BWRs are equipped with a screw drive for the control blades which allows fine motion with acceptably small steps from the PCI point of view, while GE plants have a notch drive with 15 cm steps, with the result that each control blade movement induces larger local power changes.

FRAMATOME ANP introduced POWERPLEXIII/MICROBURN-B2 core design and monitoring software in 2002. This software improves the capability to predict local power distribution changes during control blade movements. It also more accurately accounts for fuel conditioning/deconditioning states during long operating intervals with minimal control blade movement [8.9].

For CANDUs, all normal operations are carried out so that all bundle power conditions in a core do not exceed thresholds values.

Today, these rules are less important owing to the use of remedies such as Zr liner fuel in BWRs or CANLUB coating in CANDUs.

8.1.3. Optimizing the loading pattern

Optimizing the loading pattern and undertaking fuel reshuffling strategies can mitigate fuel degradation.

For example, for reactors which are susceptible to grid to rod fretting failure, some utilities have developed fuel shuffling strategies that mitigate vulnerability to grid to rod fretting failures until more robust fuel designs are in place. Elements of this strategy include development of a minimum peripheral assembly power ratio (ratio of fuel assembly power in the third cycle over power in the second cycle) in an effort to minimize the effect of fuel rod creep down.

As shown in Section 6 on PWRs, FA bow is strongly dependent on FA location in the core. Burnup has only a mild effect on global deformation of an FA. The bow of one assembly does not evolve in the same direction throughout its life. It is affected by its successive position in the core. Thus, some utilities have developed fuel shuffling strategies that mitigate vulnerability to fuel assembly bow by changing the FA of a core quadrant at each cycle.

In BWRs, channel bow is due to neutron flux gradient during irradiation and depends on core location history and channel exposure. As a result of neutron flux gradients across fuel cells, the fluence accumulated by a channel has an uneven planar distribution. Since irradiation growth depends on both material properties and fluence, channels bow to accommodate resulting uneven growth. Small variations in heat treatment, material composition or texture tend to accentuate differential growth and bow. Fuel management can partly account for these effects and help to prevent excessive bow or unacceptable interference using control blades. In several plants, specific ‘channel management procedures’ have been developed and applied. An example of such methodology is described in Ref. [8.10].

8.1.4. Guidelines for prevention of severe degradation

Fission product release from failed fuel and severe degradation can be minimized by reducing the power level of failed rods. In BWRs, it is often possible to locate the region or regions in the core that contain failed fuel by using the flux tilting method. Once those regions have been identified, it is possible to reduce the power of fuel

assemblies in those areas through selective placement of control blades. This practice has mainly been applied to locate leaking Zr liners or other sensitive fuel as early as possible in order to prevent severe degradation.

A set of guidelines or recommendations has been formulated by EPRI to prevent the formation of secondary defects, and to limit the consequences of degradation should a secondary failure occur [8.11]. The guidelines recommend: (a) close monitoring of primary failures using key indicators; (b) prompt identification of the core location containing the failures using flux tilting; and (c) local power suppression. GE [8.12] has also issued similar guidelines. The mitigation effectiveness provided by power suppression is greatest when applied to relatively small, tight defects. For this reason, detection and suppression should be undertaken as early as possible once the presence of a leaking fuel rod is indicated in a core.

In extreme cases, plant derating or early shutdown may be advisable in order to mitigate the effects of extensive fuel degradation and fuel loss. The effectiveness and impact of various mitigating actions on subsequent degradation of failed barrier and non-barrier BWR rods are discussed in Ref. [8.13].

To prevent further fuel degradation, additional limits on normal operational ramp rates are imposed by some utilities when failed fuel is present, based on their own experience.

Fuel rod failures in PWRs have generally been less severe with respect to secondary degradation leading to large levels of radioactivity release into coolant than in BWRs. This may be a consequence of different thermal-hydraulic conditions, cladding materials and the environment inside a failed rod, which exist in BWRs and PWRs. Reference [8.14], related to past French experience, shows good stability of reinserted defective fuel rods in EDF PWRs.

Another point of particular importance is assessment of impact of load following operation on the processes leading to secondary degradation. Although several authors and fuel vendors recommend stopping load follow to mitigate secondary degradation, statistical information taken from French experience, where reactors operate with frequency control, daily load follow, and extended reduced power operation, does not show a significant impact of this mode of operation on secondary degradation in PWRs. This is in agreement with modelling, which supports the conclusion that load follow operations favour the ingress of water or steam into fuel cladding, thus delaying the time in which the critical internal rod conditions of oxygen starvation (and hydrogen excess) required for massive secondary hydride formation are reached. A heavily hydrided region is brittle and may break when exposed to stress. The stress that produces a break can either be self-induced as a result of cladding volumetric expansion associated with the phase change to zirconium hydride or it can be caused by pellet cladding mechanical interaction (PCMI). PCMI might be an effect of oxidation, volume expansion of a hydrided region as well as thermal expansion of a fuel column resulting from a power increase and/or reduced thermal conductivity.

To avoid degradation in CANDUs, the channel containing a fuel defect is identified and refuelled at-power without derating [8.15].

8.1.5. Water chemistry controls

PWRs and BWRs

Although fuel failures attributable to primary water chemistry conditions have only occurred in isolated cases in the last decades, effective control of water chemistry in the primary system is crucial for maintaining good fuel performance. In addition, because of the possibility of intrusion from connected systems such as the spent fuel pool, control of water chemistry in these connected systems is also important.

The primary water chemistry concern for fuel performance is the potential for corrosion related cladding failure and associated excessive hydrogen pickup with resultant cladding embrittlement. Operating experience shows that the likelihood of excessive cladding corrosion depends greatly on reactor water chemistry conditions. For example, CILC fuel failures have occurred in BWR plants with high concentrations of copper in the feedwater and/or chemical transients. Other chemistry intrusions during operation can similarly increase the chances of failure, especially if existing cladding and water chemistry conditions are conducive to corrosion. Corrosion related cladding failures have affected BWRs much more than PWRs. Recent experience has shown, however, that at high burnup, PWR cladding corrosion can exceed its design limits, especially when the fuel is utilized under relatively high duty conditions and at high water temperatures.

The boundaries of acceptable regimes of water chemistry, cladding materials, operating temperature and heat flux have not yet been fully defined, except for specific local applications. However, the Electric Power Research Institute (EPRI), together with utility and industry specialists, has developed a set of water chemistry guidelines. These guidelines take into account the current state of knowledge on how water chemistry affects the entire system, including fuel. Water chemistry specifications are also provided by most nuclear steam supply system (NSSS) vendors and fuel suppliers.

In recent years, several water chemistry changes, such as elevated pH/lithium concentrations and zinc injection in PWRs, hydrogen water chemistry (HWC), noble metal chemical addition (NMCA) and zinc injection in BWRs have been introduced, or are being considered to optimize light water reactor plant performance through a reduction in radiation fields and minimization of irradiation assisted stress corrosion cracking (IASCC) or intergranular stress corrosion cracking (IGSCC) of plant components. These water chemistry changes for plant material protection and dose rate reduction associated with a trend to increase 'fuel duty' can have potential or known implications for fuel performance, which is why they need to be monitored to ensure they will not compromise fuel rod integrity. To this end, the effects of chemistry changes on the corrosion and hydriding behaviour of fuel cladding and crud deposition on fuel rods have been evaluated in an extensive fuel surveillance programme involving vendors, utilities and different organizations such as EPRI. A summary of results obtained in this programme is given in Refs [8.16–8.22].

In BWRs, no fuel impact was found with hydrogen water chemistry (HWC) alone [8.2, 8.17]. The combination of NMCA and Zn has been found to lead to thick tenacious crud in some cases [8.2, 8.18]. These results have led to recommendations on limits of noble metal additions (NMCA) and zinc injections to minimize risks to fuel [8.2, 8.19].

Lithium concentrations in PWRs have been increasing with increasing cycle length, leading to increased startup boron concentrations of 1200 ppm to 1400–1700 ppm or more (depending on enrichment and the amount of burnable poisons in the fuel design). To maintain a high pH (>6.9) and minimize plant shutdown radiation dose rates, a startup lithium concentration of ~3.5 ppm may be required. The fuel surveillance programme concluded that no corrosion increase of fuel rods can be attributed to elevated Li operations [8.2, 8.20]. Concerning zinc injection, there is wide and successful experience in low and medium duty cores. Measurements carried out in different plants after zinc injection show no effect on oxide thickness [8.21] and [8.22]. For high duty cores, it is recommended to proceed cautiously with zinc injection, as experience is more limited for such cores [8.23].

WWERs

The goals of water chemistry control in WWERs and PWRs are common in some aspects. However, some particular features of water chemistry control in WWER reactors should be mentioned. In order to maintain optimum pH and to limit free oxygen in the primary system, WWER reactors operate with water coolant alkalinized by a mixture of ammonia and potassium hydroxide [8.24–8.26].

The alkalinized mixture of potassium hydroxide and ammonia also provides the narrow range of pH which ensures low material corrosion. Alkali concentration is coordinated with boric acid content, as shown in Fig. 8.2. With varying concentrations of boric acid (from 7.0 to 0 g/kg), alkali concentration in the coolant circuit is maintained at a level providing a constant value of high temperature pH from 7.0 to 7.3 (7.0 to 7.4 for PWRs). Water chemistry specifications for WWER-1000 type reactors are given in Table 8.1.

Limiting the free oxygen produced through radiolysis decomposition of water is achieved by an equilibrium shift in the radiolysis reaction through the addition of hydrogen to the primary coolant. When hydrogen is directly added into the primary coolant in PWRs, in WWER reactors, hydrogen is directly obtained in the primary coolant through radiation decomposition of ammonia. Under irradiation, ammonia dissociates, forming hydrogen and nitrogen. Excess hydrogen (30–60 mL/kg) is provided to suppress coolant radiolysis, thus oxygen content is kept below 5 mg/kg.

In some reactors, a new water chemistry control with continuous hydrazine hydrate ($\text{N}_2\text{H}_4 \cdot \text{H}_2\text{O}$) dosing of the coolant has been developed to replace the standard ammonia–boron–potassium regime. During reactor power operation, hydrazine quickly decomposes into N_2 , H_2 and NH_3 .

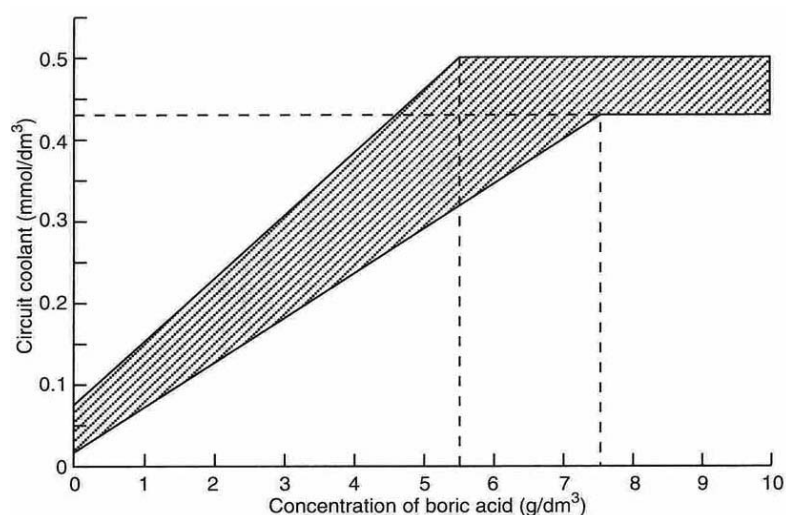


FIG. 8.2. Limits of total concentration of alkaline ions ($K + Na + Li$) as a function of boric acid concentration [8.24].

TABLE 8.1. SPECIFICATIONS FOR PRIMARY COOLANT WATER CHEMISTRY (WWER-1000) [8.24]

Parameter	Values ^a
pH (25°C)	5.9–10.3
Halogens ($Cl^- + F^-$) (mg/dm ³)	<0.1
Dissolved oxygen (mg/dm ³)	<0.005
Dissolved hydrogen (mg/dm ³)	2.7–5.4
Total concentration of ions of alkaline metals ($K + Na + Li$) as a function of boric acid concentration (mmol/dm ³)	0.02–0.5
NH ₃ (mg/dm ³)	>5.0
H ₃ BO ₃ (g/dm ³)	0–10.0
Copper (mg/dm ³)	<0.02

^a Values are relevant to standard conditions: 25°C, 0.1 MPa.

Long term operational experience without deviation of water chemistry control has not revealed any cases of fuel failure due to zirconium alloy corrosion. Oxide layers on fuel cladding irradiated at a burnup of 45–49 MW·d/kg U are usually less than 3 to 5 mm thick. The internal hydrogen content of irradiated fuel cladding is about 0.008%.

However, experience has shown that violation of water chemistry control through intrusion into the primary circuit of organics can affect fuel behaviour and induce leakage of some fuel elements as a result of mechanisms such as accelerated corrosion, friction corrosion and nodular corrosion. Fretting due to intrusion of metal debris may add to these effects. Some of these events are discussed in detail in Section 5.

CANDUs

There have been no reported cases of waterside corrosion leading to fuel performance problems in CANDU power reactors. The oxide thickness on external cladding surfaces of power reactor fuel bundles is typically less than a few micrometers.

8.1.6. Fuel loading/unloading practices

Fuel damage can also affect fuel reliability and become very costly for utilities. Interference between PWR fuel assemblies during loading and unloading operations has resulted in hang-ups between spacer grids that damaged the grids and, more rarely, the fuel rods. These difficulties led to an increase in handling phase time, and consequently outage duration. They can also generate debris in the core vessel, which has to be removed during outage. The causes of interference have included spacers prone to hang-up, bowed fuel assemblies, diminution of the gap between fuel assemblies due to the use of larger spacers to reduce assembly bow in the core, and handling practices.

The major cause of such grid damage is difficulty in positioning fuel in the core due to the bow and twist of fuel assemblies. As a remedy, some utilities and vendors have developed special tools to provide help during handling. These tools assist in positioning the bowed and twisted fuel assembly in the core and correct occasional misalignment of the bottom nozzle assembly with respect to lower internal fuel alignment pins. Appropriate actions have also been taken to improve handling practices and procedures. Fuel handling activities must be performed by well trained and experienced personnel.

Most LWRs use load control during fuel handling to monitor for indications of hang-up, and to ensure that the load limit does not exceed the pre-assigned value. Similar load limits are also applied to the fuelling machine in CANDUs to avoid fuel damage when fuelling sequences need to be switched from automatic to manual operation.

Some utilities also perform a systematic visual examination of each FA during unloading to detect damaged FAs and to limit the risk of propagation of new damage during ensuing reloading.

To avoid fuel bundle break-up in CANDUs due to cracking of the end plates, certain procedures are followed when shuffling fuel off-power from one channel to another. For example, coolant temperature should not be less than about 150°C if defective fuel is present. Above this temperature, irradiated Zircaloy components of defective fuel elements which may contain elevated levels of hydride (or deuteride) are expected to remain ductile and not crack [8.27].

Moreover, all fuel vendors have improved spacer design to be less prone to hang-up, primarily by chamfering the corners. Other improvements are expected in the future.

8.1.7. Foreign material exclusion

The main identified cause of the presence of debris in a primary circuit is lack of rigor in foreign material exclusion (FME) control. Obviously, the best way of preventing debris fretting is to keep debris out of the primary circuit. There are three main components in any debris prevention programme: preventive maintenance on debris producing components; improved maintenance and operating procedures that do not produce debris or which reduce the quantity produced, and; establishment of a debris inspection and cleanup programme [8.28]. Over the past several years, several foreign material exclusion actions have been undertaken by utilities to minimize the introduction of debris into the reactor coolant system, and guidelines were developed to consolidate successful foreign material exclusion (FME) practices and expand upon techniques for preventing debris related fuel failures.

8.1.8. Current practice for handling, storage and disposal of failed fuel

Some utilities (France, Sweden) do not have specific conditions for the handling and storage of leaking fuel. Leaking fuel assemblies are stored in the spent fuel pool without special precaution. In Japanese utilities, leaking fuel assemblies are contained in leaktight containers and kept in the cooling pond of the plant.

Defective fuel is removed from CANDU reactors using the same normal refuelling procedures that are in place for removing intact fuel. With the exception of interim storage in inspection bays, defective fuel bundles in all CANDU reactors are generally treated no differently from normally discharged fuel [8.15]. All bundles are stored under water in open stainless steel containers called baskets, trays or modules. However, defective fuel at some stations is stored in a special location in the bays. Bundles exhibiting gross deterioration may be stored in stainless steel cans with removable end plugs. These cans are not leaktight, but are designed to prevent release of fuel particles into bay water.

The current practice for WWERs is to store failed fuel assemblies in special containers after discharge. Transportation of failed fuel assemblies should also be carried out in these containers although this lowers transportation cask capacity and increases storage and transportation costs.

8.1.9. Reuse of failed LWR fuel

Experience with controlled reinsertion of leaking fuel

Today, the reinsertion of failed fuel into a core is largely discouraged throughout the world. For several years in the United States of America, an important effort has been made to eradicate fuel failure, with the ultimate goal of achieving plant operations with zero defects by 2010.

In the past, some utilities and EDF have specifically chosen a more flexible policy based upon economics without compromising safety. In spite of its broad experience [8.14], which has shown that a careful and prudent policy of reinsertion can be successful, in 2002, EDF decided to stop reinsertion of failed fuel assemblies in reactors. The main reasons put forward for this decision are the following:

- The increase in cycle length results in reintroducing failed fuel assemblies for longer duration, which increases the risk of new fuel degradation during the cycle under the effect of secondary hydriding;
- With the reinforcement of legislation related to radiological effluent releases, compliance with release limits are more difficult than in the past, in particular concerning gaseous effluents whose fission products are nearly the exclusive source;
- The increase in burnup discharge and extensive use of MOX fuel in EDF reactors raises the alpha risk in the event of defect degradation and fuel dissemination in the primary circuit;
- The behaviour of non-tight rods remains delicate to justify in an incidental or accidental situation;
- Due to improvements in fuel reliability, the number of assemblies affected was less, which appreciably reduced the economic impact of its policy;
- Finally, failed fuel represents degradation in the first fission products barrier. Its presence in a reactor does not create confidence in the nuclear power industry and can influence public acceptance of nuclear power.

Experience with repaired and reconstituted fuel

Although there has been substantial worldwide experience regarding the operation of repaired and reconstituted fuel, little has been published on the subject. Generally it seems there are no generic problems.

Siemens has accumulated fairly extensive experience owing to its early start in the field; this is described in Ref. [8.29]. By the end of 1993, repaired and reconstituted fuel assemblies had operated 474 additional cycles in PWRs and 238 additional cycles in BWRs without an enhanced failure rate. Also, no failures due to repair or fuel reconstitution were recorded during the reporting period.

In France, repair and reconstitution of failed assemblies from different fuel suppliers is now part of routine operation [8.30]. No enhanced failure rate of these assemblies was observed after repair during later cycles.

8.2. IMPROVEMENT OF QUALITY DURING MANUFACTURING

8.2.1. Evolution of a quality management system

(1) Historical background — from product control to total quality management [8.31]

The classic approach to quality control (QC) has developed into a systematic management system called ‘Quality Assurance’ (QA). QA systems gradually grew to include direct fabrication and also design, development and operation of a technical product; see for example, the ‘Code of Federal Regulations’ on ‘Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants’, issued in the United States of America in 1970. Known as ‘10 CFR 50, App. B’, these rules became a kind of basic ‘constitution’ for QA which was, in one way or another, adopted in all countries practicing nuclear energy technologies. However, at

that time the objective and the practice of quality control was rather narrow, with the intention to prevent insufficient quality of the final product — in this case, the final fuel assembly, or FA component — through systematic but random sampling, or so-called ‘reactive QC’. It was actually a ‘post-line’ QC with very limited use of statistical methods.

During the 1970s, with growing complexity in industrial production and increasing quality control costs, a new way of thinking was initiated. The objective was an active rather than a reactive quality strategy in order to prevent source of quality deficiencies and thus achieve an increase in productivity. This preventive, process oriented quality philosophy looked for ways to provide fast feedback via quality data of the relevant production steps. The realization of this idea strongly depended on the development of highly effective and automated electronic data acquisition combined with an adequately effective and, whenever possible, automated process control. It is evident that the fast development of computer technology became a key factor in this new type of quality management.

In the 1980s, process control as an essential part of customer/performance oriented quality management was recommended by ISO 9000, and important components of nuclear fuel assemblies for LWR cores had increasingly become subject to manufacturing in series, meaning the same design was used in several different nuclear plants. Further developments in quality management (QM) were jointly obtained during the 1990s by fuel vendors and utilities, see [8.32–8.41]. More examples can be found in [8.42–8.44].

In the meantime, a strong drive toward ‘Total Quality Management’ (TQM) has started everywhere in the industry [8.43]. This means company wide efforts have been made to integrate all forces and operations necessary to fulfil business objectives. This effort is customer oriented and implies continuously adjusting all quality related actions to the growing quality awareness of the customer. In understanding TQM, also in a company with complex production flow, customer relations between different production departments have to be developed and established. During the past several years, the TQM philosophy has found its way to the nuclear industry.

Table 8.2 summarizes the three major steps described above, from classical to total quality management [8.42].

(2) Modern approach to quality management [8.31]

As mentioned above, there are two main ideas guiding the implementation of modern quality management (QM):

- Process orientation;
- Continuous improvement.

The intention of ‘process orientation’ is not primarily to detect and reject nonconforming products, but to control and instantly correct processes used to fabricate these products. Process orientation implies an active attitude toward prevention: any (potential or current) problem must be detected and corrected before it can occur. The old reactive position of ‘first wait and see and then repair’ is outdated.

‘Continuous improvement’ should be seen as a constant effort to improve all steps of fabrication, i.e. manufacturing and testing. Deviations from target values must be reduced constantly. In this philosophy, it is not sufficient to meet specifications; the goal is to stay within lower and upper control limits (LCL and UCL, respectively) and as clear of these limits as possible. A process can be considered robust when relatively large variations in process parameters will have no significant influence on product properties. The centred process (Taguchi philosophy) can be evaluated by means of Cpk-values². In order to achieve a robust and centred process, it is necessary to have adequate knowledge of the entire process.

This approach, for example, is the basis for an initiative which has been introduced in General Electric’s nuclear fuel fabrication. It is called the ‘Six Sigma’ (6σ) initiative [8.45], to be applied on ‘critical to quality’ (CTQ) properties of fuel components. This 6σ (σ is a standard deviation) requirement was first developed and used by Motorola [8.46] to quantify the level of quality performance typically achieved in fabrication. Motorola study

² Cpk is a process capability index defining whether a process is centred or not, i.e. whether the average process variation value μ is in the middle of specification limits. For 6σ process width, Cpk is equal to the lower of the two values: $(USL-\mu)/3\sigma$ or $(\mu-LSL)/3\sigma$.

TABLE 8.2. STEPS OF DEVELOPMENT IN QUALITY ASSURANCE (QA) [8.42]

		Classical QA	Process Oriented QA	Total Quality Management
1	Main objective	Detection and removal of quality deficiencies	Prevention of quality deficiencies (market monitoring, product and process planning) as well as control during fabrication	Prevention of quality deficiencies
	Typical timing	Post-line control after completion of fabrication step	Initially during planning, especially during fabrication (on-line control with feedback on process) and after completion of fabrication step	During planning, fabrication and life time of a product
	Characteristics of quality strategy	Reactive, product oriented	Active, process and product oriented	Active integrative emphasis on system thinking
2	Employees involved	Quality inspection	All employees in development and fabrication, in particular the head of QA	All company employees, in particular management
	Volume and use of quality data	Evaluation of the few data which are generally subject to multiple use	Evaluation of accumulated data for different purposes: Process planning Process control Quality audits, etc.	
	Typical method of data acquisition	Manual	Computer aided CAQ through 'island solutions'	Fully computer supported CAM as CIM
	Complexity of applied procedures	Simple algorithms, graphical tools (e.g. histograms)	Sophisticated algorithms to process Q-data become applicable, also application of advanced graphical tools (e.g. boxplots)	
	Typical statistical tools	Sampling plan	Test plans for quality planning. QC cards for process control	Test plans for quality planning. QC cards for process control and service control

1 — Quality strategy

2 — Realization

results indicated that quality performance of an average company was typically at approximately the 4σ level, or on the order of 2000 to 10 000 defective outcomes per 1 000 000 undertakings (or a 2000 to 10 000 part per million (ppm) defect rate). The Motorola study also identified a select few outstanding performers, labelled 'best class', operating at a 6σ level (3.4 ppm) defect rate. It is clear that the 6σ criterion represents a point far out in the tail of a distribution and therefore is the achievement of this condition has a very low probability, or is rarely achieved.

The scale of a 1 ppm fuel failure rate might be understood from the following description: during a year not more than eight failed rods should be observed worldwide in PWRs and not more than 2.5 rods in WWERs [8.47]. A fuel failure rate equal to 1 ppm (or lower) is presently the target of fuel vendors/electrical utilities [8.48]. This target was already met by some Japanese fuel vendors [8.49, 8.50].

(3) Practical results of QA/QC improvements worldwide

The estimated world distribution of fuel failure causes in PWRs for 1987–1997 and 1995–2006 was presented in literature [8.31] and in Section 3 of this paper, respectively.

A significant reduction in the fuel failure rate for PWR fuels caused by fabrication faults was observed during the decade 1987–1997 (from 11% to 4%). This reduction was reached mainly through improvement of the quality control process and by implementing modern QM. However, from 1994–1997, fuel failure rates resulting from

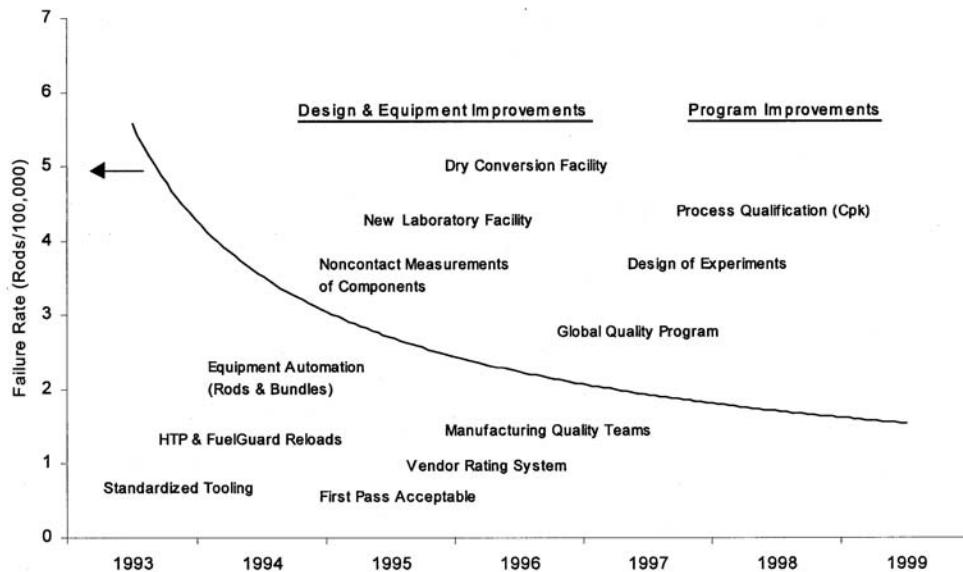


FIG. 8.3. Failure rate versus design and process improvements.

fabrication faults increased again to approximately 10%. This might be linked to the implementation of more demanding fuel operation schemes by utilities, e. g. higher burnups, longer fuel residence times, higher thermal rates. At present, the fuel failure rate component caused by fabrication has returned again to 1997 levels. This was reached by a further improvement in the fuel fabrication process, including control and cladding, and fuel materials.

Figure 8.3 [8.50, 8.51], shows reduced failure rates associated with Siemens fuel can be linked to enhanced fuel design and manufacturing technology, investment in plant processes and equipment, as well as programmatic enhancements. This comprehensive approach has provided Siemens with fuel reliability records in the industry. Siemens has three main focus areas:

- Close engineering to manufacturing controls to improve manufacturability and, in turn, reliability of fuel design;
- Continual improvement through process qualification, including all new or modified processes, equipment or tooling;
- Maintaining a global perspective in order to maximize flexibility and global manufacturing capacity, and most importantly, share lessons learned.

8.2.2. Fuel failure experiences related to fabrication defects and QMS

There were many fuel failures related to fabrication defects in the 1970s. The main causes for BWR fuel failures were local hydrides and PCI, while PWR failures were caused by collapse and rod bowing. There were some cases in which failure resulted from defects in cladding and end plug materials.

Design and quality control improvements have been effective in mitigating and preventing these fuel failures. The ratio of fabrication defects causing fuel failure has been decreasing during last 30 years, nevertheless, some fuel rods still fail today due to fabrication defects.

(1) Pellet chips

To prevent PCI failure caused by pellet chips, as is presented in Section 5, FANP has undertaken corrective actions, including improving the manufacturing process [8.52].

Manufacturing process changes involved improvement in the overall quality of fuel pellet surface conditions, in the effectiveness of the pellet visual inspection process and inspection acceptance standards and, finally, in the technology employed to fabricate fuel rods.

The following activities were directed at improving pellet surface conditions within rods:

- Active and passive scanner upgrades — improved sensitivity to pellet gaps and enrichment transitions;
- Pellet grinder wheel increase in diameter and setup, regulator wheel dressing improvements, and more frequent in-process verifications — reduces pellet end chipping at the grinders;
- Co-milling of pore former and improved roll compactor equipment — better control of pellet pore size and distribution;
- Qualification of silica addition process and optimized additive controls for all powder lots to control powder activity and grain size — better control of pellet behaviour characteristics;
- Application of press tooling wear monitoring via Con-Tracer hardware to measure the inside surface of pellet press dies for wear — reduces weakening of pellets due to worn dies;
- Elimination of an in-line ring gauge diameter check — removes the potential for pellet chipping.

Pellet fabrication is now producing fewer pellet surface condition challenges for the pellet inspection process.

The effectiveness of the pellet inspection process was enhanced by: 1) transferring pellet surface condition inspection to an off-line inspection queue which de-coupled the time available to inspect pellets from production through-put requirements; 2) upgrading environmental lighting, defect sample presentation and inspector training/qualifications; 3) introducing special pellet sheets and a tray flipping device which supported 100% surface examination, and; 4) applying a pellet inspection OverSEER system with inspection process feedback to both inspectors and pellet grinder operators. In addition, the allowable missing pellet surface defect size was significantly reduced from the previous standard. The new standard screens at 0.050 inches (1.27 mm) width and 0.025 inches (0.64 mm) depth.

New technology to fabricate fuel rods has been introduced to address the potential for pellet insertion into cladding tubes causing either pellet surface damage or clad liner upsets. Previous technology employed rod segment pushers to slide a column of pellets into a closed tube (lower end cap welded on the tubing). As successive segments were loaded into a rod, loading forces required to seat the fuel column increased. On occasion, the pellet column would buckle or wedge into the cladding during loading. This had the potential to damage pellets or gouge the inner surface of the cladding. Pellet or liner damage was difficult to reliably detect in these circumstances. As part of the integration between Siemens and Framatome worldwide fuel fabrication facilities under AREVA's joint venture, international best practice manufacturing technologies were selected to standardize operations. With respect to rod fabrication, vibratory rod loading using low amplitude, high frequency excitation was chosen to insert pellets into rods. In June 2004, FANP qualified and placed into service the vibratory rod loading system. This system has eliminated the hard loading conditions which could have damaged pellets or upset the inner surface of tubing.

The Exelon failures were a significant challenge to FANP BWR fuel reliability performance. Following extensive evaluation, several contributing factors to failures were identified. FANP introduced a comprehensive set of corrective actions to address implications of the failure mechanism and identified contributing factors. These corrective actions have produced the expected results; no additional FANP ATRIUM-9B failures have occurred in previously delivered fuel reloads and the ATRIUM-10 product continues to perform very well in a wide range of operating environments.

Another example of improvements to reduce pellet chips are those undertaken by Westinghouse [8.53].

Westinghouse has initiated company wide efforts to enhance the overall quality of UO₂ pellets by enhancing pellet pressing processes, examining pellet handling during various stages of pellet fabrication and enhancing the inspection process with a primary focus on reducing pellets with side chips. Pellet press improvements are being pursued to improve press reliability, efficiency and consistency. By using tools such as cause and effect diagrams, statistical evaluations, gage reproducibility and repeatability etc., Westinghouse has performed in-depth examinations of pellet quality after every process step. Based on these evaluations, process and inspection improvements are being pursued to modify and upgrade the process steps provoking the most pellet defects. New and improved methods and equipment have been and will continue to be implemented as a result of this work. Different sites, however, use different conversion and pelletizing processes, thus site specific improvements are being made where they will have the most impact.

Comparisons of pellet manufacturing within the BNFL group [8.53]

A benchmarking/best practices review has been conducted among all Westinghouse and BNFL sites to minimize pellet chipping, and enhance overall pellet quality and inspection processes. In addition to pellet handling, which is one of several aspects benchmarked, pellet quality variation sources were considered.

- Pellet design — chamfer angle as well as pellet L/D ratios were considered important;
- Pellet microstructure — uniform microstructure with relatively small pores was generally considered optimum for pellet strength;
- Green density — may have an impact on green pellet strength and thus on pellet chipping;
- Powder morphology — may have an impact on pellet properties. An increasing number of dendrites may improve pelletizing properties. These conclusions were applicable to wet or dry route processes;
- Add-back material — the addition of U_3O_8 to UO_2 at levels of up to 20% could strengthen green pellets. U_3O_8 particle structure and morphology was considered important;
- Press tool maintenance — the wear rate should be monitored to set alarm points before worn tooling can lead to a bad product;
- Pellet die design — the pellet ‘taper’ (the difference between top diameter and bottom diameter due to uneven pressing forces and relaxation) was believed to have an effect on pellet chipping.

These general areas of interest led to further studies, but actual areas of improvement varied between manufacturing sites since both conversion (IDR, ADU and AUC) and pellet manufacturing processes are different.

Automatic visual inspection

An alternate end chip criteria which allows inference of an end chip from the side view has been developed to facilitate rapid development of automated inspection. In preparing for automation studies, Westinghouse has identified 16 different types of defects among manufactured pellet designs UO_2 , Gadolinia, Erbium and Integrated Fuel Burnable Absorber (IFBA). Comprehensive visual standards have been prepared for all defects and studies have been undertaken in both Columbia and Västeraås on scanning of these pellets using different techniques, including 3D-laser, direct light reflection and diffuse lighting. Options for a complete automatic system were reviewed and a final design chosen.

(2) End plug welding

A review of fuel bundle manufacturing and the quality control plan adopted at the Nuclear Fuel Complex (NFC) was undertaken [8.54]. Fresh fuel in stock at each reactor site and the NFC was investigated for He leaks from fuel bundles to identify any manufacturing defects. Investigations indicated that defects could be due to incomplete end closure welds of elements. End cap sheath junctions are normally under great stress in PHWR fuel, where practically no gas plenum exists. This, accompanied by a manufacturing deficiency, especially in an end closure weld, can also cause fuel failure. Based on a systematic study, the quality control plan and welding procedures were revised.

The TIG method has been employed for plug to tube welding of a fuel rod, and X ray radiography was formerly applied as a non-destructive testing (NDT) means in order to verify the weld integrity of every fuel rod. As X ray radiography had limited capabilities in areas such as shooting time and direction, and inspection of fuel rod weld integrity is a key characteristic of regulatory inspection according to the law, JNF has developed and applied a more reliable and effective probe rotation type ultrasonic method [8.55].

RMD, BARC and NFC have successively developed a UT method to test sheath to end cap welds [8.54].

These modifications not only reduced fuel failures, but also decreased the element reject rate in helium leak tests from above 0.1% to around 0.001%.

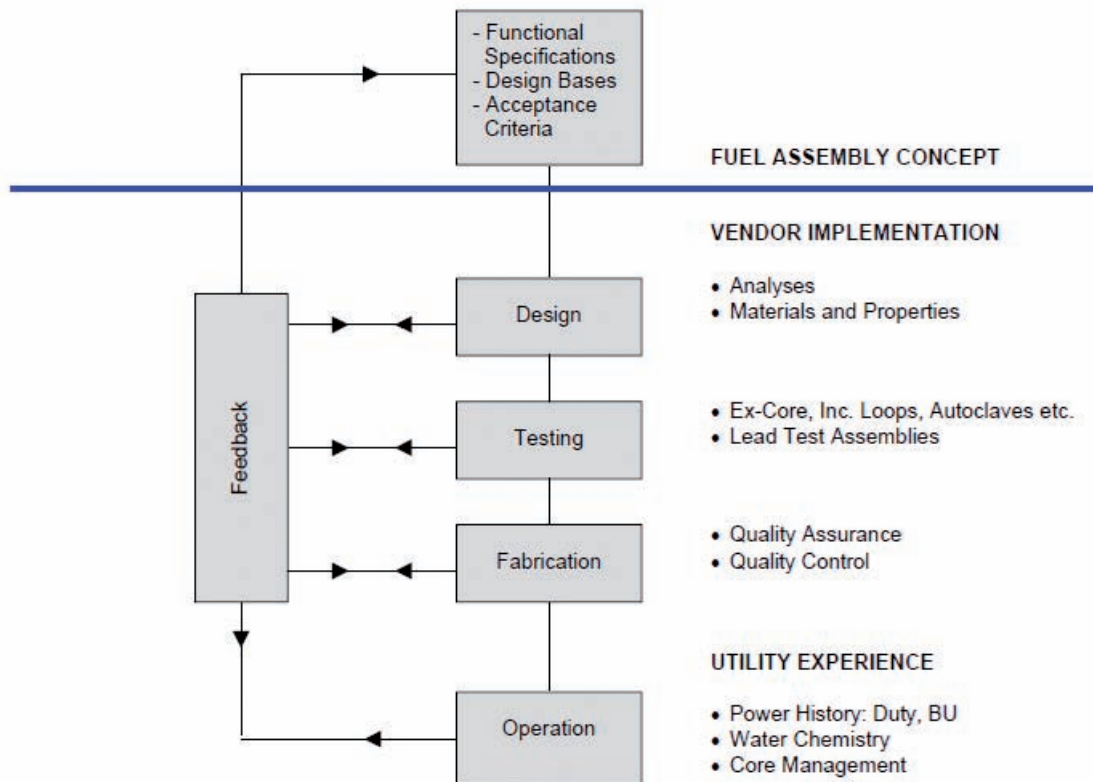


FIG. 8.4. Fuel system life cycle showing major factors influencing fuel reliability.

8.2.3. QMS and good practice for prevention of fuel failure related to fabrication defects

Quality Management Cycle (QMC) or Quality Management System (QMS) is a feedback mechanism (see Fig. 8.4) which is an essential link to overall improvement in fuel reliability. It allows fuel vendors to respond to operation related failures and other ‘out of spec’ events included within the broader definition of fuel reliability. Examples of this might be: a) design changes to a debris filter to improve its efficiency, and b) modifications to spacer grid dimensions to increase assembly rigidity and thereby mitigate grid to rod fretting or assembly bow [8.56].

8.2.4. Fuel reliability in Japan

As seen in Section 3, the fuel failure rate in Japan has been remarkably low compared to European and American rates, in the order of 10^{-6} not only during the last decade, but since the 1980s.

Although it is a known fact, it is not definitely clear why the rate is so low in Japan. Considering various situations, the following features are supposed to be the causes:

- In Japan, fuel reliability (rate of fuel failure) has been regarded as one of the most important key issues for operation of nuclear power plants since the beginning of their use;
- Power reactor operating conditions (linear heat rating, water chemistry, etc.) are milder for fuel compared to those of European and American reactors;
- In Japan, the PDCA (plan–do–check–act) cycle which is the basement of QMS has been implemented since the beginning of domestic production of nuclear fuel. As a national project, irradiation proof examination of commercial fuel bundles has been carried out and the knowledge acquired by detailed post-irradiation examinations fed back to the design, manufacture, and inspection of fuel and the operation of a plant in each development phase;

- To introduce a new design fuel, fewer numbers of Lead Test Assemblies (LTAs) are first loaded into a commercial reactor, and after fuel performance is observed during a couple of cycles, a full scale load of batch size is completed. Utilization has been carefully advanced;
- Use of improved and robust fuel designs. For example, there is no grid to rod fretting failure with Japanese fuel. This might be because of a difference in design (the use of nine grids instead of eight in a 12 foot fuel assembly) and quality control of grid spacers, where spring forces on all grid locations are routinely checked;
- In general, employees' workmanship (QC mind) is highly connected to Japanese culture, including belief in meticulousness and cleanliness. For example, the cleanliness of Japanese plants is remarkable. Therefore, debris fretting failure seldom occurs;
- In the manufacturing process, pellets are dealt with very carefully and there are few pellet chips. According to Mr. Murota, one other factor is efforts undertaken to reduce the incidence of 'random' failure, when cause cannot be readily identified. It is thought that the majority of such random failures are caused by fabrication defects. It has been shown that the number of these fuel failures is practically eliminated through Japanese total quality control [8.57]. The following three items are the basis of this approach:

(a) Manufacturing technology

Through powder processing/pelletizing technology, precision process machinery, precision welding technology, automated equipment such as robots, mechatronics control technology, micro electro technology, computers, etc., precision, mechanization, and automation are adopted to ensure that manufacturing processes are stabilized.

(b) Workmanship

In production activities, such as operation, maintenance, improvements in manufacturing, inspection, analysis and physical transportation, full use is made of management tools, such as QC and IE. Any production abnormalities are monitored and corrected. For example, use is made of knowledge and experience regarding the relationship between pellet appearance characteristics and processing conditions, the influence of weld parameters to weld bead colour tone or to the relationship between a slight flaw or foreign substance on the surface of a fuel rod and the place/cause of such an occurrence.

(c) Quality intention mindset

- Workers are imbued with the quality mindset 'downstream is a customer for upstream';
- Uniform quality between and within lots;
- Practical fabrication specifications are more stringent than design specifications, ensuring that there is always plenty of margin.

REFERENCES TO SECTION 8

- [8.1] ELECTRIC POWER RESEARCH INSTITUTE, Fuel Reliability Improvement Guidelines, Rep. Tr-100659, EPRI, Palo Alto, CA.
- [8.2] UNITED STATES NUCLEAR REGULATORY COMMISSION, Standard Technical Specifications General Electric plants, BWR/6, Vol 1, Rev. 3, NUREG-1434, June 2004.
- [8.3] UNITED STATES NUCLEAR REGULATORY COMMISSION, Standard Technical Specifications Westinghouse plants, Vol 1, Rev 3., NUREG-1431, June 2004.
- [8.4] VON JAN, R., et al., Fuel performance aspects for load following operation of light water reactors, Atomkernenerg.-Kerntechn. **48** 3 (1986) 156.
- [8.5] PROVOST, L., DEBES, M., "EDF PWR fuels operating experience and high burn performances", TopFuel 2006, Salamanca, 2006.
- [8.6] POTTS, G.A., PROEBSTLE, R.A., "Recent GE BWR fuel experience", Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, 1994, ANS, LaGrange Park, IL (1994) 87.
- [8.7] VON JAN, R., "Strategy and experience of PCI protection and load following operation in KWU LWR plant", Proc. IAEA Tech. Comm. Mtg on Power Ramping, Cycling and Load Following Behaviour of Water Reactor Fuel, Lyon, 1987, IAEA-IWGFPT/28, IAEA, Vienna (1988) 110.
- [8.8] OLSSON, S.L., "Operation with monosequence in ASEA-ATOM BWR", Proc. Spec. Mtg on Pellet-Cladding Interaction in Water Reactors, Risø, 1980, IAEA-IWGFPT/8, IAEA, Vienna (1981) 71.

- [8.9] POWERS, C., et al., “Hot cell examination results of non-classical PCI failures at La Salle” TopFuel 2005, Kyoto, 2005.
- [8.10] KARVE, A., et al., “A methodology for calculating and a process for mitigate channel distortion and cell friction”, TopFuel 2006.
- [8.11] YANG, R.L., et al., “EPRI failed fuel degradation R&D program”, Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, 1994, ANS, LaGrange Park, IL (1994) 435.
- [8.12] ARMIJO, J.S., “Performance of failed BWR fuel”, *ibid.*, p. 410.
- [8.13] KENNARD, M.W., HARBOTTLE, J.E., SUNDERLAND, D.J., “Identification and mitigation methods for failed BWR fuel”, *ibid.*, p. 503.
- [8.14] BOURNAY, P., “Management of failed fuel during operation — French policy and experience”, Proc. Int. Top. Mtg on Light Water Reactor Fuel Performance, West Palm Beach, 1994, ANS, LaGrange Park, IL (1994) 477.
- [8.15] MacDONALD, R.D., FLOYD, M.R., LEWIS, B.J., MANZER, A.M., TRUANT, P.T., Detecting, Locating and Identifying Failed Fuel in Canadian Power Reactors, Rep. AECL 9714, Atomic Energy of Canada Ltd, Chalk River (1990).
- [8.16] CHENG, B.C., et al., “Fuel performance and water chemistry variables in LWRs”, Portland, 1997.
- [8.17] SHIMADA, S., “Effect of hydrogen water chemistry on Zircaloy-2 fuel cladding and structure material performance”, TopFuel 2005, Kyoto, 2005.
- [8.18] CHENG, B., et al., “Effects of noble metal chemical application on fuel performance ANS2004”, ANS 2004 Fuel Topical Meeting, Orlando, 2004.
- [8.19] YANG, R., et al., “Fuel R & D to Improve Fuel Reliability”, TopFuel 2005, Kyoto, 2005.
- [8.20] WILSE, J.T. et al., “Recent results from B&W fuel company’s fuel performance improvement”, Int. Top. Mtg on LWR Reactor Fuel Performance, West Palm Beach, 1994.
- [8.21] SÁNCHEZ, A., et al., “Fuel evaluation under zinc injection in ASCO 1/2 and VANDELLOS 2 NPP”, TopFuel 2006, Salamanca, 2006.
- [8.22] TIGERAS, A., “EDF zinc injection: analysis of power reduction impact on the chemistry and radiochemistry parameters”, Int. Conf. on Water Chemistry of Nuclear Reactor Systems, Korea 2006.
- [8.23] ELECTRIC POWER RESEARCH INSTITUTE, PWR Zinc Applications Guidelines, estimated publication date, 2007.
- [8.24] IOLTUKHOVSKI, A.G., “Features of water-chemical mode (WCM) of reactors WWER-440 and WWER-1000, Maintenance and control. Influence of WCM deviations on fuel operational features”, IAEA Training Course on WWER Fuel QA/QC and Perspectives, Obninsk, 1995.
- [8.25] DRAGUNOV, Y.G., et al., “Water chemistry in Soviet nuclear power plants”, Coolant Technology of Water Cooled Reactors, Vol. 1, IAEA-TECDOC-667, IAEA, Vienna (1992) 110.
- [8.26] PASHEVICH, V.I., et al., “Hydrazine regime for WWER-440 and WWER-1000 primary circuits”, *ibid.*, 108.
- [8.27] CARTER, J., MANZER, A.M., “Overview of defect mechanisms in CANDU fuel”, Proc. Tech. Comm. Mtg Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management, Dimitrovgrad, 1992, IAEA-TECDOC-709, IAEA, Vienna (1993) 121.
- [8.28] STRASSER, A., GINGOLD, J., “Evaluation of debris fretting failures and preventive methods”, *ibid.*, 133.
- [8.29] SIEMENS, Siemens, Fuel Assemblies — References ’94.
- [8.30] BOURNAY, P., “Inspection and reconstitution of fuel elements: EDF experience, policy and needs”, Proc. Tech. Comm. Mtg on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management, Dimitrovgrad, 1992, IAEA-TECDOC-709, IAEA, Vienna (1993) 147.
- [8.31] ONOUFRIEV, V., “Advanced methods of quality control in nuclear fuel fabrication”, Proc. Int. Conf. on Characterization and Quality Control of Nuclear Fuels, Hyderabad, 2002, Nuclear Fuel Complex, Allied Publishers PVT Limited, New Delhi (2002) 65–83.
- [8.32] GRIZE, Y.-L., SCHMIDLI, H., “Techniken zur statistischen Qualitätskontrolle bei der Herstellung von Brennelementen”, PreussenElektra (1997).
- [8.33] BAUR, K., et.al., “Qualitätssicherung bei der beschaffung von Brennelementen”, Proc. Conf. KTG Subgroup Fuel Elements, Karlsruhe (1995) 237–252.
- [8.34] BAUR, K., et.al., “Qualitätssicherung bei der Brennelementfertigung”, Proc. Annual Meeting on Nuclear Technology ’98, Munich, 1998.
- [8.35] BAUR, K., GRIZE, Y. L., Quality management and quality assurance in the manufacture of nuclear fuel elements: new trends and challenges, *Kemtechnik* **63** 3(1998) 92–97.
- [8.36] HOLZER, R., NILSON, F., “Qualitätssicherung im Siemens Be-Geschäft. Zielsetzung und Umsetzung in einem teils sich wandelnden, teils erstarrten Umfeld”, Proc. Conf. KTG Subgroup Fuel Element, Karlsruhe (1995) 183 (published separately).
- [8.37] SIEBERT, H. J., et. al., “Statistische Prozeßregelung und moderne Methoden des Qualitätsmanagement in der Brennelementfertigung”, Proc. Annual Conf. on Nuclear Technology ’97, Aachen, 1997.
- [8.38] MICKO, F., HALLDEN, E., “Total quality approach at ABB atom nuclear fuel division”, Proc. Conf. KTG Subgroup Fuel Elements on Fabrication and Quality Assurance of Uranium and Mixed Oxide Fuel for Light and Water Reactors, Karlsruhe (1995) 111–117.
- [8.39] PROEBSTLE, R. A., GALAN, J. I. M., “Zero Leaker Initiatives”, *ibid.*, 15–35.
- [8.40] DARMAGAN, P., “Quality Assurance in the Fuel Fabrication”, Transaction TopFuel ’95, Vol. 1 (1995) 309–319.
- [8.41] VIARD, R., et. al., “Progress in manufacturing, process control and associated criteria”, Proc. Conf. KTG, Subgroup Fuel Elements, Karlsruhe, 1995, 45–56.

- [8.42] WEIDINGER, H., "Modern requirements to quality assurance and control in nuclear fuel fabrication", IAEA Training Course on WWR Fuel Design, Fabrication, Performance and Back End of the Nuclear Fuel Cycle, Bratislava, 1999.
- [8.43] INTERNATIONAL ATOMIC ENERGY AGENCY, Advanced Methods of Process/Quality Control in Nuclear Reactor Fuel Manufacture, Vienna, IAEA-TECDOC-1166, IAEA, Vienna (2000).
- [8.44] WEIDINGER, H., "Statistical methods to assess and control processes and products during nuclear fuel fabrication", IAEA Training Course on WWR Fuel Design, Fabrication, Performance and Back End of the Nuclear Fuel Cycle, Bratislava, 1999.
- [8.45] POTTS, G. A., "Recent GE BWR fuel experience", Proc. Int. Topical Mtg on Light Water Reactor Fuel Performance, Portland, 1997, ANS, La Grange Park (1997) 269.
- [8.46] HARRY, M., in The Nature of Six Sigma Quality, Motorola Inc. Publication.
- [8.47] INTERNATIONAL ATOMIC ENERGY AGENCY, Review of Fuel Failures in Water Cooled Reactors, TRS No. 388, IAEA, Vienna (1998).
- [8.48] MARTIN GALAN, J.I., "Total quality project initiatives and fabrication improvements at ENUSA factory", Specialist Mtg on Nuclear Fuel and Control Rods: Operating Experience, Design Evolution and Safety Aspects, Madrid, 1996, Proc. NEA Int. Workshop, NEA/CSNI/R (1997) 2.
- [8.49] DOI, S., et. al., "Advanced fuel design and performance for burnup extension", Proc. Int. Conf. on LWR Fuel Performance, Park City, 2000, ANS, La Grange, IL (2000) 434.
- [8.50] ITAGAKI, N., et. al., "Experience and development of BWR fuel supplied by NFI", Proc. Int. Conf. on LWR Fuel Performance, Park City, 2000 (2000) 483.
- [8.51] ADKISSON, D. A., HILL, D. J., HOFF, A., "Manufacturing controls for improved fuel reliability", Int. Top. Mtg on Light Water Reactor Fuel Performance, 2000.
- [8.52] POWERS, C., DEWES-ERLANGEN, P., BILLAUX, M., PERKINS, R., RUZAUSKAS, E., "Hot cell examination results of non-classical PCI failures at La Salle", Proc. 2005 Water Reactor Fuel Performance Mtg, Kyoto, 2005.
- [8.53] BORELL, S., WIDEGREN, H., SHAH, H., "Improved fuel performance by enhanced UO₂ pellet fabrication and inspection methods", TopFuel 2006, Salamanca, 2006.
- [8.54] RAVI, M., PRASAD, P. N., BAJAJ, S. S., BHARDWAJ, S. A., "Effect of design, manufacturing and quality on fuel performance", Proc. Int. Conf. on Characterization and Quality Control of Nuclear Fuels, Hyderabad, 2002, Nuclear Fuel Complex, Allied Publishers PVT Limited, New Delhi (2002) 307–317.
- [8.55] NISHINA, Y., INATANI, M., KAMATA, N., "Application of ultrasonic inspection technique on fuel rod seam weld", Specialist Mtg On Nuclear Fuel And Control Rods: Operating Experience, Design Evolution And Safety Aspects, Proc. of NEA Int. Workshop, Madrid, 1996, NEA/CSNI/R (1997) 2.
- [8.56] HARBOTTLE, J., KENNARD, M.W., "Towards improved fuel reliability", Int. Top. Mtg on Light Water Reactor Fuel Light Water Reactor Fuel Performance, 2000.
- [8.57] MUROTA, K., "Fabrication technology for light water reactor fuel", 9th Nuclear Fuel Summer Seminar, 1994, Japanese Nuclear Society, (in Japanese).

9. CONCLUSIONS

The present publication responds to growing awareness of a need for open discussion and concerted action to promote effective improvements in nuclear fuel reliability, as was formulated in more general terms at the 9th biennial general meeting of the World Association of Nuclear Operators (Chicago, September 2007) by WANO Chairman William Cavanaugh: “Meeting the unprecedented demands of the nuclear renaissance will require operators not only to take on the individual responsibility to guarantee the safety of their own fleet, but also to assume a collective responsibility to work together to continually upgrade the safety of operating nuclear power stations worldwide. The public demands no less from us.”

The current study covers about 96% of water reactor fuels, thus providing the most representative picture of fuel reliability available in open literature and presumes the opportunity for a number of conclusions.

Statistics for fuel failures (Section 3) seem to reveal a contradiction in incentives: on one hand, fuels should operate in more challenging conditions, with up-rate power outputs but increased failure probability, and on the other hand there is an aspiration to have fully reliable fuels with ‘zero failure rates’ to a practical ‘ppm’ extent. This balance varies from country to country and is dependent both on achieved levels of technological maturity and on local levels of economically and publicly acceptable risk.

Fuel rod failure rates in LWRs have been significantly (but not monotonically) reduced since 1987, on average to levels of 10^{-5} between 2003 and 2006. In CANDUs, element failure rates have been near 10^{-5} and have remained at this low level over the reporting period, reaching 5×10^{-6} between 2003 and 2006. However, the fuel failure rate has not markedly decreased during the last decade, with a relatively large number of failures still occurring in a few plants. Moreover, signs that new failures are increasing were observed in the early 2000s. It is interesting to note that a majority of the mechanisms causing earlier fuel failures are still prevailing, mostly in combination with new contributing factors like higher burn-ups and power ramps.

In a few cases, adequate design and manufacturing solutions have led to considerable improvements. Clad collapse (PWRs) and failures due to excessive rod bow or differential length growth have practically disappeared. Failures from baffle jetting (PWRs) temporarily increased during the 1980s, but have been reduced now to nearly zero. Hydriding caused by moisture in fuel, which had earlier been a major source of failure in all LWR types, has been essentially eliminated. Other manufacturing related failures are low in absolute numbers, accounting for 5%.

Important current issues (2003–2006) in fuel performance include grid to rod fretting in PWRs (52% of all identified root causes) and corrosion by itself or in combination with crud deposits in BWRs (46%). Debris fretting continued at significant levels (28% in BWRs and 9% in PWRs), as well as PCI/SCC in BWRs (12%). Despite all efforts to find adequate remedies, there seem to be problems that continue to challenge the industry. Besides traditional rod failures, new fuel assembly related issues have appeared that can seriously affect plant operations: fuel assembly bow and its consequences on incomplete control rod insertion (IRI), axial offset anomaly (AOA), and crud deposits on fuel. Handling damage of PWR fuel is often related to assembly bow with the consequence of spacer damage during loading or off-loading. A variety of incidents involving fretting wear have been reported.

Failure rates

- (1) The existing non-uniformity in approach regarding fuel failure data and rate calculations can lead to misinterpretation of statistical results. The current study proposes a methodology of failure rate assessment on the basis of fuel reloads which more realistically reflect fuel reliability than calculations based on fuel inventories.
- (2) Despite continuous fuel material upgrades, and design and quality assurance procedures implemented within the last decade aiming at improving fuel reliability, failure rates have oscillated in most countries (with the only exception being Japan, which has a stable and very low failure rate).

Failure mechanisms

- (3) No new failure mechanisms have been revealed during the last decade, but experimental simulation and theoretical modelling indicate the possibility of material degradation in more challenging operating conditions, e.g. hydriding of BWR fuel rods in power ramp tests.
- (4) The latest design improvements in LWR fuels have effectively tackled, but in some cases still not fully resolved, problems related to structural behaviour of fuel assemblies.

Failure analysis and management

- (5) In the current study, the methods of data collection and methodology used for statistical calculations were applied mainly to fuel leakage failures. At the same time, it was demonstrated that non-leaker failures can cause serious problems for nuclear unit operations. Deeper analysis of assembly and core related issues is needed.
- (6) Incidents involving massive failures should be specifically considered and taken into account.
- (7) An approach to the ambitious 'zero rate' target will require complex technical and organizational measures including deep analysis of feedback at all stages of the fuel quality assurance circle (R&D, design, fabrication, and operation).

ABBREVIATIONS

ADOPT	advanced doped pellet technology
ADU	ammonium di-uranate
AECL	Atomic Energy of Canada Limited
AFR	away-from-reactor (spent fuel storage)
AGR	advanced gas cooled reactor
ALARA	as low as reasonably achievable
ANF	Advanced Nuclear Fuels Corporation
ANS	American Nuclear Society
AOA	axial off-set anomaly
ASPR	axial shape power control rod
ASTM	American Society for Testing and Materials
AUC	ammonium uranyl carbonate
BARC	Bhabha Atomic Research Centre
BNFL	British Nuclear Fuel Limited
BOC	beginning of cycle
BU	burnup
CAM	computer aided manufacturing
CANLUB	Canadian lubricant (graphite coating)
CAQ	Computer aided quality
CFR	Code of Federal Regulations (USA)
CHC	corrosion hydrogen cracking
CIC	control of cladding integrity
CILC	crud induced localized corrosion
CIM	computer integrated manufacturing
CIPS	crud induced power shift
CNNC	China National Nuclear Corporation
CSNI	Committee on the Safety of Nuclear Installations (OECD/NEA)
DAD	defective assembly detection
DCP	distinctive crud pattern
DEI	dose equivalent iodine
DFBN	debris filter bottom nozzle
DHC	delayed hydride cracking
DNB	departure from nucleate boiling
EDC	(test) — expanding mandrel (test)
EDF	Electricité de France
ECT	eddy current testing
EOC	end of cycle
EOL	end of life
EPMA	electron probe micro-analysis
ERU	enriched reprocessed uranium
FA	fuel assembly
FR	fuel rod
FRED	fuel reliability database (EPRI)
FRI	fuel reliability indicator
GE	General Electric
GIF	Generation IV International Forum
GT	guide tube
GW·d/t U	burnup, measured in gigawatt days per tonne uranium
HALS	hydrogen assisted localized shear
HEU	high enriched uranium

HGC	hydrogen gas cracking
HM	heavy metal
HTP	high thermal performance
HWC	hydrogen water chemistry
IASCC	irradiation assisted stress corrosion cracking
ICRP	International Commission on Radiological Protection
IDR	integrated dry route (powder process)
IGSCC	intergranular stress corrosion cracking
INB	Industrias Nucleares Brasileiras
INPO	Institute of Nuclear Power Operations
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles (IAEA)
IRI	incomplete (control) rod insertion
IS	inner surface
ISO	International Standards Organization
JMTR	Japanese material test reactor
JNES	Japan Nuclear Energy Safety Organization
JNFL	Japan Nuclear Fuel Limited
KAERI	Korean Atomic Energy Research Institute
LCL	lower control limit
LHGR	linear heat generation rate, in W/cm or kW/m or kW/ft
LOCA	loss of coolant accident
LPD	low pressure drop
LTA	lead test assembly
LTP	low temperature process
LWR	light water reactor
MADB	Minor Actinide Property Database (IAEA)
Magnox	magnesium non-oxidizing
MOC	middle of cycle
MOX	mixed oxide (fuel)
NAC	Nuclear Assurance Corporation (International)
NDT	non-destructive testing
NFC	Nuclear Fuel Complex (Hyderabad, India)
NFCIS	Nuclear Fuel Cycle Information System (IAEA)
NGS	nuclear generating station (used in Canada)
NMCA	noble metal chemical addition
NPP	nuclear power plant
NRC	(United States) Nuclear Regulatory Commission
OECD/NEA	Organisation for Economic Co-operation and Development Nuclear Energy Agency
OS	outer surface
PCI	pellet–cladding interaction
PCIOMR	pre-conditioning interim management recommendations
PCI-SCC	pellet–cladding interaction/stress corrosion cracking
PCMI	pellet–cladding mechanical interaction
PDCA	Plan-Do-Check-Act
PIE	post-irradiation examination
PLT	pin loading tension
PRIS	Power Reactor Information System (IAEA)
QA	quality assurance
QC	quality control
QM	quality management
QMC	quality management cycle
QMS	quality management system
R/B	release to birth rate ratio

RCCA	rod cluster control assembly
RIA	reactivity initiated accident
RIAR	Research Institute of Atomic Reactors (Dimitrovgrad, Russian Federation)
RTP	ramp terminal power
SCC	stress corrosion cracking
SCIP	Studsvik Cladding Integrity Programme
SCRs	spacer capture rods
SEM	scanning electron microscopy
SEU	slightly enriched uranium
SG	spacer grid
SPLIT	split propagation laboratory investigation test
SPP	secondary phase particles
SRA	stress relieved annealed
SRP	standard review plan
SS	stainless steel
TD	theoretical density
TEM	transmission electron microscope
TIG	tungsten inert gas (welding process)
TQM	total quality management
TSSd	terminal solid solubility for dissolution
TSSp	terminal solid solubility for precipitation
TWGFPT	Technical Working Group on Water Reactor Fuel Performance and Technology (IAEA)
UCL	upper control limit
USW	upset shape welding (process)
UT	ultrasonic testing
WANO	World Association of Nuclear Operators

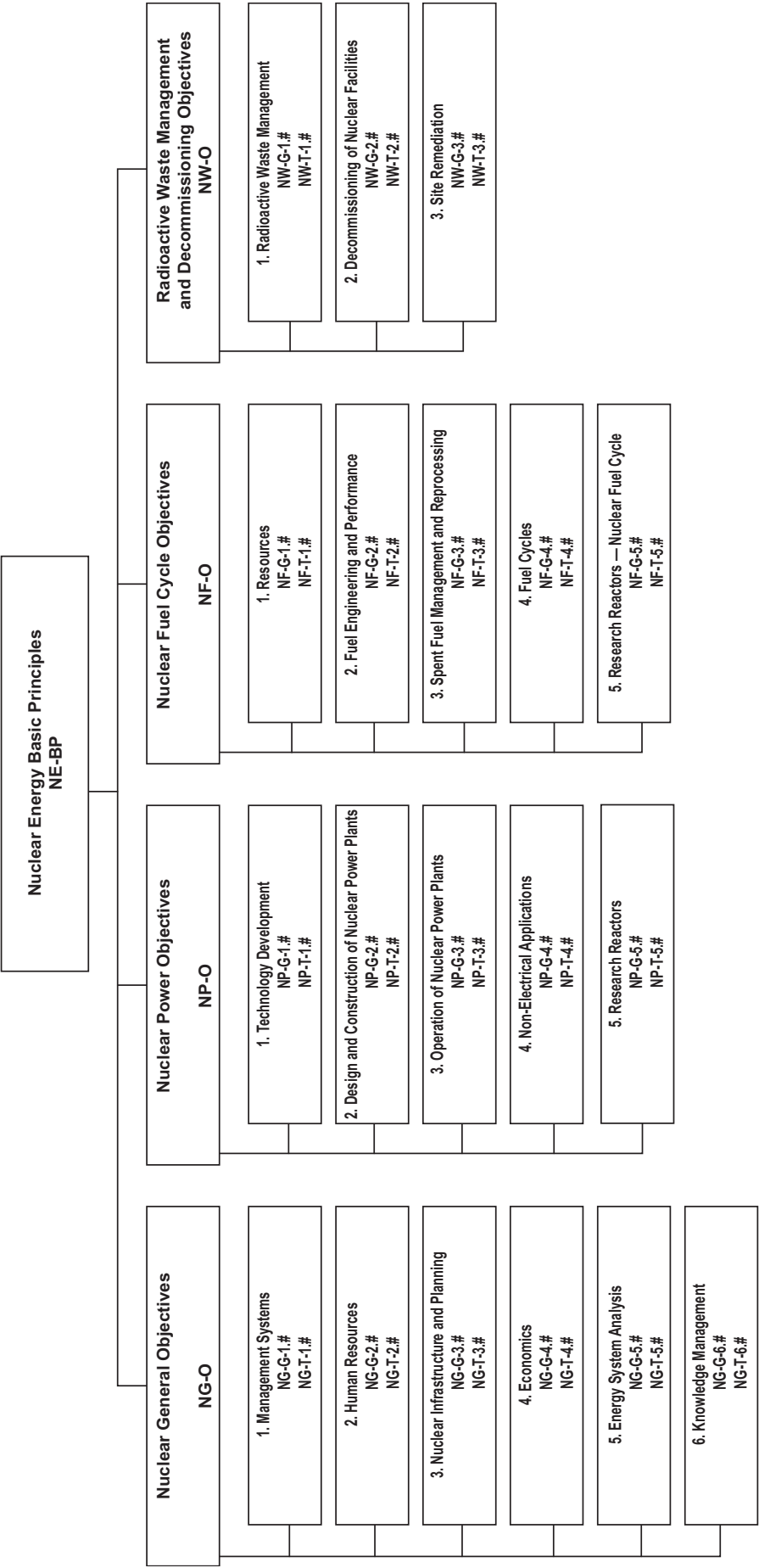
CONTRIBUTORS TO DRAFTING AND REVIEW

Alvarez, L.	Comision Nacional de Energia Atomica, Argentina
Daniels, T.	Ontario Power Generation, Canada
Dangoulème, D.	Electricite de France, France
Doria, F.	Atomic Energy of Canada Ltd, Canada
Edsinger, K.	Energy Power Research Institute, United States of America
Fujiwara, S.	Federation of Electric Power Companies of Japan, Japan
Inozemtsev, V.	International Atomic Energy Agency
Ip, M.	Bruce Power, Canada
Kamimura, K.	Japan Nuclear Energy Safety Organization, Japan
Killeen, J.	International Atomic Energy Agency
Kucuk, A.	Electric Power Research Institute, United States of America
Lewis, B.	Royal Military College, Canada
Novikov, V.	All-Russia Scientific Research Institute of Inorganic Materials, Russian Federation
Onufriev, V.	Private consultant, Russian Federation
Pageau, F.	Hydro Quebec, Canada
Prasad, P. N.	Nuclear Power Corporation of India Ltd, India
Reid, P.	Atomic Energy of Canada Ltd, Canada
Sun, A.	Atomic Energy of Canada Ltd, Canada
Suk, E.	Atomic Energy of Canada Ltd, Canada
Tayal, M.	Atomic Energy of Canada Ltd, Canada
Tran, T.	New Brunswick Power, Canada

Consultants Meetings

Vienna, Austria: 13–15 February 2007;
Vienna, Austria: 27–29 November 2007;
Vienna, Austria: 27–29 May 2008.

Structure of the IAEA Nuclear Energy Series



Key

BP: Basic Principles

O: Objectives

G: Guides

T: Technical Reports

Nos. 1-6: Topic designations

#: Guide or Report number (1, 2, 3, 4, etc.)

Examples

NG-G-3.1: Nuclear General (NG), Guide, Nuclear Infrastructure and Planning (topic 3), #1

NP-T-5.4: Nuclear Power (NP), Report (T), Research Reactors (topic 5), #4

NF-T-3.6: Nuclear Fuel (NF), Report (T), Spent Fuel Management and Reprocessing, #6

NW-G-1.1: Radioactive Waste Management and Decommissioning (NW), Guide,

Radioactive Waste (topic 1), #1



IAEA

International Atomic Energy Agency

No. 22

Where to order IAEA publications

In the following countries IAEA publications may be purchased from the sources listed below, or from major local booksellers. Payment may be made in local currency or with UNESCO coupons.

AUSTRALIA

DA Information Services, 648 Whitehorse Road, MITCHAM 3132
Telephone: +61 3 9210 7777 • Fax: +61 3 9210 7788
Email: service@dadirect.com.au • Web site: <http://www.dadirect.com.au>

BELGIUM

Jean de Lannoy, avenue du Roi 202, B-1190 Brussels
Telephone: +32 2 538 43 08 • Fax: +32 2 538 08 41
Email: jean.de.lannoy@infoboard.be • Web site: <http://www.jean-de-lannoy.be>

CANADA

Bernan Associates, 4501 Forbes Blvd, Suite 200, Lanham, MD 20706-4346, USA
Telephone: 1-800-865-3457 • Fax: 1-800-865-3450
Email: customercare@bernan.com • Web site: <http://www.bernan.com>

Renouf Publishing Company Ltd., 1-5369 Canotek Rd., Ottawa, Ontario, K1J 9J3
Telephone: +613 745 2665 • Fax: +613 745 7660
Email: order.dept@renoufbooks.com • Web site: <http://www.renoufbooks.com>

CHINA

IAEA Publications in Chinese: China Nuclear Energy Industry Corporation, Translation Section, P.O. Box 2103, Beijing

CZECH REPUBLIC

Suweco CZ, S.R.O., Klecakova 347, 180 21 Praha 9
Telephone: +420 26603 5364 • Fax: +420 28482 1646
Email: nakup@suweco.cz • Web site: <http://www.suweco.cz>

FINLAND

Akateeminen Kirjakauppa, PO BOX 128 (Keskuskatu 1), FIN-00101 Helsinki
Telephone: +358 9 121 41 • Fax: +358 9 121 4450
Email: akatilais@akateeminen.com • Web site: <http://www.akateeminen.com>

FRANCE

Form-Edit, 5, rue Janssen, P.O. Box 25, F-75921 Paris Cedex 19
Telephone: +33 1 42 01 49 49 • Fax: +33 1 42 01 90 90
Email: formedit@formedit.fr • Web site: <http://www.formedit.fr>

Lavoisier SAS, 145 rue de Provigny, 94236 Cachan Cedex
Telephone: + 33 1 47 40 67 02 • Fax +33 1 47 40 67 02
Email: romuald.verrier@lavoisier.fr • Web site: <http://www.lavoisier.fr>

GERMANY

UNO-Verlag, Vertriebs- und Verlags GmbH, Am Hofgarten 10, D-53113 Bonn
Telephone: + 49 228 94 90 20 • Fax: +49 228 94 90 20 or +49 228 94 90 222
Email: bestellung@uno-verlag.de • Web site: <http://www.uno-verlag.de>

HUNGARY

Librotrade Ltd., Book Import, P.O. Box 126, H-1656 Budapest
Telephone: +36 1 257 7777 • Fax: +36 1 257 7472 • Email: books@librotrade.hu

INDIA

Allied Publishers Group, 1st Floor, Dubash House, 15, J. N. Heredia Marg, Ballard Estate, Mumbai 400 001,
Telephone: +91 22 22617926/27 • Fax: +91 22 22617928
Email: alliedpl@vsnl.com • Web site: <http://www.alliedpublishers.com>

Bookwell, 2/72, Nirankari Colony, Delhi 110009
Telephone: +91 11 23268786, +91 11 23257264 • Fax: +91 11 23281315
Email: bookwell@vsnl.net

ITALY

Libreria Scientifica Dott. Lucio di Biasio "AEIOU", Via Coronelli 6, I-20146 Milan
Telephone: +39 02 48 95 45 52 or 48 95 45 62 • Fax: +39 02 48 95 45 48
Email: info@libreriaaeiou.eu • Website: www.libreriaaeiou.eu

JAPAN

Maruzen Company, Ltd., 13-6 Nihonbashi, 3 chome, Chuo-ku, Tokyo 103-0027
Telephone: +81 3 3275 8582 • Fax: +81 3 3275 9072
Email: journal@maruzen.co.jp • Web site: <http://www.maruzen.co.jp>

REPUBLIC OF KOREA

KINS Inc., Information Business Dept. Samho Bldg. 2nd Floor, 275-1 Yang Jae-dong SeoCho-G, Seoul 137-130
Telephone: +02 589 1740 • Fax: +02 589 1746 • Web site: <http://www.kins.re.kr>

NETHERLANDS

De Lindeboom Internationale Publicaties B.V., M.A. de Ruyterstraat 20A, NL-7482 BZ Haaksbergen
Telephone: +31 (0) 53 5740004 • Fax: +31 (0) 53 5729296
Email: books@delindeboom.com • Web site: <http://www.delindeboom.com>

Martinus Nijhoff International, Koraalrood 50, P.O. Box 1853, 2700 CZ Zoetermeer
Telephone: +31 793 684 400 • Fax: +31 793 615 698
Email: info@nijhoff.nl • Web site: <http://www.nijhoff.nl>

Swets and Zeitlinger b.v., P.O. Box 830, 2160 SZ Lisse
Telephone: +31 252 435 111 • Fax: +31 252 415 888
Email: info@swets.nl • Web site: <http://www.swets.nl>

NEW ZEALAND

DA Information Services, 648 Whitehorse Road, MITCHAM 3132, Australia
Telephone: +61 3 9210 7777 • Fax: +61 3 9210 7788
Email: service@dadirect.com.au • Web site: <http://www.dadirect.com.au>

SLOVENIA

Cankarjeva Založba d.d., Kopitarjeva 2, SI-1512 Ljubljana
Telephone: +386 1 432 31 44 • Fax: +386 1 230 14 35
Email: import.books@cankarjeva-z.si • Web site: <http://www.cankarjeva-z.si/uvvoz>

SPAIN

Díaz de Santos, S.A., c/ Juan Bravo, 3A, E-28006 Madrid
Telephone: +34 91 781 94 80 • Fax: +34 91 575 55 63
Email: compras@diazdesantos.es, carmela@diazdesantos.es, barcelona@diazdesantos.es, julio@diazdesantos.es
Web site: <http://www.diazdesantos.es>

UNITED KINGDOM

The Stationery Office Ltd, International Sales Agency, PO Box 29, Norwich, NR3 1 GN
Telephone (orders): +44 870 600 5552 • (enquiries): +44 207 873 8372 • Fax: +44 207 873 8203
Email (orders): book.orders@tso.co.uk • (enquiries): book.enquiries@tso.co.uk • Web site: <http://www.tso.co.uk>

On-line orders

DELTA Int. Book Wholesalers Ltd., 39 Alexandra Road, Addlestone, Surrey, KT15 2PQ
Email: info@profbooks.com • Web site: <http://www.profbooks.com>

Books on the Environment

Earthprint Ltd., P.O. Box 119, Stevenage SG1 4TP
Telephone: +44 1438748111 • Fax: +44 1438748844
Email: orders@earthprint.com • Web site: <http://www.earthprint.com>

UNITED NATIONS

Dept. I004, Room DC2-0853, First Avenue at 46th Street, New York, N.Y. 10017, USA
(UN) Telephone: +800 253-9646 or +212 963-8302 • Fax: +212 963-3489
Email: publications@un.org • Web site: <http://www.un.org>

UNITED STATES OF AMERICA

Bernan Associates, 4501 Forbes Blvd., Suite 200, Lanham, MD 20706-4346
Telephone: 1-800-865-3457 • Fax: 1-800-865-3450
Email: customercare@bernan.com • Web site: <http://www.bernan.com>

Renouf Publishing Company Ltd., 812 Proctor Ave., Ogdensburg, NY, 13669
Telephone: +888 551 7470 (toll-free) • Fax: +888 568 8546 (toll-free)
Email: order.dept@renoufbooks.com • Web site: <http://www.renoufbooks.com>

Orders and requests for information may also be addressed directly to:

Marketing and Sales Unit, International Atomic Energy Agency

Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Telephone: +43 1 2600 22529 (or 22530) • Fax: +43 1 2600 29302
Email: sales.publications@iaea.org • Web site: <http://www.iaea.org/books>

**INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA
ISBN 978-92-0-102610-1
ISSN 1995-7807**