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REVIEW OF FUEL FAILURES IN WATER COOLED REACTORS (2006–2015)

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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-2.5

REVIEW OF FUEL FAILURES IN WATER COOLED REACTORS (2006–2015)

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FOREWORD

One of the IAEA's statutory objectives is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world." One way this objective is achieved is through the publication of a range of technical series. Two of these are the IAEA Nuclear Energy Series and the IAEA Safety Standards Series.

According to Article III.A.6 of the IAEA Statute, the safety standards establish "standards of safety for protection of health and minimization of danger to life and property". The safety standards include the Safety Fundamentals, Safety Requirements and Safety Guides. These standards are written primarily in a regulatory style, and are binding on the IAEA for its own programmes. The principal users are the regulatory bodies in Member States and other national authorities.

The IAEA Nuclear Energy Series comprises reports designed to encourage and assist R&D on, and application of, nuclear energy for peaceful uses. This includes practical examples to be used by owners and operators of utilities in Member States, implementing organizations, academia, and government officials, among others. This information is presented in guides, reports on technology status and advances, and best practices for peaceful uses of nuclear energy based on inputs from international experts. The IAEA Nuclear Energy Series complements the IAEA Safety Standards Series.

Since the 1970s, following recommendations of the IAEA Technical Working Group on Fuel Performance and Technology, the IAEA has been involved in the worldwide analysis of fuel failures in water cooled reactors under normal operating conditions (anticipated operational occurrences). 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 such failures for two reasons. Firstly, the problems and inconvenience caused by fuel failures in plant operations can be significant. Secondly, the generally accepted goal of achieving a zero failure rate requires detailed knowledge of existing failure mechanisms, their root causes and remedies.

IAEA publications have presented analyses of fuel performance in detail, including fuel failure statistics and mechanisms for 1987–2006, with more limited data for 1969–1986. The present publication summarizes fuel failure occurrences, their mechanisms and root causes, as well as fuel failure prevention and management in plant operation for 97% of all light and heavy water cooled nuclear power reactors (pressurized light water reactors (PWRs), boiling water reactors (BWRs), water cooled, water moderated energy reactors (WWERs), and Canada deuterium–uranium (CANDU) reactors and other pressurized heavy water reactors (PHWRs)) operated worldwide in 2006–2015. Data on fuel failures in 1987–2006 extracted from the above mentioned IAEA fuel failure reports were included and analysed in the present report together with 2006–2015 fuel failure data to reveal long term (1987–2015) trends in fuel performance. Apart from fuel rod 'leakers', fuel structural damage and other fuel assembly issues are also discussed at a qualitative level.

The IAEA is grateful to all the participants who contributed to this report. The IAEA officer responsible for this publication was M. Veshchunov of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The results of a fuel failure analysis were published in the IAEA's journal Atomic Energy Review in 1979 under the title The Main Causes of Fuel Element Failure in Water-cooled Power Reactors [1]. The IAEA publication Review of Fuel Failures in Water Cooled Reactors has been issued twice, in 1998 [2] and in 2010 [3]. In 1992, the IAEA held a Technical Meeting on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management [4], and in 2002 one on Fuel Failure in Water Reactors: Causes and Mitigation [5].

Reference [1] was based on the available literature and the authors' personal experience, and considered fuel failure mechanisms that had been observed in water cooled reactors in 1966–1977. References [2] and [3] were based on an analysis of fuel failure data received from the members of the IAEA Technical Working Group on Fuel Performance and Technology (TWG-FPT) and national organizations within the Member States represented in the TWG-FPT. These publications summarized fuel failure statistics and mechanisms in the time spans 1987–1994 [2] and 1994–2006 [3]. The present publication contains in-depth information on fuel operation parameters for about 95% of all water cooled power reactors worldwide that were operating in 2006–2015.

1.2. OBJECTIVE

This publication has the same two objectives as 2010's Ref. [3]: to disseminate fuel failure statistics for about 95% of all water cooled power reactors worldwide that were operating in 2006–2015, and to present an in-depth analysis including fuel operating parameters, fuel design features and fuel failure mitigation measures, also considering correlations between them.

1.3. SCOPE

This publication provides information on fuel failure (i.e. fuel rod leak rate) statistics and on fuel assembly (FA) damages observed during 2006–2015 in water cooled power reactors worldwide. Data on fuel performance have been analysed and updates on fuel operating parameters, design and materials have been included. By considering results that had been previously published by the IAEA in fuel failure reviews [1–3], almost three decades (1987–2015) of fuel performance experience in water cooled power reactors has been summarized in this publication.

The evolution of fuel failures from 2006 to 2015, including their causes, in light water reactors (LWRs), pressurized water reactors (PWRs), boiling water reactors (BWRs), water cooled, water moderated power reactors (WWERs) and pressurized heavy water reactors (PHWRs) including Canada deuterium–uranium (CANDU) reactors is presented and assessed for in-core fuel performance. Damages to FAs and the percentage of reactors that operate with zero 'leakers' are also discussed in this publication.

This publication is an update of IAEA Nuclear Energy Series No. NF-T-2.1, Review of Fuel Failures in Water Cooled Reactors [3], issued in 2010, with new fuel performance and reliability data for 2006–2015 analysed and included.

1.4. STRUCTURE

This publication has a similar structure to the previous IAEA Fuel Failure Reviews [2, 3]. Data are summarized and calculated for about 95% of all nuclear power units worldwide in operation during the period 2006–2015. Section 2 provides statistical data on fuel failures that occurred in water cooled power reactors worldwide during 2006–2015 and analyses mechanisms of fuel failures under normal operational conditions. It complements the results presented at the conferences TopFuel 2009 [6], TopFuel 2010 [7] and TopFuel 2013 [8]. Information on this subject can also be found in the IAEA publications Refs [3–5], which consider fuel failure experience from 1970 to

2006. In the present publication, the emphasis is on fuel operational experience from 2006–2015, including data on fuel failure rates and mechanisms for the time span of 1987–2015 that are grouped in four-year periods to illustrate trends over the last three decades. FA issues, such as bow and its consequences in PWRs and WWERs, axial offset anomaly in PWRs, and channel bow and control blade cracking in BWRs, are also considered on a qualitative level.

Section 3 describes changes that have occurred during 2006–2015 in comparison with 1994–2006 [3]. It considers the fuel operating environment and fuel design and their role in ensuring good economical and fuel performance in nuclear power plants. Other parts of this section also provide an overview of changes in the detection, examination and analysis of fuel failures and in fuel failure prevention and management in plant operation.

Section 4 contains major conclusions on fuel performance in water cooled power reactors during the period 2006–2015, with operating experience summarized for 1987–2015.

2. WORLD OVERVIEW OF FUEL FAILURES FROM 2006 TO 2015

2.1. METHODOLOGY OF FUEL FAILURE RATE CALCULATIONS

2.1.1. Questionnaire

During the 1994–2006 time period, data on fuel failures were collected without applying a specific format that considers fuel reloads. Recognizing the need for such data, an expert group formed by the IAEA TWG-FPT in February 2007 developed a questionnaire that was sent to its members and to some organizations to obtain data in a format that would facilitate the calculation of fuel failure rates. Among the organizations involved in this study were Atomic Energy of Canada (Canada), Électricité de France (France), ENUSA Industrias Avanzadas (Spain), the Electric Power Research Institute (EPRI) (United States of America) and TVEL Fuel Company (Russian Federation).

A questionnaire in a spreadsheet format was developed to request data on a year-by-year basis for 2006 to 2010. After 2010, following recommendations from the IAEA TWG-FPT meeting, the 2010–2015 datasets were requested in January 2016 with the purpose of supplementing the Review of Fuel Failures in Water Cooled Reactors [3] published by the IAEA in 2010. The following data were requested:

— Plant name or number of plants reloaded during each year;
— Fuel cycle characteristics (number of cycles and their duration);
— Number of leaking FAs and fuel rods (FRs);
— Failure type;
— Number of fresh FAs loaded in each cycle;
— Total number of FAs in reactor cores in the respective year;
— Enrichment and burnup of failed FAs in discharged batches;
— Plants without fuel failures;
— Other damages without cladding breach (spacer grids, skeleton, FA bow, axial offset anomaly, etc.).

The above mentioned data were requested for LWRs (with fuel reload during each specific year) and CANDUs (with on-line refuelling). For LWRs operating throughout the year, i.e. without outages, only information on the plant name, cycle characteristics, enrichment and number of FAs in core was requested.

To facilitate the task of obtaining information for the questionnaire, codes such as PWR-1, BWR-3 or CANDU-2 were used in place of the actual plant names. If data were given for multiple plants, only the reactor type was identified. Fuel failure data were given in a similar form. For CANDU and other PHWR plants the number of identified failed bundles and the total number of bundles loaded during the year were requested. This process respected the anonymity of the data providers according to their wishes.

The following failed fuel categories were included in the questionnaire under the headings ‘failure of ‘leaker’ type’ and ‘other damage’:

— Grid-to-rod fretting;
— Crud or corrosion;
— Pellet–cladding interaction/stress corrosion cracking (PCI/SCC);
— Debris or debris fretting;
— Fuel handling damage;
— Fabrication;
— Undetermined or unknown.

Other damages (without FR leak) were also identified:

— FA bow / incomplete control rod insertion;
— Axial offset anomaly, also known as crud induced power shifts;
— Spacer grid damage;
— Channel distortion (bow) in BWRs;
— Control blade cracking and leaking in BWRs.

Comparison of the above mentioned mechanisms of fuel leakage and other damages without fuel leakage observed from 2006 to 2015 with those observed from 1994 to 2006 [3] has shown that they are very similar, except for baffle jetting, which was not observed during 2006–2015. However, damage to control blades in BWRs was first observed during 2006–2015.

Some organizations involved in the analysis of fuel failures, such as EPRI, had a separate failure category entitled ‘duty related’. PCI/SCC is only one of the failure mechanisms within the ‘duty related’ category. To be consistent with the IAEA’s Review of Fuel Failures in Water Cooled Reactors published in 2010 [3], the fuel failure category ‘duty related’ will not be used in this publication. Descriptions of unusual multiple failures and new failure mechanisms, if applicable, were also requested.

It is worthwhile to emphasize that the presentation of data at the plant specific level was considered the best option. Where data were available for a fleet of reactors only, this reporting was acceptable if accompanied by the number of FAs discharged yearly and number of units with and without fuel failures in each specific year. To calculate the FR failure rate, it was necessary to know the number of FAs discharged yearly and the average number of FRs in the FA for the discharged batches. If these data were not provided, they were taken from the literature or requested separately.

This publication was not intended to form a fuel failure database on a plant-by-plant basis. The decision on the type of data presented was left up to the participants.

2.1.2. Type of data received

The analysis of all the fuel failure data submitted categorized the data into three types: (1) data provided on a reactor-by-reactor basis; (2) data provided on a station basis, for example, summarized for all units of each station as provided by Canada (4 stations with 19 plants); and (3) grouped by reactor design within the country in question. A large majority of data presented are of the first type, sometimes provided with an indication of the real plant or unit name and sometimes with an anonymized name or number.

Countries that reported data on a reactor-by-reactor basis include: Argentina, Belgium, Brazil, Bulgaria, the Czech Republic, Finland, Hungary, Japan (for the period 2006–2010), the Netherlands, Romania, the Russian Federation, Slovakia, Spain, Sweden, Switzerland, Ukraine (for the period 2010–2015), the United Kingdom and the USA (for the period 2011–2015). Countries that reported data on a station basis include: Canada, China (for the period 2006–2010) and Ukraine (for the period 2006–2009). Countries who grouped their reactors by design include: France (grouped into 3.66 m and 4.27 m PWR units), Germany (BWRs in one group, PWRs divided into three groups: 15 × 15, 16 × 16 and 18 × 18), India (PHWRs in one group and BWRs in another), the Republic of Korea (grouped by reactor types: all PWRs and all CANDUs) and the USA (for the period of 2006–2010, grouped by reactor types: BWRs and PWRs).

2.1.3. Completeness of fuel failure datasets received

Thirty countries were operating nuclear power reactors in 2015. Of these countries, 24 were TWG-FPT members: Argentina, Belgium, Brazil, Bulgaria, Canada, China, the Czech Republic, Finland, France, Germany, Hungary, India, Japan, the Republic of Korea, the Netherlands, Romania, the Russian Federation, Slovakia, Spain, Sweden, Switzerland, Ukraine, the UK and the USA. They reported on fuel performance in their water cooled reactors in 2006–2015. Norway is also a TWG-FPT Member but operates two research reactors and no nuclear power plants. Six countries (Armenia, the Islamic Republic of Iran, Mexico, Pakistan, Slovenia and South Africa) are IAEA Member States and operate nuclear power reactors ((1 WWER-440 (Armenia), 2 WWER (Islamic Republic of Iran), 2 BWRs (Mexico), 1 PWR (Pakistan), 1 PWR (Slovenia) and 2 PWRs (South Africa)), but are not TWG-FPT Members and were not involved in the data collection. Data from Taiwan, China, were not included. Consequently, it was expected that the IAEA would receive 24 completed questionnaires from the countries that are members of the IAEA TWG-FPT. Table 1 lists the participants and the completeness of the fuel performance datasets received.

During the period 2006–2015, 28 new nuclear power plant units were connected to the grid (15 PWRs, 6 WWER-1000s, 1 CANDU and 6 other PHWRs) and 21 nuclear power plant units were shut down (9 PWRs, 5 BWRs, 4 WWER-440s and 3 CANDUs) [9]. By 31 December 2015, there were 394 nuclear power plant units in operation with fuel reloads during 2006–2015 (216 PWRs, 74 BWRs, 55 WWERs and 49 CANDUs and other PHWRs). These units accumulated 3630 years of operational experience. This means that the data received from the IAEA TWG-FPT Member States cover 95.7% of all reactors that operated with fuel reloads from 2006 to 2015 and 94.8% of their reactor-years of experience. These values are close to those submitted to the IAEA to characterize fuel failure patterns during the 1994–2006 time period, with values of 93% and 95.7%, respectively [3].

It should be noted that the quality of data received by the IAEA is not uniform for all presented packages. For some countries, post-irradiation examination data and identification of the causes of fuel failure were missing. The completeness of fuel failure datasets and basic data on reactor years of experience is recorded in Table 1.

2.1.4. Fuel failure or leak rate calculation

The primary characteristic for calculating fuel failure is the number of leaking FAs in discharged batches, which are usually determined by sipping (or canister techniques in the case of WWERs). The fuel failure rate is defined as the number of failed FAs per 1000 discharged FAs for each specific year and each specific unit or units. Another characteristic is an FR failure rate, which is calculated as the ratio of the number of leaking FRs to the total number of FRs in discharged FAs, also for each specific year and each specific unit or units.

To determine failure rates, the results of sipping show the number of leaking FAs in combination with the results of fuel inspections and examinations, which reveal the number of failed FRs in FAs. In-pool ultrasonic testing can provide information on many leaking FRs without a need to dismantle the FA but is less frequently used. Hot cell post-irradiation examination of defective FRs is a more direct method to determine the failure mechanism once the failed FR is identified within the FA in the spent fuel pool, but it is more expensive and limited in its availability. Therefore, the exact numbers of failed FRs are not always known, whereas the number of failed FAs is more accurately known. Some literature provides only data specific to the number of leaking FAs, while other information is provided on FR failure rates. For comparison with data in the literature, one needs to be aware of different definitions for fuel failure rates in use (FR or FA).

Evaluation of the fuel failure rate for water cooled reactors in Ref. [2] was performed based on published information supplemented by fuel failure information requested and received for the study from the following Members of the IAEA TWG-FPT: Canada, China, France, Germany, Japan, the Russian Federation, Sweden, Switzerland and the USA.

TABLE 1. BASIC DATA FOR 2006–2015 ON REACTOR YEARS OF EXPERIENCE AND COMPLETENESS OF FUEL PERFORMANCE DATASETS USED *(including TWG-FPT non-members)*

Location	Total units in operation with fuel reloads in 2006–2015	Breakdown by type of units in operation with fuel reloads in 2006–2015	Reactor-years of experience with fuel reloads in 2006–2015	Breakdown by type of reactor-years of experience with fuel reloads in 2006–2015, where known	Number of units covered by the data received	Number of years covered by the data received	Completeness of datasets
Argentina	3	PHWR KWU type: 2 CANDU: 1	22		3	21	Incomplete
Belgium	7	PWR: 7	70		7	70	Complete
Brazil	2	PWR: 2	20		2	20	Complete
Bulgaria	2	WWER: 2	22		4 (2006) 2 (2007–2015)	4 (2006) 18 (2007–2015)	Incomplete
Canada	19	CANDU: 19	190		19	190	Complete
China	22	PWR: 18 WWER: 2 CANDU: 2	142	PWR: 105 WWER: 17 CANDU: 20	PWR: 8 WWER: 2 CANDU: 2 Total: 12 (2006–2010)	PWR: 36 WWER: 8 CANDU: 10 Total: 54 (2006–2010)	Incomplete
Czech Republic		WWER: 6	60		6	60	Incomplete
Finland	4	BWR: 2 WWER: 2	40		4	40	Complete
France	58	PWR: 58	580		58	580	Complete
Germany	8 (in 2015)	PWR: 6 BWR: 2	127		2006:17 2007: 15 2008: 12 2009: 16 2010: 15 2011: 15 2012: 11 2013: 9 2014: 9 2015: 8	127	Complete

TABLE 1. BASIC DATA FOR 2006–2015 ON REACTOR YEARS OF EXPERIENCE AND COMPLETENESS OF FUEL PERFORMANCE DATASETS USED *(including TWG-FPT non-members)* (cont.)

Location	Total units in operation with fuel reloads in 2006–2015	Breakdown by type of units in operation with fuel reloads in 2006–2015	Reactor-years of experience with fuel reloads in 2006–2015	Breakdown by type of reactor-years of experience with fuel reloads in 2006–2015, where known	Number of units covered by the data received	Number of years covered by the data received	Completeness of datasets
Hungary	4	WWER: 4	40		4	40	Incomplete
India	21	WWER: 1 BWR: 2 PHWR: 18	164	WWER: 1 BWR: 20 PHWR: 143	WWER: 0 BWR: 2 (2006–2011) PHWR: 17 (2006–2015) Total: 19	WWER: 0 BWR: 12 (2006–2011) PHWR: 143 (2006–2015) Total: 155	Incomplete
Japan[a]	43 (in 2010)	PWR: 21 BWR: 22	240	PWR: 120 BWR: 120 (2006–2010)	PWR: 24 BWR: 24 (2006–2010) Total: 48	PWR: 120 BWR: 120 (2006–2010) Total: 240	Complete
Republic of Korea	24	PWR: 20 CANDU: 4	214	PWR: 174 CANDU: 40	PWR: 20 PHWR: 4 Total: 24	PWR: 174 CANDU: 40 Total: 214	Complete
Netherlands	1	PWR: 1	10		1	10	Complete
Romania	2	CANDU: 2	18		2	18	Complete
Russian Federation	17	WWER: 17	159		17	159	Incomplete
Slovakia	4	WWER: 4	43		4	43	Complete
Spain	7 (2013–2015)	PWR: 6 BWR: 1	77		8 (2006–2012) 7 (2013–2015)	77	Complete
Sweden	10	PWR: 3 BWR: 7	100		10	100	Complete
Switzerland	5	PWR: 3 BWR: 2	50		5	50	Complete

TABLE 1. BASIC DATA FOR 2006–2015 ON REACTOR YEARS OF EXPERIENCE AND COMPLETENESS OF FUEL PERFORMANCE DATASETS USED *(including TWG-FPT non-members)* (cont.)

Location	Total units in operation with fuel reloads in 2006–2015	Breakdown by type of units in operation with fuel reloads in 2006–2015	Reactor-years of experience with fuel reloads in 2006–2015	Breakdown by type of reactor-years of experience with fuel reloads in 2006–2015, where known	Number of units covered by the data received	Number of years covered by the data received	Completeness of datasets
Ukraine	15	WWER: 15	150		15	150	Incomplete
UK	1	PWR: 1	10		1	10	Complete
USA	99 (2015)	PWR: 65 BWR: 34	990		99	990	Complete
Total	All: 394	PWR: 216 BWR: 74 WWER: 55 CANDU: 29 Other PHWR: 20	3630		377	3441	
Total in %	100		100		95.7%	94.8%	

Note: The source of all data included in Table 1 is the answers given to the IAEA questionnaire.

[a] Following the accident at the Fukushima Daiichi nuclear power plant on 11 March 2011, all nuclear reactors in Japan were progressively shut down owing to safety concerns. Since the accident, there have been several attempts to restart several units, e.g. two reactors of the Sendai nuclear power plant were restarted on 11 August 2015 and 1 November 2015, respectively. In addition, according to Ref. [9], 10 BWR and 3 PWR reactors were shut down from January 2009 to April 2015. The contribution of nuclear energy to electricity production dropped from 24.93% in 2010 to 0.52% in 2015. Fuel failure data given in this table are for 2006–2010.

[b] This table does not include the 2 PWR and 4 BWR units in Taiwan, China. No fuel performance data was provided for 9 new Chinese PWRs in 2011–2015.

For the calculation of the FR failure rate, the following expression was used for the evaluation in Ref. [2]:

$$R \approx rD / N \qquad (1)$$

where

R is an annual FR failure rate (leaking FRs/intact FRs in core);
N is the number of FRs in the core for all operating reactors with or without refuelling in the respective year;
D is the number of leaking FAs found and discharged from all operating reactors in the respective year;

and r is the average number of leaking FRs per leaking FA, equal to 1.1 for BWR, WWER-440 and CANDU fuel and 1.6 for PWR and WWER-1000 fuel.

The *r* values were selected based on LWR experience in the USA [10]. Average FR failure rates over several years (and/or regions) were calculated by summing the annual (regional) values of *D* and *N*, respectively, rather than by averaging the annual (regional) *R* values.

For the review of fuel failures from 2006–2015, the same methodology [10] was used as for the IAEA 2010 fuel failure review. For the 490 failed PWR FAs and the 137 failed BWR FAs in 2006–2015, exact numbers were reported for the failed FRs with an indication of the cause of failure. Figure 1 shows these results. The average number of failed FRs per failed PWR FA is 1.3 and per BWR FA is 1.1.

The average number of failed FRs in failed CANDU bundles was 1.0 (compared with 1.1 in the 2010 IAEA review [3]), which agreed with the provided CANDU datasets. Information provided on failed WWER FAs with indication of the number of failed FRs included in the time frame of 2006–2015 indicated 66 failed FAs of WWER-1000s in Bulgaria, China, the Czech Republic, the Russian Federation and Ukraine with 70 leaking FRs. WWER-440s had 3 failed FAs with 3 leaking FRs in Finland, i.e. 1 leaking FR per failed FA. This value was also used for the calculation of WWER FR failure rates.

2.1.5. Percentage of units operated free of FR 'leakers'

Another way to present the fuel reliability trend is to evaluate the percentage of units experiencing an outage and demonstrating zero fuel failures. The percentage of reactors 'free of defects' for a specific year is the number of reactors free of defects with an outage in that year divided by the total number of reactors with an outage in the same year.

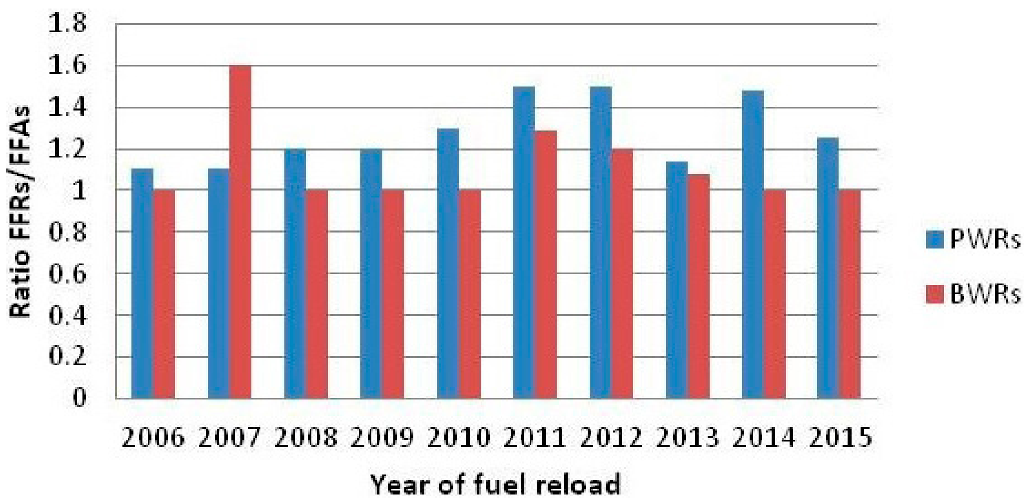


FIG. 1. Ratio of the number of failed FRs to the number of failed FAs in PWRs and BWRs (based on post-irradiation examination results).

2.2. OVERVIEW OF FR FAILURES BY REACTOR TYPE

2.2.1. PWRs

2.2.1.1. Evaluation of PWR fuel leak rates

Worldwide and regional FA failure rates for the USA (65 units), France (58 units) and the rest of Europe (Belgium, Germany, the Netherlands, Spain, Sweden, Switzerland and the UK, altogether 27 units in 2015), Japan (24 units in 2006–2010) and the Republic of Korea (20 units) are presented in Fig. 2. The FR failure rates are indicated in Fig. 3. The world data also include data on 12 Chinese PWR units in 2006–2010.

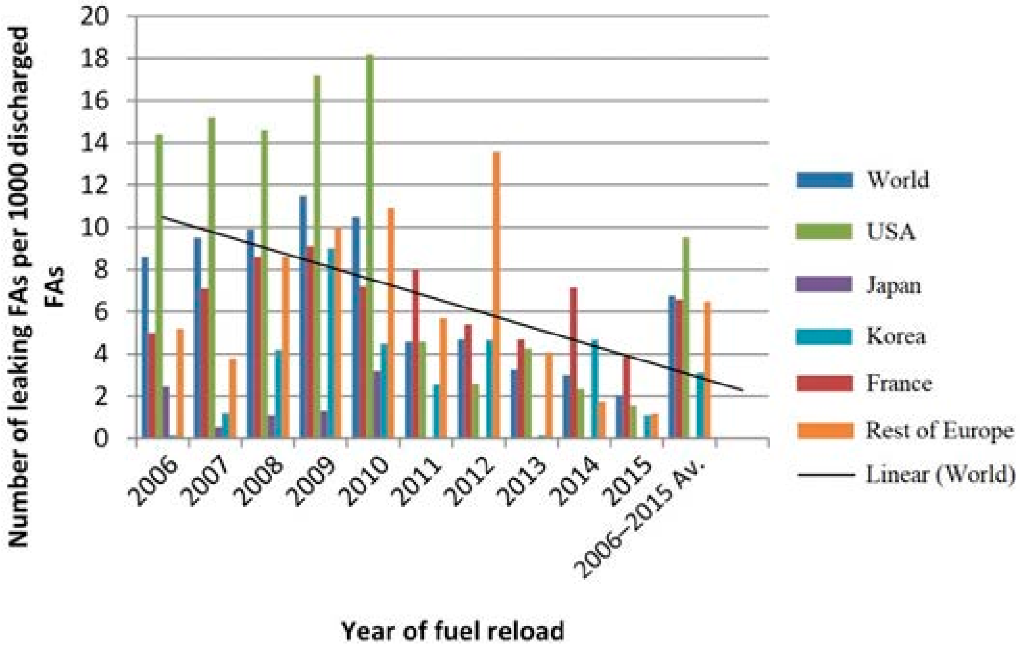


FIG. 2. PWR FA failure rates in 2006–2015. Linear extrapolation is for world averages.

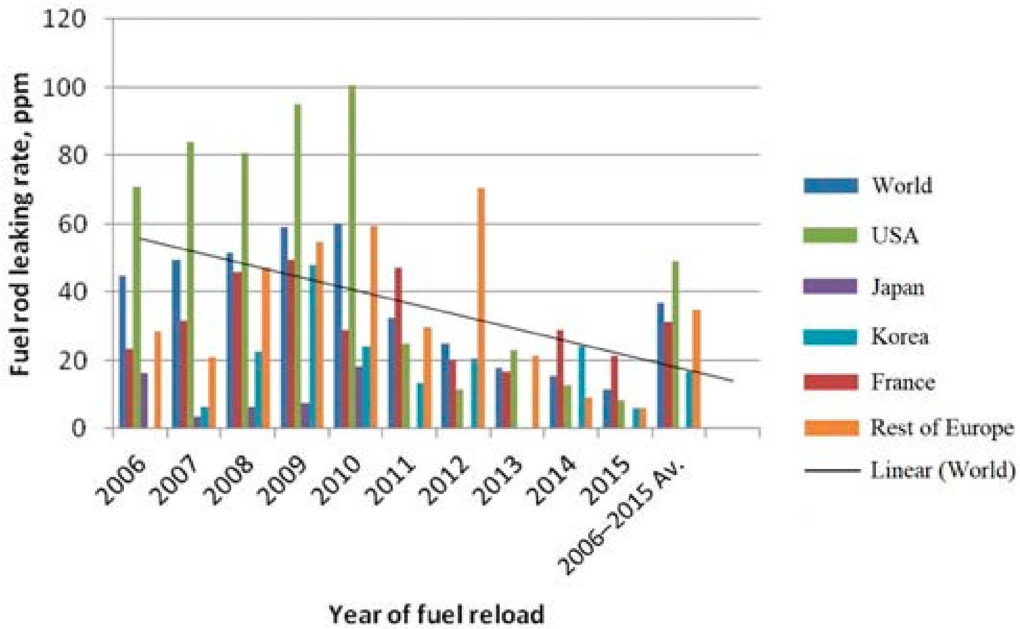


FIG. 3. PWR FR failure rates in 2006–2015. Linear extrapolation is for world averages (ppm means 1×10^6).

Figures 2 and 3 and Table 2 demonstrate an overall downward trend in the rate of fuel failure in PWRs from 1994 to 2006 and from 2006 to 2015. Data for the time frame 1994–2006 in Table 2 were taken from page 20 and table 3.3 of Ref. [3]. The calculation methodology is described in Eq. (1).

To further observe the long term trends in the PWR FR failure rate, the world average data for 1987–2015 were grouped in 4 year periods (Fig. 4). Figure 4 clearly shows a decrease in the FR failure rate in PWRs over the time frame of 1987–2015. Data for the time frame 1987–1994 were taken from table 3.7 and fig. 3.25 on page 37 of Ref. [3]. The calculation methodology is described in section 3.3.2 of Ref. [3].

TABLE 2. PWR FA AND FR FAILURE RATES FOR PWRs, AVERAGES FOR 1994–2006 AND 2006–2015 [3]

Region	FA failure rate in 1994–2006 ($\times 10^{-3}$)	FA failure rate in 2006–2015 ($\times 10^{-3}$)	FR failure rate in 1994–2006 ($\times 10^{-6}$)	FR failure rate in 2006–2015 ($\times 10^{-6}$)
World	13.8	7.1	86.8	36.7
USA	20.9	9.5	131.6	49.1
France	8.8	6.7	56.9	31.3
Rest of Europe	16.0	6.5	108.1	34.7
Republic of Korea	10.6	3.1	40.5	16.5
Japan (2006–2010)	0.5	1.7	3.7	10.3

Note: FA and FR failure rates are reported as 1 in 1000 (1×10^{-3}) and 1 in 1 000 000 (ppm), respectively.

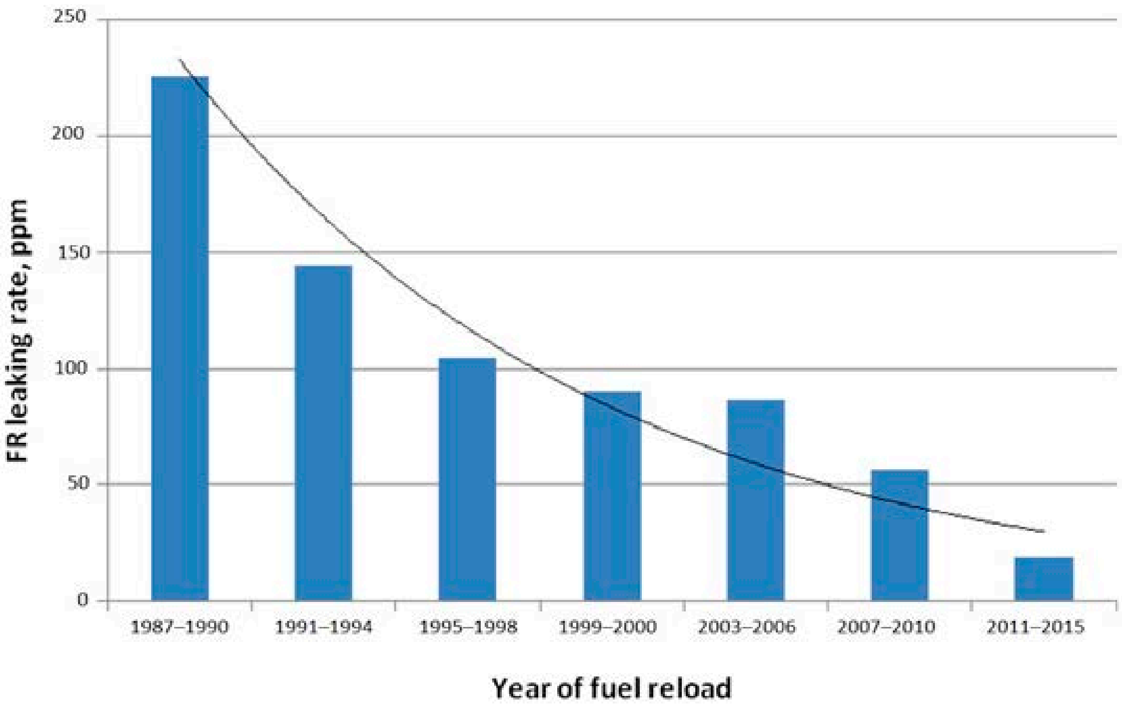


FIG. 4. World average PWR FR failure rates grouped in 4 year periods for the time frame 1987–2015 [3].

2.2.1.2. Distribution of fuel failure causes in PWRs

For the 490 failed PWR FAs worldwide in 2006–2015, the exact numbers and mechanisms of failed FRs were reported in Section 2.1.3 and are summarized in Table 3. Values for 1987–1994 and 1995–2006 were taken from table 3.8 (page 38) of Ref. [3]. Data were grouped in 4 year periods. The category 'unknown/undetermined' includes cases in which investigation results did not provide a conclusive cause and mechanism for the fuel failure. The observed increase in the unknown/undetermined fuel failures during 2011–2015 is likely because post-irradiation examination is still ongoing.

To better compare the evolution of failure mechanisms in 1987–2015 with one another, the statistical data in Table 3 were calculated on the basis of post-irradiation examination results (with the unknown/undetermined cases not being considered) and are summarized in Fig. 5 to show the share of PWR fuel failure causes observed in 2006–2010 and 2011–2015.

Figure 5 and Table 3 show that grid-to-rod fretting has been the main identified fuel failure mechanism in PWRs from the mid-1990s to 2015 with an ~88% share in the early 2000s and a share of ~59% during 2006–2015. The debris-related fretting fuel failure mechanism prevailed in the late 1980s and early 1990s with a share of ~50%, but decreased significantly to ~15% in the mid-1990s, growing again to ~34% in 2010–2015.

This reduction is a result of the significant effort to improve the design and materials of FA components, e.g. improvement to the design of spacer grids. The reduction in debris related failures seems to be more complicated and may be related to the cleaning of the primary coolant systems and better debris filters for FAs.

2.2.1.3. Percentage of PWRs free of fuel leak

Using 2006–2015 fuel performance data received from members of the IAEA TWG-FPT, the percentage of PWR units free of a fuel 'leaker' was calculated (see Fig. 6). The average value for 2006–2015 for all PWRs was 81.7% (76.6% in 2000–2006), while it was 79.4% for PWRs in the USA (62.7% in 2000–2006), 78.5% for France (75.6% in 2000–2006) and 82.3% for the rest of Europe (68.6% in 2000–2006). These data have shown a constant increase in PWR units operating defect free in the 2006–2015 time period.

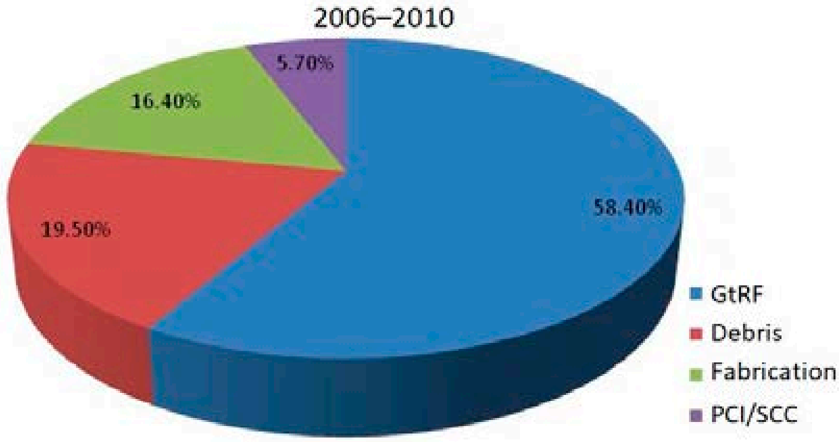
2.2.1.4. Major observations on PWR fuel failures in 2006–2015

Major observations on PWR fuel failures in 2006–2015 are as follows:

— The world average fuel failure rate for 2006–2015 is 7.1 per 1000 discharged FAs, i.e. approximately half the rate during 1994–2006 (13.8).

TABLE 3. PWR FUEL FAILURE MECHANISMS WORLDWIDE FROM 1987 TO 2015 [3]

	Proportion of fuel failures each mechanism is responsible for (%)						
	1987–1990	1991–1994	1995–1998	1999–2002	2003–2006	2007–2010	2011–2015
Grid-to-rod fretting	16.6	42.7	73.0	87.6	78.0	58.4	57.9
Debris	55.6	46.7	14.5	7.1	13.9	19.5	33.7
Fabrication	20.8	6.7	9.5	3.4	7.2	16.4	8.4
Crud or corrosion	0	0	2.2	1.5	0	0	0
PCI/SCC	0	0	0	0	0.9	5.7	0
Handling	2.8	3.8	0.8	0.4	0	0	0
Baffle jetting	4.2	0	0	0	0	0	0
Unknown/undetermined	50.0	48.0	26.7	14.6	33.2	28.4	40.2



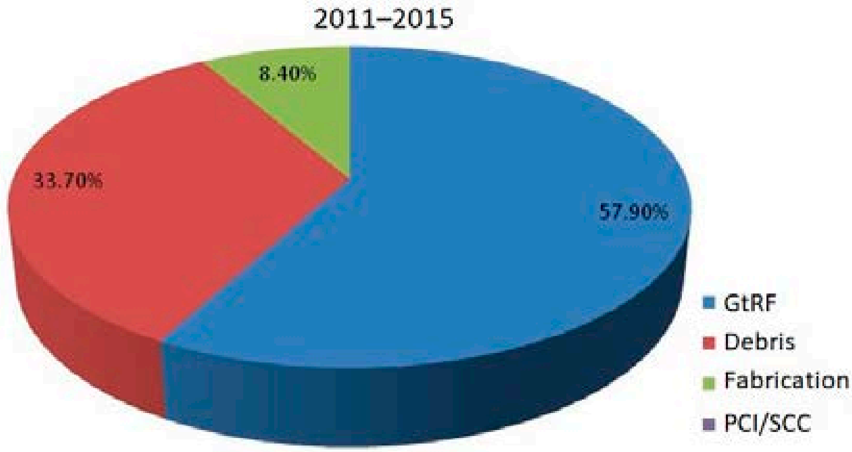


FIG. 5. PWR fuel leak causes worldwide in 2006–2010 (top) and 2011–2015 (bottom). GtRF — grid-to-rod fretting.

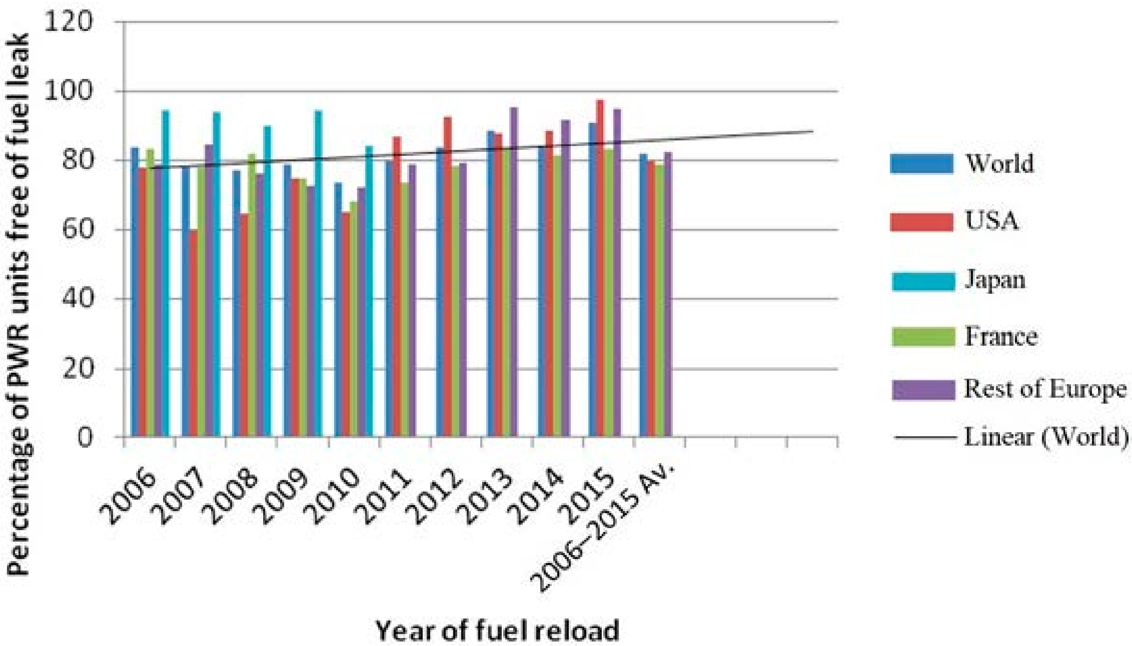


FIG. 6. Percentage of PWR units with zero fuel 'leakers'. Linear extrapolation is for world averages.

— The fuel failure rate (per 1000 discharged FAs) ranges from ~10 during 2006–2010 to ~3.5 during 2011–2015 and decreased from 2011 (4.6) to 2015 (2.0).
— No multiple failures (≥10 failed FAs in one unit within one cycle) were observed in PWRs during 2007–2015 (see Section 2.4) and the number of significant (more than 5 but less than 10 failed FAs in one unit in one cycle) decreased from 5 during 1994–2006 to 4 during 2006–2015 (see Section 2.4).
— Grid-to-rod fretting has been the main identified fuel failure mechanism in PWRs from the mid-1990s to 2015 (see Table 3).The share of unknown/undetermined fuel failures has increased from 28.4% to 40.2% between 2006 and 2015. This is likely attributable to the incompleteness of post-irradiation examination programmes.
— The percentage of PWR units that operated with zero fuel leakage was 81.7% worldwide during 2006–2015 (with a value of ~91% in 2015) and 76.6% during1994–2006. A steady improvement was demonstrated.

2.2.2. BWRs

2.2.2.1. Evaluation of BWR fuel leakage rates

Worldwide and regional FA failure rates for the USA (34 BWR units as of December 2015), Europe (Belgium, Finland, Germany, Spain, Sweden and Switzerland, with 14 units total in 2015) and Japan (24 units during 2006–2010) are presented in Fig. 7. The FR failure rates are presented in Fig. 8. The data also include six units that were shutdown in the period 2006–2015 (four in Germany, one in Spain and one in the USA); see Table 1. The data for Japan refer to the period 2006–2010, i.e. before the Fukushima Daiichi accident in 2011. Data were not collected for two units in Mexico and four units in Taiwan, China.

A similar approach to that taken with the PWR and BWR world average data for 1987–2015 was used, which was to group by 4 year periods (Fig. 9 and Table 4). BWR data also clearly demonstrate a decrease in the FR failure rate in BWRs over 1987–2015. Data for the time frame 1994–2006 were taken from tables 3.4 and 4 of Ref. [3].

Figures 7 and 8 demonstrate to a certain extent a decreasing trend in the number of fuel leaks in BWRs (see also Table 4 for 1994–2015). Some increase of fuel failure rates in European BWRs was observed in 2010 and 2013 (see Figs 7 to 9) because of several PCI and corrosion related failures in Germany and debris fretting failures in Sweden. For both PWRs and BWRs, FA failure rates were half as high during 2006–2015 as they were during 1994–2006. During 1994–2015, the FA failure rate was about a third as high in BWRs as it was in PWRs.

Data for the time frame 1987–1994 (Fig. 9) were taken from table 3.7 and fig. 3.25 of Ref. [3]. FR leakage rates were recalculated to be consistent with the 'new reload' model. The increase in the fuel failure rate during 2003–2006 was due to significant numbers of crud or corrosion related fuel failures in some BWRs in the USA, and it reached its highest value in 2003 [3].

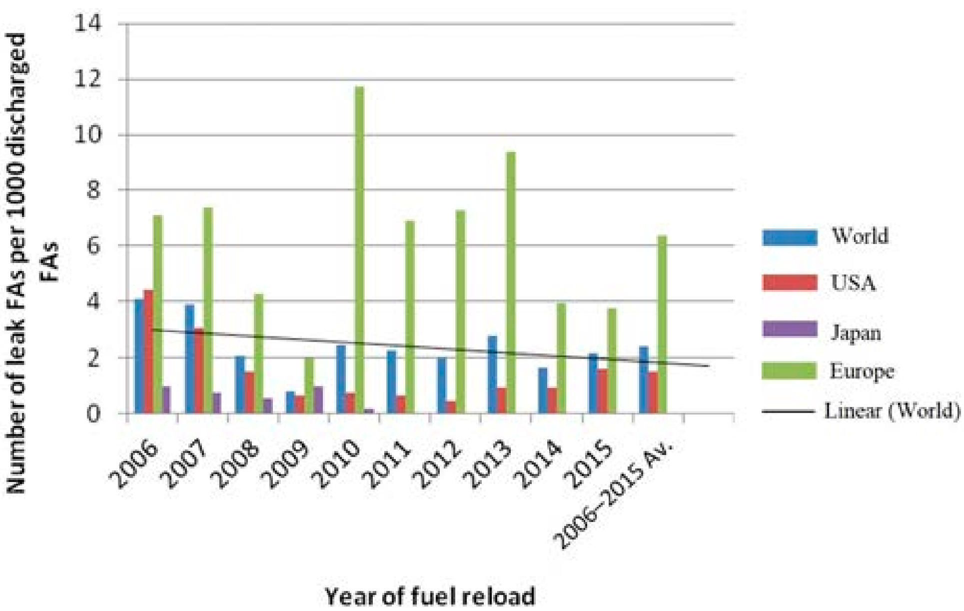


FIG. 7. BWR FA failure rates in 2006–2015. Linear extrapolation is for world averages.

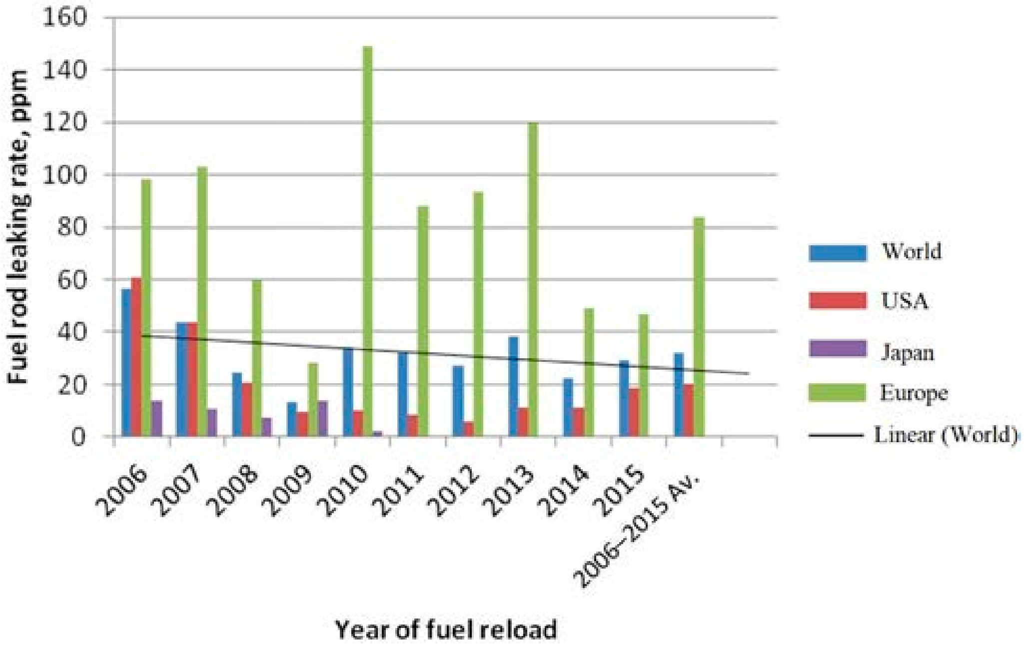


FIG. 8. BWR FR failure rates in 2006–2015. Linear extrapolation is for world averages.

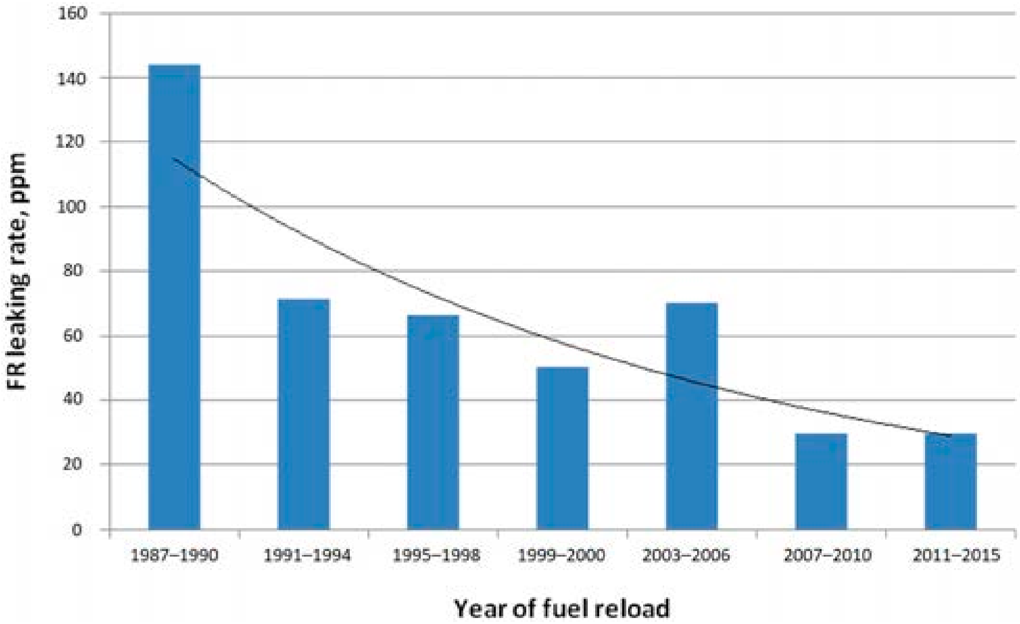


FIG. 9. World average BWR FR failure rates grouped in 4 year periods for 1987–2015 [3].

TABLE 4. BWR FA AND FR FAILURE RATE AVERAGES FOR 1994–2006 AND 2006–2015 [3]

Region	FA failure rate ($\times 10^{-3}$)		FR failure rate (ppm)	
	1994–2006	2006–2015	1994–2006	2006–2015
World	4.4	2.4	64.7	32.1
USA	5.4	1.5	78.9	20.0
Europe	6.8	6.4	101.4	83.5

2.2.2.2. Distribution of FR leakage causes in BWRs

For the 137 failed BWR FAs worldwide during 2006–2015, the exact numbers and mechanisms of failed FRs were reported in Section 2.2.2 and results are summarized in Table 5. Values for 1987–1994 and 1995–2006 were taken from table 3.9 of Ref. [3]. For BWRs, in the same way as for PWRs, data were grouped in 4 year periods. Fuel failure mechanisms are listed in Section 2.1.1. As for the analysis of PWR fuel failure mechanisms, the statistical data in Table 5 were calculated on the basis of post-irradiation examination results (with the unknown/ undetermined cases not being considered) for better comparison of the evolution of the failure mechanisms in BWRs in the time period 1987–2015.

Debris related failures have increased from an ~30% share in the 1990s to an ~49% share during 2011–2015 (see Fig. 10). In contrast to previous data reported from 1994–2006, the percentage of failure caused by crud or corrosion has decreased, while PCI/SCC occupied the second position. Fabrication related failures accounted for ~1% of all failures in 2006–2015 (~10% of all failures in the 1990s).

2.2.2.3. Percentage of BWR units free of fuel 'leakers'

Figure 11 shows the percentage of leak free BWR units worldwide. The average values for the world, the USA and Europe during 2006–2015 were 77.8%, 75.1% and 82.3%, respectively. For 1994–2006 these percentages were 77.5%, 62.0% and 59.4%, respectively. This demonstrates an improvement in the USA and Europe.

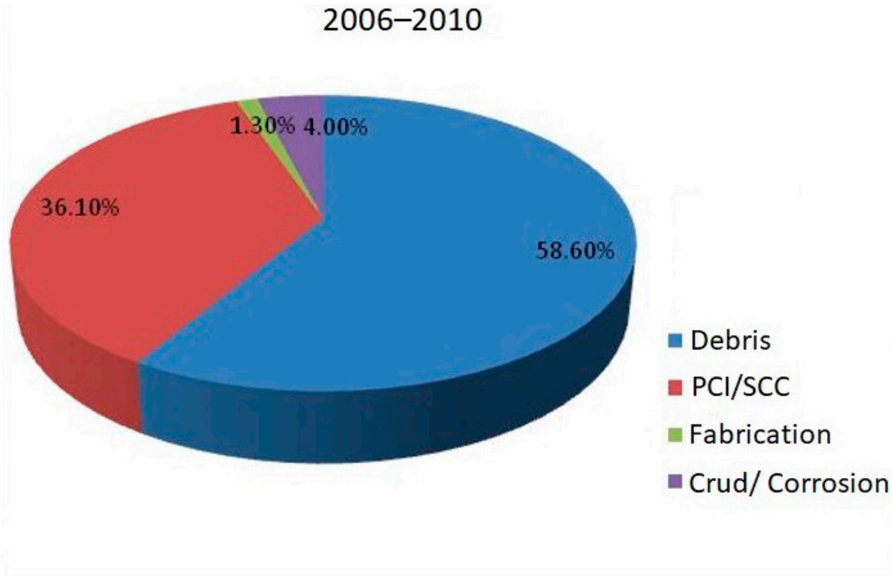
2.2.2.4. Major observations on BWR fuel failures in 2006–2015

Major observations on BWR fuel failures in 2006–2015 are as follows:

— The world average (2006–2015) FA failure rate is 2.4 per 1000 discharged FAs in comparison with 4.4 for 1994–2006. Thus, a significant improvement in BWR fuel performance has been made during the last decade.
— Only one case of significant fuel failure, due to corrosion, was observed in BWRs in 2006–2015, compared with two cases of multiple and two cases of significant fuel failure in BWRs due to crud or corrosion in 1994–2006.
— Similar to the rate in the 1994–2006 time period, the FA failure rate (2.4 per 1000 discharged FAs) is about a third of the failure rate for PWR fuel for 2006–2015 (6.8 per 1000 discharged FAs).
— Debris fretting (61.8%), crud or corrosion (7.6%) and PCI/SCC (28.1%) remained the major causes of fuel failures during 2006–2015, as compared with shares of 41%, 42% and 12.6%, respectively, for these mechanisms during 1994–2006. For these calculations, only occurrences with identified fuel failure causes were taken into account.

TABLE 5. BWR FUEL FAILURE MECHANISMS WORLDWIDE FROM 1987 TO 2015 [3]

	Fuel failure cause (%)						
	1987–1990	1991–1994	1995–1998	1999–2002	2003–2006	2007–2010	2011–2015
Debris fretting	17.5	50.5	39.6	53.4	32.2	58.6	66.0
Crud or corrosion	42.3	4.4	46.8	23.1	52.9	3.9	13.2
PCI/SCC	27.7	34.1	9.9	11.5	14.2	36.2	18.9
Fabrication	10.1	11.0	3.7	11.5	0.7	1.3	1.9
Handling	2.4	0	0	0	0	0	0
Unknown/ undetermined	1.62	32.5	24.0	27.8	13.6	29.2	26.4



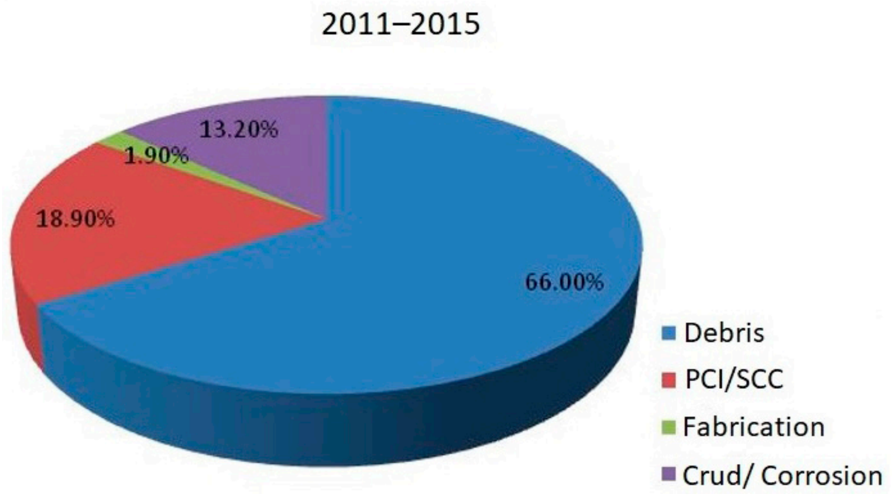


FIG. 10. BWR fuel leak causes worldwide in 2006–2010 (top) and 2011–2015 (bottom).

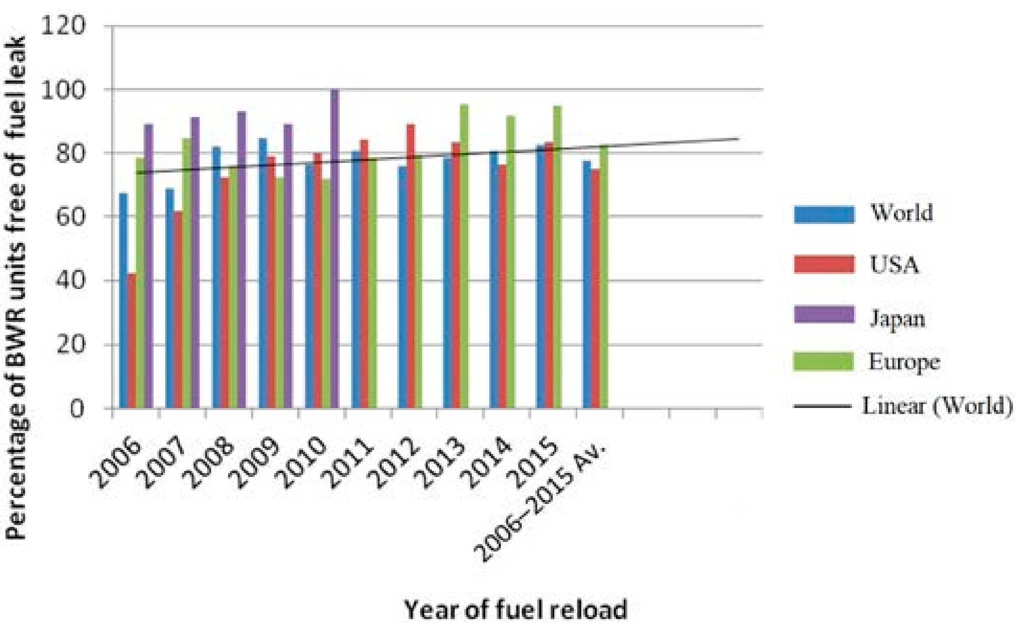


FIG. 11. Percentage of BWR units with zero fuel 'leakers'. Linear extrapolation is for world averages.

— Lacking Japanese data in 2011–2015 because of the Fukushima Daiichi accident (the percentage of Japanese units with zero fuel 'leakers' was over 90% in 1994–2010), only data for the USA and Europe were analysed. These data confirm an improved performance for BWRs (in the USA and Europe) and an increased number of units that operated with zero fuel 'leakers' during 2011–2015 in these countries.

2.2.3. WWERs

2.2.3.1. Evaluation of WWER fuel leak rates

Figure 12 presents FA failure rates for 30 WWER-1000 units operating in Bulgaria (2 units), China (2 units, data only for 2007–2010), the Czech Republic (2 units), the Russian Federation (11 units) and Ukraine (13 units); 18 WWER-440/213 units operating in the Czech Republic (4 units), Finland (2 units), Hungary (4 units), the Russian Federation (2 units), Slovakia (4 units) and Ukraine (2 units); and 4 WWER-440 units of an older design operating in the Russian Federation — a total of 52 units. Fuel failure data are missing for 1 WWER-440/270 unit (in Armenia), 2 WWER-1000 units for 2011-2015 (in China), 1 WWER-1000 unit for 2015 (in India) and 1 WWER-1000 unit for 2014–2015 (in the Islamic Republic of Iran). The average FA failure rate for the time frame of 2006–2015 is 21.5 for WWER-1000, 5.3 for WWER-440/213 and 17.4 for WWER-440/230.

A comparison of WWER FA fuel failure rates observed during 1994–2006 (32.1 for WWER-1000s and 4.7 for WWER-440/213s) [7] and the above mentioned data for 2006–2015 (21.5 and 5.3) shows a reduction of fuel failure rates for WWER-1000s during the last decade. Increases in fuel failure rates were observed for all operating WWER-1000s and for four WWER-440 units of older design in 2011. Owing to the corrective actions taken (see Section 2.2.3.2), the number of leaking FAs of older design significantly decreased in 2015.

There were 10 failed FAs in South Ukraine nuclear power plant Unit 1 (burnup of 15–49 GWd/tU) and 5 failed FAs in Khmelnitsky nuclear power plant Unit 2 (burnup of 16–53 GWd/tU) in 2011. It was the opinion of the TWG-FPT experts that such a high level of FA leakage was caused by debris damage from foreign objects. It was noted that during previous scheduled preventive maintenance of South Ukraine nuclear power plant, work to repair steam generators was conducted.

In the Russian Federation, there were 14 failed FAs in Kalinin nuclear power plant Unit 3 (2 FAs with burnup of 24 GWD/tU, 2 FAs with burnup of 39 GWd/tU and 10 FAs with burnup of 48–52 GWd/tU) and 5 failed FAs in Kalinin nuclear power plant Unit 1 (with burnup of 42–54 GWd/tU) in 2011. Fuel failures in Kalinin nuclear power plant Unit 3 were attributed to the effect of poor quality washing of the primary circuit and those in Kalinin nuclear power plant Unit 1 to debris related failures because of spacer grid rim damage. In addition, three failed FAs were observed in Temelin nuclear power plant Unit 1 and four in Unit 2 in 2011. FA failure rates for all WWER-1000s

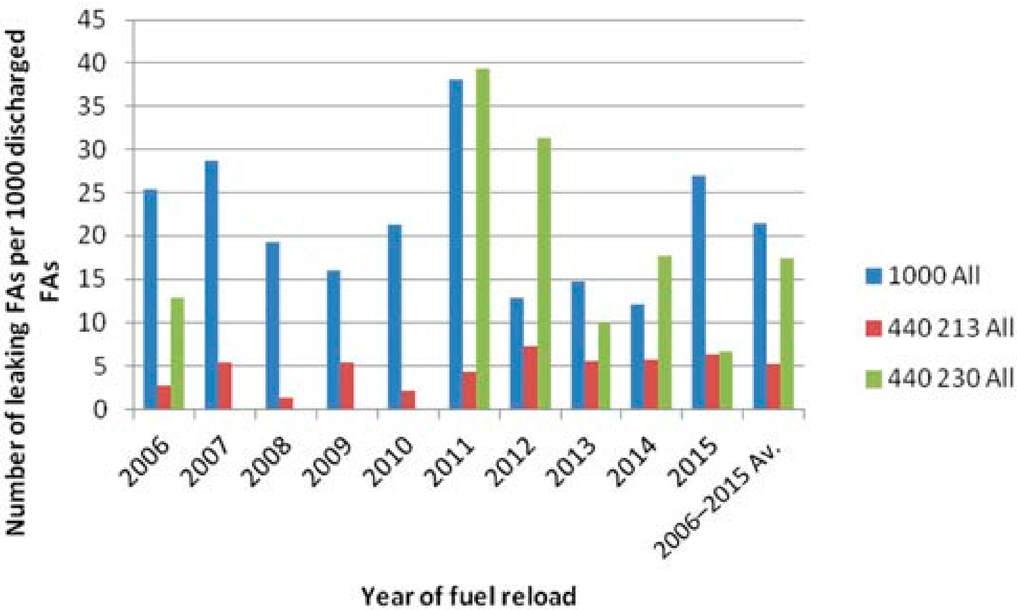


FIG. 12. WWER FA leak rate in 2006–2015.

(30 units), those operated in the Russian Federation (11 units) and in Ukraine (13 units) are presented for the years of 2006–2015 in Fig. 13.

FR failure rates for WWER-1000s and WWER-440s for the period of 2006–2015 are given in Fig. 14. The average FR leak rates during 2006–2015 accounted for 69.7 ppm for all WWER-1000s and 37.1 ppm for all WWER-440/213s. During 1994–2006, the average values were 134 ppm and 34.5 ppm, respectively. This was a ~50 % reduction of the FR failure rate during the last two decades for WWER-1000s and only a slight change for WWER-440/213s.

Figure 15 presents the FR leak rate for all WWER-1000 and WWER-440/213 units operated during 1987–2015 averaged over 4 year intervals. The FR leak rate curve reveals a slight increase in 2001, due to failures of FAs in a WWER-1000 in Balakovo Unit 1. Explanation of the FR leak rate increase in 2011 is given above, and the increase in 2001 was due to failures of FAs in WWER-1000s in Balakovo Unit 1, Balakovo Unit 2 and Rovno Unit 3, affecting 9 FAs, 9 FAs and 11 FAs, respectively.

2.2.3.2. FR leak causes in WWERs

Occurrences of increased FR leaks observed in WWER-440s (Novovoronezh Units 3 & 4) during 2003–2010 were associated with corrosion products deposited on the surface of the FR and debris damage. The subsequent lowering of the FR leaking rate observed after 2010–2011 was a result of actions to reduce the vibration of the core to an acceptable level (reduction of coolant flow and the introduction of a vibration resistant FA design) and a decrease in the formation of corrosion deposits on the FAs (additional cleaning of the coolant circuit, ultrasonic cleaning of FAs and others). The number of leaking FAs in three units of older design in the Russian Federation in 2015 decreased to zero.

Reports presented by Russian Federation specialists at the 10th and 11th International Conferences on VVER Fuel Performance, Modelling and Experimental Support in 2013 [11, 12] and 2015 [13–17] indicated that five leaking FAs from the Zero Failure Level [13] project were examined in hot cells at the Research Institute of Atomic Reactors [13, 14]. A total of seven leaking FAs are stored in the inspection and repair facilities at the Kalinin and Temelin nuclear power plants. The cause of failure for four of those inspected ‘leakers’ was identified as debris fretting because of the presence of foreign material in the primary circuit.

Twelve leaking FAs were operated in WWER-1000s and examined at the Research Institute of Atomic Reactors. Debris damage by foreign objects was the failure cause for 10 of those FAs. Grid-to-rod fretting around the anti-vibration grid and hydriding were the cause of failure for the other two FAs. All 12 leaking FAs had one defective FR.

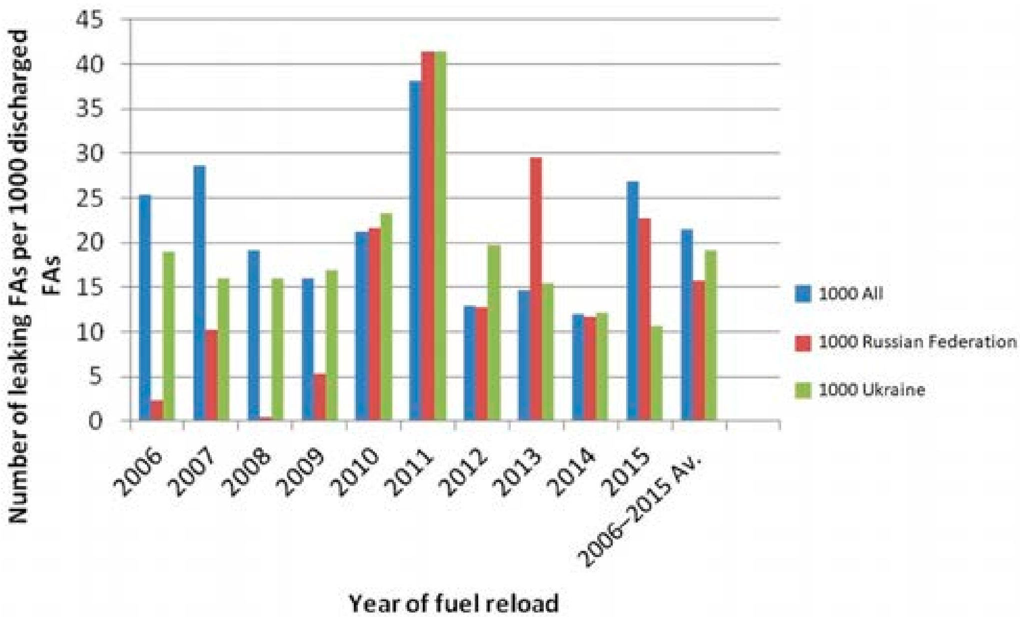


FIG. 13. WWER-1000 FA leak rates in 2006–2015.

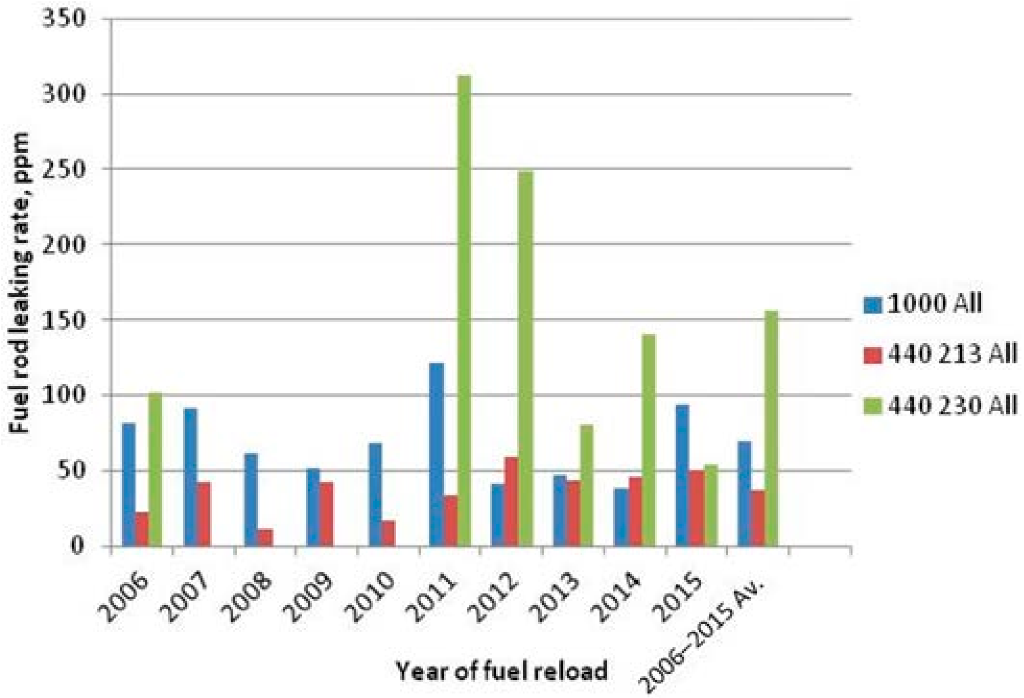


FIG. 14. WWER FR leak rates in 2006–2015.

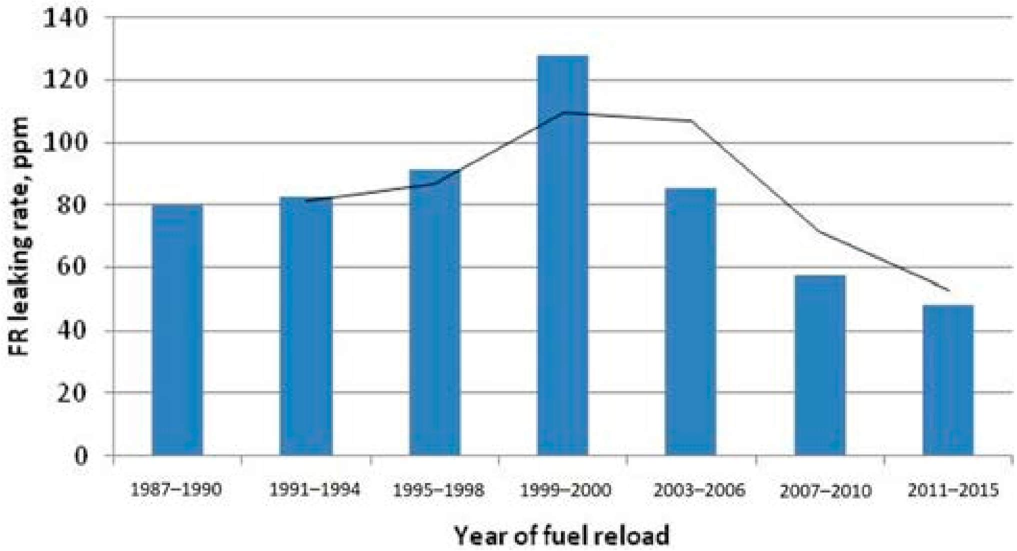


FIG. 15. FR leaking rate for all WWER-1000 and WWER-440/213 units operated in 1987–2015, averaged and grouped in 4 year time intervals.

2.2.3.3. Percentage of WWER units free of fuel 'leakers'

Figure 16 presents the percentage of WWER units that operated during 2006–2015 without fuel failures. The calculation methodology is described in Section 2.1.5. The average values are 64.2% (WWER-1000) and 78.7% (WWER-440/213).

A comparison of the percentage of WWER units that operated without fuel failures in the time frame 1994–2006 (43.4% for WWER-1000s and 79.4% for WWER-440/213s) [7] with the previous data for 2006–2015 (64.2% and 78.7%, respectively) shows an increase in the number of WWER-1000 units that operated with zero fuel 'leakers'. The trend was flat but at a higher level for WWER-440/213s during the last decade.

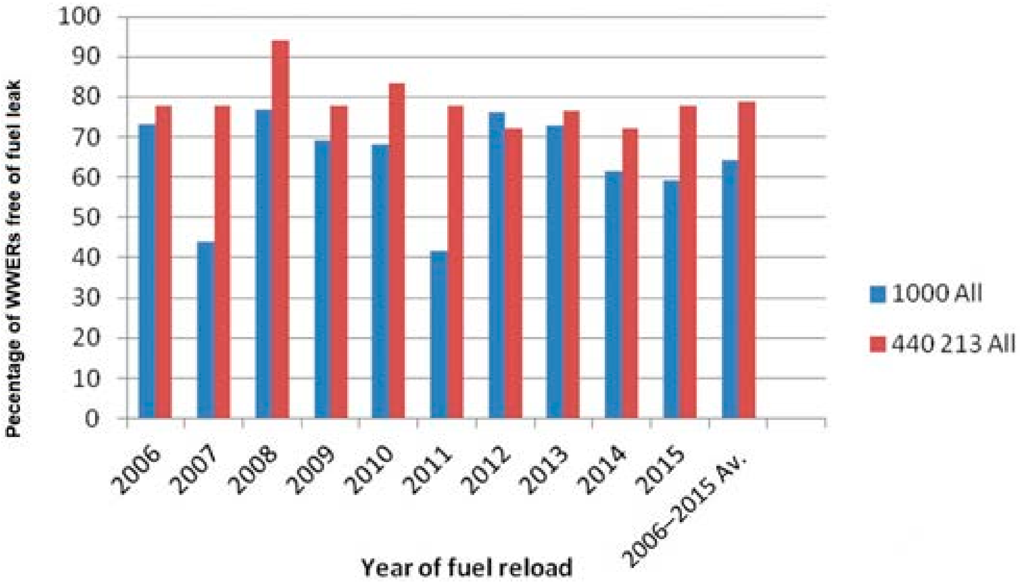


FIG. 16. Percentage of WWER units (all WWER-1000s and WWER-440/213s operated in 2006–2015) with zero fuel 'leakers'.

References [16, 17] describe the operational results of the new generation WWER-1000 FAs TVSA, TVS-2 and TVS-2 M manufactured at Novosibirsk Chemical Concentrates Plant and loaded in Bulgaria, China, the Russian Federation and Ukraine in 2007–2012. In the data gathered for this publication, the percentages of WWER units that operated without fuel failures in 2007, 2008, 2009, 2010, 2011 and 2012 were indicated as 35%, 63%, 64%, 78%, 76% and 91%. The respective values calculated from all WWER data provided for the present report for the same years were: 44%, 77%, 69%, 68%, 42% and 76%.

2.2.3.4. Major observations on WWER fuel failures

Major observations on WWER fuel failures in 2006–2015 are as follows:

- Average FA failure rates for the WWER-1000s operating worldwide were 21.5 per 1000 discharged FAs during 2006–2015 and 32.1 per 1000 discharged FAs during 1994–2006. The highest number of failed WWER-1000 FAs — 51 — was observed in 2011:
  - Bulgaria: 2 units with fuel reload, 1 failed FA;
  - Czech Republic: 2 units with fuel reload, 7 failed FAs;
  - Russian Federation: 8 units with fuel reload, 21 failed FAs;
  - Ukraine: 12 units with fuel reload, 22 failed FAs.
- Four cases of multiple and twelve cases of significant fuel failures were observed in WWER-1000s during 2006–2015 in comparison with four multiple and four significant fuel failure cases during 1994–2006. One multiple failure that occurred in 2006 in Temelin Unit 2 and one significant failure that occurred in 2006 in Temelin Unit 1 were reported after the publication of the IAEA's 2010 fuel failure review [3]; these failures have been included in this publication.
- Regarding information provided to the IAEA, the major identified fuel failure cause for WWER-1000s is debris fretting damage.
- The average percentage of units with reloads and zero fuel failures during 2006–2015 is 64.2% (WWER-1000) and 78.7% (WWER-440/213).
- The average FA failure rates for all 18 units with WWER-440/213 reactors are 5.3 during 2006–2015 and 4.7 during 1994–2006 per 1000 discharged FAs, i.e. somewhat similar levels to those observed in 1994–2015.
- One case of significant fuel failure was observed in a WWER-440/213 during 1994–2006. No failure was observed in these reactors during 2006–2015.
- Little information regarding fuel failure causes for WWER-440/213s was submitted to the IAEA during 2006–2015.

2.2.4. CANDU reactors and other PHWRs

2.2.4.1. Evaluation of fuel leak rates in CANDU reactors and other PHWRs

Recent data presented in this section describe fuel 'leaker' experience for 27 CANDU units: 1 unit in Argentina, 18 units in Canada, 2 units in China (with data provided only for 2006–2010), 4 units in the Republic of Korea, and 2 units in Romania. Data from the other 18 PHWRs in India are also reported here. The KANUPP CANDU unit in Pakistan did not provide any data. Data for Atucha Units 1 and 2 (PHWRs of Kraftwerk Union (KWU) design) were received. Atucha Unit 2 was connected to the grid in June 2014, i.e. essentially only one unit (Atucha Unit 1) was operating during the time frame 2006–2015. Therefore, data from Atucha have not been included nor considered in this publication. Data for Bruce A Units 1 and 2 were provided and considered in the report but were not included in the calculations as these units were reconnected to the grid just prior to the end of 2012.

The results of the fuel failure rate calculation for CANDUs and other PHWRs (number of leaking fuel bundles per 1000 discharged bundles) are presented in Fig. 17. The average number of leaking bundles per 1000 discharged bundles for 2006–2015 was 0.097 (CANDUs in Canada), 0.11 (for all 27 CANDU units for which data is available) and 0.79 for PHWRs in India. In 1994–2006, these data were 0.1, 0.35 and 1.5, respectively. A comparison of these data shows low fuel failure rates in Canadian units in 1994–2015 and an improvement of fuel performance in CANDUs outside Canada and in the Indian PHWRs during the last decade.

During 2006–2015, two fuel defect excursions were observed in CANDU plants, the first at Cernavoda Unit 2 in 2008 when 52 bundles (80 failed fuel elements altogether) failed because of debris fretting and faulty fabrication welds. The second excursion was at Bruce Power Units 1 and 2 and was due to debris fretting resulting from refurbishment activities on these two units, specifically to feeder cutting material retained in the heat transport system.

A visible improvement has been seen during the last decade for the Indian PHWRs. The increase in the fuel failure rate in Indian PHWRs in 2011 was due to PCI/SCC, handling damage and fabrication defects and that in 2015 due to fabrication defects and handling damage.

Fuel element leaking rates for CANDUs operated in Canada, all CANDUs and Indian PHWRs are presented in Fig. 18. The average fuel element leaking rate for 2006–2015 was 3.6 ppm for units in Canada (3.5 ppm during 1994–2006) and 4.59 ppm for all CANDU-6 units operated worldwide (10.4 ppm during 1994–2006). For Indian PHWRs the leaking rate was 42.6 ppm in 2006–2015 and 80 ppm in 1994–2006. The long term tendency in the evolution of fuel element leaking rates during 1987–2010 is shown in Fig. 19.

2.2.4.2. Evaluation of fuel leak causes in CANDU reactors and other PHWRs

Primary failures and their root causes observed in CANDU reactors during 1969–2005 were considered in the IAEA's Review of Fuel Failures in Water Cooled Reactors [3]. In the information obtained in the preparation of the present publication from Member States operating CANDU reactors in 2006–2015, two major causes, fabrication (mainly faulty welds or flaws) and debris, were mentioned as fuel element leak causes for CANDU-6 units in Canada, China and Romania. Some fuel failures that occurred at a Canadian CANDU station, as they were observed principally in heavier bundles, were suspected to be attributed to a manufacturing cause related to the chipping of fuel pellets during loading with high pellet loading forces along with low clearance tolerances for the sheath (the term used in Canada for FR cladding) [18]. The other cause identified in the Canadian reactors was debris failure. Microcracks on a peripheral FR were found in one of the Republic of Korea's CANDU units in 2008. PCI/SCC, debris fretting, fabrication defects and handling damage were causes of failure in the Indian PHWRs.

2.2.4.3. Percentage of CANDU and other PHWR units free of fuel 'leakers'

There is little information on the number of Canadian CANDU units that reported zero 'leakers' since data were provided on a 'station by station' basis. Some stations have multiple reactors; the breakdown for each reactor within a multi-unit station was not always clear. The share of units reporting zero fuel failures in operation in Argentina, China, the Republic of Korea and Romania (9 units in total) is 49% for 2006–2015 with ~22% for 1998–2006. The share of Indian PHWRs reporting zero 'leaker' operations for 2006–2015 was 41.5%, while during 1994–2006 it was ~36%.

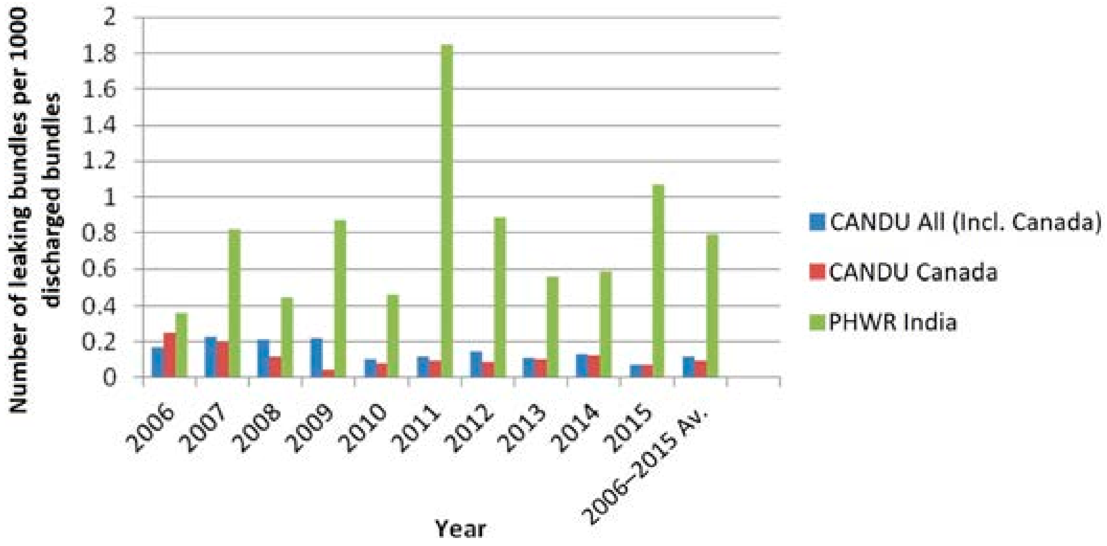


FIG. 17. CANDU and other PHWR fuel bundle failure rates.

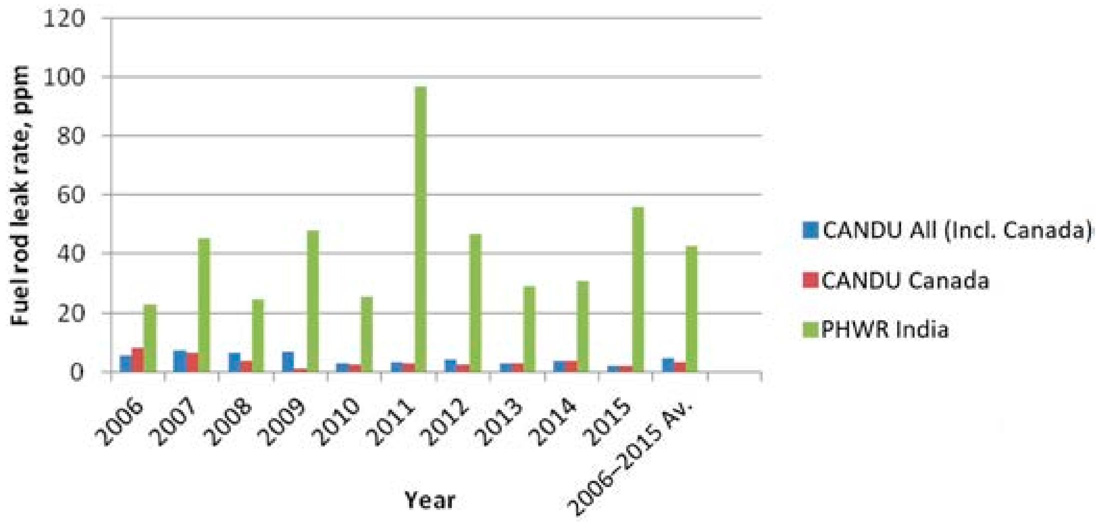


FIG. 18. CANDU and other PHWR fuel element leaking rates in 2006–2015.

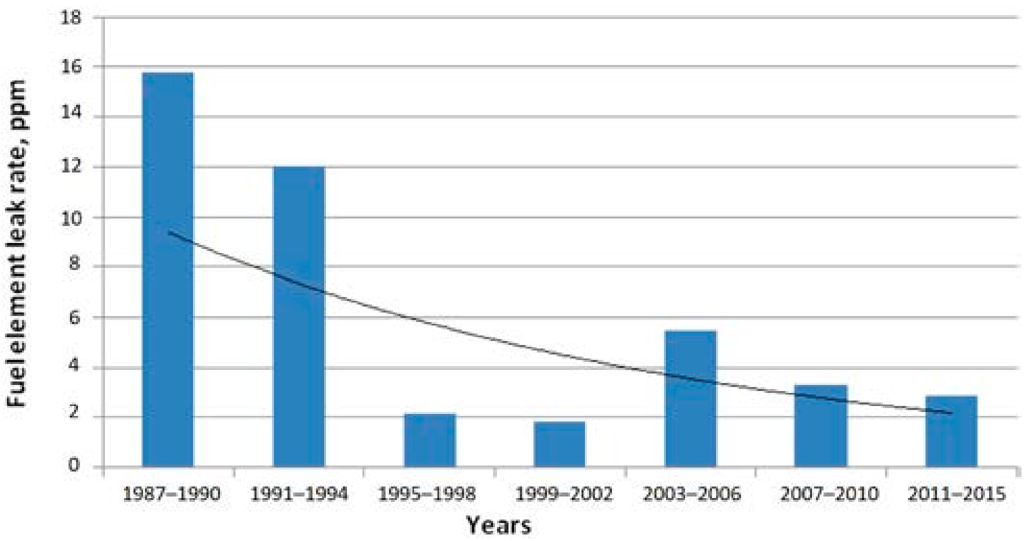


FIG. 19. CANDU fuel element leaking rate in ppm.

2.2.4.4. *Major observations on CANDU and other PHWR fuel failures*

Major observations on CANDU and other PHWR fuel failures in 2006–2015 are as follows:

— During 2006–2015, the world average fuel failure rate was 0.116 (0.35 in 1994–2006) leaking FA per 1000 discharged FAs for all CANDUs worldwide. The corresponding rates for Canadian CANDUs and Indian PHWRs were 0.097 (0.1 during 1994–2006) and 0.79 (1.5 during 1994–2006), respectively.
— The failure rate of CANDU fuel has been relatively stable and at a low level for the time period of 1994–2006 with several exceptions, including 96 fuel bundles that failed in Wolsong Unit 1 in 1996 because of insufficient baking of the CANLUB graphite coating and 52 fuel bundles that failed in Cernavoda Unit 2 in 2008 because of fabrication defects and debris fretting.
— The major fuel defect mechanisms observed in all CANDUs were debris fretting and incomplete end welds.
— Identified fuel failure causes in Indian PHWRs were related to debris, fabrication defects, PCI/SCC and handling damage.
— The percentage of units operated with zero fuel 'leakers' in CANDUs and PHWRs was higher during 2006–2015 (CANDUs: 49% and PHWRs: 42%) than in the period 1998–2006 (CANDUs: 22% and PHWRs: 36%).

2.3. DATA ON FA STRUCTURAL DAMAGE WITHOUT FR LEAK

Quantitative information on FA structural damage without FR leak submitted to the IAEA was limited, except for data from the USA where EPRI runs the Fuel Reliability Database (FRED) [19]. CANDUs have experienced occasional endplate cracks and breaks from acoustic vibrations. Data for these damages in PWRs (from Belgium, China (only 2006–2010 data provided), France, Germany and the USA) and in BWRs (from the USA) for the time frame of 2006–2015 are presented in Table 6. The most important issues for PWRs**,** as they were reported, include:

- Spacer grid damage with a determined number of damaged spacer grids or unknown number of damaged spacer grids;
- FA bow or distortion;
- FA bow or distortion resulting in spacer grid damage;
- Debris fretting;
- FA bow with spacer grid damage and incomplete rod insertion as a final stage;
- Handling damage that sometimes resulted in spacer grid damage and grid-to-rod fretting;
- Axial offset anomaly.

In PWRs, incidents of spacer grid damage were reported by Belgium (8 occurrences in 2006–2008); China (18 occurrences in 2008–2010); France (57 occurrences in 3.66 m (900 MW(e)) units and 199 occurrences in 4.27 m (1300 +1450 MW(e)) units during 2006–2015); Germany (184 occurrences in 16 × 16 units during 2006–2014 and 9 occurrences in 18 × 18 units during 2007–2013); Spain (2 occurrences in 2007); and Sweden (2 occurrences in 2006). The USA reported about 94 spacer grid damage incidents due to FA bow during 2011–2015, 8 spacer grid damage occurrences with an unknown number of damaged grids during 2011–2014 and 12 occurrences of debris fretting during 2011–2015. Incomplete rod insertion cases were reported by China for one PWR unit in 2008, France (11 occurrences in 3.66 m units during 2011–2015), Germany (difficulties were noticed in 2008 in 18 × 18 units) and USA (5 occurrences in 2006, 2008 and 2011). Axial offset anomalies were observed in the USA in 2006, 2009, 2010 and 2014 (9 occurrences in total).

For BWRs the most important issues were:

— Channel distortion or bow;
— Control blade cracking;
— Incomplete rod insertion;
— Incomplete rod insertion and inoperable control blade;
— Debris fretting.

TABLE 6. FA STRUCTURAL DAMAGE WITHOUT FR LEAK IN 2006–2015

	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	Total (2006–2015)
PWRs in the USA											
No. of units affected	Not rep.	Not rep.	Not rep.	Not rep.	Not rep.	4	6	5	6	4	25
FA bow						4			3		7
FA bow with unknown no. of distorted FAs									1		1
FA bow resulting in spacer grid damage						15	37	16	16	10	94
FA bow resulting in debris fretting						0					0
Spacer grid damage										1	1
Spacer grid damage with an unknown number of damaged spacer grids						1	3	2	2		8
Grid-to-rod fretting										1	1
Debris fretting						10	3	1		2	16
Handling damage						1		1	1		3
Incomplete rod insertion	2		1			0					3
Axial offset anomaly	5			2	1				1		9
Unknown											
PWRs in France											
Spacer grid damage in 3.66 m units	6	10	9	8	6	12	2	2	1	1	57
Spacer grid damage in 4.27 m units	28	45	32	30	29	7	16	9	1	2	199
Incomplete rod insertion in 4.27 m units						1	2	2	5	1	11
PWRs in Belgium											
Spacer grid damage	6	1	1								8
Debris fretting	1	2									3

TABLE 6. FA STRUCTURAL DAMAGE WITHOUT FR LEAK IN 2006–2015 (cont.)

	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	Total (2006–2015)
PWRs in China											
Spacer grid damage			7	10	1						18
Incomplete rod insertion			1								1
PWRs in Germany											
Spacer grid damage in 16 × 16 units	12	1	12	12	64	18	43	17	5		184
FA bow in 16 × 16 units	1		1		1						3
Debris fretting 16 × 16 units									2		2
Spacer grid damage in 18 × 18 units		6		1				2			9
Debris fretting in 18 × 18 units	1		1					2	1		5
BWRs in the USA											
No. of units affected						5	6	5	4	5	25
Channel distortion or bow	5	4	2	5	3						19
Control blade cracking and leaking involving a known number of control blades	5	4	2		3		3	3		6	26
Control blade cracking and leaking involving an unknown number of control blades						3	2		3	1	9
Incomplete rod insertion						3		4	3	8	18
Control blade cracking and leaking involving a known number of control blades or inoperable control blade							1				1
Incomplete rod insertion or inoperable control blade						2	1				3
Debris fretting							1				1

Note: Not rep. — Not reported.

For BWRs the most important issues observed during 2006–2015 included channel bow/distortion from a combination of bow, bulge and twist leading to control blade interference, control blade cracking, inoperable control blade or incomplete rod insertion. High channel bow on 4 LTAs was noticed in a Swiss BWR's units in 2010. Channel bow or distortion was also observed in several BWRs in the USA. Basic data were obtained from the EPRI FRED [19]. For those 34 BWRs that operated in the USA during 2006–2015, there were 19 occurrences of channel bow; 26 occurrences of control blade leakage (with 9 cases with an unknown number of cracked control blades); 3 occurrences of incomplete rod insertion and inoperable control blade; 1 occurrence of control blade cracking and inoperable control blade and 1 occurrence of debris damage. No information on WWER FA structural damage without FR leaking during 2006–2015 was submitted to the IAEA.

2.4. DATA ON MULTIPLE AND SIGNIFICANT FAILURE INCIDENTS

Multiple or massive failures were defined in the 2010 IAEA Review of Fuel Failures in Water Cooled Reactors [3] as incidents with failure of ≥10 FAs in one cycle (LWRs) or one year (CANDUs, PHWRs). Incidents with yearly failures of less than 10 FAs, but with 5 or more failed FAs, were defined as significant events. The definition of a fuel failure category (single, significant or multiple) requires the presentation of fuel failure data on a reactor-by-reactor basis. As indicated in Section 2.3, data of this format were provided by Argentina, Belgium, Brazil, Bulgaria, the Czech Republic, Finland, Hungary, Japan (for the period 2006–2010), the Netherlands, Romania, the Russian Federation, Slovakia, Spain, Sweden, Switzerland, Ukraine (for the period of 2010–2015), the UK, and the USA (for the period of 2011–2015) (see Tables 7 and 8). Regarding reports for 2006–2015 from Members of the IAEA TWG-FPT, 5 fuel failure cases were recognized as multiple or massive events (Table 7) and 31 cases as incidents with significant fuel failures (Table 8).

The reported incidents of multiple failures (≥10 FAs in one cycle, or in one year for CANDUs) are difficult to establish because not all incidents were reported and only a few have been published. As shown in Ref. [2], from 1987 to 1994 there were 20 such occurrences: 16 for LWRs (10 PWRs, 5 BWRs and 1 WWER) and 4 for CANDUs. The primary causes of massive fuel failure were grid-to-rod fretting and debris in PWRs, crud induced localized corrosion and debris in BWRs, and crud deposition in WWERs.

As shown in the IAEA's Review of Fuel Failures in Water Cooled Reactors [3] published in 2010, from 1994 to 2006 17 massive fuel failures were reported: 16 for LWRs (10 PWRs, 2 BWRs and 4 WWER-1000s) and 1 for CANDUs. The primary causes of massive fuel failure were grid-to-rod fretting and debris in PWRs, crud induced localized corrosion and debris in BWRs, and crud deposition in WWER-1000s. This publication collected data on five incidents of multiple failure during 2006–2015: three in WWER-1000s, one in a PWR and one in a CANDU. Fuel failure causes were fabrication and debris fretting both for WWERs and for CANDUs. Despite the nature of the multiple fuel failures in the time periods of 1987–1994 (20), 1994–2006 (17) and 2006–2015 (5), the one conclusion that might be drawn is that the frequency of multiple fuel failures has continuously decreased during the last three decades.

TABLE 7. INCIDENTS OF MULTIPLE FUEL FAILURE DURING 2006–2015
(multiple fuel failures are defined as involving ≥10 failed fuel assemblies)

Year	Reactor type and location (and plant name where known)	Number of failed FAs	Main failure mechanism
2006	WWER-1000, Temelin Unit 2 Czech Republic[a]	10	Fabrication
2006	PWR, USA	Not indicated	Grid-to-rod fretting
2008	CANDU, Cernavoda Unit 2, Romania	52 (80 FRs)	Fabrication and debris fretting
2011	WWER-1000, Kalinin Unit 3, Russian Federation	14	Unknown, debris fretting suspected (see Section 3.2.2.1)
2011	WWER-1000, South Ukraine Unit 1, Ukraine	10	Unknown

[a] Reported by the Czech participant in the TWG-FPT on 26 January 2016.

TABLE 8. INCIDENTS OF SIGNIFICANT FUEL FAILURE DURING 2006–2015
(significant fuel failures are defined as those involving 5–9 failed fuel assemblies)

Year	Reactor type and location (and plant name where known)	Number of failed FAs	Main failure mechanism
2006	WWER-1000, Temelin Unit 1, Czech Republic [a]	6	Fabrication
2006	PWR, Tihange Unit 3, Belgium	5	1 failed FR in failed FA, bottom plug weld
2007	PWR, France	5 (6 FRs)	Fabrication
2007	PWR, Angra Unit 1, Brazil	5	Debris fretting
2007	WWER-1000, Temelin Unit 2, Czech Republic	5	Fabrication
2007	CANDU, Cernavoda Unit 2, Romania	9 (16 FRs)	Fabrication, debris fretting
2008	PWR, France	8 (8 FRs)	Fabrication
2008	WWER-1000, Temelin Unit 1, Czech Republic	7	Fabrication
2008	WWER-1000, Temelin Unit 2, Czech Republic	7	Fabrication
2009	PWR 16 × 16, Germany	5 (5 FRs)	Grid-to-rod fretting affecting 2 FRs, debris fretting affecting 1 FR, grid corner fretting affecting 2 FRs
2009	WWER-1000, Temelin Unit 2, Czech Republic	5	Fabrication
2009	CANDU, Embalse, Argentina	8	Unknown
2009	CANDU, Cernavoda Unit 1, Romania	8 (8 FRs)	Fabrication
2010	PWR 16 × 16, Germany	7	Grid-to-rod fretting affecting 1 FR, debris fretting affecting 1 FR, grid corner fretting affecting 5 FRs
2010	WWER-1000, Temelin Unit 2, Czech Republic	5	Fabrication
2011	BWR, Germany	6 (9 FRs)	PCI affecting 2 FRs, corrosion affecting 7 FRs
2011	WWER-1000, Kalinin Unit 1, Russian Federation	5	Debris fretting
2011	WWER-1000, Khmelnitsky Unit 2, Ukraine	5	Debris fretting
2011	PWR, France	7 (7 FRs)	Grid-to-rod fretting
2011	PWR, France	6 (9 FRs)	Grid-to-rod fretting
2011	PWR, USA	7 (12 FRs)	Grid-to-rod fretting affecting 7 FRs
2012	PWR, USA	6 (20 FRs)	Grid-to-rod fretting affecting 6 FRs
2012	PWR 16 × 16, Germany	6 (6 FRs)	Grid corner fretting affecting 5 FRs debris fretting affecting 1 FR
2012	WWER-440/179, Novovoronezh Unit 3, Russian Federation	7	Unknown

TABLE 8. INCIDENTS OF SIGNIFICANT FUEL FAILURE DURING 2006–2015
(significant fuel failures are defined as those involving 5–9 failed fuel assemblies) (cont.)

Year	Reactor type and location (and plant name where known)	Number of failed FAs	Main failure mechanism
2012	WWER-1000, Kalinin Unit 1, Russian Federation	6	Debris fretting
2013	PHWR KWU, Atucha1, Argentina	5	Unknown
2013	WWER-1000, Kalinin Unit 3, Russian Federation	7	Debris fretting
2013	PWR, USA	7	Grid-to-rod fretting
2015	PWR, USA	5 (7 FRs)	Grid-to-rod fretting
2015	WWER-1000, Temelin Unit 1, Czech Republic	7	Unknown
2015	WWER-1000, Temelin Unit 2, Czech Republic	7	Unknown

[a] Reported by the Czech participant in the TWG-FPT on 26 January 2016.

The main failure mechanisms observed and identified in significant fuel failure occurrences include:

— PWRs: 9 (grid-to-rod fretting), 4 (debris fretting) and 3 (fabrication);
— WWER-1000s: 6 (fabrication) and 4 (debris fretting);
— CANDUs: 2 (fabrication and debris fretting).

2.5. CONCLUSIONS ON FUEL FAILURES IN LWRs AND PHWRs FROM 2006 TO 2015 AND COMPARISON WITH THOSE OBSERVED IN 1987–2006

Fuel failure rates in all water cooled reactors (PWRs, BWRs, WWERs, CANDUs and other PHWRs) decreased during 2006–2015 in comparison with the period of 1987–2006. The most stable improvement trend (a constant reduction of fuel failure rate, mitigation of multiple and significant failures) was observed for PWRs. Failure rates in Canadian CANDUs did not drop during the last 5 years as observed for all other plant types, but the CANDUs had already achieved a low level of failure occurrence in 1994–2006.

Fuel failure mechanisms in PWRs and CANDUs remain similar to those observed previously. Regarding structural damage to the FA skeleton, the frequency of FA bow and incomplete rod insertions in PWRs and WWERs has significantly decreased. Spacer grid damage has been an issue for PWRs during fuel handling operation in 2006–2015. In BWRs occurrences of channel bow were observed. The channel bow issue will be addressed by changing channel materials and operational strategies. A dominant fuel failure cause in PWRs is grid-to-rod fretting and in BWRs, WWERs and CANDUs, debris fretting.

The percentage of all PWR and BWR units in USA and Europe, all WWER-1000s and all CANDUs in Argentina, China, the Republic of Korea and Romania, and PHWRs in India that operated with zero fuel ‘leakers’ increased during 2006–2015 in comparison with the period 1994–2006.

3. UPDATES ON FUEL DESIGN, MATERIALS AND OPERATIONAL PERFORMANCE

3.1. INTRODUCTION

Fuel management is finding a balance between the economic demands to burn fuel for longer periods of operation to reduce fuel cycle costs and the need to avoid or reduce, as much as possible, any risk of in-core fuel failures. For LWRs, the key economic drivers related to batch reloading are longer operational cycles, shorter outages, longer times of fuel operation in core and high fuel burnup and minimization of FR leakage [20]. The introduction of advanced fuel materials and designs is an important factor in providing an additional margin to improve fuel burnup. The direct method to increase burnup is to use more enriched fuel. The use of in-core structural materials with lower neutron absorption has a positive impact on increased burnup but this approach has been practically exhausted.

Regarding the increase in fuel enrichment, there are strict criticality limitations imposed at fuel fabrication facilities, with a maximum uranium enrichment level of 5%. Similar criticality issues also relate to other aspects of the fuel cycle: enrichment, nuclear power plant design and operation, reprocessing, and interim spent fuel storage.

For on-power reloading, an increase in fuel enrichment (i.e. the use of slightly enriched fuel instead of natural uranium dioxide) in PHWRs is also a useful approach to improve fuel cycle economics [20].

3.2. FUEL TYPES IN OPERATION IN 2006 AND 2015

An overview of the distribution of fuel types (in terms of the FR lattice) for operating plants up to the year 2006 is given in Table 9. Data for the year 2015 were taken from responses of the Members of the IAEA TWG-FPT to the IAEA questionnaire (see Section 2) and from the literature, such as Ref. [21]. The data for the year of 2006 were from the IAEA 2010 fuel failure review (table 2.1 in Ref. [3]).

In 2015, all BWRs except for those in Japan and India operated with a core FA of the 10 × 10 type. Transitioning from a 7 × 7 fuel design to a 10 × 10 fuel design over the years has allowed the reduction of the FR’s linear heat generation rate and the fuel temperature, fission gas release and cladding corrosion rates. In 2006, 33 PWRs and 2 BWRs in the European Union were operating with batch loadings of mixed oxide (MOX) fuel [3]. For this purpose, 10 210 kg of Pu was recovered from reprocessed LWR spent fuel and used for the fabrication of MOX fuel for loading. In 2015, 10 780 kg of Pu was recovered [22].

In the PWRs operating in 2015, 70 % of the fuel had a 17 × 17 lattice. The fuel type in WWERs is a hexagonal FA with 126 FRs (WWER-440) or 312 FRs (WWER-1000). In CANDUs, 37-element bundles are used in 24 units while 28-element bundles are only used for the 6 units of the Pickering generating stations. The 17 PHWRs (16 units in India and 1 unit in Pakistan) use 19-element bundles. The 2 Siemens-KWU units in Argentina also use 37-element fuel bundles.

The CANDU fuel types are mature products and therefore their main design features have remained essentially unchanged. Nevertheless, extended technology programmes to further improve fuel performance and to evaluate the use of advanced fuel (enriched uranium, recovered uranium from PWR FRs and thorium) were carried out by Atomic Energy of Canada as well as by national programmes in Argentina, India, the Republic of Korea and Romania. The most recent fuel design is a 43-element bundle with two different FR diameters, called CANFLEX, which has not yet been implemented in CANDU reactors. Some Canadian reactors have been converted to a modified version of the 37-element bundle (i.e. 37M bundle) with a smaller centre FR to compensate for flow bypass in an ageing fuel channel, which is the most recent design change implemented in Canada.

TABLE 9. FUEL TYPES IN OPERATION IN 2006 AND 2015 [3, 21]

Type of plant	FR lattice	No. of plants in operation		Comments on 2015 data
		2006	2015	
BWR				
	7 × 7	2	2	India, 2 units of Tarapur plant
	8 × 8	5	0	—
	9 × 9	33	0	—
	10 × 10	53	50	Plants of several vendors, mainly GE
Total		93	52[a]	
PWR				
	14 × 14	24	15	Westinghouse and CE plants
	15 × 15	26	20	Plants of several vendors
	16 × 16	28	24	Plants of several vendors
	17 × 17	133	142	Plants of several vendors
	18 × 18	3	2	Siemens (KWU) plants
Total		214	203[b]	
WWER				
	126 hex	27	23	WWER-440
	312 hex	27	33	WWER-1000
Total		54	56	
CANDU and other PHWR				
	19 circ	13	17	16 in India, 1 in Pakistan
	28 circ	6	6	Pickering plant (6 units) in Canada 3 in Argentina, 13 in Canada, 2 in China, 2 in India, 4 in the Republic of Korea, 2 in Romania
	37 circ	22	26	
Total		41	49	
Grand total		402	360[c]	

[a] 22 units in Japan and 4 units in Taiwan, China, not included.
[b] 21 units in Japan and 2 units in Taiwan, China, not included.
[c] 43 units in Japan and 6 units in Taiwan, China, not included.

3.3. OPERATING ENVIRONMENT

3.3.1. Fuel burnup and fuel enrichment

According to Ref. [23], the batch average fuel burnup in LWRs has steadily increased from 1970 (18–23 GWd/tU) to 2005 (40–42 GWd/tU). Fuel enrichment was within 3.8–4.2%, i.e. comfortably under the regulatory limit of 5% ^{235}U. Data on batch average burnup and fuel enrichment for 2006–2015 were provided for the present publication (see Section 2) for about 50% of all operating PWRs and BWRs and 40% of the WWER-1000s.

Figures 20 and 21 were prepared using these data. It follows from Fig. 20 that there is a very slight increase in the burnup trend as observed in 2006–2015 compared with a constant trend before 2005.

As shown in Fig. 21, the average fuel enrichment levels (for about 50% of the PWRs and BWRs and ~40% of the WWER-1000s) as reported by Members of the IAEA TWG-FPT for 2006–2015 increased from 4% in 2006 to 4.5% in 2015 for PWRs and WWER-1000s and from 3.5% in 2006 to 4% in 2015 for BWRs.

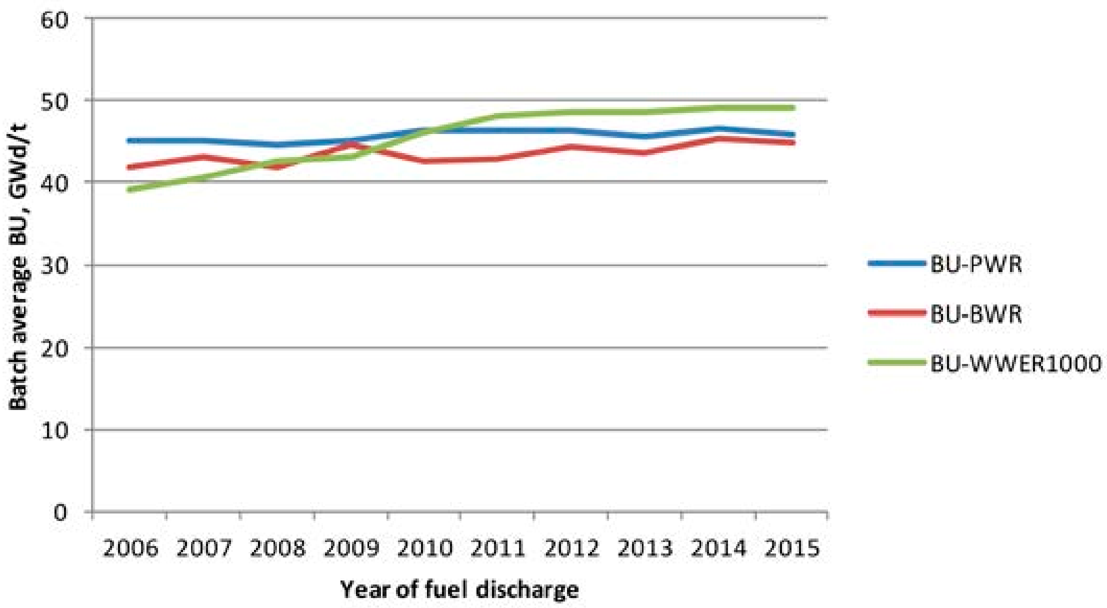


FIG. 20. Dependence of batch burnup averaged for PWRs, BWRs and WWER-1000s as reported by Members of the IAEA Technical Working Group on Fuel Performance and Technology for 2006–2015.

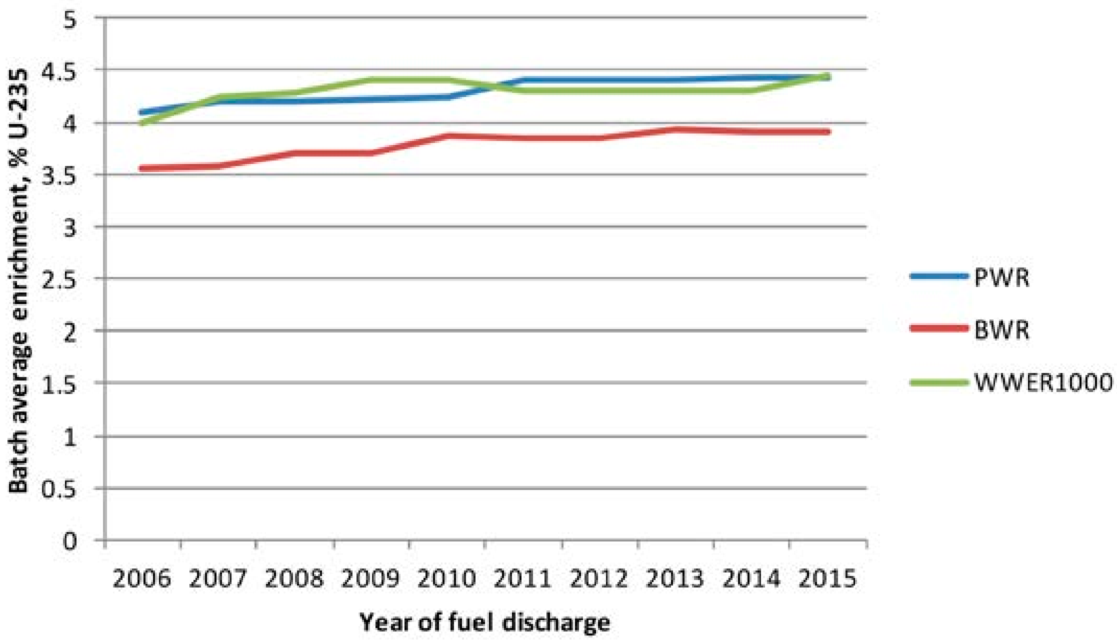


FIG. 21. Average fuel enrichment (percentage of ^{235}U) for PWRs, BWRs and WWER-1000s as reported by Members of the IAEA Technical Working Group on Fuel Performance and Technology for 2006–2015.

3.3.2. Fuel cycle length

For the PWRs in the USA, the average cycle lengths were 16.9 months, 17.7 months and 17.5 months in 2006, 2010 and 2015, respectively. For BWRs, cycles of 19.3 months, 20.4 months and 21.5 months were observed in 2006, 2010 and 2015, respectively. Even though the majority of BWRs in the USA operate with 24 month cycles, there are still a few BWRs operating with 18 month cycles. These data have demonstrated a trend for a modest increase of the cycle length in reactors in the USA. For reactors in France, the fuel cycle lengths are 12 months in 3.66 m units and 18 months in 4.27 m units. The Republic of Korea and Spain utilize 18 month cycles for PWRs and 24 month cycles for BWRs. These cycle lengths were unchanged during the period 2006–2015. For the WWER-1000s, units in the Russian Federation operated with 18 month cycles and those in Ukraine operated with 12 month cycles.

3.4. FUEL DESIGN CHANGES AND STRUCTURAL MATERIAL MODIFICATIONS

3.4.1. PWRs

It was discovered that in some FA designs, FR vibration could occur. It was noted that grid-to-rod fretting sometimes occurred in multiple FRs in some FA designs. The unfavourable vibration and flow conditions may have been created by plant design features such as baffle plates and specific flow paths. Flow testing revealed that FA vibration could also occur in some FA designs when they were subjected to axial flow. The vibration was eliminated by the redesign of the mixing vanes that were intended to increase turbulence in certain areas of the FA. This redesign also reduced grid-to-rod fretting. Changes to some spacer grid designs improved the local flow conditions that caused FR vibration. An additional margin was added in some designs by increasing the contact area (potential wear surface) on the FR support surfaces in the spacer grids. Testing confirmed a significant reduction of the wear volume of FR cladding in this case [24]. In-reactor experience has proven the effectiveness of this design change, reducing this failure mechanism to almost zero.

As for debris fretting, the first defence was to incorporate a debris filter into the bottom nozzle of FAs, to capture foreign material before it entered the first span of the FR. A protective grid such as a Guardian design protective grid that captured 93% of debris was added to enhance the capability of the FA filtering [24]. The Westinghouse design includes the Standardized Debris Filter Bottom Nozzle, and a robust protective grid located above the Standardized Debris Filter Bottom Nozzle, which provides an additional debris barrier resulting in increased fuel reliability. Another level of debris protection is provided by an oxide coating that protects the bottom 15 cm of each FR, thus increasing wear resistance over uncoated cladding. These features have improved fretting resistance and diminished fretting wear. The GRIP bottom nozzle with an improved debris filter is provided with with AREVA's new GAIA FA.

Regarding FA distortion, which is still an issue for some PWRs, especially for those loaded with 4.3 m FAs and managed with cycle lengths of 18 months or more, fuel design changes have been implemented to mitigate the following:

— Creep behaviour;
— Stiffness of the structure;
— Axial loading applied to the FA.

Creep behaviour should be improved with the current deployment of assembly skeletons made of quaternary alloys (Zr+Nb+Sn+Fe). Stiffness has been addressed by increasing the thickness of the guide tubes.

Axial loading should be optimized to minimize the compression of the structure without permitting the lifting of the FA due to the hydraulic force of the coolant fluid. In France, flow optimization has led to the removal of one spring leaf on the top nozzles for 1450 MW(e) plants (previous top nozzles had five spring leaves where new ones have only four). This was possible owing to sufficient flow rate margins on 1450 MW(e) plants to reduce the risk of FAs lifting from the lower core plate.

3.4.2. BWRs

Similar to that observed in PWRs, a slight increase in the debris related fuel failure rates during the last decade was observed for BWRs (see Table 5 and Fig. 10). New fuel design features were implemented to prevent debris failures but they have not been significantly successful.

The implementation of barrier claddings with an increased Fe content, greater stringency in fuel pellet quality and slower ramp rates have reduced but not eliminated PCI and missing pellet surface, which are causes of fuel failure [25]. By around 2004, the only fuel failure mechanism that was continuing to occur was debris fretting. Foreign material exclusion programmes were developed and implemented by utilities and different fuel vendors through fuel design features [26, 27]. For example, the development and implementation of the lower tie-plate ‘Defender’ decreased the size of debris that could enter the fuel bundle, thus lowering the failure rate of fuel manufactured by GNF by a factor of five [24]. AREVA’s FUELGUARD technology has developed to further protect FAs against modern debris forms and is now in its third generation.

Since 2000, there have been observations of channel–control blade interference caused by the distortion of Zircaloy 2 fuel channels that were susceptible to both shadow-corrosion-induced and fluence-gradient-induced bow. The replacement of Zircaloy 2 as the channel material with Zircaloy 4 was not a universal solution. Although Zircaloy 4 was less susceptible to shadow-corrosion-induced bow than Zircaloy 2, Zircaloy 4 had the same or even slightly worse fluence-gradient-induced bow performance. Fuel vendors are pursuing a transition to new materials to address the channel distortion issue. GNF is in the process of replacing Zircaloy 2 and Zircaloy 4 as the channel material with NSF alloy (1% Nb, 1% Sn, 0.35% Fe), resulting in a significantly reduced control blade interference problem [28].

3.4.3. WWERs

WWER plants can be divided into the original version WWER-440 with channelled FAs containing 126 FRs, and the WWER-1000 series, which has been in operation since the mid-1980s and uses unchannelled fuel (with one exception) and contains 312 FRs. Both reactor designs use hexagonal FR matrix and honeycomb type spacer grids (Fig. 22). Zr-1Nb alloy FRs with a 9.1 mm diameter and annular fuel pellets are used in both types. Developments in WWER fuels comprise the use of E635 alloy with higher resistance to irradiation induced growth, creep and corrosion for guide tubes, and E110 alloy for FR cladding and spacer grids.

Units 3 & 4 of Novovoronezh nuclear power plant are WWER-440s of the V179 design and Units 1 & 2 of Kola nuclear power plant are WWER-440s of the V230 design. FAs of the first generation WWER-440s were upgraded to achieve increased resistance to vibration and fuel enriched to 3.82 % of ^{235}U for 4 or 5 years of fuel operation. Units 3 & 4 of Kola nuclear power plant are WWERs of the newer V213 design using FAs of the second generation and fuel enriched up to 4.87 % for 6 years of fuel operation.

Two different design solutions for the WWER-1000 FA have been introduced (Fig. 22): the TVSA assembly [29], which includes a cage on the outside to provide structural rigidity, and the TVS-2 design [30], which uses guide tubes of increased thickness. 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.

Within the past years R&D works have resulted in the development, implementation and successful operation of new types of WWER nuclear fuel:

— WWER-440: fuel of the second generation with higher ^{235}U enrichment;
— WWER-1000 modifications of TVSA and TVS-2, such as TVS-ALFA, TVSA-PLUS, TVSA-12 and TVS-2M.

Further development and optimization of the design of WWER FRs and FA as well as design and fuel materials are currently under way.

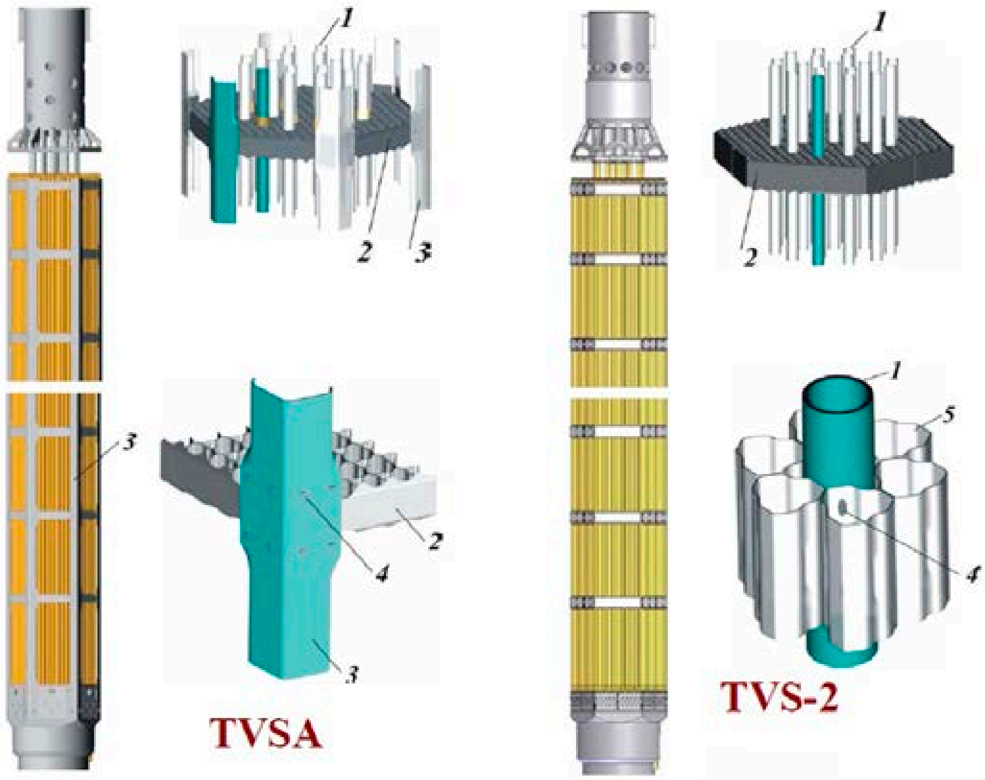


FIG. 22. WWER-1000 FAs with a rigid skeleton TVSA (left) and TVS-2 (right). Legend: 1 — guide tubes; 2 — spacer grid; 3 — angles; 4 — resistance spot welding, 5 — spacer grid cell. Figure courtesy of the Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences [29].

3.4.4. CANDU reactors

CANDU reactors use relatively short fuel bundles (50 cm long) in horizontal fuel channels employing on-line refuelling. The fuel elements have thin collapsible cladding, and no structural components such as spacer grids and support rods are required for the bundle. Bundle types with 28 fuel 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 the end cap profiles and bearing pad positions to account for different fuel handling systems and channel configurations. The Bruce and Darlington reactors in Canada were designed to use 'fuel-against-flow' loading machines, while the other plants 'fuel-with-flow'. However, in 1993 the Bruce units were derated because of safety considerations during a postulated inlet header break accident in which a flow reversal would have resulted in a sudden reactivity increase as low burnup bundles shifted toward the centre of the core (which does not occur in CANDU reactors that fuel-with-flow). Changes were introduced to use long bundles for gap management and reverse the fuelling direction. In 2002, after changes were introduced, coolant gamma activity levels were observed to increase to the point where all four Bruce B units were experiencing a higher than normal number of fuel failures because of debris fretting from mechanical damage to bearing pads [31]. The cause of debris, as shown in out-of-reactor testing, was due to the latch cut-out and bundle carrier orientation, where a subsequent change in the carrier orientation in 2005 greatly reduced the incidence of damage [31].

Generally, debris damage and fabrication flaws are the key mechanisms of FR failure in CANDUs, although the failure level has generally remained at a constant low value in the Canadian reactors. The CANDU fuel design has evolved very slightly with the introduction of a 37R-long and 37M bundle [32]. The 'long' bundle is 13 mm longer to address fuel channel creep. In the modified 'M' bundle, the central element is slightly smaller in size than the other elements as compared with a regular (R) bundle and is used to improve the critical heat flux and safety characteristics of the bundle. These different types of bundles are being introduced in several stations.

Debris failures continue to be a predominant cause of fuel failure in Canadian CANDU reactors. There have not been any design changes to the fuel bundles for failure mitigation since failure rates remain generally low.

More recent fabrication failures were assessed through hot cell examination to determine the root cause. This investigation indicated a need to adhere to appropriate tolerances and pellet loading forces during fuel fabrication to eliminate pellet chipping, especially with the fabrication of heavier bundles [33].

3.5. EVALUATION OF COOLANT ACTIVITY

If a reactor operates with defective FRs, reactor coolant can enter the fuel-to-clad gap and fission products (notably the volatile species of noble gas and iodine) can be released into the primary coolant, resulting in an increase in activity levels at the plant. The high pressure coolant can enter through the defect site where the UO_2 may potentially oxidize locally, which can enhance the fission product release, and lower the incipient melting temperature and thermal conductivity of the hyperstoichiometric fuel [34]. Iodine release can also occur on reactor shutdown if the temperature in the fuel-to-clad gap falls below the saturation temperature allowing liquid water to dissolve the soluble iodine species giving rise to an ‘iodine spike’ [35, 36].

The onset of failure is usually detected with an increase in activity in the reactor coolant or off-gas. The coolant activity can be analysed to determine if it is due to a new fuel failure or an increase in the size of previously known fuel failures, or if it can be attributed to other causes such as changes in reactor power. Various methods have been developed to analyse trends in the activities of individual fission products, including physically based models, as well as codes for fuel failure monitoring. A review for BWRs, PWRs and CANDUs is provided in Refs [3, 37].

A number of codes have been developed for fuel failure monitoring in order to estimate the number of failures, which typically involve an analysis of noble gas and iodine radioisotopes during steady state conditions, including:

— DIADEME: used by the French Atomic Energy Commission [38, 39] (based on the release to birth ratios of noble gas and iodine in the primary coolant and iodine release during a transient).
— FPA (fission product analysis): used by Westinghouse for PWR reactors [40] (based on iodine and noble gas isotopes).
— CHIRON: developed by the Electric Power Research Institute for PWRs and BWRs [41] (based on a combination of seven noble gas nuclides and five iodine nuclides).
— MERLIN: used by Électricité de France [42, 43] (based on seven noble gas nuclides and five iodine nuclides).
— Visual_DETECT for CANDU reactors, which numerically fits a fission product release model to steady state coolant activity data, where the fitting parameters provide for a characterization of the number of fuel failure(s), average defect size and amount of in-core tramp uranium contamination based on well characterized data derived from in-reactor loop experiments at the Chalk River Laboratories [44].

Other techniques proposed for estimating the numbers of failures in CANDUs are given in Refs [45, 46].

In most PWRs, fuel failure detection is based on both ^{133}Xe activity and the $^{133}Xe/^{135}Xe$ ratio, as ^{133}Xe is the most sensitive radionuclide that can be measured in the primary coolant in the case of a fuel failure. A failure is suspected in France, for example, if one the following criteria is reached: $^{133}Xe > 1000$ MBq/t, $^{133}Xe/^{135}Xe > 0.9$ (see Fig. 23), or if a peak of ^{131}I or ^{133}Xe is detected following a transient.

As soon as a failure is detected, a reinforcement of the activity surveillance of the primary circuit is conducted. For instance, the frequency of alpha activity measurements may be increased from one month to one week. Sipping tests will also have to be performed during the unloading of the core.

A specific surveillance of UO_2 release during an operating cycle is also considered in France based on activity measurements of ^{134}I. It may be interesting to know whether a fuel failure is releasing uranium or plutonium oxide into the primary circuit (fuel washout) to anticipate countermeasures that can be put in place during the next outage to minimize alpha contamination. To distinguish the activities of ^{134}I due to tramp uranium and those released from fuel, some empirical criteria, based on operational feedback, have been defined to characterize uranium or plutonium oxide release.

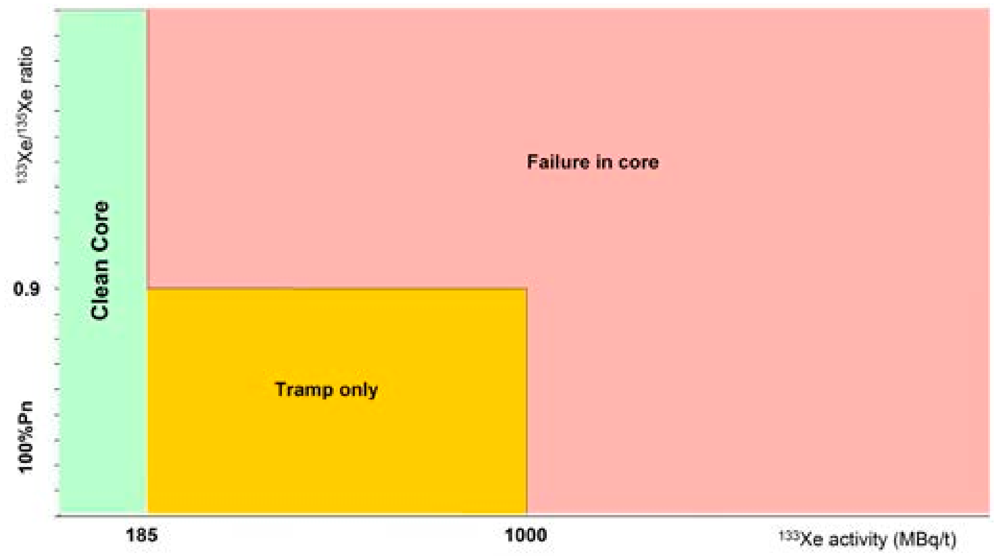


FIG. 23. Fuel failure detection diagram used in France, based on ^{133}Xe and ^{135}Xe. Figure courtesy of Électricité de France.

Thus, in France, a fuel failure is considered to have occurred if: $\Delta A(^{134}I) > 200$ MBq/t and $A(^{134}I)_{EOC} / A(^{134}I)_{EOCtramp} > 1.5$ or $1.2 < A(^{134}I)_{EOC} / A(^{134}I)_{EOCtramp} < 1.5$ with a slope change in the evolution of ^{134}I after fuel failure is observed, where:

— $\Delta A(^{134}I) = A(^{134}I)_{EOC} - A(^{134}I)_{BOC}$ = difference of ^{134}I activity between end of cycle (EOC) and beginning of cycle (BOC);
— $A(^{134}I)_{EOCtramp}$ = ^{134}I activity at EOC due to tramp uranium (forecast according to a theoretical formula).

Some improvements to identify FR failure for UO_2/MOX, based on the ^{135}Xe to ^{85m}Kr activity ratio in the primary coolant, have been performed in recent years. Theoretical calculations performed at Électricité de France show that this ratio is in the range of 10–14 for MOX and 4–8 for UO_2 fuel (see Fig. 24).

A less usual type of fuel failure, a so-called ‘weak leak failure’ or ‘pure gas leaker’ can occur. The ^{133}Xe release of these weak ‘leakers’ is not proportional to the failed FR’s power but to the tiny size of the primary defect. Classical assessment methods, some of which are described above, cannot be employed. The main quantity to monitor is the ^{133}Xe activity such that the activity level can be compared with past activity levels. Reference [47] provides details of the behaviour of these leakers, which can be difficult to detect.

For the WWER design, methods have also been advanced to assess defective FR parameters in operation based on a monitoring of primary circuit coolant activity with application of a design code to interpret data [48–50]. Two codes, RELVVER and RTOP-CA, have been developed in the Russian Federation for WWER applications:

— The RELVVER code is based on analytical models for fission product release from fuel pellets. Recoil and knockout release mechanisms are considered. Release by diffusion is not taken into account. Hence, the range of RELVVER applicability is limited by fuel temperature. The rate of mass transfer inside a leaking FR and the rate of fission product release into coolant are specified by the user. These rates are defined as the corresponding coefficients in balance equations. Additional validation was performed for the last version of the code (RELVVER-UNI) in 2015–2016 up to the burnup of 72 MWd/kgU (the average burnup in FR). Recently, RELVVER has mainly been applied as a design basis tool.
— RTOP-CA is a more mechanistic code which employs similar models to the other codes and accounts for such physical phenomena as: radiolysis assisted fuel oxidation, thermal conductivity degradation with burnup, changes in the fission product diffusivity in hyperstoichiometric fuel, fission product release from the fuel, mass transfer in the failed FRs and release into the primary circuit [51–53]. The RTOP-CA code was validated against in-pile and out-of-pile small scale experiments and experiments with refabricated FRs in research reactors (using artificial defects in cladding). Activity monitoring data in operating WWER units were also used for validation.

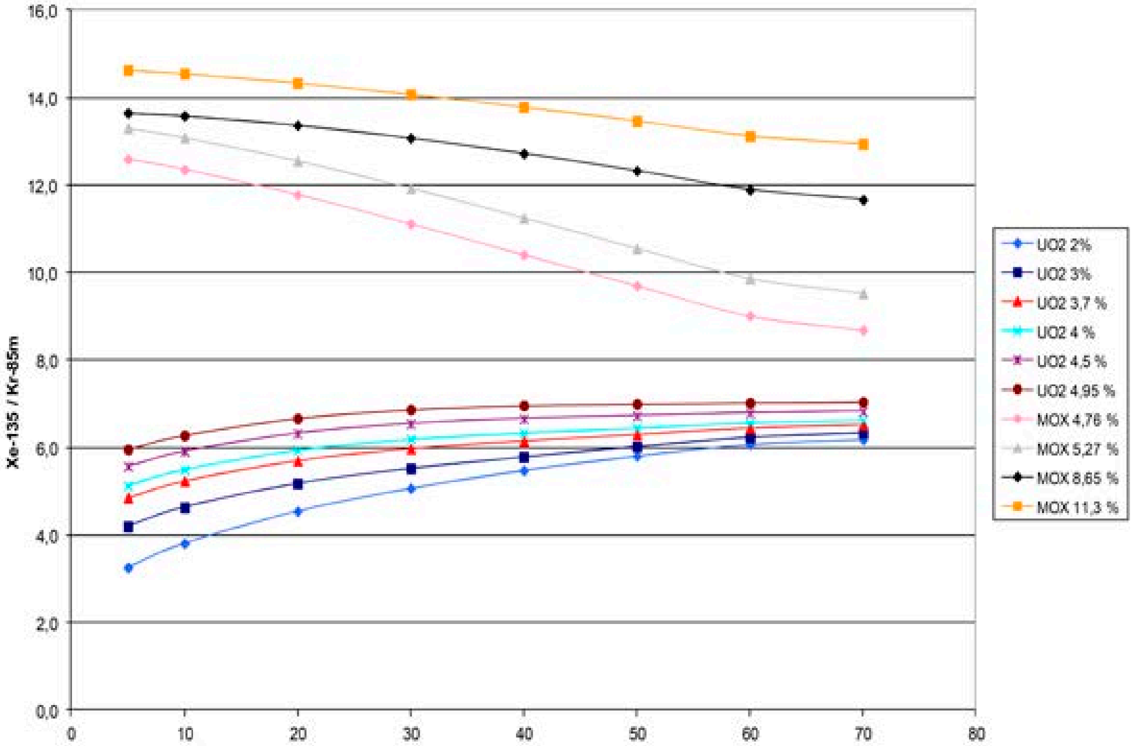


FIG. 24. UO_2/MOX fuel failure determination based on $^{135}Xe/^{85m}Kr$ ratio. Figure courtesy of Électricité de France.

Although these treatments are available for a detailed fuel failure diagnostic analysis, they require a multi-isotopic analysis of either noble gases or iodine. However, a simpler technique has been proposed that establishes a threshold value below which a unit has a high probability of operating without defects. For instance, the Institute of Nuclear Power Operations and the World Association of Nuclear Operators have introduced a fuel reliability indicator based on a measurement of the ^{131}I coolant activity level corrected for tramp uranium effects. The indicator is calculated monthly using Eq. (2) [54]:

$$FRI = [A_{131} - k \times A_{134} \times (L_N/LHGR) \times (100/P_0)]^{1.5} \quad (2)$$

where

- A_{131} is the average steady state activity of ^{131}I in the coolant normalized to a purification constant of 2×10^{-5} 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} is the average steady state activity of ^{134}I in the coolant normalized to a purification constant of 2×10^{-5} s^{-1};
- L_N iis the linear heat generation rate used for normalization (18 kW/m);
- $LHGR$ is the actual average linear heat generation rate at 100% power (kW/m);

and P_0 is average reactor power as a percentage at the time the activities were measured.

The above equation for the calculation of a fuel reliability indicator was already mentioned in the 2010 IAEA fuel failure review [3]. In this publication, the exponent of 1.5 in Eq. (2) is used for normalization to a representative heat generation rate, L_N, where athermal diffusion of fission products through the fuel matrix predominates. This methodology, however, is currently being assessed through a first principles analysis for CANDU fuel [55]. Higher linear fuel ratings are experienced in CANDU fuel compared with PWR, BWR and WWER defective FRs where vacancy-enhanced diffusion (for oxidized fuel) and high temperature intrinsic diffusion may occur in addition to athermal diffusion that could affect the scaling.

3.6. FUEL FAILURE IDENTIFICATION

The identification of leaking FAs after shutdown is usually carried out by testing the assemblies under transient conditions (e.g. change of vertical position or heating up of the assembly). During these transients increased activity release from the leaking FRs can be detected. Different sipping methods can be applied (e.g. in-core sipping, telescope sipping, canister sipping) in the reactor vessel, in the spent fuel pool or in the mast of the refuelling machine. A more precise identification of leaking FRs in failed FAs and their failure mechanisms and root causes is conducted either by methods of poolside inspection or hot cell examination. More details can be found in Ref. [3].

3.6.1. Sipping in PWRs and BWRs

In most PWRs, fuel failure detection during outages is based both on in-mast sipping and in-cell sipping devices. In-cell sipping is performed in the fuel building and is used either on high residual power FAs, for which in-mast diagnosis is uncertain and cannot be used to classify an FA as a 'leaker' or non-leaker, or on low residual power FAs, after repair, to check the efficiency of the repair.

In-cell sipping in most PWR plants uses the wet sipping technique, based on heating the water around the FA to produce fission gas expansion. French in-cell equipment includes both a gas loop to measure ^{133}Xe radionuclides and a water loop to measure ^{134}Cs and ^{137}Cs radionuclides. Xenon-133 is measured on-line during the heating of the assembly, with caesium measurements processed off-line.

In most European countries, in-cell devices are fixed in the concrete structure of the fuel building. In the USA, mobile devices are used for in-cell sipping based on the vacuum sipping technique. Mobile devices generally include only a gas loop, as a water loop is more difficult to implement in mobile equipment.

In-cell sipping (either wet or vacuum sipping) is very reliable and is complementary to in-mast sipping. On the other hand, in-cell sipping requires more time to test an FA than in-mast sipping. Another drawback of the in-cell equipment, especially for fixed devices (such as those used in France and the rest of Europe) is the maintenance of the equipment, which is more important than for in-mast equipment and can be difficult.

Both vacuum and wet sipping techniques are also used in BWRs. However, vacuum sipping, because of its better accuracy, is used more frequently in BWRs than wet sipping [3]. In-mast sipping is another frequently used technique in BWRs for the identification of leaking FRs.

3.6.2. Flux tilting in BWRs

In a case of primary fuel failure in a BWR, it is important to locate the 'leaker' to protect it from power increases. Traditional flux tilts at low power may damage the failed FR and a milder flux tilt at higher power has therefore been developed. Because of reduced load on failed fuel, the probability of a defect's growth is not significant. This method still detects failures with high accuracy and at less operational cost. Recommendations on optimal flux tilting regimes in BWRs can be found in Ref. [56].

3.6.3. Failed fuel detection in BWRs

Off-gas monitoring provides the simplest method for monitoring fuel integrity in BWRs. Except for decay, fission gases that escape from failed FRs or from fission of tramp uranium are quantitatively transferred (at a rate higher than 98%) by the steam system from the primary coolant to the main condenser. The off-gas system maintains the main condenser vacuum at less than 13.79 kPa by extracting non-condensable gases. These non-condensable gases include the filling gas (He), coolant radiolysis gases and fission product gases. At time zero, the design basis release rate for the noble gas nuclides is approximately 92.5 GBq per second. Most of these nuclides decay rapidly, and after 30 minutes the activity is reduced to approximately 3.7 GqB per second. The normal off-gas grab sample analysis is based on six nuclides, ^{85m}Kr, ^{88}Kr, ^{87}Kr, ^{133}Xe, ^{135}Xe and ^{138}Xe. A fuel failure in a BWR is often first identified by a steep increase in the off-gas monitor response. In addition, most BWRs augment noble gas analysis with reactor water analysis of major iodine and caesium nuclides. In some cases, analyses of other soluble radionuclides (strontium and neptunium) are performed.

Historically, various isotopic activity ratios have been used to provide an indication and confirmation of the presence of failed fuel. Generally, isotopic ratios may provide better indications of the presence of small defects or changes in the defect release characteristics than activity concentrations alone. Any ratios that accentuate the difference between short lived and long lived nuclide release rates can be used as an indication of fuel integrity. A significant change in the magnitude of these ratios is a clear indication of a fuel failure. Both noble gas and iodine isotopes serve the same purpose. Xenon-138 is the shortest lived of the commonly measured off-gas nuclides, while ^{133}Xe is the longest lived. The most common ratio utilized in BWR fuel integrity assessments is the ^{138}Xe to ^{133}Xe ratio. Values of the ^{138}Xe/^{133}Xe ratio <100 strongly indicate the presence of defective fuel in the core. Values greater than 300 have been found to be indicative of a defect free core. The theoretical ratio of ^{138}Xe to ^{133}Xe should be 448 for a pure recoil release with no new fuel failure. It should be noted that in the presence of a high recoil level, a small release of ^{133}Xe from a new failure may not significantly change the ^{138}Xe/^{133}Xe value. Other isotope ratios have also been used for the indication and monitoring of fuel failures.

3.6.4. Failed fuel detection in WWERs

The visual inspection of FAs in the mast of the refuelling machine using a camera enables the assessment of the following:

— FR damage;
— The corrosive condition of FRs and FA components;
— Any deformation of FRs and FAs, including bow, elongation, etc.;
— Fretting signs;
— FA damage due to refuelling operations;
— The presence of foreign objects.

In most WWER units, the detection of leaking FAs during reactor outages is performed in two stages. The first stage is the sipping test in the mast of the refuelling machine. When an FA is extracted out of the core, air is blown into the lower part of the mast and the activity of ^{133}Xe is monitored. If the recorded activity exceeds a certain level (3σ or other), the FA is considered to be ‘suspected’ of leaking. The second stage is a mandatory test of all the suspected FAs in special casks for FR cladding integrity control in the spent fuel storage pool. Two methods are applied for the leakage test.

A traditional method is as follows. Water pressure in the leakage test cask is raised, held for several minutes and then dropped. After mixing of water in the circuit of the leakage test cask, one water sample is taken to measure ^{131}I activity. The ‘suspected’ FA is declared to be leaking (or sound) by comparing its activity to the activity of several other (usually more than 10) FAs which were not marked down as ‘suspected’ when tested in the mast of the refuelling machine. An FA is declared to be leaking without comparison to other FAs if its ^{131}I activity exceeds a ‘gross failure’ limit indicated in the technical specifications by the fuel vendor.

A second technique has been applied in recent years in most WWERs in the Russian Federation. It uses pressure cycling in the leakage test cask. Water pressure in the leakage test cask is raised and reduced repeatedly over a period of about 20 minutes. During this time several water samples are taken to measure the activity of ^{131}I, ^{133}Xe, ^{134}Cs, ^{136}Cs, ^{137}Cs, ^{140}Ba and ^{106}Ru. By comparing activities in different samples, it is first determined whether the FA is leaking or not. Secondly, it is determined whether the leakage is minor or severe. The corresponding criteria were developed for different designs of WWER fuel by using the RTOP code [57]. The pressure cycling technique provides a means of estimating the equivalent hydraulic diameter of the primary defect in cladding and to reveal the occurrence of coarse secondary defects in the cladding of a leaking FR.

3.6.5. Failed fuel detection and localization in CANDU reactors

CANDUs have two systems for locating fuel defects in the core: (i) the delayed neutron system and (ii) the feeder scanning system. The Bruce and CANDU-6 reactors employ a delayed neutron system, while Darlington reactors have a feeder scanning system. The Pickering reactors have no failed fuel location system. The location of a fuel defect can only be known after the fuel has been discharged from the core. Detailed descriptions of delayed neutron and feeder scanning systems are presented in Refs [58–60].

3.7. FUEL FAILURE MECHANISMS

3.7.1. Primary failures and their causes observed in 2006–2015

Table 10 indicates fuel failure (leak) mechanisms as well as root causes identified during the 2006–2015 period. In 2006–2015 these mechanisms accounted for the following proportion of fuel failures in PWRs: grid-to-rod fretting (~58%), debris fretting (~26.6%), fabrication deficiencies (~12.4%) and PCI/SCC (~2.8%). In BWRs, fuel failure mechanisms include debris fretting (~62.3%), PCI/SCC (~27.6%), crud or corrosion (~8.6%) and fabrication deficiencies (~1.2%). At the same time debris fretting has been the main cause of FR leaking in WWERs. In CANDUs, debris fretting and manufacturing defects (incomplete welds and incorrect pellet tolerances) were the main fuel defect types observed in 2006–2015.

Fuel failure mechanisms observed in 1994–2006 and not observed from 2007 onwards include: baffle jetting in PWRs, primary hydriding (owing to moisture, other contamination in pellets or cladding due to manufacture) in all reactor types and clad collapse in PWRs. No new mechanisms of fuel failure in reactors operated at normal conditions have been reported for the period 2006–2015. Fuel failure root causes that appeared and were observed during the last decade are considered in Sections 3.7.1.1 to 3.7.1.6.

3.7.1.1. Fretting due to grid spring cracking in PWRs

In some French PWRs, fretting, due to the breaking of FA grid springs, has led to fuel failures. FA grid springs are internal components of the grids that maintain the FRs in the grid cells by means of dimples.

The root cause of the breaking of the springs, which were made of Inconel alloy 718 as part of the AREVA AFA 3G fuel design, is irradiation assisted stress corrosion cracking. The rupture is always intergranular and can affect different parts of the spring.

Grid spring cracking mainly affects bottom grids, and in some cases top grids that are submitted to lower neutron fluxes. A high neutron flux may cause relaxation of the internal stresses in the springs and decrease the risk of stress corrosion cracking, while a low flux may induce viscoplastic strain and the priming of the crack. According to this most likely scenario, failures occur at the end of the lifetime of FAs and mainly affect assemblies close to the baffle during their first cycle (core positions with low neutron flux). The sharp edge of the broken ligament can wear out the cladding with vibrations, causing a through-wall defect.

The use of material less sensitive to crack initiation will be implemented by AREVA for grid spring manufacturing to reduce the occurrence of grid spring cracking.

3.7.1.2. Seal weld defects in PWRs

In past years, some failures have occurred due to welding defects affecting the top and bottom end plugs. The defects were due to accidental contamination during welding operations in the manufacturing plants.

In France during the last decade, seal weld defects have also led to fuel failures on AREVA fuel. It has been established that such defects only occur at the upper end plug, which undergoes a second pass in the sealed welding chamber. Figure 25 illustrates such a defect.

As for end plug welding defects, the problem of seal weld defects has been solved by AREVA by replacing laser and tungsten inert gas welding processes with the upset shape welding process. The upset shape welding process does not melt the alloys and can be implemented in hyperbaric conditions.

3.7.1.3. Thin chips in PWRs

In French reactors, some very small through-wall cracks were observed in recent years on failed FRs of first cycle FAs (see Fig. 26). The leakage generally occurred during the first three months of the cycles. AREVA named this kind of defect ‘early leakers’.

AREVA root cause analyses concluded that those failures were caused by internal debris (thin chips or shavings) generated during FR insertion in the skeleton of the FAs. Since 2010, the FRs have been lubricated with water during their insertion into the FA skeleton, which has addressed the problem.

TABLE 10. PRIMARY FR FAILURE MECHANISMS AND RELATED ROOT CAUSES

FR failure mechanisms	Root causes	Related areas			Found in plant designs:			
		Manufacturer	Operation	Design	BWR	PWR	WWER	CANDU
Cladding wear due to vibration	Insufficient FR support (design/manuf. related)	✓		✓		✓		
Grid-to-rod fretting	Rod vibration by fluid elastic instability/ cross-flow		✓	✓		✓		
	Flow induced FR/FA vibration		✓	✓		✓		
	Grid cell damage during handling		✓			✓	✓[b]	
	Grid spring cracking	✓		✓		✓		
Debris fretting	Debris circulating in the coolant	✓	✓	✓	✓	✓	✓	✓
	Cladding thin shaving	✓				✓		
	Baffle bolts cracking		✓[a]			✓		
	Cladding chipping (shaving)	✓				✓		
Corrosion	Crud or corrosion		✓	✓	✓			
Manufacturer defects	Welding defects (end plugs or seal)	✓			✓	✓		✓
	Pellet chipping	✓		✓				✓
PCI/SCC	Normal ramps, missing pellet surface	✓			✓	✓		✓
Pellet–cladding mechanical interaction	PCI at high burnup		✓		✓			
Dryout	Loss of coolant wetting		✓		✓[b]			

[a] In-reactor equipment wear.
[b] Isolated event.

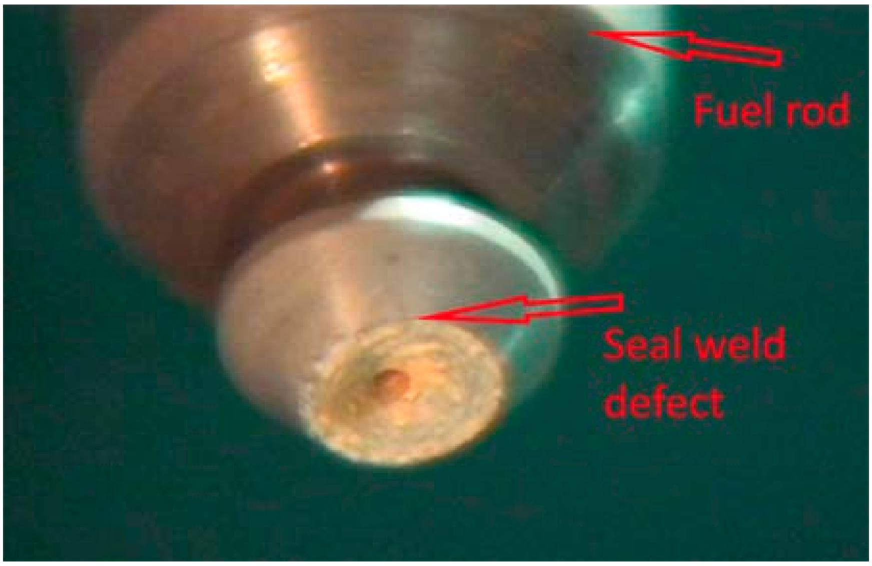


FIG. 25. Seal weld defect on a FR in an French reactor. Figure courtesy of AREVA NP.

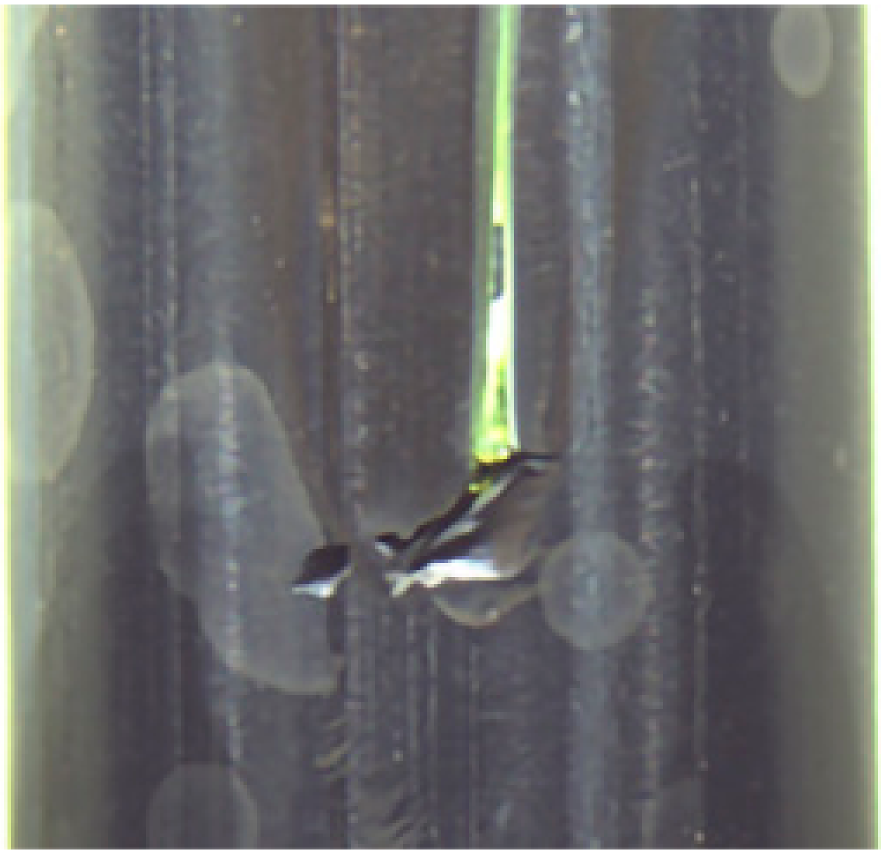


FIG. 26. Early leaker (first cycle) on a FR of an French reactor. Figure courtesy of AREVA NP.

3.7.1.4. Fretting due to cracking of baffle bolts in PWRs

A bolt head or bolt lock tab may detach from the reactor internals if the baffle bolt is degraded. During normal operation, the separation of a bolt head or a bolt lock tab can introduce loose parts or foreign material into the reactor coolant system. The loose parts or foreign material could impact FAs and potentially lead to fuel failure. Two plants in the USA have detected degraded baffle bolts. Operating experience indicates that the baffle bolts are more susceptible to degradation in older Westinghouse four-loop reactors that have a ‘down-flow’ configuration. The baffle bolts are now made of Type 347 stainless steel. The NRC’s initial risk informed assessment of the issue determined that degraded baffle bolts do not warrant the immediate shutdown of any plant. The issue is not a significant safety concern [61].

3.7.1.5. Manufacturing defects: Pellet chipping in CANDU reactors

Improvements in the manufacturing process have reduced failure rates over the years. However, such failures still occur, mainly at the end-cap welds. Another type of event that occurred in this reporting period related to manufacturing was pellet chipping when there was limited free volume in CANDU fuel elements. Pellet chipping arose due to high pellet loading forces and low diametral clearance tolerances in the sheath in the high mass fuel bundles [62].

3.7.2. Other physical processes with defective CANDU fuel

3.7.2.1. Iodine induced stress corrosion cracking as a failure mechanism for power ramped fuel

Iodine stress corrosion cracking depends on certain factors such as iodine concentration, oxide layer type and thickness on the fuel sheath, irradiation history, metallurgical parameters related to the sheath such as texture and microstructure, and the mechanical properties of the zirconium alloys [63, 64]. A comprehensive thermodynamics study has shown that as the mole fraction of I increases, ZrI_4 becomes the more dominant species [65]. It has been further suggested that the influence of CANLUB in CANDU fuel, used to mitigate iodine stress corrosion cracking occurrences, may be due to impurities found in the CANLUB interlayer coating (such as Na [65]).

As shown in Fig. 27, the following mechanistic processes occur in the fuel-to-sheath gap: (i) fission product release by diffusion of the long lived and short lived species of I (and Cs) from the fuel matrix to the free surfaces of the fuel; (ii) deposition and formation of CsI at the free surfaces of the fuel, (iii) radiolysis of the CsI due to energy loss of fission fragment recoils to produce I_2 for gas phase diffusion in the fuel-to-sheath gap to the crack tip, (iv) gettering of the iodine by the Na impurity in the CANLUB and (v) a ZrI_4 reaction through a Van Arkel process at the crack tip for crack penetration and propagation [65].

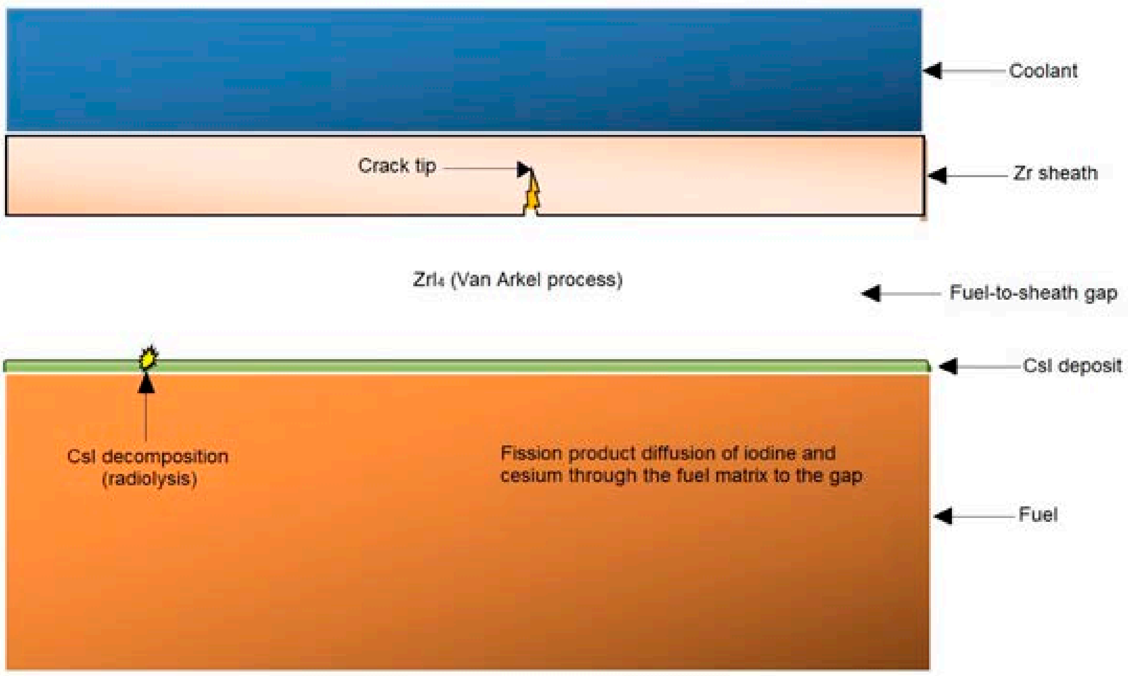


FIG. 27. Schematic of the iodine stress corrosion cracking process for crack penetration. Figure courtesy of the Royal Military College, Canada.

In this treatment, the simple empirical treatment of Penn-Lo-Wood [66] is used to account for the crack initiation time; however, a more mechanistic treatment is still needed to fundamentally couple any chemistry kinetics model to a fundamental stress analysis or crack nucleation model. This chemical kinetic combined model has been benchmarked against 335 cases of experimental and commercial power ramp experience.

3.7.2.2. Physicochemical properties of defective fuel

With the presence of steam in the fuel-to-sheath gap of a defective FR, the consequential effect of fuel oxidation will affect both the thermal performance of the FR and the associated fission product release into the primary coolant [66]. The performance of defective FRs can be affected by degraded fuel thermal conductivity with continued fuel oxidation, a lower incipient melting temperature for hyperstoichiometric fuel (see Fig. 28), and fuel restructuring (i.e. columnar grain growth) [67]. Fuel loss from defective fuel can also result from grain boundary oxidation under the defect site with coolant erosion that yields tramp uranium contamination in the reactor. The release of volatile fission products can be further enhanced with a greater diffusional mobility in the hyperstoichiometric fuel.

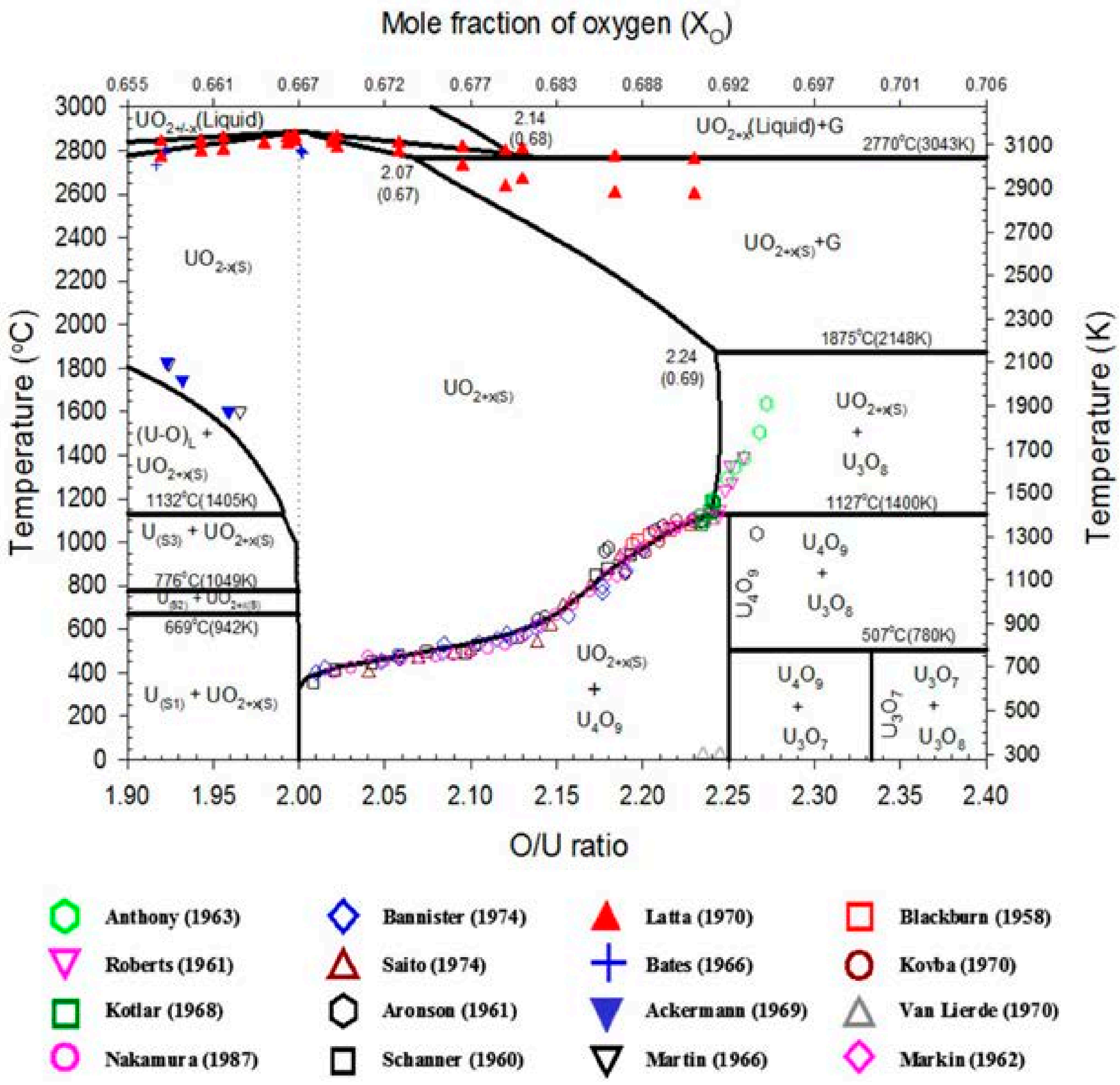


FIG. 28. Calculated phase diagram for the U-O system, highlighting the UO_{2+x} region, shown against experimental phase boundary determinations. Figure courtesy of the Royal Military College, Canada [66].

Oxidation of the fuel and sheath, in addition to H_2O (or D_2O) radiolysis, leads to hydrogen (deuterium) production in the gap, where a critical H_2/H_2O or D_2/D_2O ratio, usually at locations far away from the defect, can lead to sheath hydriding (deuteriding). Hydriding occurs through the following reaction:

$$H_2(g) + Zr(\alpha) \rightarrow ZrH_2(\delta)$$

The diffusion of hydrogen through ZrO_2 is very slow to the extent that hydriding is only likely to occur on a bare area of the cladding or where the protective oxide has broken down. The molar volume of the δ hydride phase is 17% greater than that of the original phase of zirconium (α-Zr) in the sheath. A hydride blister eventually grows in a 'sunburst' shape. The increases in molar volume of the sheath make it brittle and can lead to secondary defects associated with fission product release. Thermodynamic models of the Zr-H and Zr-O-H systems yield the terminal solubility and partial pressure concentrations required to form hydride in the cladding [68, 69]. As such, it can be demonstrated that virtually pure H_2 is required to produce a secondary hydride.

3.8. STRUCTURAL DAMAGE TO FUEL

3.8.1. FA stuck under upper internals in PWRs

Three events of FAs stuck under the upper core plate during the upper internal lift-off occurred in France, one in 2008 and two in 2009. Figure 29 illustrates such an event, in which two FAs were seen to be stuck after removal of the upper internals.

The situation was recovered using a two stage approach to:

- Secure the assemblies by positioning two metallic girders, one on each side of the FAs, across the vessel (on the vessel flange);
- Dislodge the assemblies from the upper core plate using a specific tool.

Two root causes were identified to explain the different events. The first was the presence of a significant gap between FAs after reloading the core. This root cause was responsible for two events of FA lift-off in 2008 and 2009 in the same unit. The gap between the two assemblies occurred in the following ways:

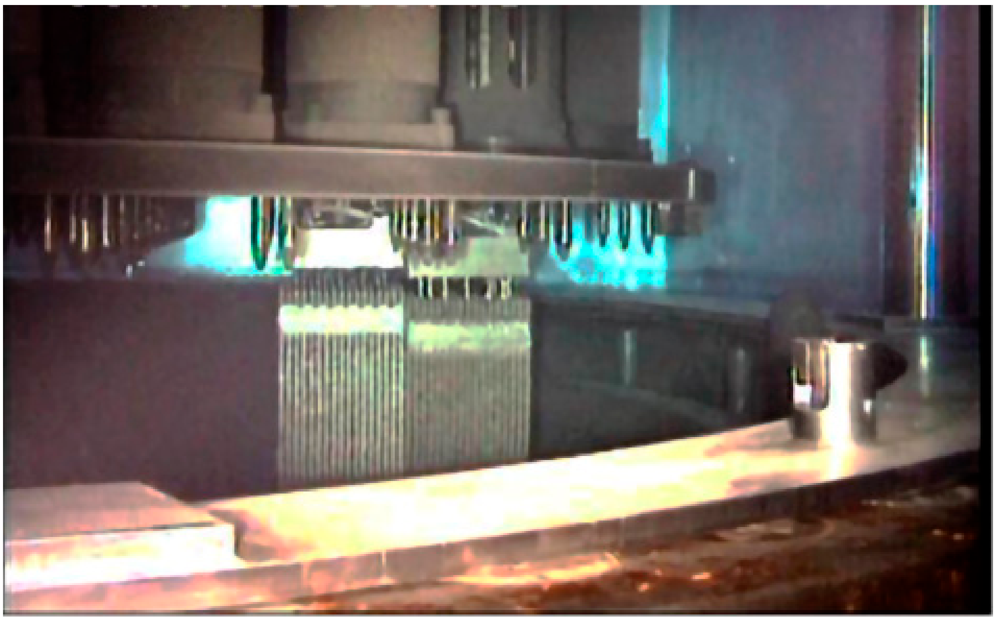


FIG. 29. Two FAs stuck under an upper core plate in a French plant in 2008. Figure courtesy of Électricité de France.

— In the first event in 2008, a ball was lost from a ball bearing of the refuelling tool and positioned under the foot of one assembly at the lower core plate. The ball caused a tilting of the head of the FA leading to a gap between this assembly and its neighbour.
— For the second event in 2009, a significant bowing of an FA caused the tilting of its top nozzle.

When the upper internals were put in place, one upper core plate pin jammed into the hole instead being inserted into the FAs. In France, new procedures of core mapping after reloading were defined after this event to prevent shifts of the S-holes greater than 7.5 mm. New requirements for the televisual inspection of the lower core plate were also defined to improve debris detection, with the aim of fully cleaning the lower core plate before reloading. To minimize the foreign material exclusion risk, inspection of the bottom nozzles was also carried out at each unloading of the core.

The second root cause identified was a metal embossing inside the S-hole of the top nozzle induced seizing between the pin of the upper core plate and the S-hole. In this specific case, no significant gap existed between FAs once the core had been fully loaded. Nevertheless, one FA was again seen to be stuck under the upper core plate. As a consequence of this event, a televisual inspection of the S-holes of top nozzles is systematically performed before core reloadings in French reactors. In cases where metal embossing is suspected, the FA is not reloaded and has to be repaired.

3.8.2. Damage to spacer grids in WWERs

Damage to the 'flat' rims of TVSA-ALFA spacer grids during the loading and unloading of FAs during maintenance was observed in 2009, 2010 and 2011 at Kalinin nuclear power plant Unit 1, a WWER-1000 (Fig. 30). The damage resulted in the generation of foreign material objects which led to an increase in the fuel failure rate. The root cause was defined as the poor design of the spacer grid's rim. After changing the rim's design in 2010 there were no incidences of damage to spacer grid rims of the TVSA-ALFA FAs [70].

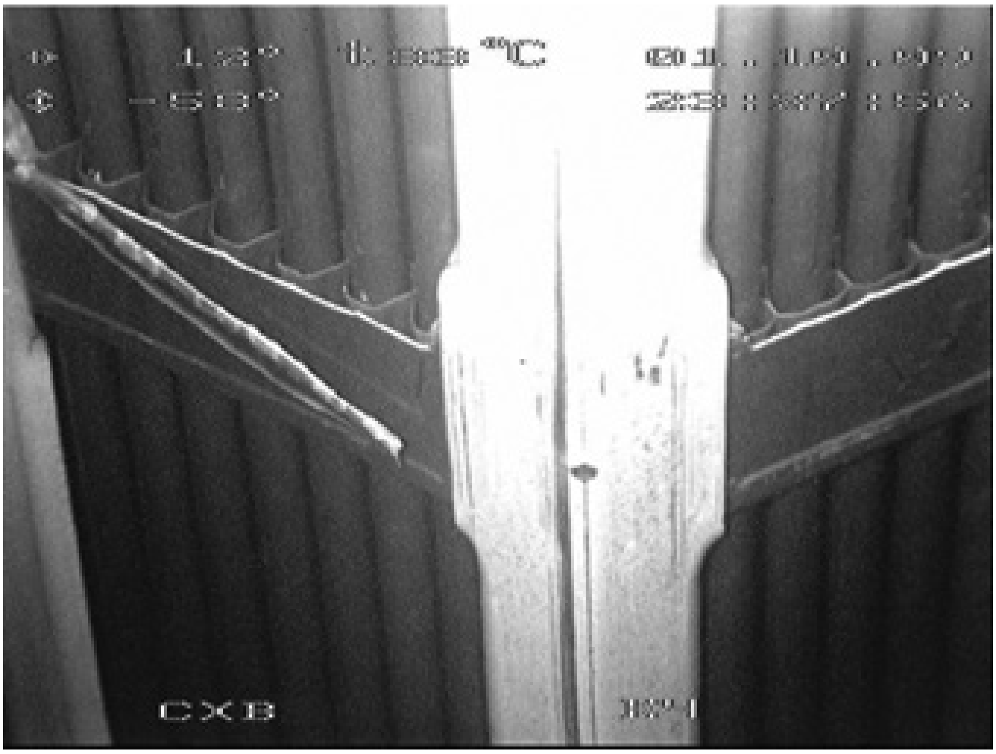


FIG. 30. Damage to the fourth spacer grid of the TVSA-ALFA FA at Kalinin nuclear power plant Unit 1. Figure courtesy of the Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences [70].

3.8.3. Structural damage to fuel in CANDU reactors

The structural damage of CANDU fuel may occur during operation and could lead to a sheath breach resulting in higher coolant activity levels or cause structural damage to the bundle components, which could cause difficulty in discharging bundles from the channel. The mechanisms of structural damage are detailed in Sections 3.8.3.1–3.8.3.5.

3.8.3.1. Longitudinal ridging

The thin CANDU fuel sheath is designed to 'collapse' on the pellets under the coolant pressure in the reactor because of its elastic instability at high external pressures. Longitudinal ridge(s) can form in the sheath with plastic deformation concentrated at the tip of the ridge(s) owing to to a combination of high external coolant pressure, high internal clearances and minimum wall thicknesses.

The 'critical collapse pressures' for CANDU fuel sheath longitudinal ridging were determined by out-of-reactor tests, and a correlation was developed to predict critical collapse pressure for an element design and its operational conditions. Tests were performed covering various metallurgical conditions and operating conditions, which included hydrostatic test conditions. Sufficient safety margins are achieved by maintaining the coolant pressure below the critical collapse pressure.

3.8.3.2. Axial sheath collapse

The thin sheath of CANDU fuel is susceptible to large radial deformations under coolant pressure if it is not supported by the pellets inside the fuel element; i.e. if there is an axial gap between the pellets (or between the end-pellet and the end-cap). CANDU fuel elements have an axial gap and part of this may be distributed between the pellets and between the end-pellet and the end-cap on either or both ends of the fuel element. The sheath must withstand the coolant pressure if this cumulative axial gap appears at any position in the element (see Fig. 31). Axial collapse can also occur owing to a chipped or missing pellet.

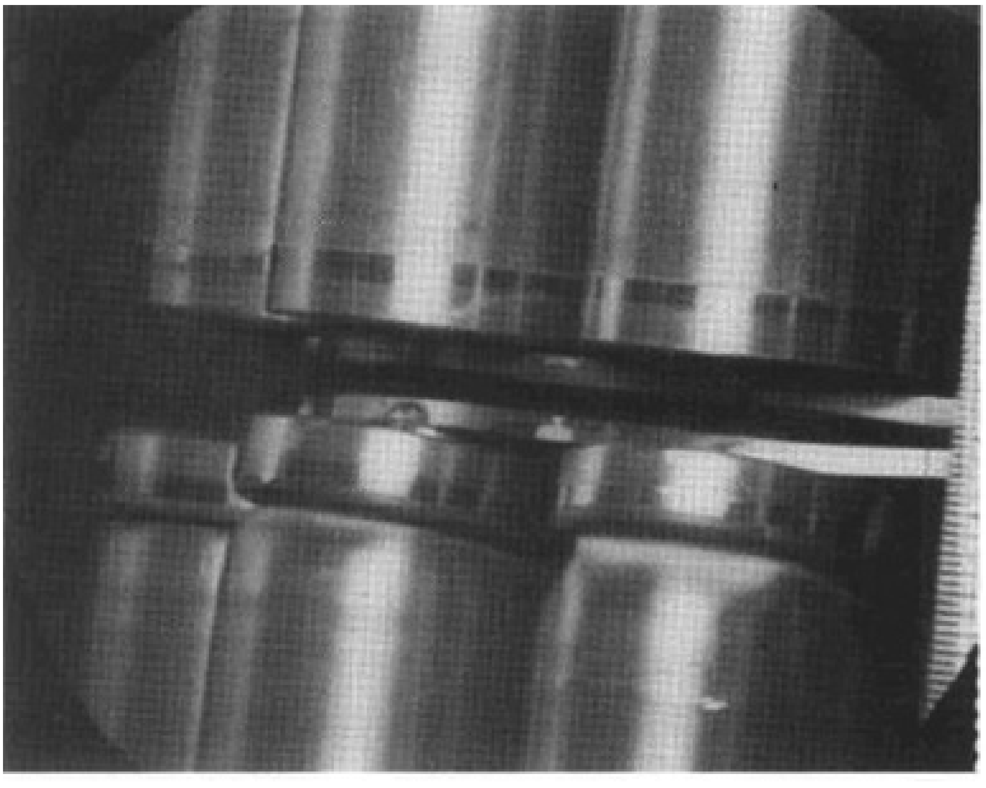


FIG. 31. Sheath axial collapse into axial gaps in a CANDU reactor. Figure courtesy of CANDU Energy.

Tests were conducted to determine the effects of external pressure on the CANDU fuel sheaths at different axial gaps. The tests showed that at small axial gaps, radial deformation starts as 'necking' or 'ridging' which increases with increasing pressure. At larger axial gaps, an instantaneous collapse of the sheath occurs when the pressure reaches a critical level; this level is termed the 'critical axial collapse pressure'. Tests were performed covering various metallurgical conditions and operating conditions, which included hydrostatic test conditions. Sufficient safety margins are achieved by maintaining the coolant pressure below the critical axial collapse pressure.

3.8.3.3. Fretting damage

Fretting damage to the sheath is caused by debris circulated through the core by the coolant. Defective elements can appear anywhere in the bundle and anywhere along a fuel channel, depending on the size of the debris. Previous experience has shown that the defects caused by debris fretting are usually single element failures. The defective elements may have local areas of swelling because of UO_2 oxidation, uranium deposition downstream of the defect and small amounts of secondary deuteride damage. The amount of secondary damage depends on the burnup or time duration in the core while the fuel is defective. Primary defect sites are usually found during inspections in the bays and by post-irradiation examination in hot cells (Fig. 32). In addition, shiny surfaces indicate where the fretting has occurred.

Fretting between the interelement spacer pads and between the bearing pad and pressure tube during operation can lead to sheath failure or damage. Fretting of the appendages and the pressure tube are caused by flow induced vibrations while the bundle resides in the axial flow region and the cross-flow region during normal operation. The wear rate is increased when the bundles reside for extended periods in the cross-flow region and are subjected to acoustic pulsations.

3.8.3.4. Fatigue damage

Fatigue in fuel structural components results from the cumulative effect of strain cycles in the components. The failures can occur when a bundle remains in the cross-flow region for extended periods or is subjected to acoustic vibrations. The critical area where fatigue failure can occur in a bundle is the end-plate web and the bundle assembly weld.

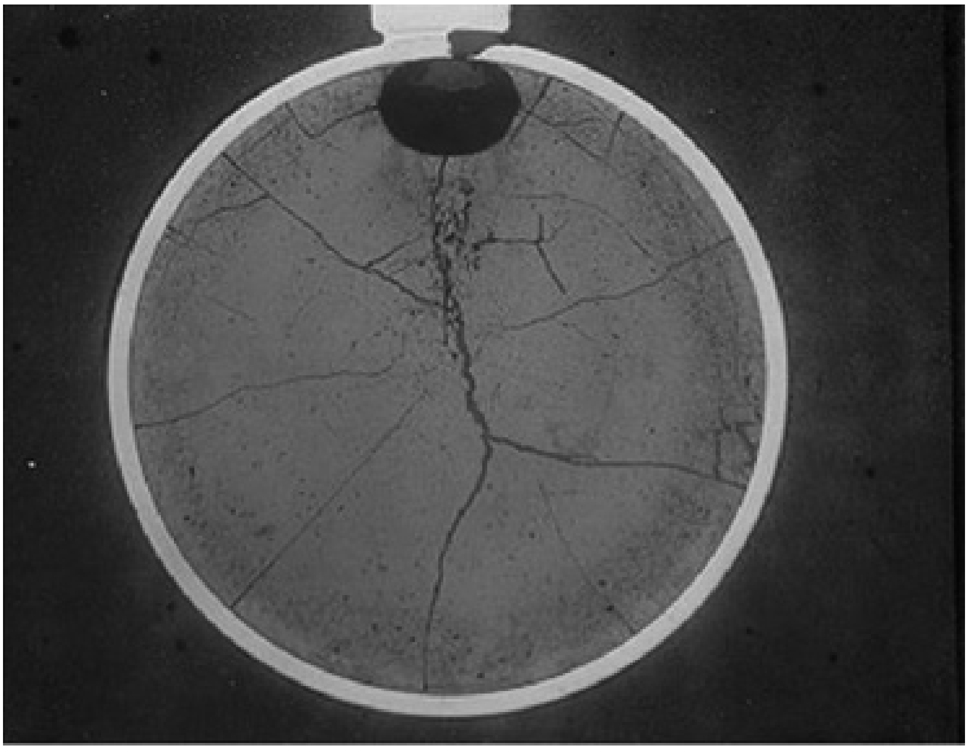


FIG. 32. Primary defect under a pad followed by UO_2 erosion. Figure courtesy of SNC-Lavalin, CANDU Energy.

The fuel bundle experiences cyclic loads produced by coolant flow during its residence in the inlet and outlet cross-flow regions of the fuel channel during refuelling (a residence of typically less than 10 minutes) and during residence in the fuel channel (typically hundreds of days). Fuel bundles may remain in the cross-flow region for extended periods for various reasons. The accumulated fatigue life fractions can cause failure of the endplate and the endplate-to-assembly weld (Fig. 33). Out-of-reactor tests have been conducted and operational guidelines and limits are provided to avoid this failure mechanism.

3.8.3.5. Other fuel damage mechanisms

Other potential fuel damage mechanisms include delayed hydride cracking, refuelling impact, sliding wear, corrosion, crevice corrosion and crud formation.

3.9. FUEL FAILURE PREVENTION AND MANAGEMENT

3.9.1. Coolant activity and radiochemistry operational technical specification limits

If a nuclear power unit operates with leaking FRs in the core, activity may be released into the coolant. These units can operate with leaking FRs if the specified limits are not reached. The most important indicators to be measured and used to limit the operation with leaking FRs are:

— Iodine-131 activity in the primary coolant;
— The sum of the activity of iodine isotopes;
— Radioactive noble gas release;
— The uranium concentration in the primary coolant.

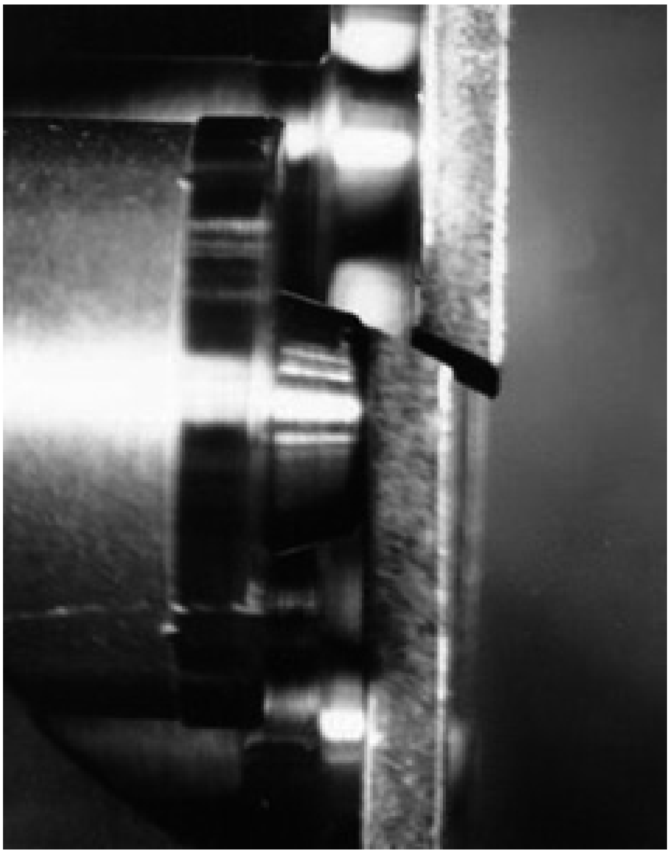


FIG. 33. Endplate fatigue crack. Figure courtesy of SNC-Lavalin, CANDU Energy.

3.9.1.1. PWRs

In France, radiochemical operational technical specifications that were defined in 2003 for 1300 MW(e) plants to avoid new massive fretting failures such as in cycle 8 of Unit 3 of Cattenom Nuclear Power Plant were reviewed in 2013. Because no further fretting failures have been observed in French plants, new limits were defined, regarding such aspects as:

— Radioactive gas release to the environment for the parameter sum of gas activities (^{133}Xe + ^{133m}Xe + ^{135}Xe + ^{138}Xe + ^{85m}Kr + ^{87}Kr + ^{88}Kr);
— Radiological consequences of a class 4 steam generator tube rupture accident for the dose equivalent iodine;
— Radioprotection (limitation of the fissile material in the primary coolant) based on ^{134}I.

For each parameter, two thresholds are determined that imply different actions for the plant operation:

— The ‘reinforced surveillance’ threshold: When this limit is reached, the load follow is stopped and the periodicity of the radiochemistry measurements is increased.
— The ‘shutdown’ threshold: Reaching this limit implies the shutdown of the plant within 48 hours or 8 hours, depending on the activity.

For French 1300 MW(e) units, the new limits are summarized in Table 11. They will soon be extended to all French nuclear power plants.

3.9.1.2. BWRs

The triggers for plant shutdown are specified in a plant’s technical specification, based on Revision 1 of NUREG-0473 [71]. For BWRs, the requirements for most plants are (a) the gross radioactivity (beta and/or gamma) rate of noble gases (sum of six) measured at the main condenser air ejector which has a limit of ≤ 3.7 MBq/sec/MWt (100 μCi/sec/MWt), and (b) the primary coolant dose equivalent iodine which has a limit of ≤ 7.4 kBq/g (0.2 μCi). A trigger point for a plant shutdown below the technical specification requirements may not be firmly established to allow utilities to evaluate specific issues associated with a given failure scenario and to allow adequate time to plan for a mid-cycle fuel maintenance outage or early refuel outage. Typically, the trigger values established for each action level by a plant will depend on plant specific factors.

Utilities have developed action plans typically with 3–5 action levels beyond failure free operation to track the condition of failed fuel and to implement actions to minimize the overall effect of a fuel failure on plant effluents, personnel exposure and plant operation. The failed fuel action plans are required beyond the routine radiochemistry reviews during failure free operation. The conditions that trigger the action levels described above are typically based on coolant and/or off-gas fission product activity levels that may be corrected for pre-failure tramp activity. For BWRs, this is typically the off-gas sum of six activities or the fuel reliability indicator. Some utilities use only one indicator for each action level, which was the standard approach when action plans were first

TABLE 11. NEW RADIOCHEMISTRY OPERATIONAL TECHNICAL SPECIFICATION LIMITS FOR FRENCH 1300 MW(e) UNITS

	Reinforced surveillance threshold	Shutdown within 48 h threshold
Sum of noble gas activities	10 000 MBq/t (0.3 μCi/g)	150 000 MBq/t (4 μCi/g)
Dose equivalent iodine	4000 MBq/t (0.1 μCi/g)	20 000 MBq/t (0.5 μCi/g)
I-134	A + 1000 MBq/t	A + 3000 MBq/t

Note: $A = A_0 (1 + k \cdot \mathrm{BU})$, where A_0 is ^{134}I activity at beginning of cycle, $k = 0.7 \times 10^{-4}$, or 0.25×10^{-4} , or 0 and burnup, BU, is in MWd/t.

developed. Some utilities have since introduced multiple trigger points to address a wide range of failure release characteristics they have experienced. When the activity release rate exceeds a certain action level, an increasingly high level of plant management is involved to access and manage the fuel failure issue. When the activity release reaches a situation (Level 3 or 5) in which the overall plant operation may be significantly impacted or the plant operational technical specifications may be exceeded, planning and actual plant shutdown to inspect and retrieve the failed fuel may be executed.

In comparison with PWRs, some BWR plants implement flux tilting and power suppression when leaking FRs are present. Operational technical specification limits and necessary actions for BWRs in several countries are given in Ref. [72]. For instance, if the BWRs at the Olkiluoto Nuclear Power Plant in Finland reach the level of $^{131}I > 2.2 \times 10^6$ Bq/kg for a cumulative time of 800 hours or $^{131}I > 4.4 \times 10^7$ Bq/kg, the plant will shut down.

3.9.1.3. WWERs

The safety of nuclear fuel during its operation in nuclear power plants in the Russian Federation is ensured by compliance with the requirements established by the regulators. According to these requirements, operational limits and the limits of safe operation of WWER-1000 cores in the presence of damaged FRs have been calculated and are currently used based on the specific activity of the primary coolant. These limits are based on the sum of iodine isotope activities (Table 12).

Upon reaching the specified operating limits for the specific activity of the primary coolant, measures (the reduction and stabilization of unit power) should be taken to reduce the coolant activity below the operating limit. If measures to reduce coolant activity below the operational limit have failed or the limit of safe operation with leaking FRs has been reached, the unit shall be immediately shut down. A reactor restart is only possible when the integrity of all FAs has been confirmed, defected FAs have been discharged and the cause of fuel damage has been determined.

3.9.1.4. CANDU reactors

For Canadian CANDU reactors, the licence limit value for the coolant activity concentration of ^{131}I has continually decreased over the years and is quite restrictive. Moreover, once a limit has been reached, the operator has 24 h for completion of the defect location, defuelling and confirmation activities prior to a required shutdown of the unit.

3.9.2. Foreign material exclusion practices

3.9.2.1. PWRs

In recent years, several foreign material exclusion actions have been reinforced to minimize the risk of fuel failures in reactors. Multiple televisual inspections are performed during outages, with high definition cameras, allowing the identification of debris at different locations:

— On (and sometimes below) the lower core plate;
— In the vessel bottom head (once the lower internals have been removed);
— Under the bottom nozzles of the FAs;
— In the FA bundles.

TABLE 12. OPERATIONAL LIMITS AND THE LIMITS OF SAFE OPERATION OF WWER-1000 CORES

Parameter	Operational limit	Limit of safe operation
Specific coolant activity based on the sum of iodine isotopes I-131–I-135 Bq/g	37 000 Bq (1×10^{-3} Ci)	185 000 Bq (5×10^{-3} Ci)

Note: The conversion factor of 1 Ci = 37 GBq.

Associated with these inspection facilities, different retrieving tools have been developed to eliminate debris as soon as it is detected (see Fig. 34).

A foreign material exclusion programme is applied for various activities within the nuclear industry including: basic design of equipment, fuel manufacturing, operating and maintenance. Nevertheless, it takes time to eradicate legacy debris present in the primary circuits that are still responsible for fuel failures in PWRs.

3.9.2.2. BWRs

Foreign material exclusion practices in BWRs are similar to those in PWRs. However, the possibilities of introducing debris are higher in BWRs than in PWRs, since BWRs operate with a direct cycle.

3.9.2.3. WWERs

The following measures to reduce the probability of damage to the FR claddings due to the presence of foreign objects in the primary circuit coolant have been developed and used at WWERs:

- — Unloading of FAs for inspection and cleaning of reactor internals and pressure vessel;
- — Cleaning the spent fuel pool water by installing a filter;
- — Examining the boundary fitting bodies of the primary circuit;
- — Inspecting the defected FAs and anti-debris filters;
- — Analysing the location of defected FAs to identify patterns of their occurrence;
- — Excluding foreign materials from the primary coolant circuit;
- — Avoiding winding materials becoming entangled with loose equipment;
- — Checking consumables being brought in and out of the control zones;
- — Using visual aids to confirm the absence of loose parts.

FIG. 34. Example of a gripper tool (left) used on PWRs to retrieve debris from an FA (right). Figure courtesy of Électricité de France.

3.9.2.4. CANDU reactors

Foreign materials or debris within the primary circuit of CANDU reactors can be circulated through the core by the coolant. In new (and refurbished) CANDU reactors, fuel defects can occur because of debris being trapped within fuel bundles. Depending on the size and location of the debris, the coolant velocity can cause the element to vibrate, interact with foreign material and damage the fuel sheath. The main source of debris found within the primary circuit comes from the construction stage of new reactors. Other sources of debris can be made available to the primary circuit after start-up when reactors are shut down for routine maintenance and inspection of primary circuit components.

To reduce the risk of fretting defects during the initial start-up in CANDU reactors, foreign material exclusion is carried out by introducing strainers. The strainers are normally installed in specific channels during commissioning to remove debris.

3.9.3. Water chemistry

3.9.3.1. PWRs

State of the art water chemistry programmes help ensure the continued integrity of reactor coolant system construction and fuel cladding materials, ensure satisfactory core performance and support the industry trend toward reduced radiation fields. Controlling the chemistry variables (dissolved oxygen, lithium, pH) in the primary coolant concentration is critical to control phenomena such as axial offset anomaly or uncontrolled crud and corrosion in the fuel that can lead to undesired fuel performance [73].

3.9.3.2. BWRs

BWR water chemistry has evolved from essentially pure, relatively oxidizing water to the current programmes, which include: hydrogen injection for intergranular stress corrosion cracking mitigation, depleted zinc oxide addition to minimize shutdown dose rates and noble metal chemical application (NobleChem) or on-line NobleChem [74] to mitigate intergranular stress corrosion cracking and to avoid main steam operating dose rates caused by moderate hydrogen injection in some plants. The objectives of improving BWR water chemistry control are to extend the operating life of the reactor piping, vessel and internals and balance-of-plant materials and turbines, while controlling costs to retain economic viability. The injection of hydrogen and application of noble metals cause water chemistry on the structural material surface in the reactor environment to change from oxidizing to reducing.

On-Line NobleChem is an improvement on conventional noble metal chemical application that can be implemented while a BWR plant is operating and generating power. It uses platinum and has no impact in fuel performance or fuel reliability.

3.9.3.3. WWER-1000s

Water chemistry control and specifications for primary coolant water chemistry in WWER-1000s were described in detail in the IAEA's 2010 fuel failure review (table 8.1, in Ref. [3]). Regarding updated specifications introduced into practice by Rosenergoatom in April 2013 [75], changes of water chemistry parameters in comparison with those given in Ref. [3] are not significant.

3.9.3.4. CANDU reactors

There have been no reported cases of waterside corrosion leading to fuel performance problems in CANDU power reactors. The oxide thickness on the external cladding surfaces of power reactor fuel bundles is typically less than a few micrometres. Deposits previously observed on sheathing from discharged CANDU fuel in Canadian reactors have not resulted in fuel failure [76]. Moreover, as part of the refurbishment and return to service of several units at the Bruce Nuclear Generating Station, hot conditioning was used to prepare the primary heat transport system for operation. In this process, a magnetite (iron oxide) is introduced on the primary heat transport system carbon steel surfaces [64] before the reactor produces power, where iron is also expected to precipitate on other surfaces such as the fuel. Five bundles discharged from Unit 4 in 2010 and Unit 1 in 2012 were examined by post-irradiation examination in hot cells, where it was verified that the deposits had had no impact on fuel performance [77].

Individual CANDU plants maintain their own chemistry practices and operational guidelines. Guidelines for these values in general for the primary heat transport system [78] are shown in Table 13. Water is kept alkaline to control and minimize the corrosion of system components.

3.10. GUIDELINES FOR THE PREVENTION OF SEVERE DEGRADATION

3.10.1. Secondary hydriding degradation in PWRs

In PWRs, secondary hydriding degradation is that which occurs during the cycle after the primary defect occurrence (see Figs 35 and 36), for any burnup ranges. Secondary damage generally leads to specific fission product release (increase of ^{134}I) associated with fuel material release into the coolant.

In French PWRs, fuel failure degradation prevention is being addressed using radiochemical operating technical specifications that define progressive limits for the following parameters:

— The sum of gas isotopic activities;
— Dose equivalent iodine;
— Iodine-134 activity.

For each parameter, three progressive thresholds, based mainly on operating feedback, give indications on how fuel failures evolve, and define suitable actions to mitigate fuel degradation.

The first is a reinforced surveillance threshold. Reaching this limit leads to an increase in the frequency of radiochemical measurements (gamma spectrometry) to better understand the behaviour of fuel failures. Load following is stopped to stabilize the power and analyse whether the activity increase is due to a defected fuel degradation or to plant power changes.

The second and third thresholds in French radiochemical operational technical specifications are shutdown thresholds (referring to shutdown within 48 hours and 8 hours, respectively). These limits indicate that a significant fuel degradation event is happening in the primary circuit, which requires a quick plant shutdown to avoid any non-compliance with safety analysis (for example, a class 4 steam generator tube rupture accident).

TABLE 13. TARGET CHEMISTRY PARAMETERS IN THE PRIMARY HEAT TRANSPORT SYSTEM OF CANDU REACTORS

Parameter	Typical specification range
pH_a	10.1–10.4
[Li+]	0.35–0.55 mg/kg (ppm)
$[D_2]$	3–10 mL/kg
Conductivity	0.86–1.4 mS/m (dependent upon LiOH concentration)
Dissolved O_2	<0.01 mg/kg
$[Cl^-]$, $[SO_4^{2-}]$	<0.05 mg/kg
Isotopic	>98.65% D_2O
Fission products	ALARA (monitoring ^{131}I indicative of fuel failure)

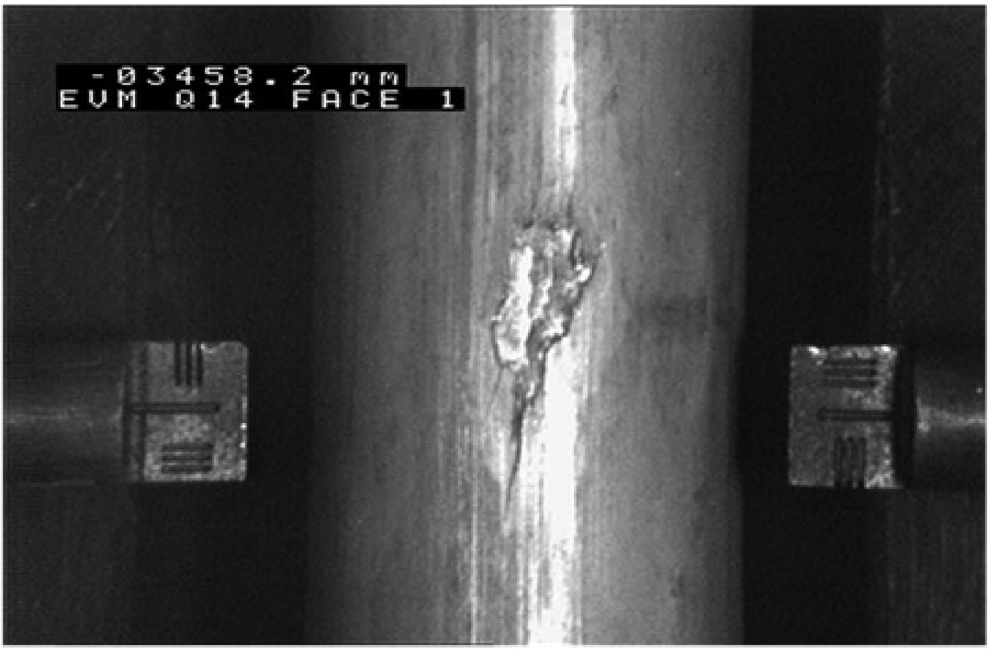


FIG. 35. Example of sunburst due to secondary hydriding. Figure courtesy of AREVA NP.

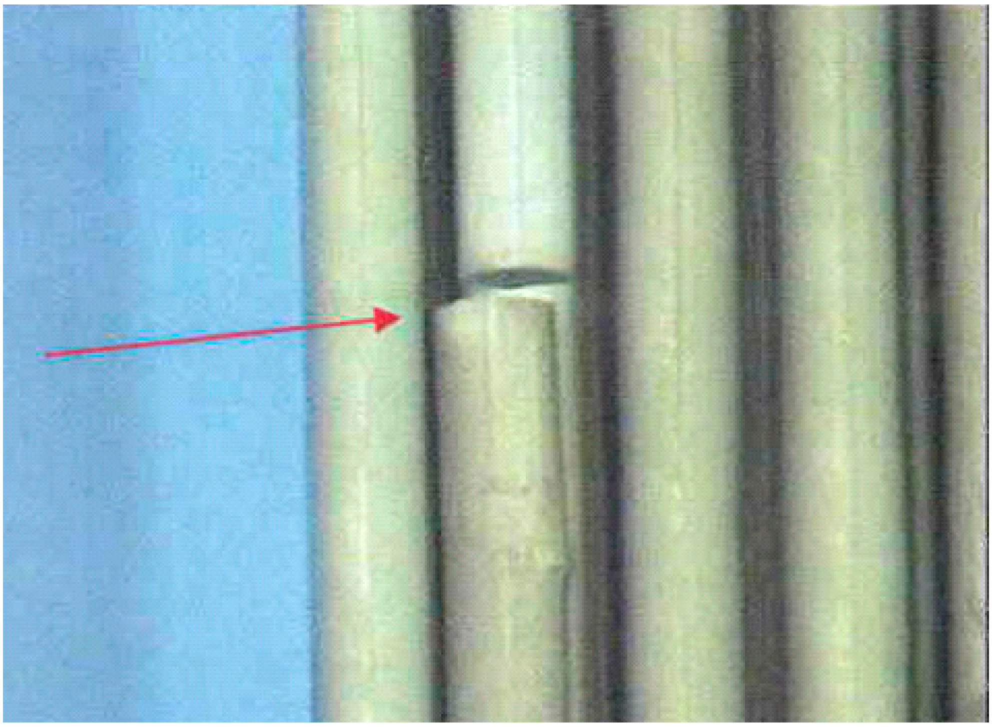


FIG. 36. Example of full circumferential rupture due to secondary hydriding. Figure courtesy of Électricité de France.

Plant shutdown within 48 hours gives enough time to prepare specific arrangements for an unplanned outage (purification and/or degassing the primary circuit). Shutdown within 8 hours aims at minimizing the consequences of high activities due to gas release into the environment as quickly as possible.

3.10.2. Secondary hydriding degradation in BWRs

Secondary fuel failures still occur when the leaking fuel stays in the core without any immediate power suppression. BWRs, in contrast to PWRs and WWERs, have a capability to identify the leaking bundle during operation and reduce the power of the bundle to prevent further degradation of the leaker. Reference [3] provides information on fuel management during operation when a secondary fuel failure is detected by radiochemistry indicators. Once the failure is located from the radiochemistry data, the power in the cell is suppressed by inserting the control rod to avoid degradation of the failed FR. The success of this technique is very high in avoiding cracks or rupture of the FRs during operation until the end of cycle. BWRs use power suppression quite often to avoid secondary hydriding and further damage to the leaker.

3.10.3. WWERs

The safety of nuclear fuel operation is ensured by the fact that a reactor should be shut down and FAs with leaking FRs unloaded from the core if the operational limit for the specific activity of reference fission products in the primary coolant is reached or exceeded (Fig. 37).

Currently, nuclear power plants with WWER-1000s apply a practice according to which even leaking FAs with a low degree of defects are not loaded into the reactor core for additional burning. This provides a general decrease in activity levels in WWERs and reduces the risk of significant secondary degradation of leaking FRs.

If FR leakage is detected during WWER unit operation, there are general guidelines to minimize the risk of secondary fuel degradation:

- It is necessary to minimize changes in the reactor power.
- If manoeuvres are necessary, power changes must be carried out slowly if possible.

Leaking FRs allow direct contact between UO_2 and coolant and the resulted secondary defects could cause significant FR failure. For their identification, the cycling pressure method is recommended (see Section 3.6.4), which allows the estimation of the size of the defect, thus defects that are permeable to fission gas and those that

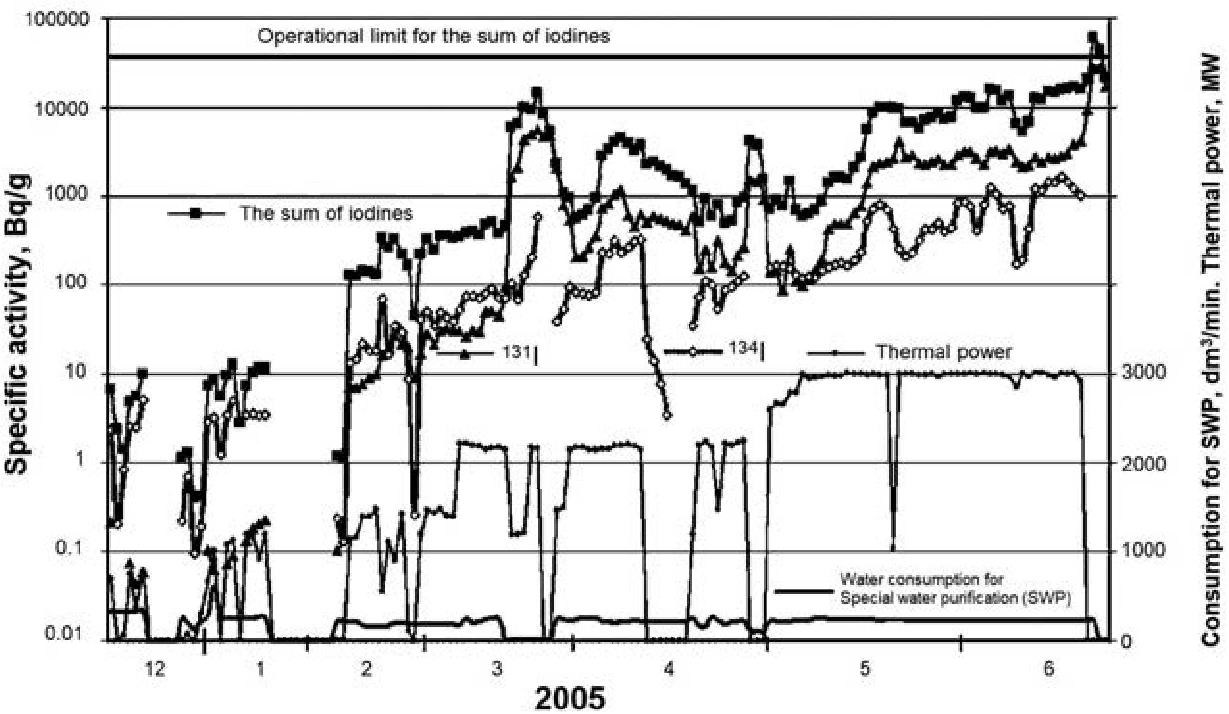


FIG. 37. The specific activity in the primary coolant during operation of Unit 3 of the Kalinin nuclear power plant with a secondary defect in the leaking FR during the first six months of 2005. Figure courtesy of the Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences [79].

are not can be distinguished. Defected FAs (and FRs) require 'pencil case' storage, as well as special technology for transportation and storage.

3.10.4. CANDU reactors

The severe degradation of CANDU fuel is addressed through design, operation practices and feedback from the in-bay, and post-irradiation examination of defected and unfailed elements as detailed in Sections 3.10.4.1. and 3.10.4.2.

3.10.4.1. Design

In the design stage, the common practice is to develop the design requirements, followed by assessments and analyses to ensure that they are met based on the fuel acceptance criteria defined for all fuel damage mechanisms. The intent here is to ensure that there is sufficient margin from the damage limit. The fuel acceptance criteria are built with a focus of addressing the following margins: operating, design and safety.

3.10.4.2. Operations

Normally, defected elements or bundles with gross defects releasing fission products are identified by activity concentrations determined through coolant chemistry sampling (or coolant grab samples). Operational and alarm limits are based on selected noble gas isotopes. When the alarm limits are reached, the reactor power is reduced slowly to minimize the risk of transient iodine release and to allow time for iodine inventories to decrease by decay.

As previously mentioned (see Section 3.6.5), some CANDU utilities use two independently operated failed fuel systems: the delayed neutron monitoring system and the gaseous fission product monitoring system (referred to as a feeder scanning system in Section 3.6.5). The delayed neutron system is used when failed fuel has been detected in the reactor, either by online coolant activity monitoring or by grab sample measurements. This system is used when the reactor is operating at high power to determine which fuel channel contains the failed fuel. Defective fuel must later be confirmed by inspection. The gaseous fission product monitor is a computer controlled, high resolution gamma ray spectrometer, which operates continuously measuring the gamma ray activity of gaseous fission products (i.e. ^{133}Xe, ^{135}Xe, ^{88}Kr and ^{131}I).

Defective fuel elements with small defects operating in low power positions may be below the threshold sensitivity of the defective fuel monitors. Also, some small defects may become plugged because of oxidation of the Zircaloy. These types of fuel failures are considered inconsequential to the reactor operation.

In some fuel defects, deterioration of the defect could result in fission product gases, UO_2 and Zircaloy being released to the primary coolant. The UO_2 and radioactive fission products that are lost from the defective element are not a hazard either to the public or to the station operators. However, they do contribute to radiation levels in the plant and to the maintenance costs of cleanup systems. Therefore, it is desirable to avoid gross deterioration of defective fuel elements. When a defect is detected and located by the monitoring system, the fuel bundle can be discharged from that channel using the on-power fuelling system. It may not be possible to remove suspected defects immediately owing to uncertainty in the monitor signals, fuelling machine availability, monitor availability and local reactor flux conditions. There is a relationship between the rate of deterioration of defective fuel and the surface heat flux (or element power).

Generally, the higher the power of the defective bundle, the sooner the bundle should be discharged. The maximum number of defective fuel elements in the core is indirectly controlled by shutdown limits that are based on coolant activity levels. As a guideline, the following steps are taken by certain CANDU utilities for the removal of defective bundles:

— A delayed neutron signal is collected before and after the channel is fuelled.
— A gamma spectrum analysis of both the primary heat transport system loops is completed before and after the fuel channel is fuelled. This enables the detection of short lived isotopes.
— The defuelling of bundles is performed in pairs. The local area alarm gamma monitor identifies a defective bundle based on airborne gamma fields.
— The suspected pair is confirmed by wet sipping and set aside for in-bay examination.

3.10.4.3. Feedback from in-pool inspection and post-irradiation examination

CANDU utilities have an established fuel programme with the intent of systematically processing and integrating information and conducting regular cross-discipline reviews to ensure the safe operation of nuclear reactors. Some CANDU utilities perform in-pool inspection on defected bundles or elements and unfailed elements. In cases where the root cause of fuel failure cannot be identified during in-pool inspection, the defected elements are removed from the bundle and sent for post-irradiation examination. In the hot cells both non-destructive examination and destructive examination are performed to determine the root cause of the defects. The non-destructive examination consists of element profilometry, gamma scanning and a destructive examination with element puncture and fission gas analysis. The destructive examination further consists of hydrogen/deuterium analysis, scanning electron microscope examination and energy dispersive spectrometry, metallography and ceramography examinations. Unfailed elements are also examined to ensure consistency with the design and operating limits. Corrective actions are initiated where the root cause of defected elements is determined (whether it is related to manufacturing, operational or handling aspects) to address the deficiencies or inconsistencies identified.

4. CONCLUSIONS

The current fuel failure study covers 97% of water cooled reactors (PWRs, BWRs, WWERs and CANDUs and other PHWRs) operated worldwide from 2006 to 2015. Fuel performance data (i.e. including the annual number of failed FAs and FRs, the root cause of fuel failure and FA structural damage without subsequent FR failure) and fuel operating environment data (burnup, core residence time and cycle length of operation for failed assemblies and FRs) were collected as responses from country members of the IAEA TWG-FPT to the IAEA questionnaire. Using fuel performance data collected by the IAEA in earlier studies (1987–1994 fuel failure data in Ref. [2] and 1994–2006 fuel failure data in Ref. [3]) FR leak rates and analysis of fuel failure causes in the present study were extended to a time frame of three decades (1987–2015). Data on the failure rate of FAs are available for the time span 1994–2015.

4.1. FUEL FAILURE AND LEAK RATES

The FR failure rates in LWRs have significantly reduced over time (but not smoothly) from 1987 to 2015. Figures 1–19 in Section 2 show a stable decrease in the number of FR failures in PWRs and BWRs (Figs 4 and 9), a relative increase in the failure rate in WWERs in 1999–2002 (see Fig. 15) followed by a decrease, as well as a sharp decrease in 1995 for Canadian CANDUs and a continued low level up to 2015 (see Fig. 19). For 9 CANDUs operated outside Canada and 18 PHWRs operated in India, there was a significant reduction in the fuel failure rates as observed in 2006–2015 in comparison with those seen in 1994–2006.

FA failure rates (i.e. expressed as the number of FAs with defected FRs identified in 1000 discharged assemblies) during a given year as averaged for 1994–2006 and 2006–2015 showed a decrease from 13.8 to 7.1 in PWRs, from 4.4 to 2.4 in BWRs, from 32.1 to 21.5 in WWER-1000s, from 0.1 to 0.1 (stable) in Canadian CANDUs, from 0.35 to 0.11 for all CANDUs (including those in Canada), and from 1.5 to 0.79 in Indian PHWRs. The reduction in the fuel failure rate is a result of the implementation of more challenging fuel operating conditions by the reactor operator and fuel improvements and modifications made by the fuel designer and vendor, as well as random occurrences from possible deviations in water chemistry, debris present in the primary coolant and so on. The latter factor can explain the observed trends in the rate of fuel failure reduction over the years.

4.2. NUMBER OF UNITS OPERATED WITH DEFECT FREE FUEL

The world average trend in the number of units that operated defect free is in a positive direction. For PWRs, BWRs and WWERs, these data are presented in Figs 6, 11 and 16. For CANDUs that were operated outside Canada as well as 18 other PHWRs in India these data are presented in Section 2.2.4.3.

4.3. FUEL LEAK MECHANISMS AND CAUSES

No new fuel failure mechanisms appeared during the period of 2006–2015. Some former failure mechanisms had not been reported (baffle jetting in PWRs, hydriding in all reactor types and clad collapse in PWRs) by nuclear operators. Grid-to-rod fretting was the most frequently observed FR leak mechanism in PWRs and debris fretting the most common in BWRs. Debris fretting in PWRs and PCI/SCC in BWRs were also observed but less frequently. Debris fretting was the major FR failure mechanism in WWERs. Debris fretting and occasional fabrication defects (incomplete end welds) have occurred in CANDUs with important improvements observed outside of Canada over the last decade. Some debris failures resulted from refurbishment activities in a CANDU reactor in Canada. Another failure mechanism observed in a domestic CANDU was pellet chipping with high pellet loading forces and low clearance tolerances. In PHWR reactors, fuel failures due to debris fretting, fabrication defects, handling damage and isolated occurrences of PCI/SCC were reported.

New debris sources, some of them from fuel, including its manufacturing, were identified. New root causes have been identified in PWRs: grid spring cracking, seal weld defects in upper end plugs, cladding shavings and cracking of baffle bolts. As previously mentioned, pellet chipping occurred during CANDU fuel element fabrication.

The share of fuel failures with unknown or undetermined fuel failure mechanism or cause is still a significant category.

4.4. MASSIVE AND SIGNIFICANT FUEL FAILURES IN SINGLE UNITS DURING ONE CYCLE

Seventeen occurrences of massive fuel failures were observed during 1994–2006 (10 in PWRs, 2 in BWRs, 4 in WWER-1000s and 1 in a CANDU) and 5 occurrences during 2006–2015 (1 in a PWR, 3 in WWER-1000s and 1 in a CANDU). No massive fuel failures were reported during 2007–2015 in PWRs and BWRs.

Twelve occurrences of significant fuel failures were observed during 1994–2006 (5 in PWRs, 2 in BWRs, 4 in WWER-1000s and 1 in a WWER-440. Thirty-one occurrences of significant fuel failures were observed during 2006–2015 (13 in PWRs, 1 in a BWR, 12 in WWER-1000s, 1 in a WWER-440, 3 in CANDUs and 1 in a PHWR of the KWU type). The main mechanisms of massive and significant fuel failures during 2006–2015 were given in Tables 7 and 8.

4.5. FA STRUCTURAL DAMAGE WITHOUT FR LEAKING IN 2006–2015

The most important and frequently observed issues in PWRs were: (a) spacer grid damage that sometimes resulted in grid-to-rod fretting or debris fretting without FR leak, and (b) FA bow with spacer grid damage and incomplete rod insertion as a final stage.

The most important and frequently observed issues in BWRs were: channel bow or distortion from a combination of bow, bulge and twist leading to control blade interference, control blade cracking and inoperable control blade and incomplete rod insertion.

As mentioned in Section 2.3, rather limited information on WWER, CANDU and PHWR FA structural damage without FR leaking during 2006–2015 was submitted to the IAEA.

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ABBREVIATIONS

BWR	boiling water reactor
CANDU	Canada deuterium–uranium pressurized heavy water reactor
FA	fuel assembly
FR	fuel rod
LWR	light water reactor
MOX	mixed oxide (fuel)
PCI	pellet–cladding interaction
PHWR	pressurized heavy water reactor
PWR	pressurized water reactor
SCC	stress corrosion cracking
TWG-FPT	(IAEA) Technical Working Group on Fuel Performance and Technology
WWER	water cooled, water moderated power reactor

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Key

BP:	Basic Principles
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