IAEA-TECDOC-1345

Fuel failure in water reactors: Causes and mitigation

Proceedings of a Technical Meeting held in Bratislava, Slovakia, 17–21 June 2002

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

March 2003

The originating Section of this publication in the IAEA was:

Nuclear Fuel Cycle and Materials Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

FUEL FAILURE IN WATER REACTORS:
CAUSES AND MITIGATION
IAEA, VIENNA, 2003
IAEA-TECDOC-1345
ISBN 92–0–101703–0
ISSN 1011–4289

Printed by the IAEA in Austria
March 2003

FOREWORD

At the invitation of the Government of Slovakia and in response to a proposal by the IAEA Technical Working Group on Water Reactor Fuel Performance and Technology (TWGFPT), the IAEA convened a Technical Meeting on Fuel Failure in Water Reactors: Causes and Mitigation in Bratislava, Slovakia from 17 to 21 June 2002. The meeting was hosted by the Nuclear Regulatory Authority and VUJE Trnava, Inc. Engineering, Design and Research Organization.

For many years, the IAEA has been closely involved in the analysis of nuclear fuel performance, basic fuel failure causes and failure mechanisms in water power reactors. The IAEA conducted a Technical Committee Meeting on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management in 1992 in Dimitrovgrad, Russian Federation (IAEA-TECDOC-709), conducted a study on fuel failures and published Technical Reports Series No. 388, Review on Fuel Failures in Water Cooled Reactors, in 1998 and conducted a survey on fuel failures in water cooled power reactors from 1995–1998 (CANDUs, BWRs, PWRs and WWERs) in 2000. The objective of this meeting was to analyse and discuss utility and fuel vendor experience in fuel failure cause identification and on implemented remedies to reduce the number of fuel failures and/or to mitigate fuel failure impact on NPP operation.

Fifty-three specialists in fuel design, fabrication and operation from 18 countries took part in the meeting in order to gather and discuss existing knowledge of the subject and identify the need for further efforts. Twenty-six papers were presented in five sessions covering experience with recent fuel failure events and their mitigation, and the current knowledge of fuel failure mechanisms in light water cooled power reactors. The most frequently observed events included grid-to-rod fretting failures in PWRs, severe secondary failures, especially the long axial splits and circumferential fractures observed in BWRs, axial offset anomalies in PWRs and some others. During recent years fuel performance in water-cooled power reactors has improved significantly since most early problems have been solved. Fuel failure rates are now at a low level ($<10^{-5}$ or <10 ppm) and continue to be reduced. However, fuel failure has remained a very important issue for NPP operation and economics.

The IAEA wishes to thank all the participants for their contribution to the meeting and to this publication, especially V. Petenyi of VUJE Trnava, Inc. Engineering, Design and Research Organization who co-ordinated the work of the local organizational committee. The IAEA officer responsible for this publication was V. Onoufriev of the Division of Nuclear Fuel Cycle and Waste Technology.

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SUMMARY

INTRODUCTION

The objective of this technical meeting (TM) was to review the present knowledge of the causes and mechanisms of fuel failure in water reactors during normal operational conditions. Emphasis has been given to analysis of failure causes and their mitigation by means of design as well as plant and core operation including strategies for operation with failed fuel. Some information on detection techniques (on-line monitoring and diagnostics, flux tilting, sipping techniques, etc) has also been presented.

This TM presented also the progress on the above-mentioned subjects since the last meeting held in 1992 (Dimitrovgrad, Russian Federation).

The topics covered in the papers were as follows:

- Experience feedback on fuel reliability (8 papers)
- Strategies to avoid or mitigate fuel failures (4 papers)
- Experimental studies on fuel failures and degradation mechanisms (4 papers)
- Modelling of fuel failure mechanisms (3 papers)
- Detection and monitoring during operation or outage (4 papers)
- Modelling and assessment of fuel failures (3 papers)

SESSION I: GENERAL OVERVIEW OF PRESENT STATUS OF EXPERIENCE

Chairmen: P. Darilek (VUJE Trnava Inc., Slovakia and KAERI, Republic of Korea) and J.M. Alonso Pacheco (ENUSA, Spain)

In this first session of TM a total of six papers were presented. The papers provided a comprehensive summary of fuel performance experience and practices for enhancing fuel reliability in different countries like Spain, Russian Federation, Bulgaria, Hungary and Ukraine, or from fuel vendors like Framatome-ANP. The discussion at the end of the Session was devoted to the identification of present major fuel failures root causes and remedies, and to establish the guidelines for future research efforts that ensure maximum fuel reliability.

The following **fuel failures root causes** were identified:

- Debris
- Pellet-Cladding Interaction (PCI, particularly in BWRs)
- Manufacturing defects
- Flow Induced Vibrations, that includes different mechanisms and conditions like: baffle jetting, enhanced cross flow at core periphery, enhanced cross flow at core bottom (vessel flow anomaly), mixing vane distribution and orientation in the grids

- Fuel Handling Incidents
- Poor resistance to secondary degradation (particularly in BWRs).

The following **remedies** were applied:

- Debris filter devises that retain and catch foreign particles preventing fuel rod damage. This type of remedy, in conjunction with NPP's efforts for minimising debris sources, has proved efficiency since debris is not major fuel failure mechanism anymore
- Improving mechanical performance against flow induced vibrations. That it is being accomplished by addressing two main parameters:
    - Reduced vibration response (i.e., adequate overall fuel assembly mechanical design, good grid design…)
    - More resistance to fretting wear (contacting materials, grid design…)
- Barrier cladding has proved to be effective protection against PCI failures in BWR. Iron addition to the barrier is the general accepted remedy for preventing secondary degradation of failed barrier fuel rods.
- To improve manufacturing and inspection techniques, besides the implementation of more robust and efficient quality systems as a way to eliminate manufacture related defects.

- The following further **research efforts** were recommended:

    - *Fuel surveillance. Not only failed fuel needs to be investigated but characterisation of key properties of representative fuel is required (corrosion, growth, fretting marks…).*
    - *Minimise number of failed fuel cases where no inspection is accomplished.*
    - *In case of PWR that needs to operate under load follow condition, PCI resistance needs to be satisfactorily addressed.*
    - *Improve understanding of Flow Induced Vibration mechanism and effects including:*
        - *Extensive inspection of potentially affected fuel;*
        - *Better analysis of reactor specifics and circumstances;*
        - *Representative fuel assembly testing loops and procedures;*
        - *More powerful analytical tools and models.*
    - *Exchanging fuel failure experience more openly.*

SESSION II: MITIGATION OF FAILURES BY DESIGN AND MANUFACTURING

Chairmen: W. Klinger (Framatome-ANP, Germany) and A. Bykov (NAEC, Ehergoatom, Ukraine)

The session showed that fuel vendors face a task to provide a fuel design that would assist avoiding main actually observed fuel failure mechanisms. Information on fuel behaviour from operation, PIE, and modelling is the basis for taking a decision on modification of fuel rod (FR) or fuel assembly (FA) design. Papers presented demonstrated that major fuel failure causes are very similar for different reactor types. Of course, failure frequency for each cause depends on specific reactor type, because of different fuel rod and assembly design, water chemistry, coolant flow, thermal and mechanical loads, etc. Major modifications that have been and are being done by fuel designers/producers are presented below.

Debris catchers/filters: The purpose of a filter is to catch foreign objects to avoid cladding damage, but with only minor impact on pressure drop. Since the early 90ies, fuel vendors have developed several debris filter designs. Therefore, most of current fuel deliveries include a debris filter. With regard to Swedish presentation on Triple Wave debris filter, the discussion showed that cladding damage by metallic debris is the most severe failure cause in BWRs worldwide, making 40–50% of all failures. Debris filters were first used in PWR FAs and demonstrated their efficiency drastically reducing the number of debris fretting related failures. It is expected that advanced filters in BWRs will be very effective also. Debris filters in WWER FAs are now under testing in commercial reactors.

Stiffening FA skeleton: To achieve an improved stiffness, the FA design should have an optimized Guide Tube (GT) wall thickness ("as thick as possible"), it should have a thick-walled dashpot, and should dispose of a stiff SG/GT (SG-Spacer Grid) connection (i.e. direct welding or comparable technique). Such modified FA designs have been implemented during the last years. In addition, new low-growth materials with reduced susceptibility to stress relaxation have been introduced.

Fretting: Factors influencing rod/spring vibrations were understood and, to significant extent, eliminated. Analysis of grid-to-rod fretting in Angra-I PWR in Brazil showed that deficiency in FA design resulted in significant number of fuel rod fretting failures. During discussion of the root failure cause, participants agreed upon the fuel vendor's comment that the transfer of so-called "proven for plant type A fuel design" to plant of type B was a fuel failure prerequisite in Angra-I plant. Even minor changes in fuel design or operating conditions may result in lowering fuel failure threshold, i.e. worsening fuel reliability. This factor has to be taken into account.

PCI: PCI was the major failure mechanism in BWR fuel in the past, causing most of the fuel failures until the mid 90ies. Introduction of cladding materials with increased PCI resistance resulted in a drastic reduction of such failures.

Accelerated Corrosion: Fuel failures due to accelerated corrosion have been observed in few PWR plants, mainly in combination with unfavorable coolant chemistry. To avoid such fuel failures, advanced cladding types have and are been implemented with improved corrosion resistance.

Dismountable Fas: Repair of failed fuel before reinsertion for further use reduces the activity release into the coolant and is even required by authorities in some countries. Fuel vendors developed fuel assemblies with dismountable top or/and bottom end pieces. Repair of such fuel assemblies can be done by replacement of the failed rod by a dummy rod or a matching uranium rod. Today, a large experience exists with handling, repair and reuse of failed assemblies.

Mitigation of secondary failures: Formation of secondary defects results in increased activity release into the coolant and has even caused early shutdown of some plants. Measures to prevent the growth of primary defects and formation of secondary defects were discussed. During the last years fuel vendors have introduced advanced cladding types with improved resistance against secondary degradation of failure.

Reactors of different design: The operating experience in reactors of different generations showed significant differences in fuel performance even for the same fuel type. For example, reactor WWER-440 has several modifications (V-179, V-213, V-230, V-270) with different operation features and, respectively, with different fuel failure rates.

Mixed cores: Because of ongoing FA design modification, many plants are operating with FAs from one vendor, but of slightly different design. Moreover, FAs supplied by different vendors may operate in one core, so-called mixed core. This practices was described in Swedish paper for Ringhals NPP where interaction between different type FAs had an impact on fuel performance. The effect of mixed cores have to be taken into account in evaluation and modelling of performance of lead test FAs (they operate in a core surrounded by FAs of "old" design).

Recommendations for future work: From the utility standpoint of view, the operational reliability of FAs is one of the most important aspects. Regarding future efforts to improve the fuel reliability, the participants agreed on the following:

- *Grid-to-rod fretting due to fluid induced vibration is still the major PWR fuel failure cause that requires further R&D effort, both in FA design and structural material improvement and also in modeling.*
- *Possible problems related to mixed cores should be taken into account in evaluation and modelling. In addition the performance should be ensured by lead test FAs.*
- *The possible negative effects of fuel failures in NPPs (e.g. losses in case of early plant shut-down) justify further investments into the improvement of rod/FA design, fuel and structural materials.*

SESSION III: EXPERIMENTAL STUDIES OF FUEL FAILURE AND DEGRADATION MECHANISMS

Chairmen: Yong-Soo Kim (KAERI, Republic of Korea) and A.V. Smirnov (SSC RF RIAR, Russian Federation)

The failures of nuclear fuel rods in nuclear power plants were reported and respective failure causes have been investigated by using Post-Irradiation Examination (PIE) techniques (Republic of Korea, Japan, Russian Federation, Switzerland-United States of America) and modelling (Russian Federation). The consequences of such defects are fatal. The coolant can let into the fuel rod, the coolant flashes into steam, and then complicated processes such as steam oxidation of UO_2, oxidation and hydriding of the cladding inner surface, restructuring of UO_2. Fuel failures led to increasing activity levels in the reactor coolant system and the reactor prematurely shut down.

According to the published data, currently the causes of PWR fuel rod damage are as follows:

- Fretting of fuel rod claddings under the spacer grids (grid-rod fretting) — 40–45 % of cases;
- Interaction between claddings and debris in the coolant flow (debris-fretting) — 40–45 of cases;

- Causes conditioned by violation of the fuel production technology (primary hydriding, welding defects, primary cladding tube defects) — less than 5%;
- Undetermined causes and — the rest.

Similar proportions take place in the WWER reactors. The difference consists that percent of fuel rod failure due to grid-rod fretting is significantly lower. The interaction between fuel column and cladding (PCI) are not the cause of WWER fuel rod failure under the design-basis conditions. The failure thresholds of high burnup WWER and PWR fuel rods by PCI-mechanism are situated above 400 W/cm.

BWR fuel rods have increased PCI resistance and provide large flexibility in reactor operation by allowing faster power ascension rates. Nevertheless on same cases fuel rods failed after 3 cycles of operation and an approximate burnup of 26 MWd/kg U, after a power transient following a control blade manoeuvre.

In that way two main causes are responsible for fuel failure, namely:

- Debris-fretting because of primary coolant contamination;
- And grid-to-rod fretting because of FA vibration.

Elimination of first cause is water purification on primary coolant and development of debris filter. But the experience of primary coolant purification shows that the amount of fuel failure cases after purification can be increased. It is necessary to develop the purification criteria and specifications. To remove the second cause it is necessary to develop the FA design that will resist to vibrations.

In any case there are the tasks of identification of failed fuel rod, finding the cause and repair of FA. These tasks might be solved by using inspection and repairing stands.

TM marks importance of modelling of behaviour of the damaged fuel. Thus, it is necessary to develop the following models:

- *Primary defect formation at residual moisture in the fuel rod as a result of manufacturing*
- *Degradation of cladding properties at high burnups.*

SESSION IV: MITIGATION OF FAILURES/DEGRADATION BY PLANT OPERATION

Chairmen: J.A. Perrotta (IPEN/CNEN-SP, Brazil) and H. Hayashi (NUPEC, Japan)

The title of this session implies two items. The first one relates to the plant operational procedures to mitigate any fuel failure, and the second one relates to plant operational procedures to avoid degradation of leaking fuels and to mitigate fuel pellet wash-out. This session included two papers concerning modelling of BWR cladding mechanical behaviour at power ramps and high burnup and analysis of development of secondary failures in BWR claddings. These papers and following discussion allow to summarize the status in the area.

Mr. Suzuki presented the analysis of the deformation behaviour of BWR fuel segment rod during power ramp test using fuel performance code FEMAXI-6 which incorporated fission gas bubble swelling and pellet-cladding bonding models. Importance of the pellet and cladding properties, especially gas bubble swelling and creep rate of cladding, and also the effect of bonding to generate biaxial stress mode in the cladding was shown. It was estimated that circumferential stress at RTP would be about 300 MPa and it is sufficient for crack penetration after formation of radially oriented crack. It was also shown that FEMAXI-6 code is capable to provide useful information on cladding mechanical loading at high burnup.

Mr. Rudling presented an overview on operation and design strategies to minimize degradation of failed BWR fuel. Primary fuel failure causes were presented and scenarios for secondary failure development were discussed including circumferential cracks (breaks) and long axial cracks. Attention was paid to the tendency of primarily failed BWR rods to degrade depending on fuel design and reactor operation. Recommendations were given for avoiding load follow, fast power increase rates after power decreases, flux tilting and power suppress in order to increase resistance to the degradation of failed rod. Also, a new model "BwrFuel Release" for analysis of fuel failures which is capable to separate activity released from tramp uranium from that released from the defect(s). Model works well for variety of water chemistry regimes presently used in BWRs.

Plant operational procedures to mitigate fuel failure

Plant operation with impact on fuel failure mitigation is mainly related to power ramp and load following. PCI fuel failure is the main mechanism associated to this operational mode. Operational factors affecting PCI fuel failure during power ramps are: burnup accumulated prior to the ramp; maximum rod power during the ramp; power increment beyond the pre-irradiated power level; average power ramp rate; and dwell time at high power. Procedures for power maneuvers have been established by fuel designers and vendors.

PCI failures in PWR have not been any more a problem for many years. Due to the use of chemical reactivity control and less inserted control rods, local power perturbation in fuel rods is minimized. Moreover, some plants are equipped with automatic control system for power distribution, using input data from in-core instrumentation, and they can operate in power ramp and load following with very little restriction. For plants without this automatic control system, more restricted procedures are used, limiting power ramp rates above a threshold value of the reactor power. These values vary from vendor to vendor and can be as low as (for situations with no previous power conditioning) 20 to 40% full power as threshold power, and 3% full power per hour as power ramp rate. Studies have being done and changes in power ramp rate and threshold power have being recommended, improving current plant start-up and load following strategies, and plant economics.

BWRs are more subjected to PCI failures, when compared to PWR, due to the fact that control blades are used for compensation of reactivity changes with burnup increase and for power changes. This induces more distortions in power distribution of adjacent fuel rods. Procedures are needed to impose limitations on the power increase rate during start-up after refueling or after control rod sequence exchange and on the control rod withdrawal speed in the high power region when the fuel is not preconditioned. Procedures are established by vendors and are similar in nature but dependent on the design of the control blade driving mechanism. These procedures have being important to the mitigation of PCI failures in BWRs, although design changes, as the introduction of barrier cladding, has significantly reduced the onset for PCI fuel failure.

Research efforts have been done for both PWR and BWR fuel in order to improve knowledge and data for high burnup fuel performance under power ramp and power cycling. Data for burnup higher than 40 MWD/kgU for BWR and 50 MWD/kgU for PWR have been published recently. Threshold power for the onset of vulnerability to PCI failure can be obtained from these data. The modeling of PCI during power ramp, considering the influence of fission gas swelling on the observed cladding strain, has to be developed (see, for example, IAEA-TECDOC-1179, IAEA-TECDOC-1233) and taken into account for establishing plant operational procedures.

Plant operational procedures to avoid degradation of leaking fuels and to mitigate fuel pellet wash-out

Normally, utilities do not reinsert failed fuel in reactor, although some utilities, e.g. EDF, allow reinsertion of leaking fuel assemblies, subject to a sipping test criterion. Fuel failure can affect plant operation. Limits for coolant or off-gas activity during normal operation must be within technical specification for failed fuels. Continuous operation of failed fuels can lead to a further fuel degradation and induce release of significant amounts of radioactive products to the coolant.

After the instant of fuel rod cladding failure, coolant enters through the breach and begins to oxidize the fuel pellets and the inside surface of the cladding. The production of hydrogen within the fuel rod leads to hydriding of the zircaloy and degradation of the cladding. Oxidation of the UO_2 fuel increases the fuel temperature and diffusivity of fission gases and volatile fission products. Continuous operation may culminate in large breaches of the cladding (transversal break or axial split) and fuel particles escaping to the coolant system. The mechanisms responsible for cladding breach occur at secondary locations away from the primary defect. The extent of physical deterioration can vary depending on the time the primary defect was formed, the type and location of the primary defect, the fuel rod design, and the operating power history. Particularly for axial split formation it can initiate from a heavily localized hydrided region at a certain distance away from the primary defect (debris failure pattern) or propagate from a primary defect (PCI failure pattern).

Due to differences in coolant conditions (pressure, steam fraction) and cladding material (composition, microstructure), the degradation of failed fuel is more severe in BWR rods than PWR rods. There have been several incidents of BWR fuel failures using high purity Zr-liner cladding that have resulted in high off-gas plant contamination. There is a consensus that the high purity Zr-liner contributes heavily to the failed fuel degradation.

By understanding the physical and chemical processes of fuel rod degradation, it was possible to evaluate potential mitigating actions, which could lessen the consequences of a leaking fuel rod (mainly concerned to an axial split formation). Design and plant operational actions were taken. Use of alloyed Zr-liner (with Sn or Fe) to reduce the rate of corrosion and the concomitant hydriding of the cladding during post failure operation is one of the design actions taken for advanced fuels. One plant operational action is to decrease the fuel rod power level and to avoid power changes in order to minimize stresses imposed on embrittled cladding material (PCMI) required to initiate and propagate an axial split.

Concerning the plant operational procedures, there are significant differences between PWRs and BWRs. In PWR the power reduction is done throughout the core, whereas local power suppression through control rod insertion is not possible. On the other hand, BWR operators have the option of inserting control blades, which gives chance for addressing operational guidelines. In general these guidelines consist of: identifying the suspect leaking fuel rod location in the core using power suppression testing; decreasing the local power level to minimize the rate of secondary degradation; limiting the extent and rate of power changes in the suspect location. Based on these general guidelines, fuel vendors have established recommendations keeping some specificity for each fuel and reactor design.

EPRI also performed a comprehensive program to investigate the phenomena associated to fuel degradation, to propose operational guidelines for mitigating the effects of failed rods during a reactor cycle, and to develop a model to help predict the behavior of failed fuel rods. The "Defective Fuel Element Code-T (DEFECT)" code, that is capable of modeling the complex physical and chemical processes involved in failed fuel rod degradation, has been developed. The code includes models for the simulation of thermal, mechanical and chemical processes within an operating LWR fuel rod that has steam into the pellet-cladding gap due to cladding breach. It considers the coupling of fuel thermomechanical behavior models with an axial gas transport model for hydrogen evolution in order to determine the formation of secondary defects and propagation of axial cladding cracks. The models were based on industry-wide knowledge gained from laboratory tests, hot cell PIE, and fuel experience with BWR fuel failures. The code can be used to evaluate operating strategies, which mitigate the degradation of a fuel rod, and the effectiveness of various fuel rod design modifications proposed to overcome the problem.

Recommendations on future work:

- *Further development of models for simulation of PCI/PCMI rod failures at high burnup in order to verify the margin's decrease related to the actual threshold and procedures for power ramping and load follow is needed.*
- *Experimenters and modelers should be encouraged to develop models and codes for simulating failed fuel rods degradation.*

SESSION V: DETECTION AND MONITORING
Chairmen: J. Schunk (NPP Paks, Hungary) and V. Petenyi (VUJE, Slovakia)

Eight papers of this session were devoted to detection and monitoring of fuel failures in reactor cores. Detection and monitoring are essential tools for fuel failure determination during normal or transient operation and outages. The main aim is to measure and determine different parameters (mainly from primary coolant) in order to calculate fuel failure characteristics and other important parameters affecting the safe and reliable operation. Based on these results decision are made to change or reload failed fuels.

The main findings resulted from paper presentations and discussion are as follows:

1. Fuel reliability at Jaslovske Bohunice NPP:

    - long term (15 years) experience on fuel failures are given,
    - the root cause of higher failure rate at V-230 than at V-213 is not found, but many possible reasons are given (dummy assemblies implementation, in-core parameter differences technological/physical differences, etc.),

2. Failed rods diagnostic and primary circuit contamination levels determination thanks to the DIADEME code:

   - correlation between primary activity concentration and fuel failure parameters are found,
   - characterization of failed fuel rods (quantity of transuranium, number of failed rods, defect size, burn-up of leaking rods, UO_2 or MOX discrimination) is given,
   - extrapolation of alpha-activities to the end of cycle to prepare maintenance operations can be done,
   - predictions of primary activity levels due to reload of defective fuel assemblies can be estimated by PROFIP code.

3. Disadvantages of means and methods of fuel failure detection:

   - critical evaluation of present evaluation method was given.

4. Fuel failures at Paks NPP:

   - fuel performance examination by evaluation of activity concentration in primary coolant (during normal and transient operation) was carried out together with adopting spiking model for WWER 440 and using micro/radio analytical examinations,
   - the fuel failure rate evaluation was done by expert system and steady state model.

5. Defected fuel monitoring at Cernavoda NPP:

   - summary of defect investigation at limit 1 by on-line detection and location system are given based on gaseous fission product monitoring system (Xe-133, 135, Kr-88, I-131) and delayed neutron system (I-137, Br-87), with a good correlation between them.

6. Regulation of the fission product activity in the primary coolant and assessment of defective fuel rod characteristics in steady-state WWER-type reactor operation:

   - determination of maximum permissible level of fuel rod failure and fission product activity concentration in primary coolant was given,
   - the reliability of the assessment was increased by using the TIMS code.

7. Summary of technical development on the on-line monitoring and fuel failure evaluation system at Temelin NPP:

   - an easy-to-use on-line gamma-spectrometry system is used for collection and calculations of complex information on fuel performance data (number and type of defects),
   - different codes (PES, PEPA) need some fine-tuning to Temelin fuel.

8. Fuel failure diagnostics in normal operation of NPPs with WWER-type reactors:

   - reliability of failed fuel monitoring and detection systems was considered and found to be appropriate,
   - but failure due to manufacturing reasons reduce the reliability of failure diagnostics.

Conclusions and recommendations for future work:

- *Detailed PIE are required and highly recommended to establish to check irradiated fuel conditions and characterize failures, find root causes in order to avoid future failures and reduce fuel failure rate.*

- *A surveillance programme should be introduced and implemented during complete refueling outages to ensure the requested cleanliness in primary systems.*

- *Different codes should be harmonized in order to be able to compare results gained with them. Further improvements are also necessary to adjust these codes to multiple failure description and characterization.*

- *Results from modelling and real statistical data should be clearly differentiated.*

- *Fuel design and manufacturing process upgrading is necessary to minimize failures from manufacturing reasons.*

PANEL SESSION
Chairmen: D. Parrat (CEN Cadarache, France) and P. Rudling (ANT, Sweden)

1. CURRENT SITUATION OF THE UTILITIES

In many countries, the production and distribution of electric power has experienced a dual evolution over the last decade. Firstly, the construction and commissioning of new nuclear power plants decrease dramatically. Secondly, the liberalisation and deregulation of the electrical market has forced the nuclear utilities to become more competitive.

In order to improve effectiveness in the field of the fuel cycle economy, operators and fuel vendors are considering a variety of means to enhance plant performance and to reduce costs by introducing measures such as:

- fuel burnup extension,
- power uprates,
- more aggressive loading schemes,
- decrease trends for system materials Intergranular Stress-Corrosion Cracking (IGSCC),
- decrease in activity buildup,
- longer fuel cycles,
- plant life extension.

This trend has taken place in the last several years, and is likely to continue in the long term.

As a consequence of this development, fuel isotopic composition of the irradiated fuel has strongly evolved, with a presence of fission gases, alpha and long half-life fission products specific activities more important at the end of life. On the other hand, larger cladding stresses result from fuel-clad interaction, external corrosion, or grid-rod interaction. This changed situation could impact the fuel reliability and, if necessary, the associated plant operational surveillance or the fuel management.

In this context, fuel failures become an important point for the nuclear utilities, which play an important role as a driving force for the R&D efforts related to fuel performance. They are now facing numerous inputs, from a technical or an organizational point of view, and have to deliver two main outputs:

- sustained safety margins,
- a preserved place on a deregulated market, involving a stronger competition.

This situation is shown on the diagram, see Fig. 1. One can identify following parameters including inputs and outputs:

1.1. Inputs

- Regulatory bodies, which can formulate specific requests, or define new operating technical specifications. An important point concerns the fuel performance in accidental conditions.
- Political environment: the current situation of the nuclear industry leads to consider the "political risk" at least as important as the technical risk (suspension of plant operation for example if the plant is operated inappropriately).
- Economy: this parameter is now become a key-point, which presents many incentives.
- Usage of MOX fuel assemblies in some countries, which is accompanied by a specific management and operational surveillance.
- Intermediate spent fuel storage, which necessitates on-site facilities.
- External scientific support, coming from R and D institutions, universities, independent experts.
- The drainage of competence that has been seen over the last decade in the nuclear area. This situation has occurred both at the utilities as well as at the fuel vendors due to that few new young engineers have the confidence to start to work in the nuclear field while the experts are retiring. To ensure safe and economical operation of the plants, it is crucial that staff both at nuclear utilities and fuel vendors participate in teaching and training classes.

1.2. Outputs

- Maintaining safety margins during normal operation and accidental conditions, by means of:
    - new fuel designs,
    - new fuel management schemes,
    - R and D on fuel performance in accidental situations (RIA, LOCA…) to assess the mechanisms involved
    - new codes and methodologies (including the treatment of uncertainties).*

- Maintaining competitiveness on a deregulated market by:
    - continuous improvements in the fuel design and fuel operation that lead to less fuel malfunctions -> less cost for inspection, repair or reconstitution,
    - higher discharge burn-up -> lower back-end cost,
    - longer cycles and shorter outages -> lower outage costs and easier staff management,

- power up-rates and more aggressive loadings (which may lead to higher peaking factors and more local boiling in PWRs) -> improved neutron economy,
- load follow and remote control -> better price for electricity if it is needed,
- improved chemistry : Noble Metal Chemistry (NMCA) and Hydrogen Water Chemistry (HWC) in BWRs, and Zn injection in PWRs -> less cracking in piping and less inspection and repair costs,
- less fuel failures and degradation, {Zn, Fe, O} dosage in BWRs, Zn dosage in PWRs, increased LiOH content in PWRs -> lower activity build-up, with several favourable consequences in the plant management (lower surveillance and repair cost, lower cost for wastes disposal, better achievement of the ALARA principle,…).

2. CURRENT ISSUES FOR FUEL FAILURES IN NORMAL OPERATION

These evolutions or improvements in the fuel or in the plant management have involved a new repartition of fuel failures

2.1. "Non failure" situation

It is worthwhile to notice that some external causes could affect the fuel reliability, and shall be taken into consideration. These external causes are:

- fuel handling damages,
- fuel assembly bowing,
- Axial Offset Anomalies (AOA),
- new water chemistry regimes, such as Noble Metal Chemical Addition (NMCA) or LiOH increase.

2.2. Primary failures

The meeting put the stress on the following root causes:

2.2.1. Fretting

- due to debris in the coolant,
- due to grid-rod interaction (baffle jetting, cross flow, manufacturing defects, improper design towards the high neutronic fluence).

2.2.2. Pellet-Cladding Interaction (PCI)

- manufacturing defects (missing pellet surface in liner or non-liner fuel, fuel chip),
- improper fuel surveillance codes.

2.2.3. Hydride assisted cracking

- a new failure mechanism was noted where a crack initiated at a massive hydride layer at the clad outer surface propagated through the whole cladding thickness during ramp testing. This failure mechanism may potentially limit high burnup operation specifically in BWRs since hydriding will become more pronounced at higher burnups and simultaneously the pellet-cladding gap will become smaller. During power ramping, e.g. by pulling a control rod, PCMI may result in fuel failures.

2.2.4. Other manufacturing defects

- primary hydriding,
- weld crack or incomplete plug welding,
- clad flaw or defect.

2.2.5. Crud Induced Local Corrosion Failures (see § 6.4)

2.3. Secondary failures and fuel degradation

The main consequence of a fuel degradation is the release of fissile material into the coolant and an uranium core contamination. Experience feedback has shown that the natural removal of this contamination (spent fuel assemblies discharge, cleaning of the primary circuit walls) takes several years. Another consequence is a potential problem to extract the failed rod, which can breaks if the clad is severely hydrided.

3. TOOLS TO MANAGE THE PRESENCE OF FUEL FAILURES

Participants agree that the annual fuel rod failure rate is the most suitable and the mostly applied quantity to evaluate the PWR fuel reliability by the utilities. An improvement could be to define a normalization between ”failed rods” and ”failed rods with degradation”. For BWRs the situation is not comparable since there is a vast variation in activity release from a failed rod. If severe degradation occurs, the plant may have to shut down just to remove one leaking rod. For BWRs, the most relevant measure is the total off gas activity and uranium contamination.

Some scientific progresses to do, and experience feedback or devices to use, have been highlighted during the meeting, due to their interest to mitigate the consequences of failures on plant operation:

3.1. R&D support

Some topics coming from the R and D support shall be investigated, in order to have in hands a more reliable predicting of the failure evolution during the current cycle:

- better mechanistic understanding of the root cause development or of the degradation mechanism,
- better assessment codes,
- better out-of-pile tests to verify behaviour of fuel and clad under given conditions.

3.2. On-line assessment

Determination of the failure root cause is an important point to identify and anticipate a potential generic problem on the fuel. Several tools are commonly used in plants for that, and some of them shall be improved or have to be more extensively used. It is notably the case for determination of failed fuel characteristics under operation, by means of:

- on-line primary gamma activities measurements,
- more accurate and reproducible sampling methods,
- more accurate assessment and diagnosis methods and codes.

3.3. Outage inspection

On the other hand, complementary information are gained due to inspection methods to assess failures during outage:

- in-mast or in-cell qualitative sipping test,
- quantitative sipping test in some countries,
- individual inspection of failed rod, after extraction,
- ultrasonic testing and Eddy Current testing if rods are individually reachable.

3.4. Doubtful cases

For a doubtful case, or when importance or potential consequences justify it, it is useful to send the failed rod to a Hot Cell Laboratory for complementary non-destructive and destructive testing (see § 6.2).

4. REMEDIES TO PREVENT FAILURE OCCURRENCE

Some remedies or methodologies have been identified during the meeting to prevent failures:

- more robust fuel designs (e.g. spacers),
- plant modifications (e.g. elimination of debris, flow conversion, change to Ti-condensers),
- better manufacturing control,
- better verification of new fuel designs : the first point is of course to verify that a specific fuel design is suitable for the considered purpose. But a second point, which should be as important as the first one, is to assess the limits of the new design in all respects related to the operating conditions.

5. POTENTIAL ISSUES LINKED TO NEW FUEL DESIGNS AND NEW MATERIALS UTILIZATION

New fuel designs or new materials utilization may potentially lead to some operating problems:

- As mentioned in § 4, incomplete verification of margins related to new fuel designs may cause problems in non-typical operating conditions.

- The fuel assembly integral behaviour should be verified, in order to avoid interaction between two parameters, apparently not linked after a first analysis.
- New problems could emerge, even if they were not real problems in the old designs.

6. INSPECTION, RESEARCH AND DEVELOPMENT EFFORTS NEEDED TO SOLVE POTENTIAL ISSUES

The current status of the failures in water reactors, and the trends observed in the fuel management, lead to formulate some recommendations:

6.1. On-site inspection needs

It is highly recommended to perform on-site inspections in order to:

- identify failed fuel assemblies,
- identify number of failed rods and of defects,
- identify the root cause(s) for the defect(s),
- eliminate as soon as possible certain primary failure causes (e.g. debris),
- prevent potential generic root causes.

For these purposes, several tools are available on-site: in-mast sipping, in-cell sipping, underwater visual inspection (assembly and individual rod), ultrasonic testing and Eddy Current testing (if assembly design permits it).

It has to be underlined that dismountable assemblies permit individual rod extraction and inspection, and facilitate the assembly repair.

Moreover, some participants underlined the interest to develop a non-destructive technique for quantitative analysis of hydrogen in fuel rods.

6.2. Hot Cells post-irradiation examination needs

In some cases, complementary examination in hot cells permits a more complete assessment:

- definite failure root cause determination or confirmation,
- irradiated materials properties verification (e.g. behaviour of defective fuel or clad in specific situations or in accidental conditions).

6.3. Technology and experience feedback transfer between different reactor types

During the meeting, it has been remarked that some problems have been solved in one type of reactor, and that remedies could be usefully considered in another type still facing a similar problem. Some examples have been highlighted:

- cladding liner technology developed for BWRs: interest for PWRs and WWERs,
- Zr/Nb material behaviour in WWERs experiences: interest for PWRs,
- material behaviour experiences in more PWRs demanding conditions (higher burn-up, longer cycles, mixed cores,…): interest for WWERs.

Moreover, participants guess to exchange experience feedback on “how to” avoid primary failure and fuel degradation (other subjects can be also discussed).

6.4. Mechanistic understanding

A need for a mechanistic understanding of the failed fuel rod behaviour has been also underlined, to develop or adapt models. There are numerous cases were the fuel vendor has made a "quick-fix" of a problem by essentially making some change in their manufacturing process but without any mechanistic knowledge behind the change. Therefore a small change in operation, applied water chemistry regimes or material manufacturing process will make the problem to reoccur.

The crud induced corrosion problems both in PWRs and BWRs is a good example. These types of failures occurred already 20 years ago and they still reoccur. The only way to resolve this problem is to get a mechanistic understanding of the crud induced corrosion process. As long as we have not this understanding, this type of failure will reoccur.

The following material related issues clearly needs to mechanistically understood to resolve the issue:

- crack initiation and propagation in the cladding,
- release of fission products out of defective fuel during transients,
- internal volumes of a failed rod description.

6.5. Early integration of R&D results

Intensive R and D programmes often accompany new improvements on fuel or assembly design. These programmes concern generally only a very specific topic (e.g. clad external corrosion, grid-rod fretting, pellet-cladding interaction, fission gases retention in the fuel matrix…). It is important that results gained thanks to these programmes can be early presented to other designers (reactor water chemists, material people, fuel people), to integrate them and avoid supplementary R&D tests.

6.6. R&D needs for experiments

New designs behaviour (fuel, clad, assembly...) shall be assessed thanks to two main types of experiments, either out of pile, or in pile:

6.6.1. Separate effects experiments

These experiments permit to:

- enhance the database in addition to the one coming from surveillance programs (other parameters, other operating conditions)
- verify the influence of changes in the design of in the operating conditions,
- assess the behaviour during extreme conditions, to find the operating limits of the design (existence of a possible cliff-edge effect ?),
- anticipate possible industrial generic problems.

6.6.2. Global (or integral) experiments

A few experiments of this type are useful to:

- assess the performance of the global design thanks to a final verification taking into account numerous parameters and a bundle geometry as representative as possible,
- check the whole code prediction (benchmark).

6.7. Better assessment of experiments or on-site inspection results

In some cases, non-typical fuel or rod behaviour observed through in-pile experiments or on-site inspection, are simply classified as "anomaly" or "non-representative operating conditions". These cases should be analysed more attentively, because they may enable the utility to catch an emerging potential problem.

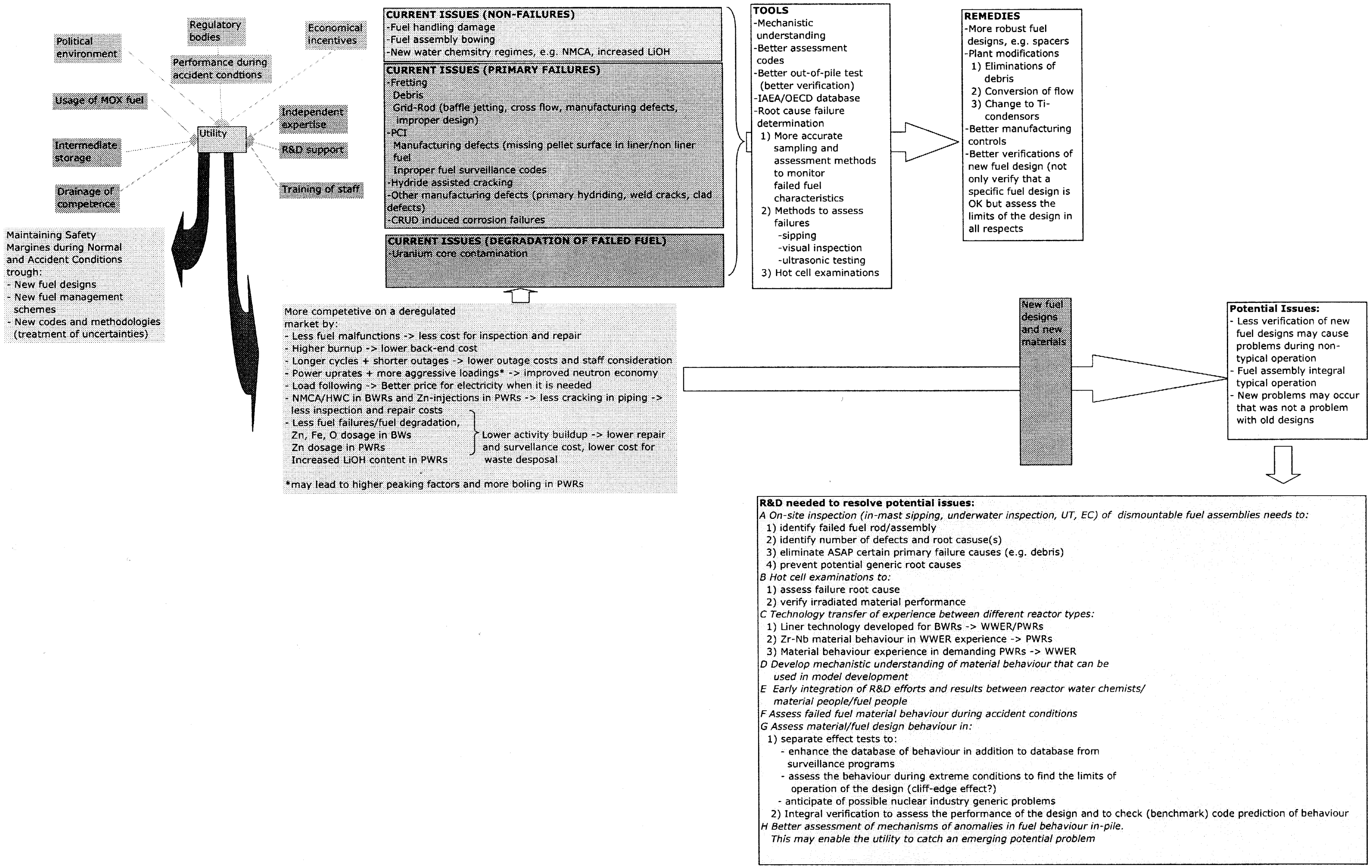

FIG. 1. Current issues, tools, remedies and R&D needed to solve potential issues.

GENERAL OVERVIEW OF PRESENT STATUS OF EXPERIENCE

(Session 1)

EXPERIENCE AND RELIABILITY OF FRAMATOME ANP'S PWR AND BWR FUEL

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Abstract

Based on three decades of fuel supply to 169 PWR and BWR plants on four continents, Framatome ANP has a very large database from operating experience feedback. The performance of Framatome PWR and BWR fuel is discussed for the period 1992–2001 with special emphasis on fuel failures, countermeasures and their effectiveness. While PWR fuel performance in most reactors has been good, the performance in some years did suffer from special circumstances that caused grid-to-rod fretting failures in few PWRs. After solving this problem, fuel of all types showed high reliability again. Especially the current PWR fuel products AFA 3G, HTP, Mark B and Mark BW showed a very good operating performance. Fuel reliability of Framatome ANP BWR fuel has been excellent over the last decade with average annual fuel rod failure rates under 1 × 10-5 since 1991. More than 40% of all BWR fuel failures in the 1992-2001 decade were caused by debris fretting. The debris problem has been remedied with the FUELGUARDTM lower tie plate, and by reactor operators' efforts to control the sources of debris. PCI, the main failure mechanism in former periods, affected only 10 rods. All of these rods had non-liner cladding.

1. INTRODUCTION

The irradiation performance of the Framatome ANP nuclear fuel products during the period 1992-2001 is presented with emphasis on fuel failures and the performance of current fuel products. Results of failed fuel examination are presented together with counter measures taken and the efficiency of measures.

2. OVERALL IRRADIATION EXPERIENCE

By December 2001, nuclear fuel fabricated by Framatome ANP in Belgium, France, Germany, and in the USA had been irradiated in 169 commercial power reactors on four continents. This fuel included more than 140,000 fuel assemblies containing over 25 million fuel rods. The maximum assembly burnups are 65 GWd/tU in a PWR and 71 GWd/tU in a BWR.

The burnup distribution of individual fuel assemblies with burnups above 40 GWd/tU is given in Figure 1. Although a large part of the fuel currently in core (which is scheduled to reach higher discharge burnups than earlier fuel designs) is still operating at low burnup, an increasing fraction of Framatome ANP fuel has already achieved burnup values beyond the former burnup targets.

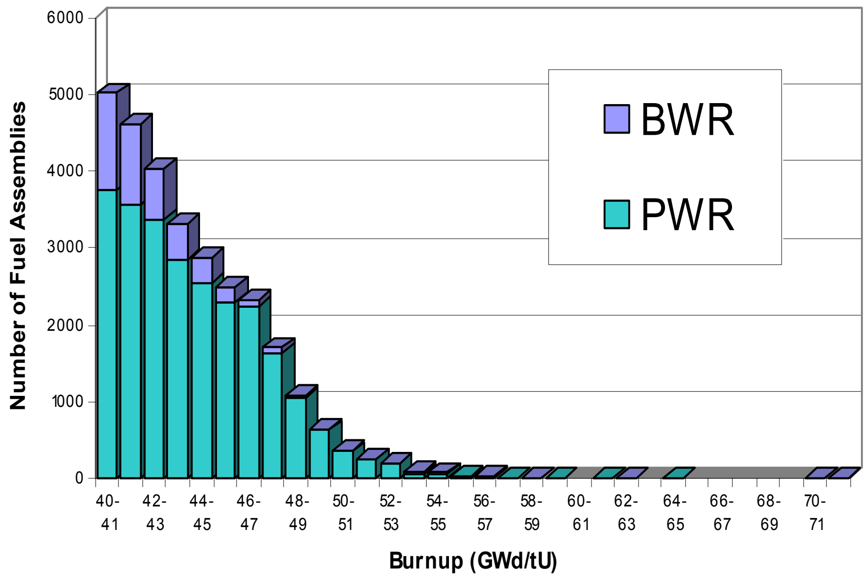

FIG. 1. Burnup distribution of Framatome ANP fuel assemblies (as of Dec. 2001).

3. FUEL FAILURE EXPERIENCE

Failure experience is presented in terms of annual fuel rod failure rate, i.e. number of failed rods during a year divided by the number of operating rods in this year. For many years, the industry-wide target to achieve was a failure rate below 10-5. Although some setback did occur recently, this target has been achieved with both, BWR and PWR fuel.

3.1. PWR fuel

Framatome ANP PWR fuel of all types had already shown high operating reliability with mean annual fuel rod failure rates around 4 × 10-5 at the end of the 80ies and a continuing downward trend.

This positive trend was interrupted in the mid 90ies, when fuel in some reactors suffered from special situations that led to a number of grid-to-rod failures, mainly in 16 × 16 fuel assemblies. Recovering from this instance, the fuel performance showed a positive trend again and the mean failure rates were less than 1 × 10-5 in the years 1999 and 2000. This was the best annual result achieved with Framatome ANP PWR assemblies (Fig. 2).

The results for 2001 took an unexpected turn due to a number of grid-to-rod fretting failures as reported hereafter, which disrupted the positive trend.

Fuel examination revealed that grid-to-rod fretting, debris fretting and fabrication deficiencies caused most failures in this period (Fig. 3).

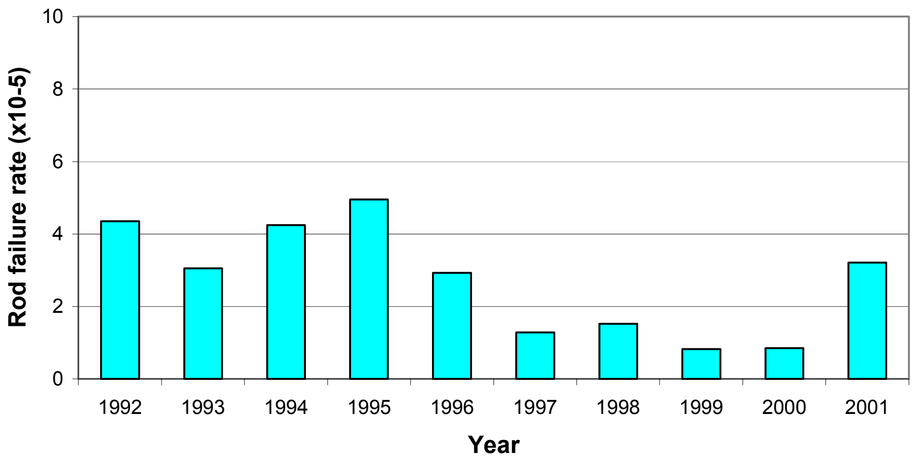

FIG. 2. PWR rod failure rates in 1992-2001.

Fuel irradiated in Germany has recovered from *grid-to-rod fretting failures due to spacer spring breakage* in the mid 90ies. The cause of failure was traced back to the combination of high stress and high stress corrosion cracking susceptibility in spacer springs, caused by improperly heated Inconel springs and the use of this spacer at the lowermost spacer position which is below the active length in the affected plants. This problem was solved by use of proven Inconel spacers outside the active region.

Cases of *grid-to-rod fretting* occurred *at the core periphery* in this period. These failures were often but not always due to baffle jetting. In a German plant for instance, grid-to-rod fretting failures have repeatedly been observed at the same core positions at the core periphery with fuel using conventional spring-and-dimple spacers. Since HTP assemblies have been loaded at these core positions in 1997, such failures were not observed again.

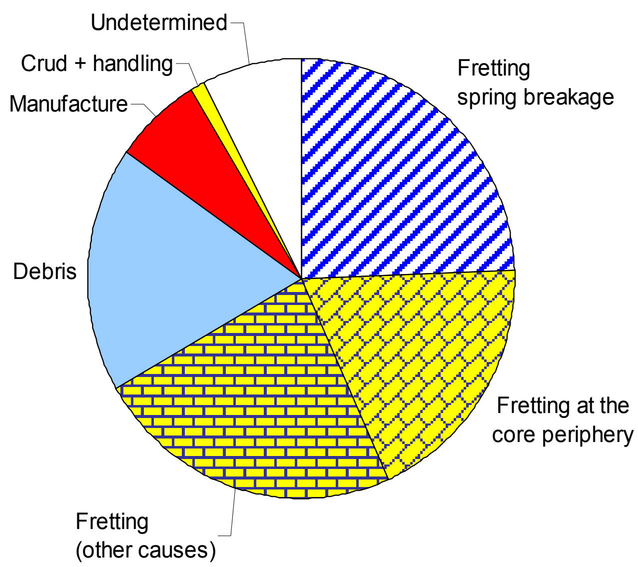

FIG. 3. PWR rod failure causes in 1992-2001.

Grid-to-rod fretting due to fuel rod vibration remains as a significant cause of fuel failure and is responsible for one quarter of PWR fuel failures. Such failures were observed last year. 28 AFA 2G fuel assemblies developed leaking fuel rods during cycle 8 of Cattenom 3 (a French nuclear plant). After the outage in early 2001, fuel examinations were performed and it was determined that the failures were caused by spacer grid fretting wear under the lower grid. Studies and hydraulic testing are still under way to fully understand the root cause. This grid-to-rod fretting is partially due to Cattenom 3 high cross flow and partially to the long residence time of some of the fuel assemblies in the most demanding positions. Cattenom 3 is a 4 loop 14 foot core plant utilizing long (18 month) irradiation cycles with a load follow and frequency control operation. Our design and manufacturing departments are working hard to deliver reinforced fuel assemblies by the end of this year.

Besides grid-to-rod fretting, *debris-induced fretting* caused most of the failures (Fig. 4). The TRAPPERTM, FUELGUARDTM and IDF anti-debris devices proved their filtering efficiency with a drastic decrease in the number of failures due to debris. To date there have been no debris failures in any PWR fuel using one of these debris retaining bottom nozzles.

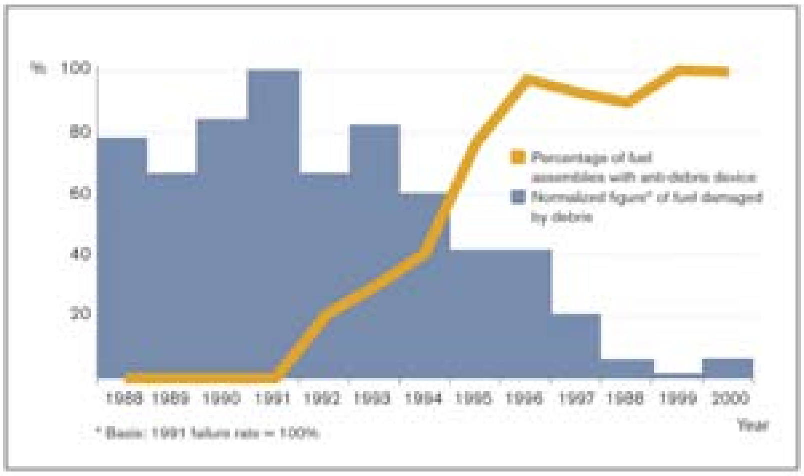

FIG. 4. Anti debris devices have proven effective.

Fuel examinations revealed that *primary hydriding* due to contamination by hydrogenous compounds was a frequent fabrication cause of leaks. One typical leak of this kind occurred at the end of 1994 in a French reactor. After two weeks of operation, high fission product activity was measured and the reactor was shut down. Ten fuel rods in two fresh assemblies were leaking. All of the failed rods had blisters just above the bottom end plug. The information gathered from the hot cell examinations and fabrication investigation (root cause analysis) demonstrated that the cause of the failures was pollution of some pellets by hydrogenous compounds. Measures have been taken in the fabrication plant to prevent repeated occurrences of pollution.

Some fuel assemblies suffered from specific problems like fabrication (end plug welding), and primary coolant chemistry (crud). A typical example of a *failure caused by welding* is described below. In 1999, one failed fuel rod was a rod that had been repaired. The seal weld cycle proceeded incorrectly and it was repaired by re-welding. The investigation proved that the root cause of the failure was this specific repair. This seal weld repair procedure has been eliminated.

There has been one incidence of *crud* related failures, which occurred in the United States. In a 1995 refuelling, nine failures and close to one hundred degraded rods were discovered. Low pH and high boron concentrations early in the cycle led to the formation of a thick layer of crud on the peripheral rods of some of the high powered fuel assemblies. It is believed that steam blankets formed into the crud layer, significantly increasing the cladding temperature and causing accelerated corrosion. The fuel rods failed as a result of the accelerated corrosion.

3.2. BWR fuel

In the 70ies and 80ies, the rod failure rates for BWR fuel assemblies fluctuated more widely, mainly due to instances with enhanced number of PCI (pellet clad interaction) failures. As discussed below, this situation has changed completely.

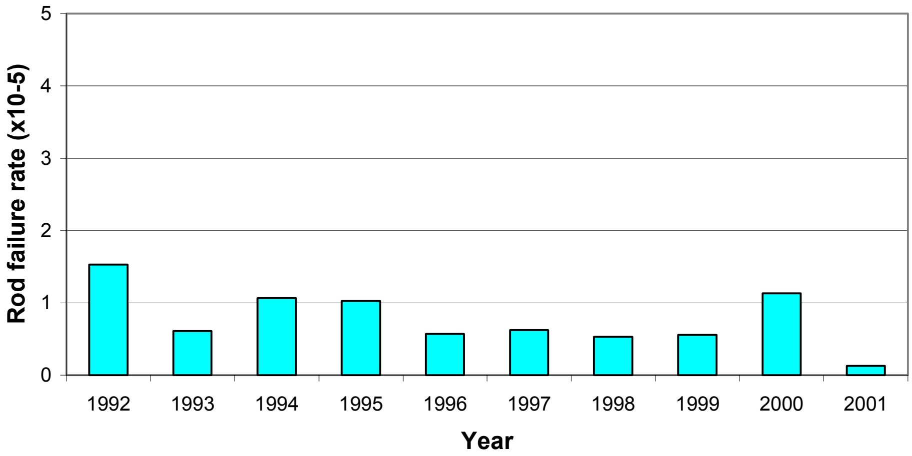

FIG. 5. Annual BWR fuel rod failure rates in 1992-2001.

In the last ten years, the performance of Framatome ANP BWR fuel has been excellent. Average annual fuel rod failure rates showed a generally decreasing tendency with values below 2 × 10-5 since 1991, and the average for the period even dropping as low as 0.7 × 10-5.

Poolside examination revealed three failure mechanisms that caused most of the BWR fuel failures.

Fretting by metallic debris was the leading failure mechanism in BWR fuel in this period, causing more than 40% of all fuel rod failures. From this, a further reduction of debris fretting failures is the best measure for a further improved BWR fuel performance. Same as in PWR fuel, the debris problem has been remedied with the implementation of debris resistant lower tie plates, and by reactor operators' efforts to control the sources of debris.

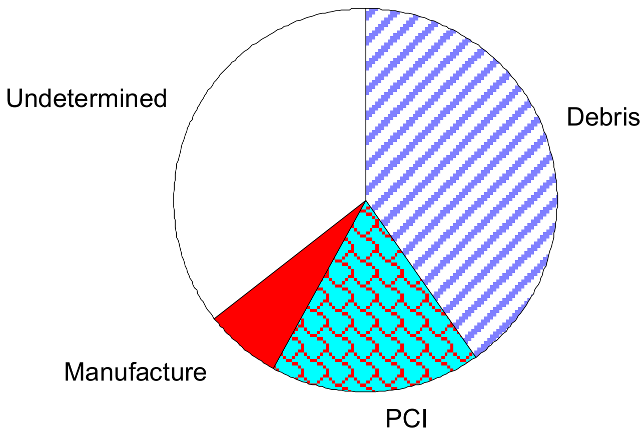

FIG. 6. BWR rod failure causes in 1992-2001.

PCI, the main failure mechanism in former periods, affected only 10 rods in the time period since 1992. All of these rods had non-liner cladding. The countermeasures initiated as a consequence of the PCI damage in 8x8 and 9x9-1 fuel (short term: administratively by operating recommendations, medium-term: improved fuel quality and clad materials) were generally successful. Just 3 failed rods confirm this within the last 5 years. Two major contributing factors in this area have been the introduction of new cladding materials and improvements in fuel pellet quality. Thanks to these advances as well as the excellent capabilities of today's core monitoring systems, the probability of fuel rod failures being caused by pellet-clad interaction (PCI) has been further reduced.

Few failures in 1-cycle fuel rods had to be attributed to *fabrication deficiencies*. Poolside examination revealed primary hydriding due to hydrogen containing material within the rods as the most likely cause of failure. Measures were taken in fuel fabrication to prevent further failures or at least reduce such failures.

Based on this situation, the most effective measure to achieve a further reduction of BWR fuel failure rates is the reduction of debris failures. The commercial implementation of debris filters in BWR fuel started with the ATRIUM 10 design.

4. CURRENT FUEL DESIGNS

Today's fuel performance depends on the operating behaviour of current advanced fuel designs. Such advanced PWR fuel products are supplied as AFA 3G and HTP assemblies in Europe and as Mark B/BW and HTP in the US. In case of BWR fuel, this is the ATRIUM 10 concept.

4.1. HTP

HTP fuel assemblies have shown an excellent operating behaviour over a 14-year period during which altogether 3,340 HTP assemblies have been in service.

With a total of seven failed rods, five of which occurred in one plant, the annual fuel rod failure rate for all HTP fuel remains as low as 3 × 10-6. Fuel exam revealed, that 5 failures were caused by grid-to-rod fretting in the outermost bi-metallic spacers, which in the meantime has been replaced for most fuel delivery batches. One rod failed due to debris, the remaining failed assembly has not yet been inspected.

Most of the operating experience with HTP fuel assemblies was gained with assemblies having a FUELGUARDTM anti debris filter. Up to now, no debris fretting failure was observed in these assemblies.

4.2 AFA-3G

Since 1997, more than 5,000 AFA 3G fuel assemblies have been irradiated in France and in 13 PWRs in others countries. All are showing very good behaviour.

Only seven AFA 3G fuel assemblies have developed failures. Three of the failures occurred in Germany as a result of spacer spring failure. Another two were due to a fabrication problem. Both problems have been addressed. The two remaining failed assemblies will be inspected in the near future. The resulting average annual fuel rod failure rate is 6 × 10-6.

All AFA 3G fuel assemblies are configured with the TRAPPER™ bottom nozzle. It consists of a perforated plate welded to the top of a ribbed supporting structure. Its effectiveness in stopping debris larger than 3.3 mm is 100%. Moreover, the cavity formed by the internal ribs prevents debris from migrating to the perimeter of the nozzle. The TRAPPER™ bottom nozzle design is based on the vast experience with the AFA 2G anti debris device. Its effectiveness is superior to the AFA 2G anti debris device. There have been no debris failures in the AFA 3G fuel assemblies.

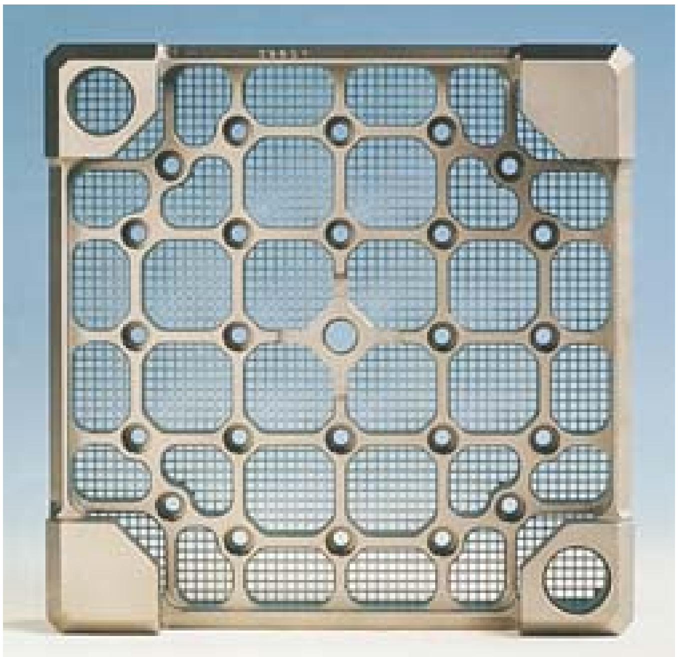

FIG. 7. Trapper anti debris device.

4.3 Mark BW and Mark B

The Mark BW, which is a 17 × 17 replacement for Westinghouse designed PWRs, was first produced on a batch basis in 1991. Since then, 2,587 fuel assemblies have been manufactured. In the first couple of years debris was the dominant failure mechanism. After the first two years of production and the introduction of the debris resistant lower end fitting, there have been no debris failures. That is no debris failures in the last 555,000 fuel rods manufactured.

Other failure mechanisms have occurred very infrequently. The failure rate for the Mark BW for the last five years, for all causes, has been 3 × 10-6.

The Mark B fuel assembly, a 15 × 15 fuel assembly designed for B&W reactors, has been in production since 1971. Since then, 8,916 fuel assemblies have been manufactured. The most common failure mechanism of the early designs was debris fretting. Since the introduction of a debris resistant design, there has been only one debris failure in the last 670,000 fuel rods manufactured.

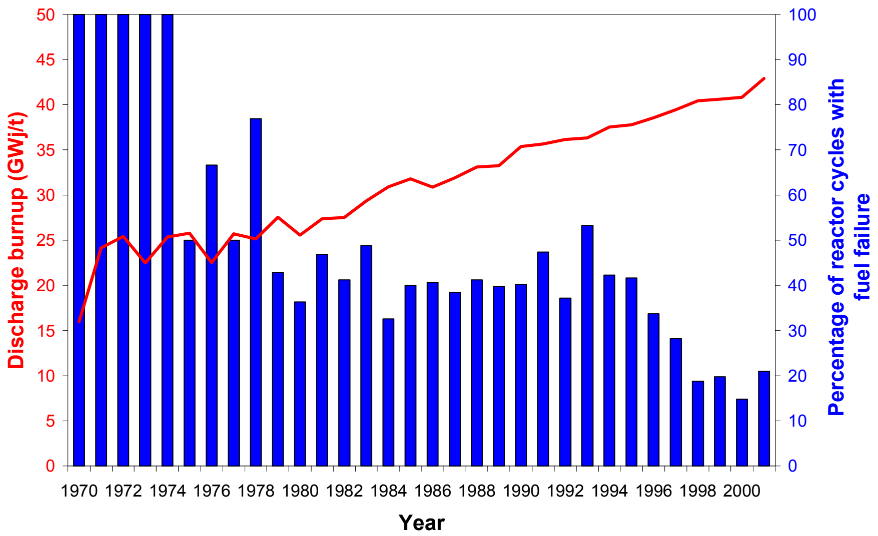

FIG. 8. Increasing discharge burnup and high operating reliability: no contradiction.

4.4. ATRIUM 10

Current BWR fuel deliveries are based on the ATRIUM 10TM fuel design. BWR cladding, which has been optimised for waterside corrosion resistance, is supplied either as Zircaloy-2 through-wall cladding or, optionally, as Fe-enhanced Zr liner cladding, which has high PCI resistance in combination with high resistance to secondary degradation in the event of primary failure from any cause.

So far, ATRIUM 10TM fuel assemblies have been supplied to a total of 19 plants in Europe, Asia, and the US giving Framatome ANP operating experience with nearly 4,000 fuel assemblies of this type and, thanks to ongoing supply contracts, the experience will continue to be enhanced. The ATRIUM 10TM fuel assemblies showed a very good operational performance with only six failed rods in total. Fuel examination confirmed debris fretting as the primary cause of failure in three of these rods. For the other three rods the cause of failure remains undetermined.

5. CONCLUSION

Feedback from the operational behaviour of more than 25 million irradiated Framatome fuel rods have provided the knowledge to deliver fuel with high reliability even under today's demanding operating conditions. Each failed fuel rod is a setback for fuel reliability. Framatome ANP follows a firm procedure to clarify the root cause of failure as a starting point for the development and implementation of effective counter measures. This proceeding resulted in improved fuel reliability in the past and will enable Framatome to reach even better fuel performance in the coming years.

SPANISH EXPERIENCE WITH LWR FUEL: GENERAL OVERVIEW

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Abstract

The operating experience in Spain regarding fuel failures is described in this paper. The operational strategies followed by the NPPs, their fuel failure history, the root causes and the remedies implemented in the form of fuel design changes, operational actions and analysis methods changes too, are discussed. The operational strategies now followed include the operation of longer cycles, nuclear power uprates, mixed cores, and a tendency to apply best estimate analysis methods. Changes associated to this strategy are the increase of the fuel discharge burnup, the use of higher enrichments and burnable absorbers, and the operation of the fuel in a more aggressive environment in general terms. These changes have an impact on the fuel safety and reliability aspects, and have led to fuel design changes like the utilization of advanced cladding materials, pellets, etc. All those changes have also obligated to upgrade the thermal mechanical models. The presentation shows the fuel failure history, including all the failures founded, both at PWR and BWR NPPs, the main root causes (crud induced localized corrosion (CILC), debris, baffle jetting, hydriding, etc.), and the design and operation solutions implemented by both the vendors and the NPPs (design, chemical control, condenser tubes change, upflow conversion, etc.).

1. INTRODUCTION

The different vintages and technologies of its nuclear power plants (NPP) characterize the Spanish nuclear park. Fuel failure history, or more strictly speaking, the phenomena associated with fuel failure, is then related to individual plants in general, because of their different nuclear characteristics (nuclear power, different technology, different core designs, etc), and with the exclusion of those plants, PWR, from second and third generation and United States (**US**) technology.

Generally, the second and third generation plants from *US* technology, have applied different operational strategies that affect core environment, and have influence on fuel behaviour, and fuel failure history.

The strategies followed by the different SNPPs are the extension of fuel cycles, the increase of nuclear power, the utilization of mixed cores, (not only due to fuel design changes, but also to new fuel designs), the increase in discharge burnup, and also the use of low leakage loading patterns.

One of the ways the SNPPs are working involves changes in design and analysis methodologies, oriented to a statistical approach. These “best-estimate” models also imply uncertainty analysis.

From nine units Spain has, seven have increased their cycle length. This change affect specially, the second and third generation *SNPPs* with *US* technology, and follow the way open by the American ones.

Table 1. Spanish Nuclear Park

Plant (number of reactors) Nuclear/Electric Power	Model (Date of construction)	Design origin
C.N. Jose Cabrera (1) [a] 510/160 MW	1 loop PWR (1968)	Westinghouse (*US*)
C. N. Sta. M. De Garoña (1) [a] 1381/460 MW	BWR-3 (1971)	General Electric (*US*)
C. N. Almaraz (2) [b] 2696/930 MW	PWR (1981-1983))	Westinghouse (*US*)
C. N. Ascó (2) [b] 2696/930 MW	PWR (1983-1985)	Westinghouse (*US*)
C. N. Cofrentes (1) [b] 2952/994 MW	BWR-6 (1984)	General Electric (*US*)
C. N. Vandellós 2 (1)[c] 2775/992 MW	PWR (1987)	Westinghouse (*US*)
C. N. Trillo (1) [c] 3010/1066 MW	PWR (1988)	KWU (Germany)

[a] 1st Generation ; [b] 2nd Generation; [c] 3rd Generation

Table 2. Current cycle lengths and uprating process of different SNPP

NPP	**Cycle length** (months)	**1st uprate** %	**2nd uprate** %	**3rd uprate** %	**Current status**	**Next future** %
GAROÑA	24 ([a])	-	-	-	100 %	-
JOSE CABRERA	12	-	-	-	100 %	-
COFRENTES	18([a])	+2 (1988)	+2.2 (1998)	+5.8% (2002)	110 %	-
ASCO 1	18([a])	+8 (2000)			108 %	+1.5 (2003)
ASCO 2	18([a])	+8 (1999)			108%	+1.5 (2002)
ALMARAZ 1	18([a])	-	-	-	100 %	+1.4 (2003)
ALMARAZ 2	18([a])	-	-	-	100%	+1.4 (2004)
VANDELLOS	18([a])	+4.5 (1999)	+1.5 (2002)	-	106%	-
TRILLO	12	-	-	-	100 %	-

([a]) Change of cycle length.

Concerning the uprating strategy, the SNPPs have followed the international trend to increase their nuclear power. We can distinguish between two kinds of uprates in Spain until the current situation. Those which implies strong charge of licensing and analysis, that are usually greater than 2%, and those less than 2%, called mini-uprates, that are usually related to the improvement of flow measurement on the feedwater system.

The evaluation of the uprating, by the CSN, is different depending on the type of uprate we are dealing with. Anyway, the *SNPPs* have followed an uprating process in different stages (see Table 2).

After the deregulation of the electricity markets in Europe, the Spanish *NPPs* are intended to reduce operating costs. One of the strategies followed by the *NPPs* is to have different fuel suppliers.

Moreover, the operational strategies have implied fuel design changes. The *SNPP* have mixed cores (see Table 3), with fuels of a same supplier but different designs, or with fuel form different suppliers too.

The safety limits are associated to the fuel design. Thus, different fuels have different safety limits. If the utilities do not change the supplier, the various analyses are internally coherent; as long as the supplier design and monitoring methods are approved, no additional action is needed.

Table 3. Mixed cores

NPP	DESIGN	FUEL VENDORS	FUEL TYPE	DEMO	Licensed Enrichment	Maximum enrichment reload	Burnup Limit MWd/Kg
COFRENTES (BWR-6)	GE	GENUSA ABB ATOM	GE 11 GE-12 SVEA 96+	No	Fuel type dependent (f.t.d.)	4.5%	f.t.d.
GAROÑA (BWR-3)	GE	GENUSA	GE-11 GE14	No	Fuel type dependent (f.t.d.)	-	f.t.d.
VANDELLOS (PWR-3 loops)	W	ENUSA	W-MAEF W-AEF W-OFA	No	4.90 %	4.7%	60 [1]
ASCO 1 & 2 (PWR-3 loops)	W	ENUSA	W-MAEF W-AEF	No	4.90	4.7%	60 [1]
ALMARAZ 1 & 2 (PWR-3 loops)	W	ENUSA [Framatome]	W-MAEF, AEF [AFA (demos)]	Yes	4.5	4.5%	60 [1]
TRILLO (PWR-3 loops)	KWU	KWU [ENUSA]	AH76 AH116 AH196 FOCUS AH216 FOCUS AH266 ENUSA (Demos)	Yes	4.3%	4.5%	
JOSE CABRERA (PWR-1 loop))	W	ENUSA	14x14 HIPAR 14X14 LOLOPAR	No	4.0 %	-	60 [1]

[1] average burnup per fuel pin rod (45-48 MWd/Kg U per fuel assembly: f.t.d.)

However, if more than one fuel vendor is involved, the utility must take appropriate action to ensure that the different methods and correlations do not carry over any inconsistencies or mismatches.

On the other hand, we can also see in table 3 that the enrichment is in some cases near the 5% in weight, and that the discharge burnup is near the licensing limit. The value of this limit is 62 GWd/Tn U.

The continuous increments of enrichment in PWR make it necessary to use burnable poisons at "beginning of cycle" (**BOC**). Initially, this need was realized by the chemical control of coolant, but now control is taken by the use of WABAS or pellets with burnable poisons mixed with fuel. This strategy has had some impact on the analytical methods used for fuel rod performance analysis.

The methods and models used for fuel rod performance analysis, and used as input for accident analysis, have suffered various changes and have become (or evolved to) statistical or best estimate cases. This is due to the better understanding of the behaviour of new materials, new designs, and the data compiled in post Irradiation examinations (**PIE**).

The Spanish NPPs have been introducing core designs with rods containing burnable poison of varying (different) contents in weight. This technique allows to improve the Chemical coolant programme and reduces the risk of fuel failure (The SNPP used a modified coolant chemistry programme).

However, the use of burnable pellets implies a penalty over the enrichment, and this affects the core design too.

The use of burnable poisons mixed into the fuel pellet reduces the conductivity. The penalty associated to this fact implies problems with the core design. The use of a new conductivity correlation for Gd pellets with a concentration up to 2%, as a best-estimate model, eliminates the penalty on the enrichment, and permits more elements containing low concentration of burnable poison. This allows more flexible core designs.

In the table 4, we can appreciate the near future plans. We can see that all the strategies discussed before are not finished yet. Mini upratings, new design elements, higher enrichment, higher discharge burnup, etc.

From a regulatory point of view, these strategies have an effect on safety limits. The use of mixed cores, the higher enrichment, and the more and more aggressive operational conditions, combined with the use of statistical or best estimate methods, all affect the traditional approach to fuel limits. Variables like DNBR or CPR, shutdown margin, reactivity coefficients, must be taken into account as a whole, because their behaviour has a synergistic interaction.

The uprating, for example, increases the linear heat generation rate (LHGR). This affects the following safety related criteria: the DNBR/CPR, the reactivity coefficient, the shutdown margin, the internal gas pressure, PCI, amongst others in normal operation as well as anticipated transients or postulated accidents.

Table 4. Near future plans

NPP	New fuels	Higher enrichment	Higher discharge burnup	New analysis methodologies	New management strategies	Other aspects
COFRENTES (BWR-6)	X	X	X	X	X	X
GAROÑA (BWR-3)	-	-	-	X	-	-
VANDELLOS (PWR-3 loops)	-	X	-	-	-	X
ASCO 1 & 2 (PWR-3 loops)	-	X	-	-	X	X
ALMARAZ 1 & 2 (PWR-3 loops)	-	X	X	-	X	X
TRILLO (PWR-3 loops)	-	X	X	X	-	X
JOSE CABRERA (PWR-I loop))	-	-	-	-	-	-

Table 5: PWR failure history during the last 10 years

YEAR	DEBRIS		FRETTING		PRIMARY HYDRIDING		UNKNOWN/ NON INSPECTED		TOTAL		
	E	KWU	E	KWU	E	KWU	E	KWU	E	KWU	ALL
93	2	0	0	0	0	0	0/3	0/1	5	1	6
94	3	0	0	0	0	0	1/4	0/0	8	0	8
95	0	0	0	0	0	0	0/6	0/0	6	0	6
96	0	1	0	0	0	0	0/3	0/0	3	1	4
97	9	0	0	0	0	4	0/3	0/0	12	4	16
98	0	0	0	0	0	1	1/3	0/0	4	1	5
99	1	0	1	0	0	0	1/0	0/0	3	0	3
00	6	0	0	0	0	0	0/0	0/0	6	0	6
01		0	0	0	0	0	0/1	0/0	1	0	1
02	2	0	0	0	0	0	0/0	0/0	2	0	2

2. FUEL HISTORY: CAUSES AND ACTIONS

Let's look now the fuel failure history of PWR. We can see in this transparency that I have made a distinction between American design NPPs and German ones, because of the differences between their fuel element designs. We have to bear (or keep) in mind that there are six American design NPPs and only one of German design in Spain.

The table 5 shows data over the last ten years. If we'd add the data of fuel failure from the years before 1992, we could observe a trend of the fuel failure. Before seeing the evolution graph, let's consider this table in more detail. The main cause of fuel failure is the existence of Debris in coolant. The second, due to Primary hydriding, has affected only the German design plant.

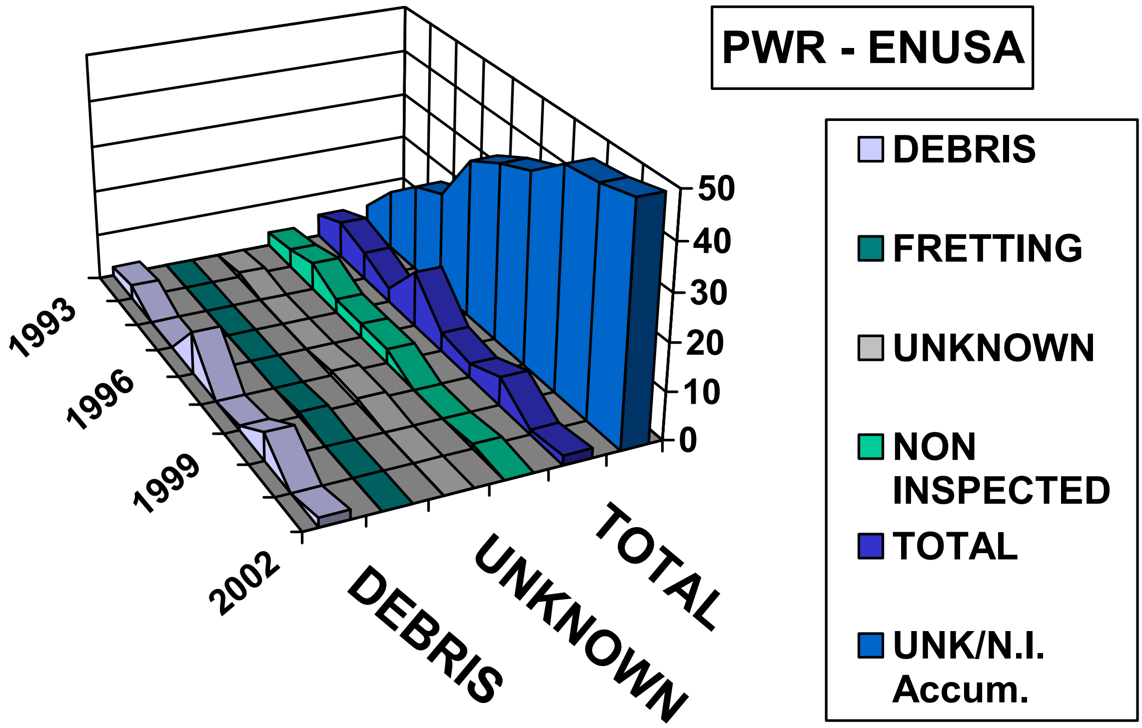

FIG. 1. Fuel failure statistics-causes for PWR-ENUSA plant.

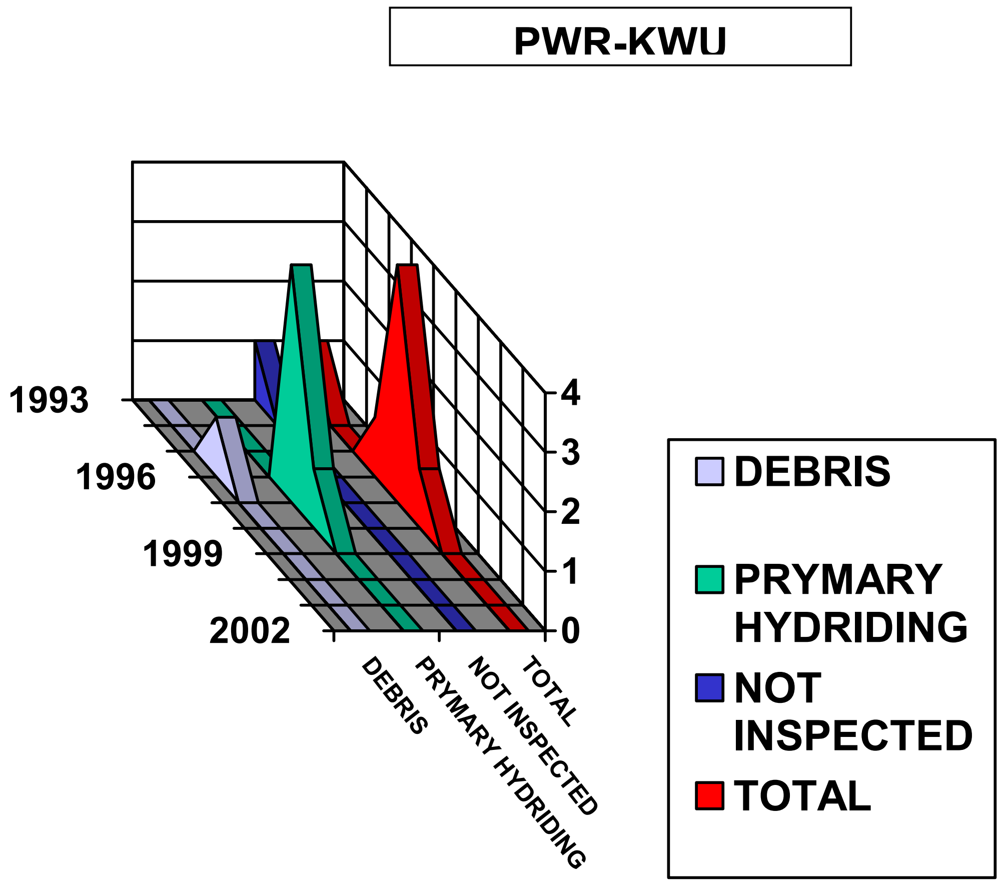

FIG. 2. *Fuel failure statistics-causes for PWR-KWU plant.*

Analysing the fuel failure graph, we can see a reduction in the frequency of fuel failure over the last four years. If we see the evolution per failure cause in Fig. 1, we observe a reduction on DEBRIS, hydriding and fretting causes, but we observe too the existence of not inspected and unknown failed elements, so we cannot conclude that the mentioned causes have been controlled.

For the KWU PWR plant, see Fig. 2, the primary hydriding has been considered the main cause on fuel failure, but the remedies seem to be sufficient to eliminate this cause.

- Concerning the main failure causes, as I mentioned before, the appearance of DEBRIS into the coolant has produced fuel failure by fretting, generally underneath the first spacer. The corrective actions taken, were to recuperate the elements by replacing the failed rods with stainless steel ones, and to make improvements on the plant maintenance activities to reduce the probability of appearance of DEBRIS and the change of fuel design in order to incorporate a DEBRIS filter in the bottom plate.

  It should be mentioned that the maximum value of failed elements, if we split these values by each plant, has happened after major modifications on plant systems or actions on the primary circuit. In any case, the evolution of this failure cause indicates to us that the problem doesn't seem to have been solved.

  In the past, when a susceptibility cause has been encountered, safety studies have been carried out to prevent fuel damage due to DEBRIS (for example, the fracture of top nozzle spring screws)

- The fuel induced vibrations fretting is identified like the cause of failed elements in the SNPPs. We can distinguish various mechanisms which can be considered in this set of failures, like manufacturing related fretting, or baffle jetting,, operational induced fretting, or vortex induced fretting. The failed element due to this cause is identified like manufacturing related cause.

  Attending to the vortex induced fretting, the existence of turbulence and flow inhomogeneities at the core bottom lead to the existence of fretting modes between spacer grids and rods, and finally to rod damage. This failure cause was discovered in the United States, and it isn't a cause of failure in Spain. The root cause analysis concluded that two facts affect this: the specific design of spacer grids, and the specific flow the elements are submitted to.

  However, the preventive actions taken by the SNPP were to use burnable poison type WABA on pheripherical configuration, to reduce the resonance frequency and eliminate breaking risks, and a modification on the assembly process into the element fabrication. This problem has been corrected by new grid design.

- The failure by baffle jetting cause affected only one plant in Spain. The rod vibration by cross flow, due to the differential pressure existing between the ascending flow of coolant through the core, and the descending flow providing internal cooling for the baffles, was corrected:

  Firstly by the substitution of both failed and damaged rods by stainless steel ones, including the introduction of fresh fuel with stainless steel rods in the susceptible positions, and the performance of mechanical peening on the baffle joints to reduce the gap size between segments and thus reduce the effective flow which can cause rod vibration.

And finally by the modification on flux sense from down-flow to up-flow in order to reduce the differential pressure with the ascending flow of coolant through the core.

- Hydriding was observed in one plant. The root cause analysis confirms the cause of this failure to be the existence of contamination during the fabrication process. The improvements on the manufacturing process, seem to be sufficient to avoid this failure cause.

- Although they aren't the direct cause of fuel failure, there are other problems which could be a problem for safety related limits. Apart from mechanical problems, like rod bowing or rod growth, the corrosion is, possibly, the only one that has an implication on the different safety criteria.

  Some inspections carried out in 1997 at a PWR plant observed the existence of a corrosion layer above the design limit This fact forced the NPP to select other material for the cladding. The original material was improved Zr-4, and the new material selected by the PWR-NPPs is ZIRLO.

  As a consequence of this inspection, the various NPPs at risk of the same behaviour developed a common inspection program PIC, together with the fuel supplier, to evaluate the new aspects of new design fuel.

Now, we'll move on BWR fuel failure history. In the same way than PWR treatment, we can see in table 6, the main causes of fuel failure over the last ten years.

This aside, the number of fuel failures, is low and seems to be stable As you can see, I have include the CILC phenomena into the table, although there hasn't been any failed elements over these years, because this failure cause was one of the main fuel failure cause in the past.

Table 6. BWR failure history during the last 10 years

YEAR	CILC	PCI	Fabrication	Unknown / Not Inspected	Hydriding	TOTAL
88-91	21	1	2	1 / 2	0	27
92	0	0	1	0 / 0	0	0
93	0	0	0	2 / 0	0	1
94	0	0	0	0 / 0	0	2
95	0	0	0	0 / 0	0	0
96	0	0	0	0 / 0	0	0
97	0	0	0	1 / 0	0	0
98	0	0	0	0 / 0	0	1
99	0	0	0	0 / 2	0	2
00	0	0	0	0 / 1	0	0
01	0	0	0	0 / 0	0	1
02	0	0	0	0 / 3	0	3

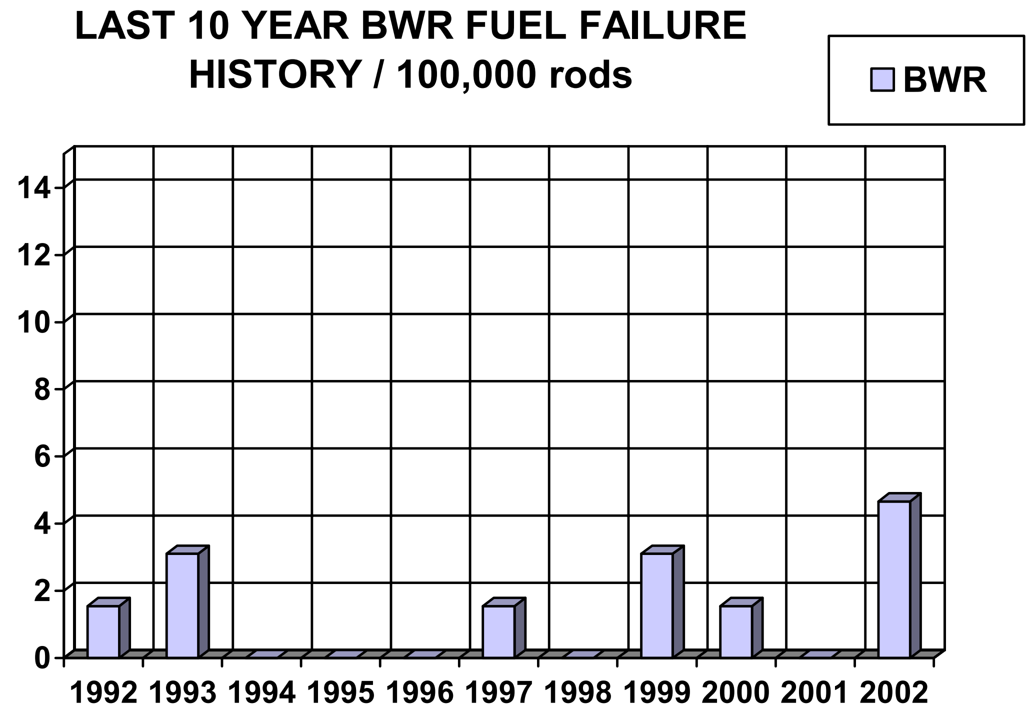

FIG. 3. BWR fuel failure history.

We can appreciate the stability of these values (see Fig. 3). Again, the inspection policy of the NPPs, prevent us to understanding the main cause of the last fuel failures, especially when the fuel reached its discharge burnup.

- Although Crud Induced Localized Corrosion (**CILC**) seems to have been solved, it was the main fuel failure cause in the past in one plant. The actions taken by the NPP to solve it were oriented to prevent the high concentrations of Copper found in primary coolant being the main reason obtained from the root cause analysis.

  The action taken was to change demineralisation process with the use of one more filter, and the physical modification in order to augment the filter surface. With this procedure, CILC, as a failure cause, seemed to be solved.

  However, recently, the NPP has implemented a design modification, in order to change the material of condenser tubes from Copper to Titanium. With this definitive solution, the origin of Copper in the coolant system disappears.

- Over the last ten years, the fabrication cause has been the most important cause of fuel failure. The solutions have been, in general, to improve the manufacturing and the inspection processes (UT, ET,. VT, etc.)

  As we can see on the table, the last failed elements haven't been inspected, and we put them on the unknown column, so we can't guarantee that the different failures have been solved.

- Other causes, like PCI or Hydriding were considered as one time events. The operational strategies (PCIOMR), and the introduction of barriers in the inner side of cladding seem to be sufficient to prevent them.

- As regards recent indications of shadow corrosion detected under the spacer, the studies performed seem to conclude the level is stable under normal operational condition. Nevertheless, there is an ongoing surveillance programme for GE-12 and GE-14 cladding material.

Concerning the operation of the plant, the strategies to operate with failed elements are in accordance with the international ones.

Thus, control is ensured through the follow up of coolant or off-gas activities. The Iodine and Xe activity level measurements serve to find or confirm the existence, type and size of fuel failure. On the other hand, the relation between Caesium isotopes serves to predict the burnup level, and then, where the probable area is, so the failed element can be located.

Apart from the limits established in Technical specifications, no other fuel safety criteria on coolant activity exist.

In the case of BWR plants, there is the possibility of screening the failed element, in order to continue with the operation and prevent the risks of secondary degradations.

In terms of fuel failure history, in conclusion, we can see that fuel failure rate of the SNPPs has been maintained over the last ten years in PWR and BWR, but the level is lower than in previous years.

The main failure cause in PWR NPPs is DEBRIS (Westinghouse plants), and the high values encountered coincided with major design modification or activities on the primary circuit, and PRIMARY HYDRIDING (KWU plant). These problems appear to have been solved, but we don't have the information on the last failures in the Westinghouse plants. This is due to the NPPs inspection policy on fuel elements with high burnup.

On the other hand, for BWR plants, the main causes were CILC and fabrication. The last year data seem to demonstrate that both of them have been solved, specially CILC problem.

3. CONCERNS AND NEEDS

With the advent of advance fuel and core design, the adoption of more aggressive fuel management modes and the implementation of more accurate (statistical or best estimate) design and analysis methods, the SNPPs have increased the level of reliability, and more aggressive conditions on the fuel, Fig. 4. Both of them affect the safety limits. The concerns about the normal operation problems seem to be solved, but the aggressive conditions, rise us doubts about the fuel behaviour under RIA and LOCA conditions.

Spain participates in most of the fuel research international programs, like HALDEN, ALPS, CABRI, Robust Fuel Programme, which intend to demonstrate the availability and safety of high burnup fuel.

These investigation programmes aren't focused on normal operation, and I won't discuss them. However, SNPPs and ENUSA as a fuel supplier, are developing a fuel inspection programme (PIC), devoted to the behaviour of the fuel under normal operation.

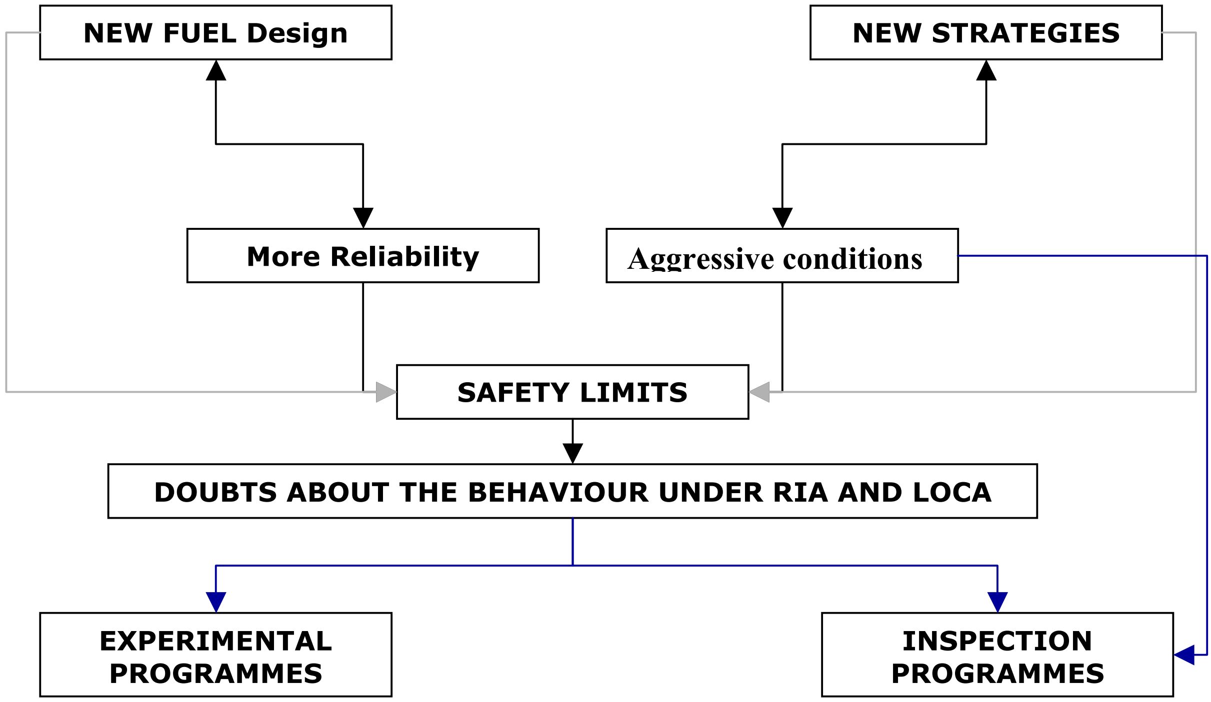

FIG. 4. Schematics of the fuel concerns, needs and programmes.

The fuel inspection programme was carried out because of high corrosion levels encountered during inspection of elements with Zr-4 cladding material in PWR plants. The values obtained were higher than the design limits. This, combined with the design changes due to rod bowing, induced the NPPs to develop an inspection programme to corroborate the data used to license the fuel design. This programme covers different operational conditions.

The programme is currently ongoing, and the first results, over fuel elements with hard operational conditions show a better behaviour of ZIRLO versus Zircaloy-4 for corrosion growth. These results also show the dependence on the operational conditions.

4. CONCLUSIONS.

An overview about the different operational strategies carried out in Spain, and the fuel failure history for both PWR and BWR NPPs, and the ways followed to eliminate or control them. We can conclude that all the operational strategies carried out imply more aggressive environmental conditions on the fuel elements.

- Fuel failure rate has exhibited a declining trend over the years in Spain, however some occurrence spikes were observed during last decade. On the other hand, existing fuel inspection policy does not guarantee investigation of all fuel failure events, in particular when failures affect to relatively highly burnt assemblies. So, *a continued fuel failure root cause analysis programme is needed for in the future.*
- The high burnup fuel behaviour is not well understood, so there is a ***need** to continue with the experimental programmes*.
- The data obtained from the fuel inspection programme reveal a dependence on operational conditions. So *the inspection programmes to follow the fuel behaviour in different operational conditions have to be maintained.*

PROBABILISTIC-STATISTICAL ANALYSIS OF WWER FUEL ELEMENT LEAKING CAUSES AND COMPARATIVE ANALYSIS OF THE FUEL RELIABILITY INDICATOR ON NPPs WITH WWER AND PWR REACTORS

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Abstract

The results of a comparative analysis of the fuel reliability indicator on NPPs with WWER and PWR reactors are presented in this report. As an exponent for a comparative reliability analysis of the WANO fuel reliability indicator is used. The fuel reliability indicator provides a general measure of the extent to which the reactor coolant activity is increased as a result of fuel damage. The analysis of fuel reliability indicator values during 1991-2001 at NPPs with WWER-1000 and WWER-440 reactors (Russia, Ukraine, Bulgaria, Czech Republic, Slovakia, Hungary, Finland) is carried out. The high level of WWER fuel reliability is scored except for cases of fuel failures in separate operating periods of some units.

1. INTRODUCTION

The results of probabilistic-statistical methods of fuel rod (FR) leaking causes analysis application are presented in the report. A probabilistic-statistical method of nuclear FR leaking causes analysis was developed by VNIIAES. These methods were originally used for the analysis of RBMK FR leaking causes. The methods allowed identifying and classifying FR leaking causes for channel-type reactors. The causes identified are design-, fabrication- and operation-related ones [1].

Over the past years, this approach has been effectively used for the analysis of WWER FR leaking events [2]. Using this method is particularly effective for the analysis of fuel element leaking causes when using the expensive examination of fuel assemblies (FAs) in shielded "hot" cells is not possible. Due to this method, it was possible to find and to explain FR leaking causes due to the excursions of water chemistry regime and appearance of slime, resins and dispersed corrosion product particles in coolant. The method was successively used in analysis of occurrences happened in Unit-2 of Kola NPP in 1991-1992 [1,3,4,5], in Loviisa NPP in 1994-1995, in Paks NPP in 1997, in Unit-4 of Novovoronez NPP in 1998.

Due to the use of probabilistic-statistical analysis of fuel element leaking causes, the specific mechanism of FR leaking was analyzed and described. This mechanism is dealt with mass transfer in the coolant turbulent flow and deposition of the dispersed particles (particles with specific inert and gravitational characteristics) on FR surface. Such particles might be corrosion products of the primary circuit, particles resulted from detraction and destruction of foreign objects in the primary circuit (a tear away mechanical details, welding "hail", accidentally forgotten wooden and organic details during the repair works, filter resin disintegration particles), as well as particles forming during chemical cleaning and not removed from coolant. Alongside with local pit corrosion, the largest dispersed particles can result in fuel rod cladding debris-type damage.

2. INPUT DATA FOR STATISTICAL ANALYSIS

Information not only on FAs with leaking FRs, but also on all similar FAs in a core (which were fabricated and operated at the same time as leaky FAs) is used as input data. Input data for the analysis are as following:

- Results of cladding tightness inspection in special canister (stand test in the cooling pool) for discharged FAs;
- Data on the isotope composition of the primary coolant during the cycle before fuel discharge;
- Data on the location of all analyzed FAs in the core for the whole time of their operation, including all reshuffles and coolant flow restrictions;
- Data on power density distribution and fuel burnup in the core for the whole time of operation for all analyzed FAs;
- Data on operation of similar, with regard to the date and technology of fabrication, FAs at other units of the same NPP and other NPPs with similar reactors;
- Information on the water chemistry regimes for the whole time of operation of all analyzed FAs, including detailed information on water chemistry during the first cycle;
- Information on reactor main parameters for the whole time of FA operation in the core and parameters specific to the analyzed assembly (in comparison with specifications), as well as pressure drop in the reactor, in main circulation pumps and steam generators, outlet coolant temperature on output from FA during the whole time of operation;
- Passport data on all analyzed FAs (serial numbers, date of fabrication, enrichment and etc).

3. ESSENCE OF PROBABILISTIC-STATISTICAL ANALYSIS METHOD OF FUEL ROD LEAKING CAUSES

The method is based on using probabilistic-statistical system structural-dynamic approach to the FR leaking causes analysis for WWER type reactors. Earlier this approach was used for the analysis of RBMK FR leaking causes and demonstrated its high efficiency.

The method takes into account that fuel element depressurization process is multistage and includes the following stages:

- Appearance of the FR cladding defect;
- Propagation of the defect till the opposite cladding surface;
- Transformation of the initial microdefect into the macrodefect;
- Appearance and growth of the secondary wall-through defects;
- Quick growth of the secondary defects till cladding failure.

Herewith, each stage of the FR leaking process is governed by specific physical mechanism and, accordingly, has own prevailing factors. New causes-factors, stipulating in the whole FA failure, are added on each new stage a cladding leaking. Root causes of a FR leaking define processes of generation and wall-through defect propagation. This is why several functions characterizing different stages of cladding defect growth are used for the analysis of fuel element leaking causes, including:

- Probability of FA operation without failure, intensities of failures and flows of FA failures on leaking functions (4 and 5 stages);

- Probability of FA operation without fuel element leakage, intensities of finding FAs with leaking FRs and flows of finding FAs with leaking FRs (2 and 3 stage);
- Normalized steady-state primary coolant specific activities of iodine radionuclides corresponding to different types of cladding defects (gas leaking - 2-3 stage, direct contact of fuel and coolant - 4-5 stage);
- Probabilities of FR leaking in loaded FAs by cycles (1 stage).

Systematic approach is used in this method, including analysis of the following systems:

- FA with its structural components, which changes define the design-related causes of FA damage;
- A technological system defining initial properties of as-fabricated FA and associated failure causes;
- A reactor, its elements (including FAs) and subsystems (including system of interaction of personnel and reactor influencing upon operational conditions) defining operation causes of FR leaking.

Interaction of these tree above-mentioned systems defines FA in-core performance.

At the analysis of system influence on the fuel element leaking processes, the impact of structural relatively stable factors (defined by the stability of elements and subsystems), and dynamic variable-factors (defined by the links between elements and subsystems) is analyzed.

Parameters defining FR leaking causes are determined by investigating correlations between response functions (effects) with structural relatively stable and dynamically changeable variable parameters including initial properties and FR operational conditions.

For revealing of the factors corresponding to direct and root causes of fuel element leaking is produce collation of parameters, defining causes of fuel element leaking, with the kit of parameters, defining known mechanisms of fuel element leaking:

(a) In accordance with design:

- Creep down of FR claddings and densification of fuel pellets;
- Fretting-corrosion of FR;
- Growth and bow of fuel rods in fuel assemblies;

(b) In accordance with fabrication:

- Internal local hydriding;
- Improper enrichment and rod filling with pellets;
- Welding defects;

(c) In accordance with operation:

- Waterside corrosion of fuel element claddings;
- Waterside corrosion of claddings accelerated by a corrosion product crud on fuel elements;
- Debris-fretting;

- A mechanical interaction of fuel with the cladding aggravated by the inner face cladding interaction with fuel fission aggressive products.

Mechanisms of fuel element leaking dealt with operation conditions are defined by all variety of factors, e.g. operation parameters, fuel element fabrication parameters, and design features. Analysis allows to define prevailing factors.

4. LEAKING MECHANISM DEALT WITH THE PRIMARY CIRCUIT WATER CHEMISTRY

Very specific fuel failure mechanism was revealed and described thanks to the use of the probabilistic-statistical analysis of fuel element leaking causes. This mechanism is connected with mass transfer in the coolant turbulent flow and deposition of the dispersed particles on fuel elements. Such particles might be corrosion products of the primary circuit, particles resulted from detraction and destruction of foreign objects in the primary circuit (a tear away mechanical details, welding "hail", accidentally forgotten wooden and organic details during the repair works, filter resin disintegration particles), as well as particles forming during chemical cleaning and not removed from coolant. In WWER reactors dispersed corrosion product may appear in the core as result of crud (on steam generator side) washout due to the pH change.

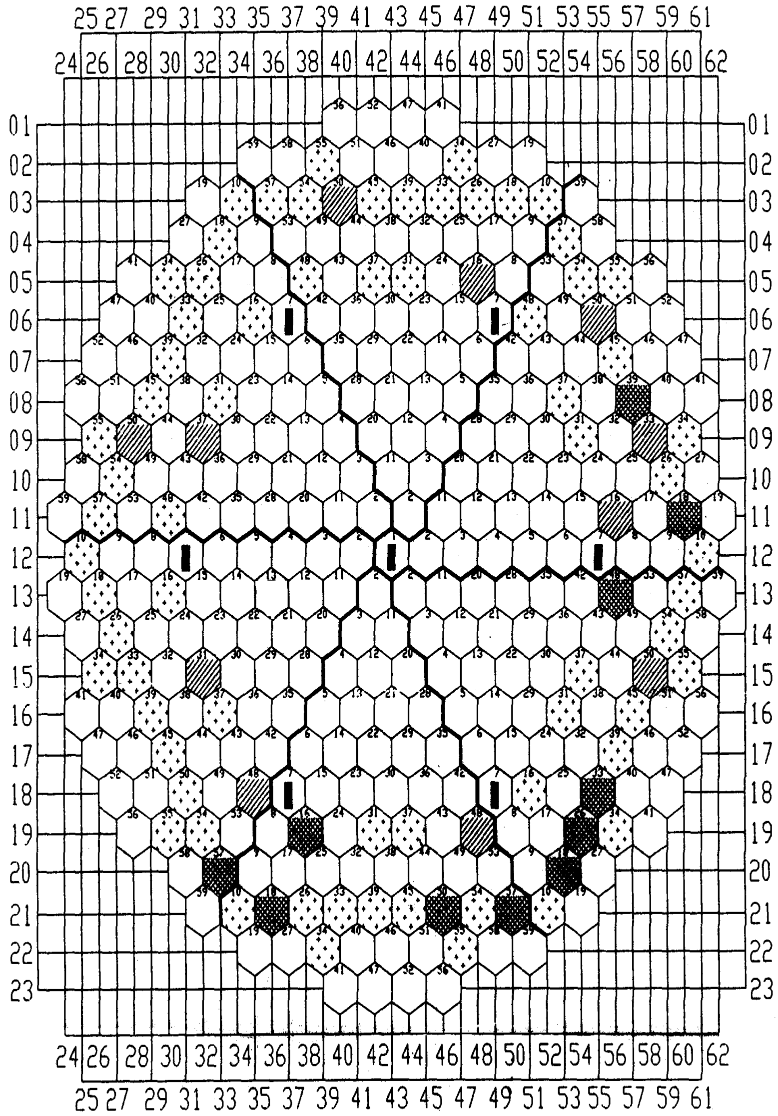
Dispersed particles due to their gravitation and inert characteristics may be redistributed in a primary coolant flow depending on its parameters. This redistribution is particularly essential at the entering to the reactor vessel, on the bottom of the vessel and in inlet to the fuel assemblies, when horizontal and vertical forming coolant velocities might be significantly changed. This is why a concentration of dispersed particles inside FAs depends on their location in a core.

Calculations have shown that, due to the specifics of WWER core design, ingress of such particles into FAs situated in the core center and periphery is of low probability. At the same time, these particles are concentrated in the ring between core center and periphery. Analysis of the location of leaking FAs in a core during the first cycle has confirmed this point of view. The ring may be a complete one, if there is a global violation of water chemistry regime, or only a part of it if particles enter into a core only from the specific steam generators or main circulation pumps. (See example on Fig.1).

The crud deposition theory tells that dispersed particles are to deposit on fuel element cladding surfaces in the areas with the increased coolant flow turbulization, namely in FA inlet and in the area of spacer grids. For depositing on cladding surface, particles should overpass wall buffer and laminar frontier coolant layers. In areas with the increased flow turbulence, thickness of frontier layers is below, and additional transverse forming flow turbulence that defining transverse velocity of dispersed particles is higher.

As far as dispersed particles have dimension of about 10-30 microns, when precipitating on a develop fuel element surfaces they cause some cladding raggedness. As a result of significant increase of a hydraulic resistance in the above-mentioned parts of fuel assemblies. In turn, this may result in redistribution of coolant flows in a core.

Dispersed particles are absorbed, to a significant extent, on surfaces of freshly loaded fuel elements. This is due to the fact that defensive oxide film on cladding surface has not yet reached its equilibrium thickness. Deposition of such particles may result in local cladding corrosion. This is bound as with reduction of defensive characteristics an oxide film, so and with formation of the porous crud resulted in so-called "wick" effect. This effect consists of concentration of aggressive water admixtures (chloride ions, alkali, etc.) in pores. This is why special attention should be paid to the maintaining a good water chemistry regime mode at start-up periods after loading new fuel assemblies in a reactor core. The biggest dispersed particles may cause debris-type failure.



Conditional indications

Fuel assemblies of party, first fuelling before 21-fuel lifetime, with leaktightness fuel elements on finish 23-fuel lifetime

Fuel assemblies of party, first fuelling before 21-fuel lifetime, with leaking fuel elements discovered on finish 23-fuel lifetime with leakage degree on 131-I more 2.0E-5 Ci/kg

Fuel assemblies of party, first fuelling before 21-fuel lifetime, with leaking fuel elements discovered on finish 23-fuel lifetime with leakage degree on 131-I less 2.0E-5 Ci/kg

FIG. 1. WWER-440 core cartogram for the 21st fuel cycle.

It is noteworthy that duration of corrosion process till formation of through-wall holes varies usually from one to three and more years depending on amount of dispersed particles in water. This is why the root causes of fuel element leaking and leakage itself, with release of fission products into coolant, are separated in time. At the same time, debris-fretting failure develops usually during one fuel cycle.

5. THE FUEL RELIABILITY INDICATOR FOR NPPS WITH WWER, PWR AND PHWR REACTORS

WANO fuel reliability indicator is used for comparison of fuel reliability at different reactors/units. The purpose of the fuel reliability indicator is to monitor industry progress in achieving and maintaining high fuel integrity, and to foster a healthy respect for preservation of fuel integrity. Failed fuel represents a breach in the initial barrier preventing off-site release of fission products.

The fuel reliability indicator (FRI) is inferred from fission product activities present in the reactor coolant. For PWRs, PHWRs and WWERs, the indicator is defined as the steady-state primary coolant iodine-131 activity (Bq/g), corrected for the tramp uranium contribution and power level, and normalized to a common purification rate.

Steady state is defined as continuous operation for at least three days prior to data collection for at a power level that does not vary more than ±5 percent. In order to obtain an indicator value for a month, the steady state power at which data is collected must be 85 percent or greater. This ensures appropriate indicator accuracy. For months where no period of steady state power was 85 percent or greater, the highest steady state power achieved should be reported. Tramp contribution is caused by fissionable material that has been deposited on reactor core internals from previous defective fuel elements or is present on the surface of fuel elements from the manufacturing process. The fuel reliability indicator for periods longer than a month is determined as the average of the most recent operating quarter monthly values.

The monthly value of the PWR, WWER and PHWR indicator is calculated as the following equations:

$$\mathbf{FRI_P = [(A_{131})_N - k\,(A_{134})_N] * [(L_n/LHGR) * (100 / P_o)]^{1.5}} \qquad (1)$$

Where: $\mathbf{(A_{131})_N}$ is the average steady-state activity of $\mathbf{^{131}I}$ in the coolant normalized to a common purification rate and expressed in Bq/g;

k is the tramp correction coefficient (a constant with a value of 0.0318). This coefficient is based on a tramp material composition of 30 percent uranium and 70 percent plutonium;

$\mathbf{(A_{134})_N}$ is the average steady-state activity of $\mathbf{^{134}I}$ in the coolant normalized to a common purification rate and expressed in Bq/g;

$\mathbf{L_n}$ is the linear heat generation rate used as basis for normalization (18.0 kilowatts per meter);

LHGR is the average linear heat generation rate at 100 percent power (kilowatts per meter) for the unit;

$\mathbf{P_0}$ is the average reactor power (percent) at the time activities are measured.

If a calculated monthly indicator value for a unit is less than $3.7*10^{-2}$ Bq/g, it is replaced by the value $3.7*10^{-2}$ Bq/g.

The average steady-state activity of $\mathbf{^{131}I}$ and $\mathbf{^{134}I}$ in the coolant normalized to a common purification rate is calculated as follows:

$$\mathbf{A_N}(\boldsymbol{i}) = \mathbf{A_M}(\boldsymbol{i}) * (\lambda_i + \mathbf{B_a}) / (\lambda_i + \mathbf{B_n}) \qquad (2)$$

Where: $\mathbf{A_N}(\boldsymbol{i})$ is the average steady-state isotopic activity of "***i***" nuclide in the coolant normalized to a common purification rate and expressed in Bq/g;

$\mathbf{A_M}(\boldsymbol{i})$ is the average measured isotopic activity of "***i***" nuclide in the coolant (Bq/g);

λ_i is the decay constant of the "***i***" nuclide (seconds^{-1});

$\mathbf{B_n}$ is the purification rate constant equal $2*10^{-5}$ seconds^{-1} and taken assumed for unity of normalization;

$\mathbf{B_a}$ is the actual purification rate constant (seconds^{-1}) defined below:

$$\mathrm{B_a} = \sum_{i=1}^{a} \frac{G_i}{M} \bullet \left(1 - \frac{1}{K_i}\right) \qquad (3)$$

G_i is the letdown flow rate (kg/sec) of the "*i*" system filters corrected to normal reactor coolant system operating temperature

M is the reactor coolant mass at normal operating temperature (excluding the pressurizer) and expressed in kg;

$K_i = A_i / A_{oi}$ is the clearing coefficient for iodine nuclides in "*i*" system filters

A_i is the iodine radionuclide specific activity upstream of the "*i*" system filters in Becquerels/gram;

A_{oi} is the iodine radionuclide specific activity downstream of the "*i*" system filters in Bq/g.

The results of WANO fuel reliability indicator (FRI) calculation of arithmetic mean values on the indicated procedure for units with WWER-440, WWER-1000, PWR and PHWR reactors are represented in Figure 2. These results are grounded on the information of the WANO Atlanta Center database. It is necessary to mark, that in the given report the results of calculation of arithmetic mean values of fuel reliability indicator, instead of median, used WANO are represented.

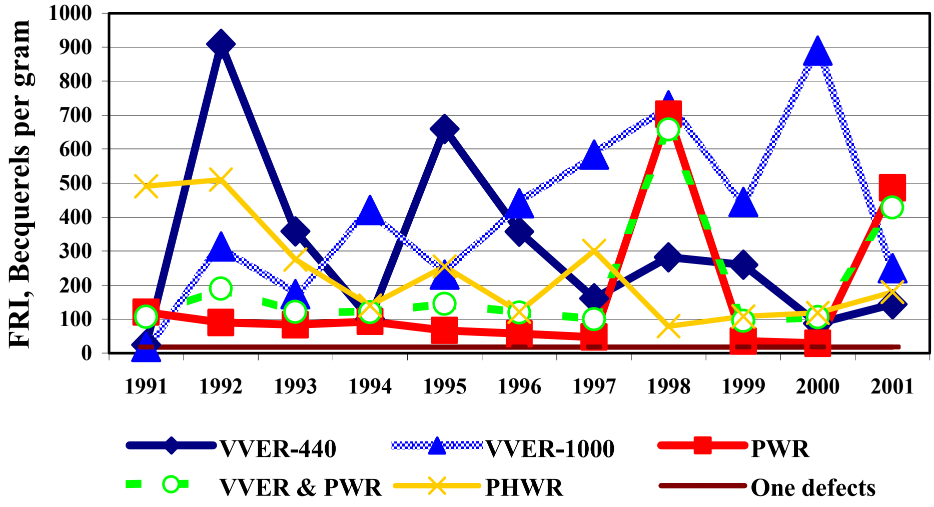


FIG. 2. Average values of the WANO fuel reliability indicator at NPPs with WWER, PWR and PHWR reactors.

For units with WWER-440, PWR and PHWR reactors the fuel reliability augmentation tendency on an FRI in 1991-2001 is scored. For units with WWER-1000 reactors, the FRI value in 1991-2001 is a little bit worse. At the same time, for all numbered types of water-cooled reactors the considerable differences of FRI average value per miscellaneous years are scored.

A reactor core containing one or more fuel rod defects is likely to produce indicator values (under steady-state conditions) greater than 19 Bq/g. The FRI average values in miscellaneous countries in 2001 and share of units in them (%) having leaking fuel rods (FRI > 19 Bq/g) are represented in Figure 3.

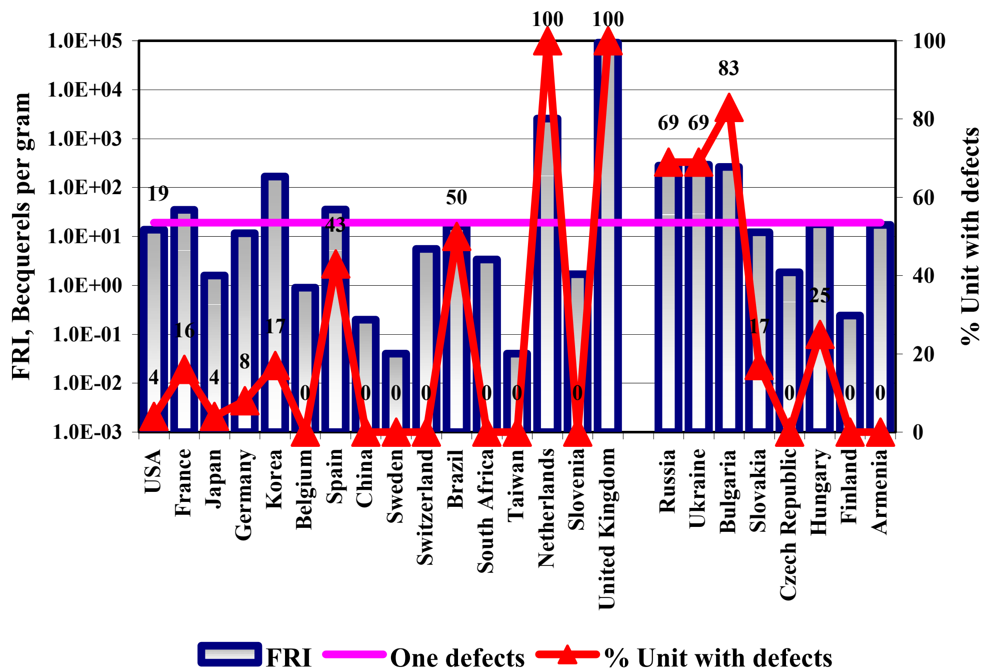


Fig. 3. Average values of the WANO fuel reliability indicator in 2001 for countries with WWER, PWR and PHWR reactors and share of units with leaking fuel rods (FRI > 19 Bq/g).

An attempt to understand considerable differences in FRI values in different countries operating WWER type reactors has been undertaken abd it is described in this paper.

1st and 2nd units with the WWER-440 of B-230 design and 3rd and 4th units with the WWER-440 of B-213 design are in operation at Bohunice NPP (Slovakia). For units 3 and 4 FRI values were lower than 19 Bq/g (Fig.4). It confirms absence of leaking fuel rods. In too time on units 1 and 2 with B-230 design reactors in batches occurred leaking fuel rods. On unit 1 the fuel rod leakage has given in FRI magnification above than 19 Bq/g in 1996-1997 (1996 - 219 Bq/g) and in 1999 (68 Bq/g). On unit 2 the fuel rod leakage is marked in 1991-1993 (maximum of 389 Bq/g in 1992) and in 1996-1998 (maximum of 269 Bq/g in 1997).

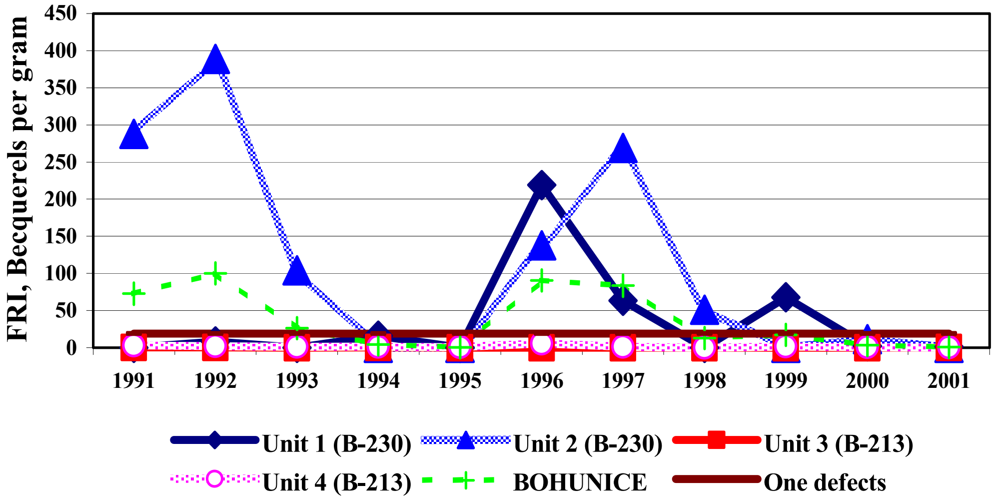


FIG. 4. Fuel reliability indicator values for Bohunice NPP (Slovakia) with WWER-440 reactors of B-230 and B-213 designs.

Four units of the WWER-440 of B-213 design are in operation at Dukovany NPP (Czech Republic). The FRI values more than 19 Bq/g speaking about availability leaking fuel rods was scored as given below (Fig.5):

- Unit 1 - in 1994-1995 (maximum of 114 Bq/g in 1995);
- Unit 2 - in 1993-1994 (maximum of 109 Bq/g in 1993);
- Unit 3 - in 1994 (24 Bq/g);
- Unit 4 - in 1993-1994 (maximum of 69 Bq/g in 1994).

Thus, the worst FRI values for Dukovany NPP (114 Bq/g, Unit 1 in 1995) was significantly lower than for Bohunice NPP (389 Bq/g, Unit 2 in1992 and 219 Bq/g, Unit 1 in 1996), that testifies to smaller amount of leaking fuel rods.

Four units of the WWER-440 of B-213 design are in operation in Paks, Hungary. The FRI values, higher than 19 Bq/g, justifying availability of leaking fuel rods, was scored as given below (Fig.6):

- Unit 1 - in 1992 (30 Bq/g);
- Unit 2 - in 1997 (20 Bq/g);
- Unit 3 - in 2001 (69 Bq/g);
- Unit 4 – in 1991 (60 Bq/g) and in 1994-1995 (maximum of 85 Bq/g in 1994).

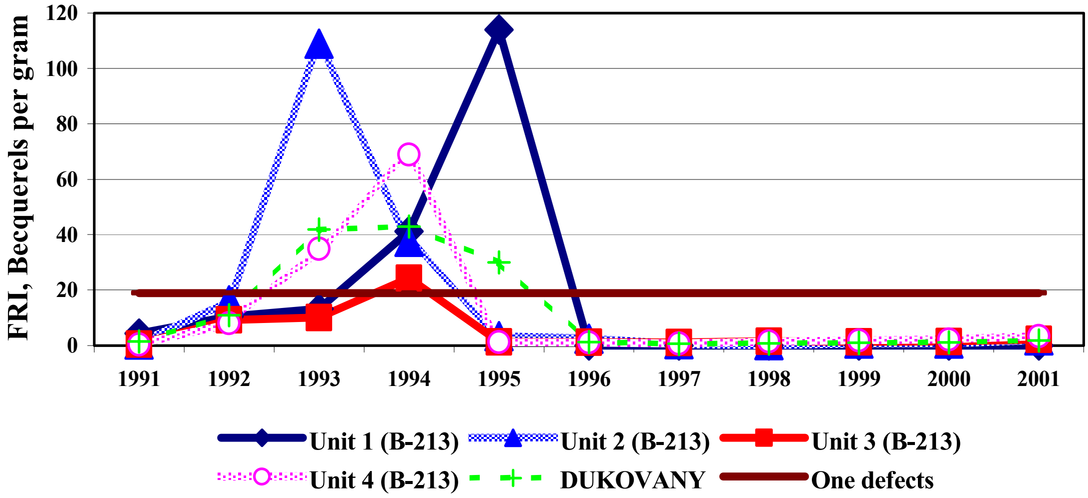


FIG. 5. Fuel reliability indicator values on Dukovany NPP (Czech Republic) with WWER-440 reactorsB-213 design.

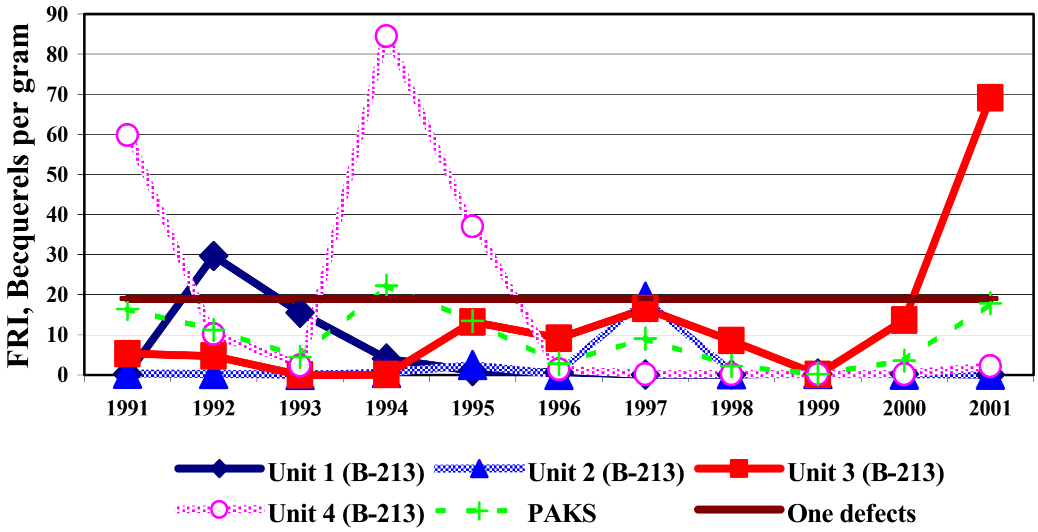


FIG. 6. Fuel reliability indicator values for Paks NPP (Hungary) with WWER-440 reactors of B-213 design.

At the beginning of the 14-th fuel cycle in September 1997, at the Unit 2 of NPP Paks during power increase the temperature measurements at the outlet of the fuel assemblies (FA) indicated anomaly of temperature distribution. Outlet temperature near loops No.2 and 3 was rather higher than in the remaining area. The reactor could not work on a rated power, as technical specifications limitative value of temperature in an outlet of most weighted fuel assemblies was reached at power 95%, and the anomaly developed the tendency to sluggish propagation. The reason of asymmetry was clogging of several fuel assemblies by an accumulated precipitation of corrosion products. Erosion caused by a foreign material in system and chemical impact during a decontamination have caused a damage of defensive oxide layer on interior surfaces of a primary loop, and then high concentration of dissolved metal ions in the coolant. The deposition of these ions has resulted in accumulation of slimes in fuel assemblies and partial clogging.

After necessary preparation, the unit was shut down for refueling outage 5 months before the schedule. All fuel assemblies in the core were discharged and cleaning of the primary circuit was carried out. Reactor started to operate with replaced fuel. These measures resulted in the further operation without fuel failure.

The FRI value on NPP Paks indicates high fuel rod reliability, with appearance of only a few rods with gas untightness.

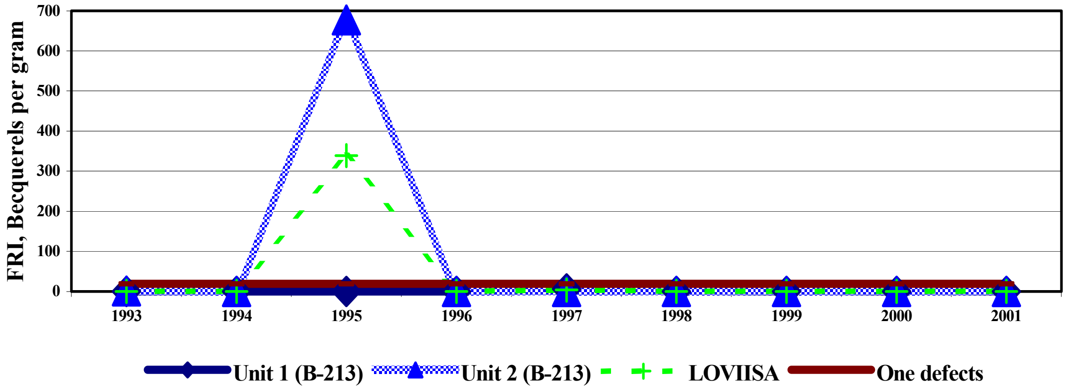


FIG. 7. Fuel reliability indicator values on Loviisa NPP (Finland) with WWER-440 reactors of B-213 design.

At Loviisa NPP (Finland) with two units of the WWER-440 of B-213 design in 1991-2001 the FRI values higher than 19 Bq/g were noticed only in 1995 (Fig.7) at the unit 2 (677 Bq/g). The cause of fuel rod leaking was bound in this case to the corrosion products remained in the coolant after a decontamination of a primary loop in 1994. Decontamination was made using the CORD-method (Chemical Oxidating Reducing Decontamination) developed by Siemens AG. The total surface area involved in the process was about 17 000 m^2, and totally 292 kilograms of iron, chromium and nickel were removed during the four cycles of the CORD-process. This case, as well as NPP Paks case, has resulted in sags of the coolant flow and temperature rise at output of fuel assemblies.

Two units (Units 1 and 2) of the WWER-440 of B-230 design and two units (Units 3 and 4) of the WWER-440 of B-213 design are in operation at Kola NPP (Russia). For units 3 and 4 FRI values in 1991-2001 were lower than 19 Bq/g. The values lower than 100 Bq/g were scored (Fig.8):

- Unit 3 in 1992-1995 (maximal in 1995 - 795 Bq/g) and in 1999 (201 Bq/g);
- Unit 4 in 1992-1995 (maximal in 1993 - 334 Bq/g).

As for other years, FRI values for Units 3 and 4 varied from 40 to 92 Bq/g, that evidenced availability of a very few rods with gas untightness and on high fuel operational reliability at these Units. At the same time FRI values for Units 1 and 2 varied from 321 to 15900 Bq/g. At these units FRI values periodically raised higher 1000 Bq/g:

- Unit 1 in 1992 (15900 Bq/g), in 1995-1997 (6600 Bq/g in 1995) and in 1999 (1220 Bq/g);
- Unit 2 in 1992-1995 (maximal in 1993 - 6820 Bq/g), in 1998-1999 (2820 Bq/g in 1998) and in 2001 (2670 Bq/g).

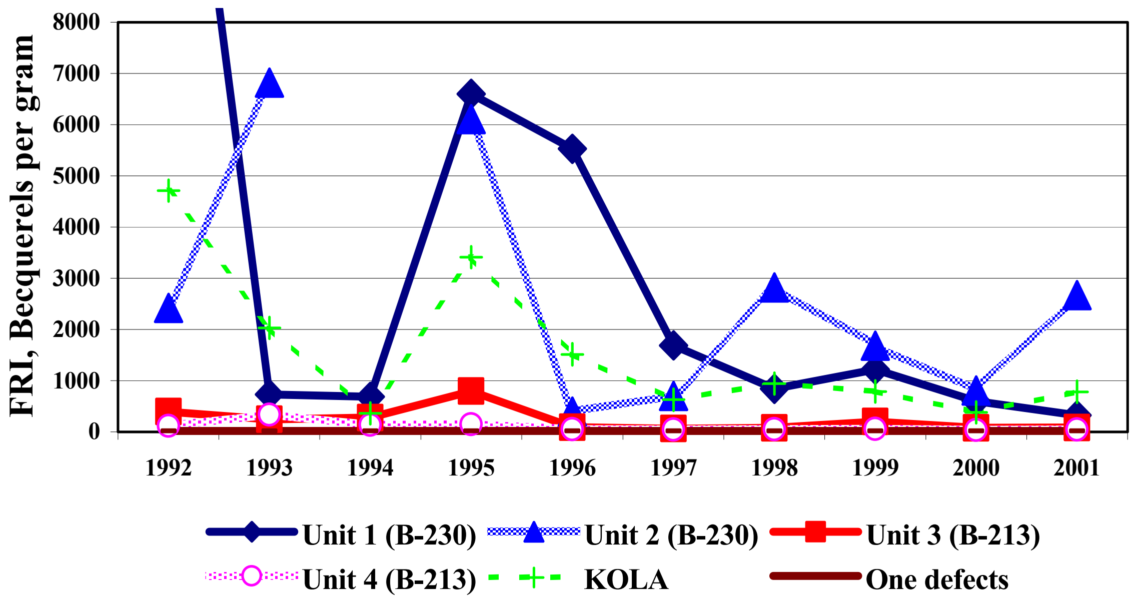


FIG. 8. Fuel reliability indicator values for Kola NPP (Russia) with WWER-440 reactors.

It indicates the availability of fuel rods with direct contact of fuel and coolant in a core. The analysis showed that activity rising was due to the appearance of foreign objects in the Unit 2 core. Two fuel failure mechanisms were noticed:

- Local pit corrosion under crud deposits;
- Debris-fretting initiated by foreign particles.

At Novovoronezh NPP (Russia) two units with WWER-440 reactors of the B-179 design (units 3 and 4) and one unit with a WWER-1000 reactor of the B-187 design (unit 5) are now in operation. The FRI value higher than 19 Bq/g was scored per following years (Fig.9):

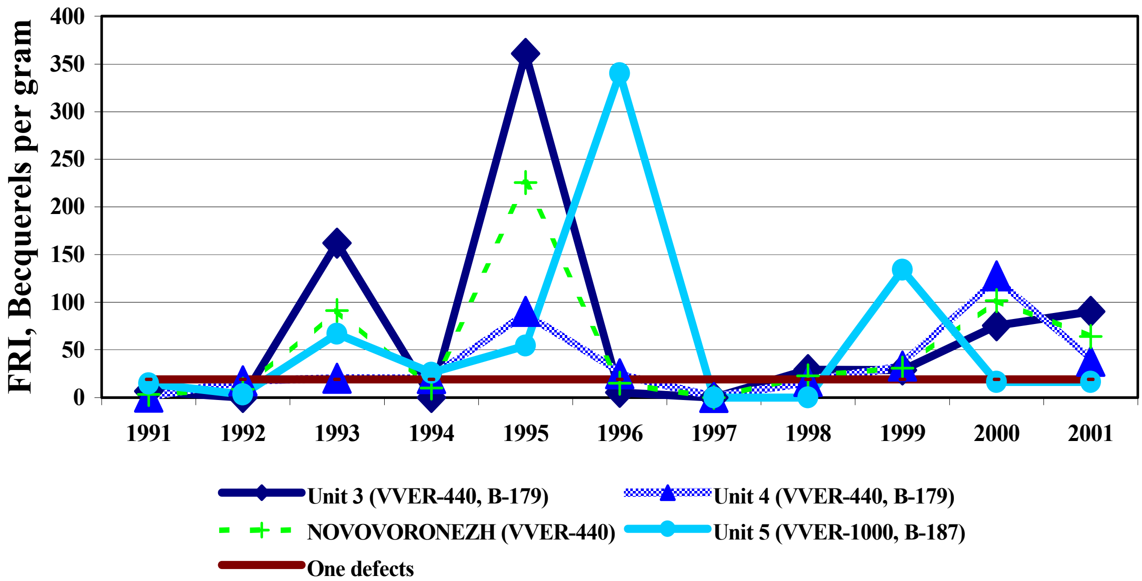


FIG. 9. Fuel reliability indicator values for Novovoronezh NPP (Russia) with WWER-440 reactors of B-179 design and WWER-1000 reactor of B-187 design.

- Unit 3 in 1993 (162 Bq/g), in 1995 (361 Bq/g) and in 1998-2001 (maximum 90 Bq/g in 2001);
- Unit 4 in 1993-1996 (maximum of 90 Bq/g in 1995) and in 1999-2001 (maximum of 127 Bq/g in 2000);
- Unit 5 in 1993-1996 (maximum of 340 Bq/g in 1996) and in1999 (134 Bq/g).

At Kozloduy NPP (Bulgaria) four units with the WWER-440 of B-230 design (units 1-4) and two units with the WWER-1000 of B-320 design (units 5 and 6) are now in operation. The FRI values higher than 19 Bq/g were scored per following years (Fig.10):

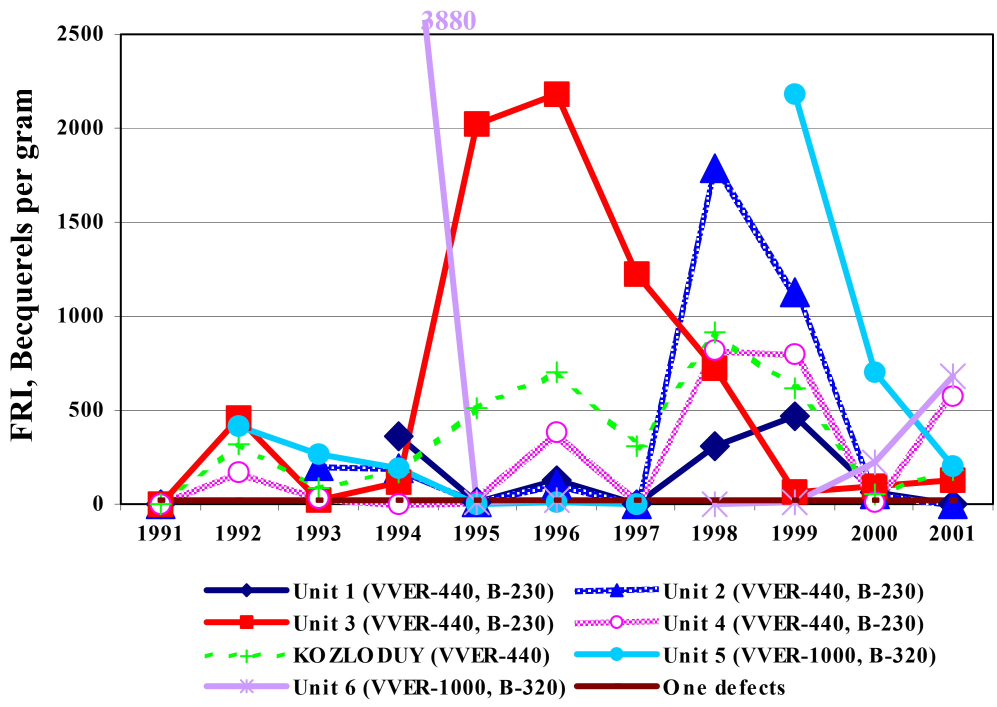


FIG. 10. Fuel reliability indicator values for Kozloduy NPP (Bulgaria) with WWER-440 reactors of B-230 design and WWER-1000 reactors of B-320 designs.

- Unit 1 in 1994 (361 Bq/g), in 1996 (126 Bq/g) and in 1998-2000 (maximum 471 Bq/g in 1999);
- Unit 2 in 1993-1994 (maximum of 204 Bq/g in 1993), in 1996 (107 Bq/g) and in 1998-2000 (maximum of 1790 Bq/g in 1998);
- Unit 3 in 1992-2001 (maximums in 1992 - 461 Bq/g, in1996 - 2180 Bq/g and in 2001 - 130 Bq/g);
- Unit 4 in 1992-1993 (maximum of 170 Bq/g in 1992), in 1996 (380 Bq/g), in 1998-1999 (maximum of 817 Bq/g in 1998) and in 2001 (574 Bq/g)
- Unit 5 in 1992-1994 (maximum of 416 Bq/g in 1992) and in 1999-2001 (maximum of 2180 Bq/g in 1999);
- Unit 6 in 1994 (3880 Bq/g) and in 2000-2001 (maximum of 682 Bq/g in 2001).

At Rovno NPP (Ukraine) two units with WWER-440 reactors of the B-213 design (units 1 and 2) and one unit with a WWER-1000 reactor of the B-320 design (unit 3) are in operation. The FRI values higher than 19 Bq/g were scored per following years (Fig.11):

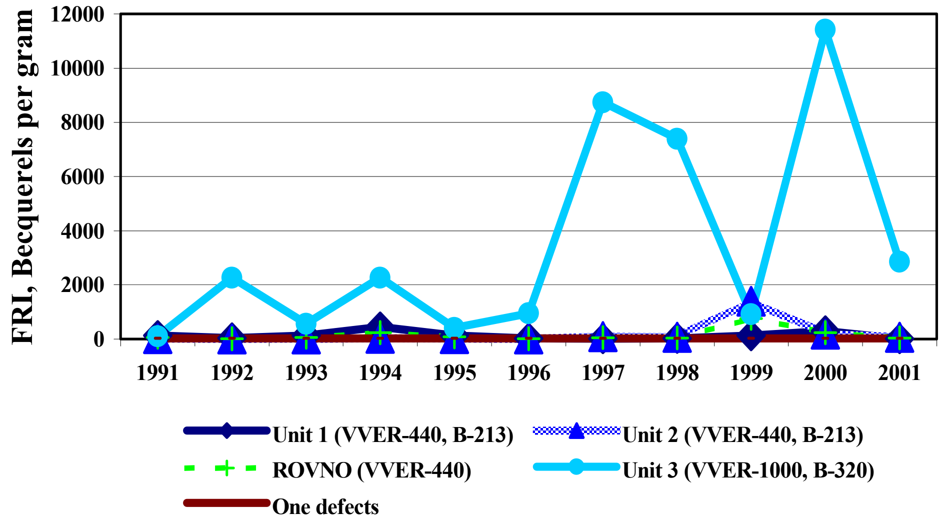


FIG. 11. Fuel reliability indicator values for Rovno NPP (Ukraine) with WWER-440 reactors of B-213 design and WWER-1000 reactor of B-320 design.

- Unit 1 in 1991-1996 (maximum of 446 Bq/g in 1994) and in 1991-1996 (maximum of 297 Bq/g in 2001);
- Unit 2 in 1994 (30 Bq/g) and in 1997-2001 (maximum of 1400 Bq/g in 1999);
- Unit 3 in 1991-2001 (maximums of 2260 Bq/g in 1992 and 1994, 8740 Bq/g in 1997 and 11417 Bq/g in 2000).

The case of maximum average value of FRI at Unit 3 of Rovno NPP might be explained by leaving in the core for this cycle the assembly with defect of fuel rod cladding of the type “direct contact of fuel and coolant”.

At Khmelnitski NPP (Ukraine) one unit with WWER-1000 reactor of the B-320 design is in operation. For this unit, FRI values higher than 19 Bq/g were scored per following years (Fig.12): in 1992-1993 (32 Bq/g in 1992); in 1995-1996 (3200 Bq/g in 1996); in 1998-2001 (maximum of 3110 Bq/g in 1999).

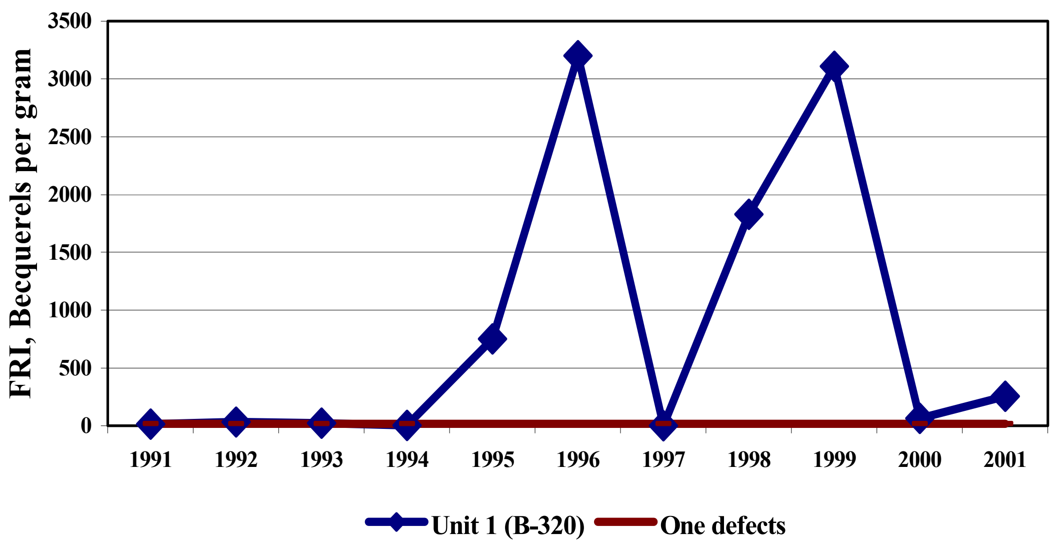


FIG. 12. Fuel reliability indicator values for Khmelnitski NPP (Ukraine) with WWER-1000.

At South Ukrainian NPP (Ukraine) three units with WWER-1000 reactors of the miscellaneous designs B-302 (Unit 1), B-338 (Unit 2) and B-320 (Unit 3) are in operation. The FRI values higher than 19 Bq/g were scored per following years (Fig.13):

- Unit 1 in 1992 (114 Bq/g), in 1994 (21 Bq/g), in 1996-1997 (169 Bq/g in 1997г.) and in 1999-2000 (803 Bq/g in 2000);
- Unit 2 in 1995-2001 (maximum of 2930 Bq/g in 1998);
- Unit 3 in 1994 (116 Bq/g) and in 2000-2001 (maximum of 972 Bq/g in 2000).

At Zaporozhye NPP (Ukraine) six units with WWER-1000 reactors of the B-320 design are in operation. The FRI value higher than 19 Bq/g were scored per following years (Fig.14):

- Unit 1 in 1992-1996 (maximum of 2490 Bq/g in 1995 and 1996) and in 2001 (41 Bq/g);
- Unit 2 in 1992 (630 Bq/g), in 1994-1996 (maximum of 46 Bq/g in 1995 and 1996) and in 2000 (1352 Bq/g);
- Unit 3 in 1992-1996 (maximum of 896 Bq/g in 1994) and in 2001 (160 Bq/g);
- Unit 4 in 1992-1993 (630 Bq/g in 1992) and in 2001 (187 Bq/g);
- Unit 5 in 1996 (169 Bq/g) and in 1998-1999 (maximum of 78 Bq/g in 1998);
- Unit 6 in 1998 (1250 Bq/g) and in 2000-2001 (maximum of 287 Bq/g in 2000).

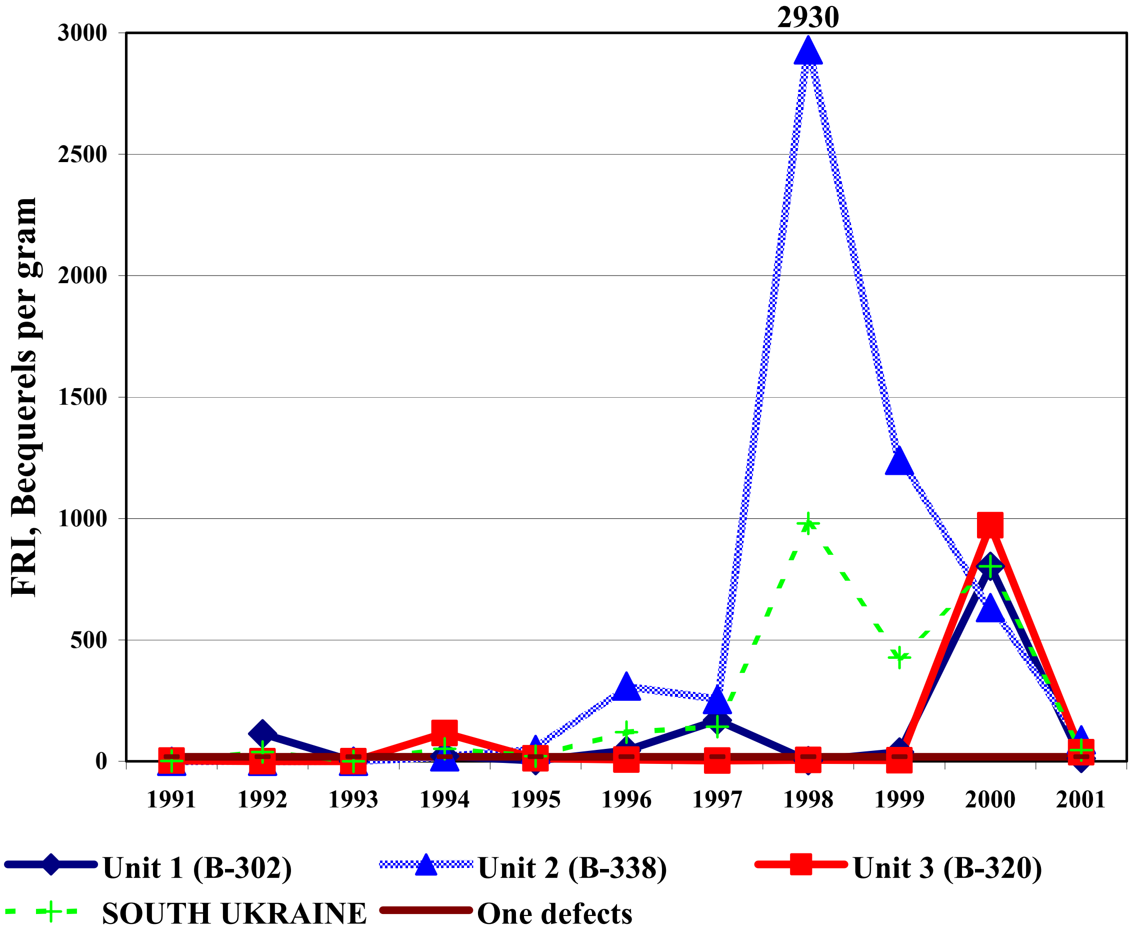


FIG. 13. Fuel reliability indicator values for South Ukraine NPP (Ukraine) with WWER-1000 reactors of B-302, B-338 and B-320 designs.

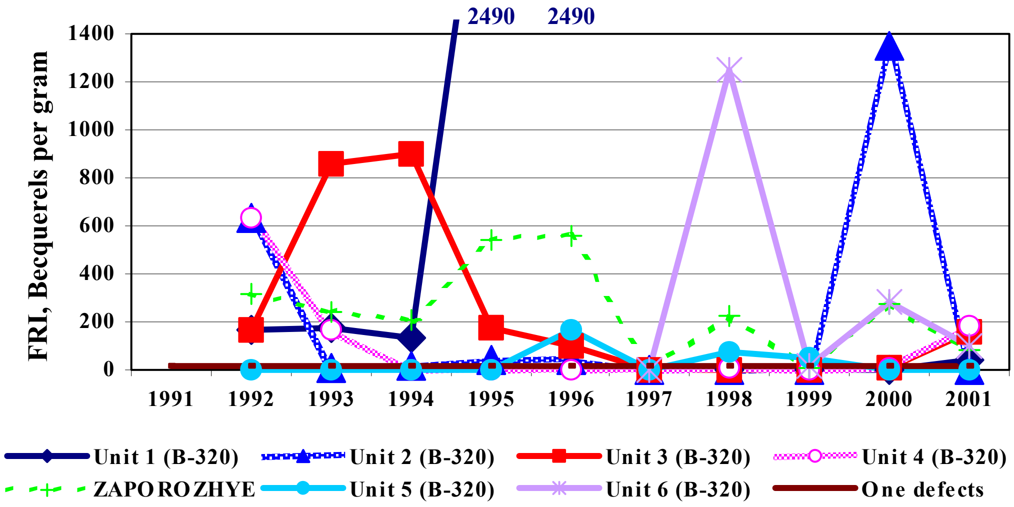


FIG. 14. Fuel reliability indicator values for Zaporozhye NPP (Ukraine) with WWER-1000 reactors of B-320 design.

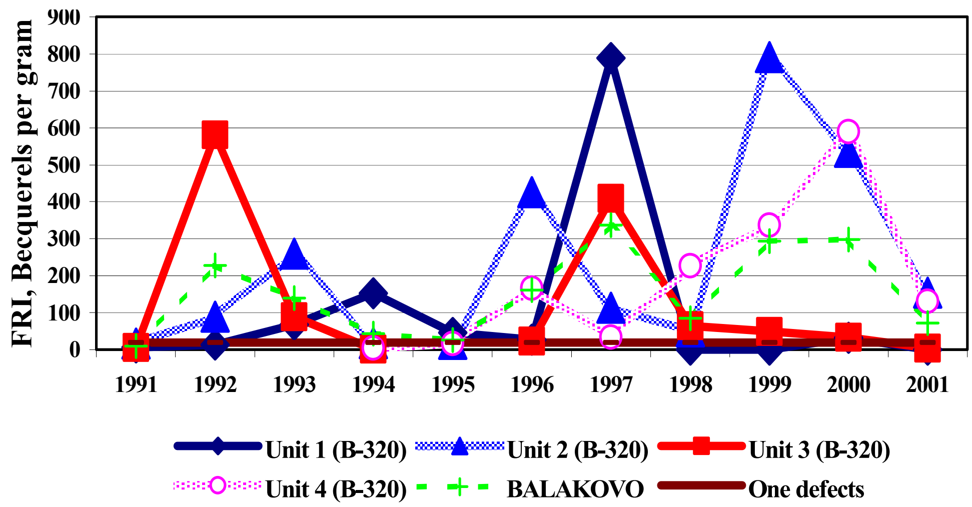


FIG. 15. Fuel reliability indicator values for Balakovo NPP (Russia) with WWER-1000 reactors of B-320 design.

At Balakovo NPP (Russia) four units with WWER-1000 reactors of the B-320 design are in operation. The FRI value higher than 19 Bq/g were scored per following years (Fig.15):

- Unit 1 in 1993-1997 (maximum of 789 Bq/g in 1997) and in 2000 (32 Bq/g);
- Unit 2 in 1992-1993 (260 Bq/g in 1993) and in 1996-2001 (maximums of 426 Bq/g in 1996 and 791 Bq/g in 1999);
- Unit 3 in 1992-1993 (maximum of 581 Bq/g in 1992) and in 1996-2000 (maximum of 409 Bq/g in 1997);
- Unit 4 in 1996-2001 (maximums of 166 Bq/g in 1996 and 590 Bq/g in 2000).

At Kalinin NPP (Russia) two units with WWER-1000 reactors of the B-338 designs are in operation. The FRI value higher than 19 Bq/g was scored per following years (Fig.16):

- Unit 1 in 1991-1994 (maximum of 229 Bq/g in 1992) and in 1999-2000 (118 Bq/g in 2000);
- Unit 2 in 1993-1996 (maximum of 372 Bq/g in 1993), in 1999 (23 Bq/g) and in 2001 (72 Bq/g).

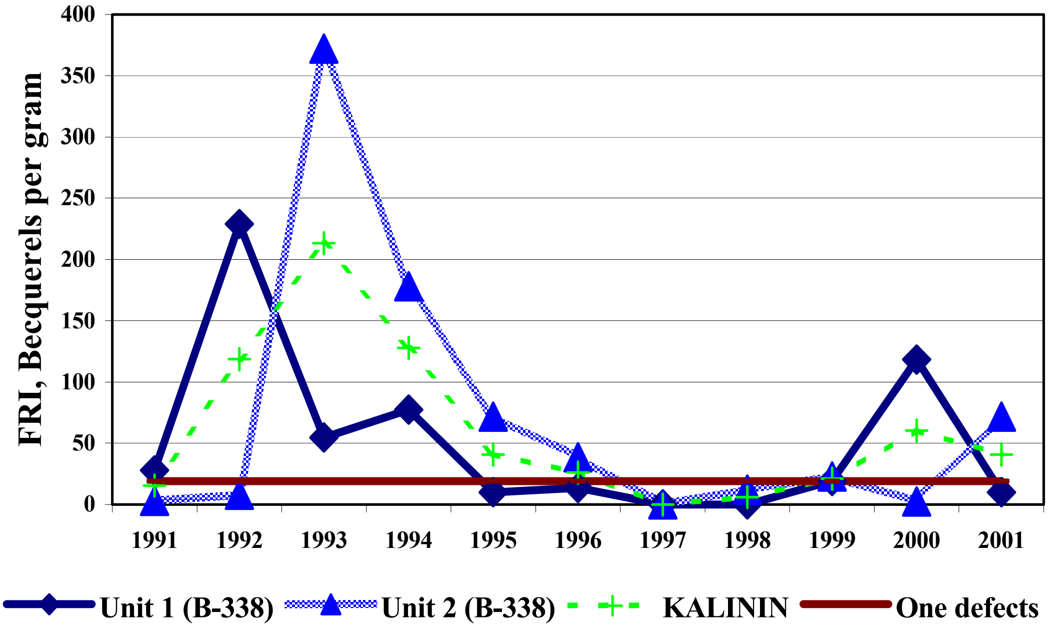


FIG. 16. Fuel reliability indicator values on Kalinin NPP (Russia) with WWER-1000 reactors of B-338 design.

6. CONCLUSIONS

The high level of WWER fuel reliability is noticed except of cases of heightened fuel failures in specific operating periods of some units. The analysis of these cases has shown, that they are caused by appearance of foreign particles in a reactor core. Two mechanisms of fuel failure were noticed:

- Local pit corrosion under crud deposits;
- Debris-fretting by foreign particles.

More often cases of increased fuel rod failure rates are scored for WWER-440 reactors of the old designs (B-179, B-230). Interior vessel surfaces of these reactors are made from carbon steel and do not have protective coating. It increases the probability of ingress of corrosion product 'slimes' into a primary coolant.

Prevention measures are recommended as following:

- Implementation of measures to improve the safety culture during maintenance and decontamination and better control of waterchemistry norms.
- Implementation of monitoring systems to control the size and concentration of dispersed impurity particles and corrosion products in the coolant.
- Installation of debris filter on fuel assemblies.

- Development and usage of means to clean primary coolant clearing from foreign objects and corrosion product ‘slimes’.

The developed probabilistic-statistical method of analysis of fuel element leaking causes has demonstrated its high effectiveness, especially when PIE of failed assemblies is impossible. The use of the probabilistic - statistical analysis for identification of fuel rod failure may save time, money and other utility’s resources.

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FUEL ASSEMBLY CHEMICAL CLEANING

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Abstract

One year ago Framatome Advanced Nuclear Power GmbH performed chemical cleaning of 170 fuel assemblies and conducted qualification measures for reuse of the fuel assemblies at the Hungarian Nuclear Power Plant Paks Unit 2 (WWER, 440 MW). The paper presents this experience.

1. INTRODUCTION

NPP Paks found a thermal-hydraulic anomaly in the reactor core during cycle 14 that was caused by corrosion product deposits on fuel assemblies (FAs) that increased the hydraulic resistance of the FAs. Consequently, the coolant flow through the FAs was insufficient resulting in a temperature asymmetry inside the reactor core. Based on this fact NPP Paks performed differential pressure measurements of all fuel assemblies in order to determine the hydraulic resistance and subsequently the limit values for the hydraulic acceptance of FAs to be used. Based on the hydraulic investigations a total number of 170 FAs was selected for cleaning.

The necessity for cleaning the FAs was explained by the fact that the FAs were subjected to a short term usage in the reactor core only maximum of 1,5 years and had still a capacity for additional 2 fuel cycles.

- The realization of cleaning has to be **cheaper than to buy** new FAs
- Δp of cleaned FAs have **to meet the hydraulic acceptance criterion**
- The cleaning concept has to **include qualification measures** for reuse of FAs:
  - – The confirmation of the cleanness of FAs with Δp
  - – The evidence of the integrity of FAs after cleaning

The prerequisites of the customer concerning the performance of cleaning were:

2. BOUNDARY CONDITIONS AND OVERALL CONCEPT

Essentially in context with the customer targets was the development of the overall concept. Certain boundary conditions had to be considered for this resulting in special requirements which had to be fulfilled, Fig. 1. The overall concept developed to meet the requirements is summarized briefly in Fig. 1.

Regarding the kind of deposits, which were corrosion products with a different quantity, the range of blocking of the FAs was between 10–65 %, a well-proven cleaning technology for safe dissolution of corrosion products was necessary. It also had to ensure that it would not cause additional damage to the FAs. The chemical cleaning process HP/CORD® UV was applied as cleaning technology.

The neutron flux and the high radiation of the FAs required consideration of the neutron reaction and of the high shielded environment of the FAs, respectively. Neutron reactions were taken into account by using boric acid as a cleaning solution. The shielded environment was realized by performing the chemical cleaning 14 m below borated water in pool no. 1.

With regard to waste treatment, the usage of NPP disposal systems had to be avoided. Therefore the minimization of waste was very important and no flushing water had to be used as a consequence. The waste was minimized by means of a bypass purification system (resins) and in situ decomposition of the chemicals.

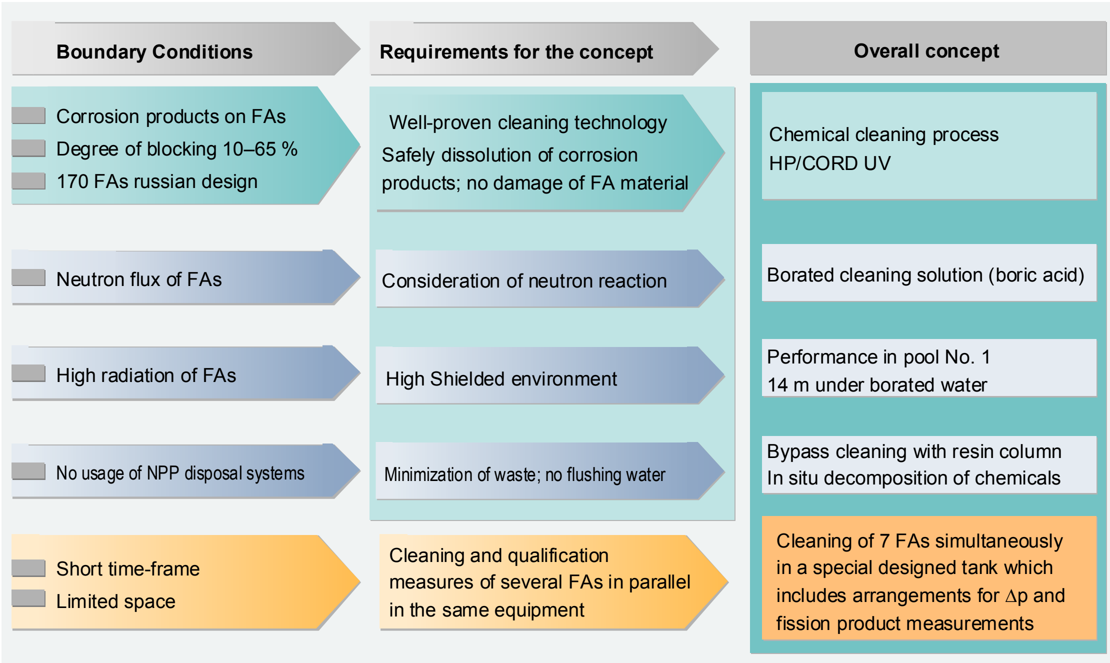


FIG. 1. Development of the overall concept.

Two additional points were of particular importance for consideration in the overall concept. Only a short time frame was available for cleaning the FAs, because NPP Paks planned reusing the FAs in the reactor core at the next outage.

Further the limited space in pool no. 1 that was available for installation of the equipment did not permit placement of separate measuring equipment in parallel to the cleaning equipment and the time frame for a post-measurement was insufficient. Therefore it was required to conduct the cleaning and qualification measures of several FAs in parallel by using the same equipment. Finally 7 FAs were simultaneously cleaned in a special designed tank in which the arrangements for differential pressure (Δp) and fission product measurements were included. Furthermore NPP Paks requested the performance of qualification tests for:

Cleaning process	Preliminary lab tests for material compatibility
Overall concept	Test cleaning of original FAs with original process engineering

The harmlessness of the chemical cleaning process HP/CORD® UV regarding the base material was not only proven by a multitude of laboratory tests but also by the practical long-term experience of more than 15 years of decon applications in NPPs. This has qualified the HP/CORD® UV process for component and system up to full system decontamination.

Independent of these facts, preliminary lab tests for material compatibility were carried out for the chemical cleaning of FAs at the NPP Paks. The results showed that the structural materials were compatible to the CORD process and that the HP/CORD® UV process would be the suitable process to dissolve the corrosion products from the FAs.

With regard to the overall concept a test cleaning of original FAs with original process engineering had to be performed to demonstrate that by applying of the overall concept, the fuel assemblies could be safely and reproducible cleaned and qualified.

Of particular importance was the approval of the concept and results obtained from the pretests, the test cleaning and subsequently the industrial cleaning by:

- Russian FA manufacturer MSZ
- Hungarian Atomic Energy Authority

3. PERFORMANCE OF CLEANING

3.1. Arrangement of cleaning equipment

The cleaning process of the FAs was carried out in the reactor hall of Unit 2 on the reactor podium.

The mobile decontamination system AMDA® (**A**utomated **M**odular/Mobile **D**econtamination **A**ppliance) represents the decontamination equipment used for cleaning the FAs.

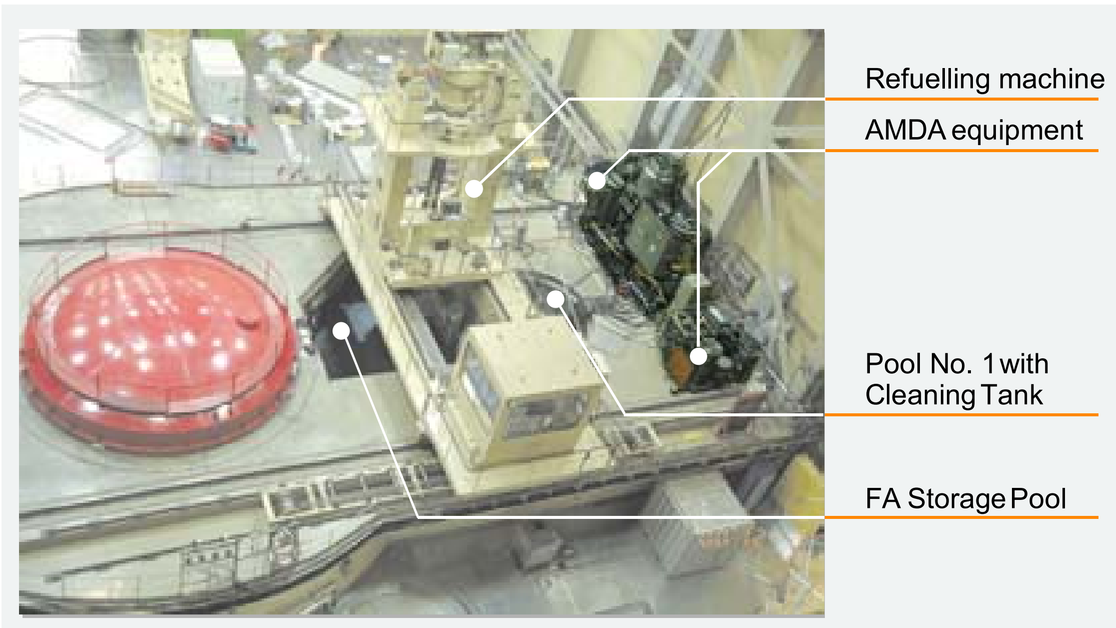


FIG. 2. Arrangement of cleaning equipment (AMDA) in the reactor hall of unit 2.

Fig. 2 gives an impression of the in situ situation for the performance of the cleaning. The AMDA equipment was positioned nearby the FA storage pool close to pool no. 1 in which the cleaning tank was inserted and connected with hoses to the AMDA. The controls of the AMDA were positioned one level below of the reactor podium in order to minimize personnel dose exposures.

The cleaning tank inside the pool was placed such that the individual tank positions could be approached by the refueling machine in automatic mode.

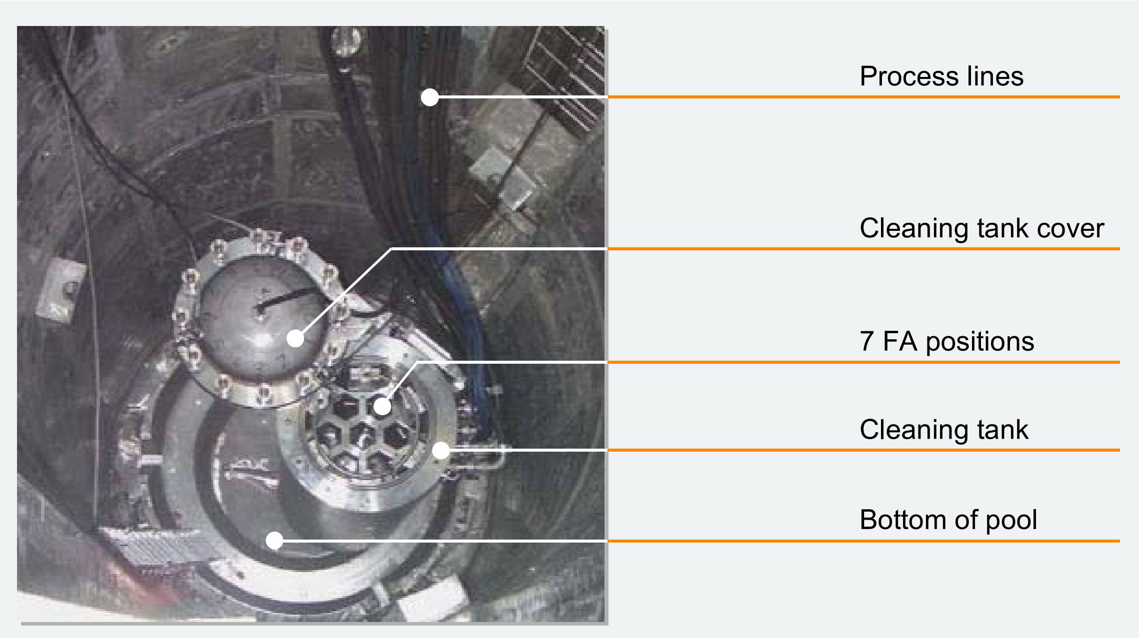


FIG. 3. Cleaning tank temporarily installed in pool no. 1.

Fig. 3 shows the cleaning tank temporarily installed at the bottom of the pool. Framatome ANP designed a special cleaning tank with 7 FAs positions to insert FAs of special Russian design (fig. 4). The cleaning tank was connected to the AMDA with process lines to feed the chemicals into the cleaning loop and consequently in the FAs. The cleaning tank remains inside the pool when opening and closing the tank cover. The prerequisite to open the tank inside the pool was that the water quality after the cleaning is comparable to the quality before cleaning.

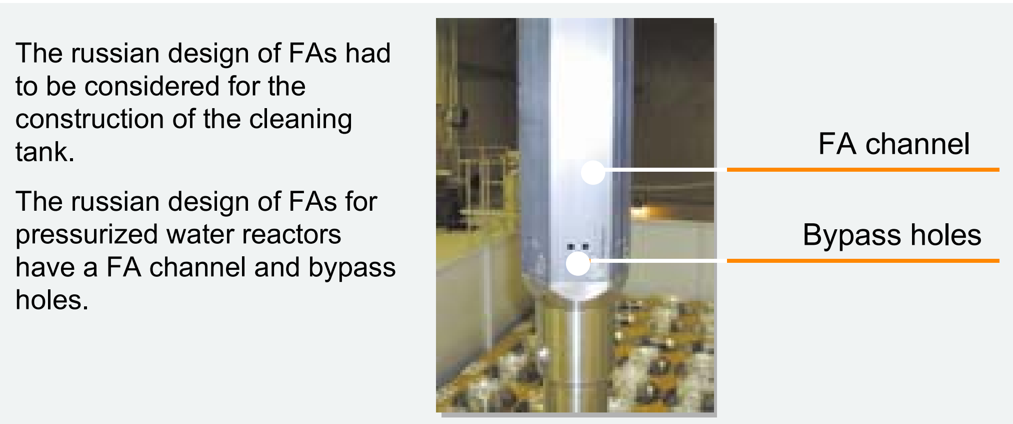


FIG. 4. Design of russian fuel assemblies.

3.2. Processes engineering

Fig. 5 shows schematically the cleaning tank positioned inside the pool and the interfacing cleaning circuit. The flow path inside the tank is illustrated for one FA. However, the tank permitted simultaneous cleaning of 7 fuel assemblies. Therefore, the tank has 7 inlet lines and one common return line.

Furthermore, the scheme illustrates the dimension of the tank in relation to the pool. The tank has a height of 4 m and is positioned 14 m under water.

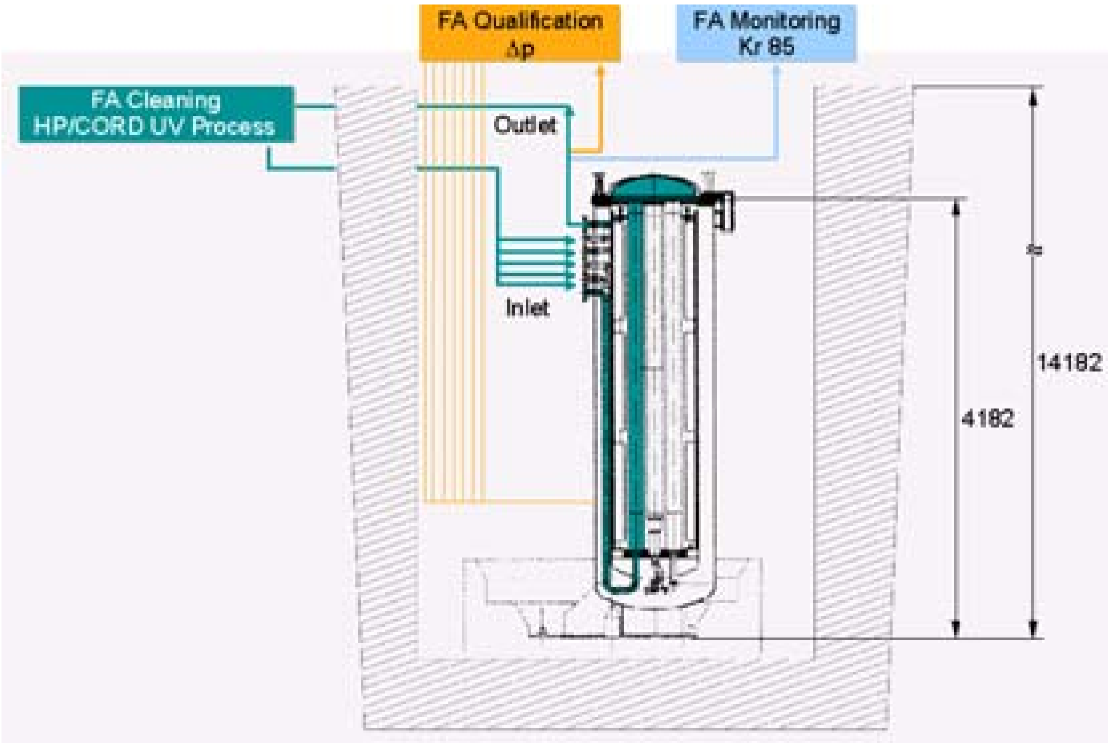


FIG. 5. Implementation of arrangements for qualification measurements.

A point of particular interest was the feasibility of qualification measurements inside the cleaning equipment.

Continuous monitoring of Kr85 was conducted in order to prove the integrity of the FAs. The fission product Kr85 was measured by using the exhaust air exiting the cleaning circuit in the outlet line.

The most important qualification measure was the performance of differential pressure measurements in order to prove the cleanness of FAs. Both measures were realized by means of the cleaning circuit just downstream of the cleaning tank.

3.3. Chemical cleaning process HP/CORD® UV

The cleaning process of the blocked FAs was carried out by using the well known and proven technology of HP/CORD® UV of Framatome ANP. The synonym HP/CORD UV stands for: **P**ermanganic acid (**H**) used for the pre-oxidation step of the **C**hemical **O**xidation **R**eduction **D**econtamination process and **U**ltra**V**iolet light utilized for in situ decomposition of the decon chemicals, subsequently. The principle of the process is illustrated in Fig. 6.

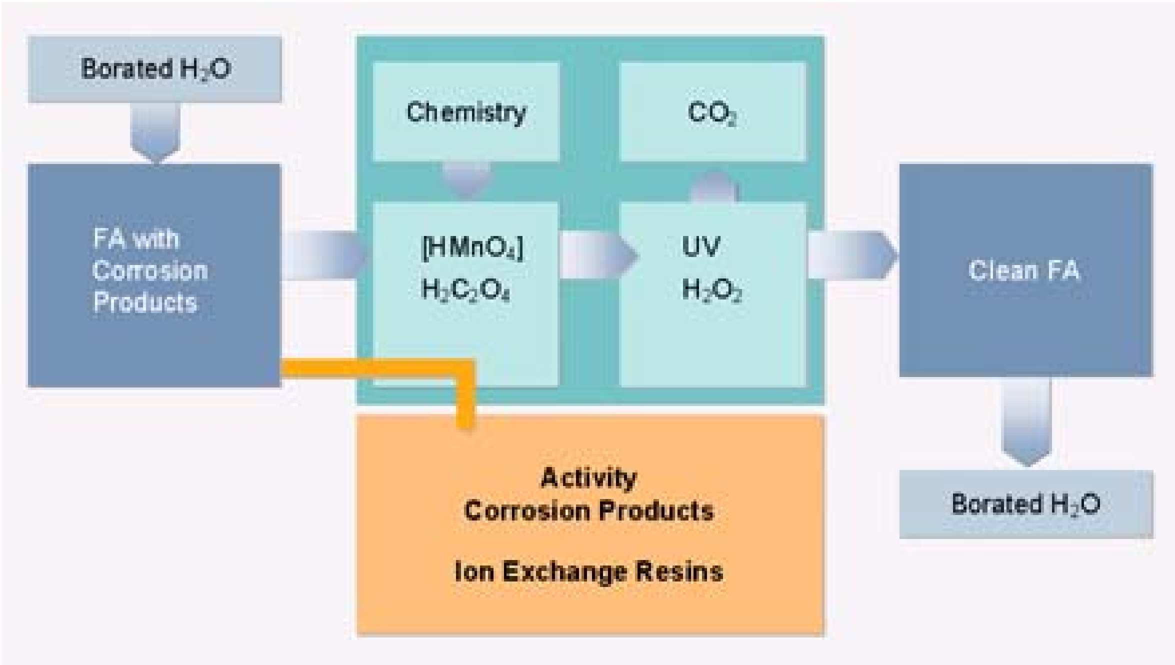


FIG. 6. Principle of the HP/CORD® UV process.

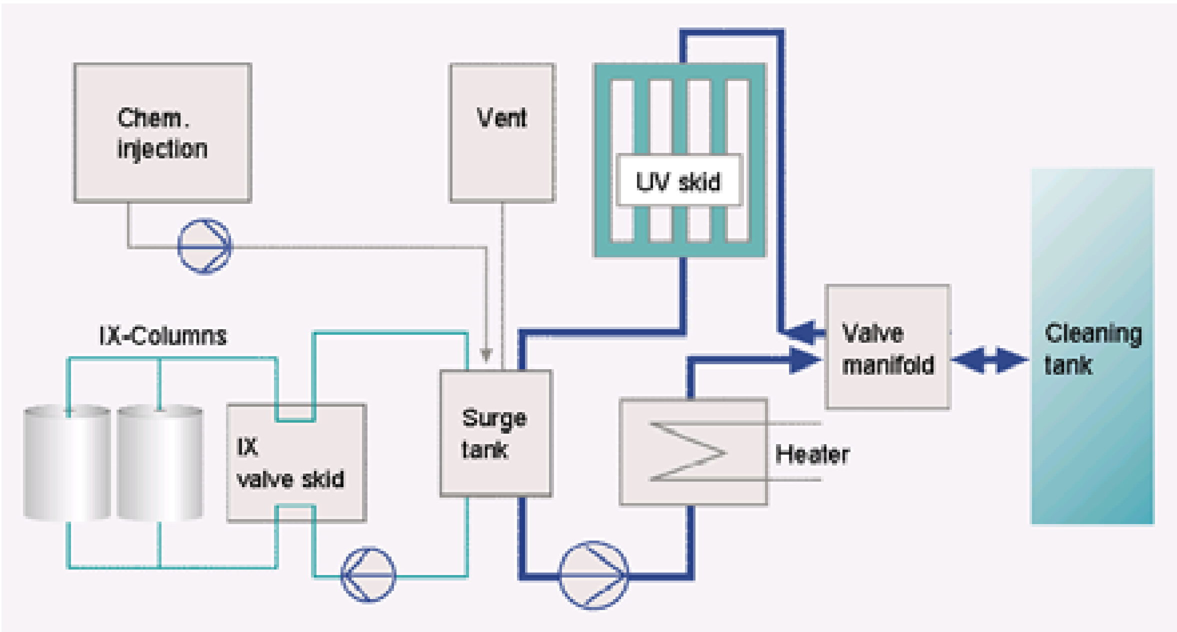


Fig. 7. Flow diagram of the chemical cleaning process.

Dissolution of the fuel assembly corrosion products is done with oxalic acid. Dissolved activity and corrosion products (Fe, Cr, Ni) are continuously removed during the entire cleaning step by a bypass cleanup path through ion exchange resins.

At the end of each cleaning step, the chemicals are decomposed to water and CO_2 with a ultraviolet light. Therefore the water quality after the cleaning is comparable to the one before.

Up to three CORD UV cycles were performed for the most blocked FAs, less blocked FAs could be cleaned with only one cycle. The pre-oxidation step was only applied during the second and third cleaning cycle, respectively.

Fig. 7 represents the flow diagram of the closed loop of the chemical cleaning process. The process consists of the main circuit containing the heater and the UV skid and a bypass circuit with the ion exchange resins.

The circuit is filled with borated water and heated to 92°C. Then the CORD chemicals are injected into the system. The chemical cleaning solution flows into the cleaning tank, dissolves the corrosion products from the FAs and returns from the tank into the UV skid for decomposition of the chemicals at the end of the cleaning step. Bypass cleanup through ion exchange resins takes place during the entire cleaning step.

Typical curves obtained during the chemical cleaning with the HP/CORD® UV process are shown in Fig. 8.

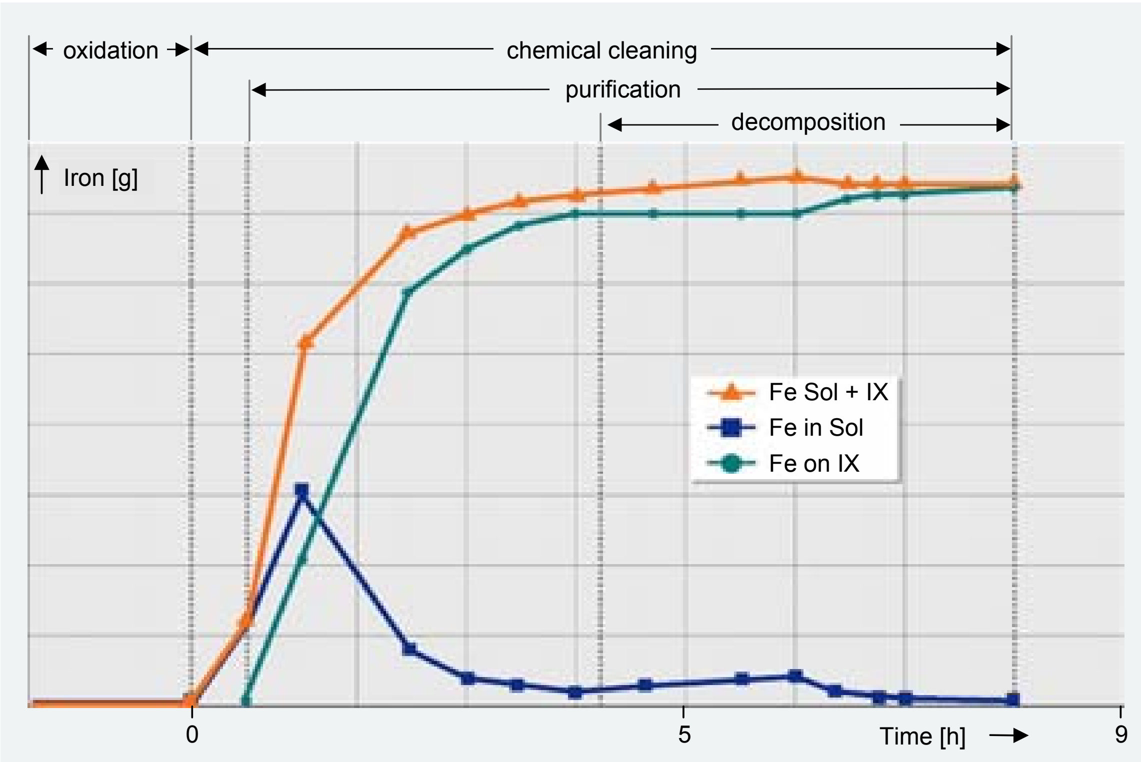


FIG. 8. Iron removal curves during the cleaning step.

The diagram represents 3 iron removal curves in dependency of the cleaning duration. The lower curve represents the iron amount presently dissolved in the cleaning solution. The green marked curve indicates the total amount of iron which is already absorbed by the ion exchange resins and the red marked curve is the total of iron still in solution plus iron already on ion exchange resins. No iron removal takes place during the oxidation step.

The overall trend of the 3 curves show that the total amount if iron removed from the FAs is fixed onto ion exchange resins after the cleaning step.

The most important advantages of the HP/CORD® UV technology with regard to cleaning FAs are:

- Very effective for desolve oxids for all reactor types and all types of water chemistry
- Produce reliable and reproductible results
- Regenerative process
- Entire cleaning is done with only one fill of water
- Waste volumes are very low
- Complete oxidative, in-situ decomposition of the cleaning acid to CO_2 at the end of the cleaning process
- No chelates in waste

3.4. Performance of qualification measurements

3.4.1. Kr85-Measurements

To prove the tightness of the FAs continuous on-line monitoring of the Kr85 isotope was conducted during the entire chemical cleaning process. The fission product Krypton 85 would penetrate into the cleaning solvent in case of a fuel rod defect and could thus be detected in the non-pressurized surge tank of the AMDA cleaning system. A typical curve produced by Kr85 monitoring of a FA batch is shown in Fig. 9. No peaks were detected, that means no damage occured on the FAs.

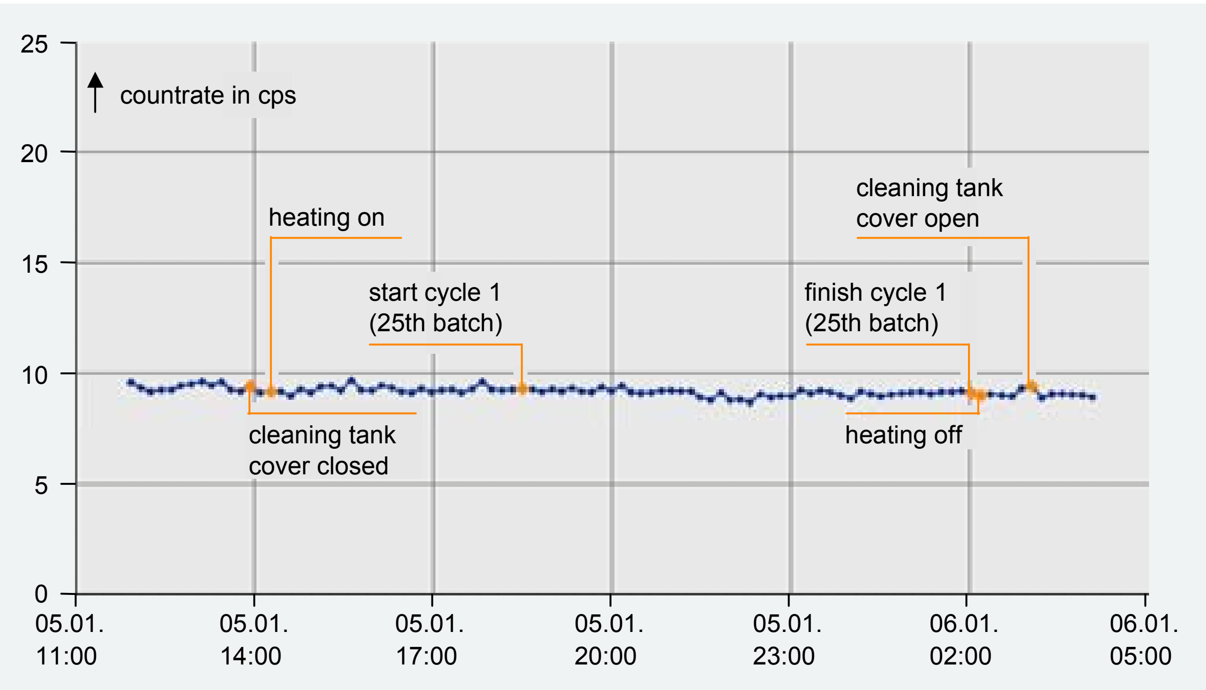


FIG. 9. Continuos on-line monitoring of Kr85.

3.4.2. Differential pressure measurements

For confirmation of the cleanness of the cleaned FAs, differential pressure (Δp) measurements were carried out by using the cleaning tank within the cleaning circuit as illustrated in Fig. 10. Borated water was used as flow medium and the Δp-measurements were done under the same operating conditions as for the cleaning.

Differential pressure was measured across the entire FA inside the cleaning tank for all 7 FAs one after the other. Therefore, for each FA a separate signal connection line was installed to the pressure transmitter that was located outside the pool on the reactor podium.

The Δp-skid consists of a pressure transmitter, multiple pressure port connections to the FAs including isolation valves and a purge water system for venting and flushing purposes. The data acquisition and on-line evaluation were performed with the AMDA equipment.

Differential pressure measurements were carried out in order to determine the relative hydraulic resistance of the cleaned FAs. The procedure for this is graphically illustrated in Fig. 11.
Δp data were measured for the blocked FAs to be cleaned before starting the cleaning procedure. To determine the hydraulic level for the cleaned FAs, Δp measurements were again done after having finished the corresponding cleaning cycles.

The individual Δp ratios of blocked FAs and a new FA, represent the degree of blocking for the FAs to be cleaned, whereas the Δp ratio of a blocked and a cleaned FA represents the degree of cleaning. In other words: The magnitude of increased Δp caused by blocking of the FAs is the required magnitude for decreasing the Δp by cleaning the FAs.

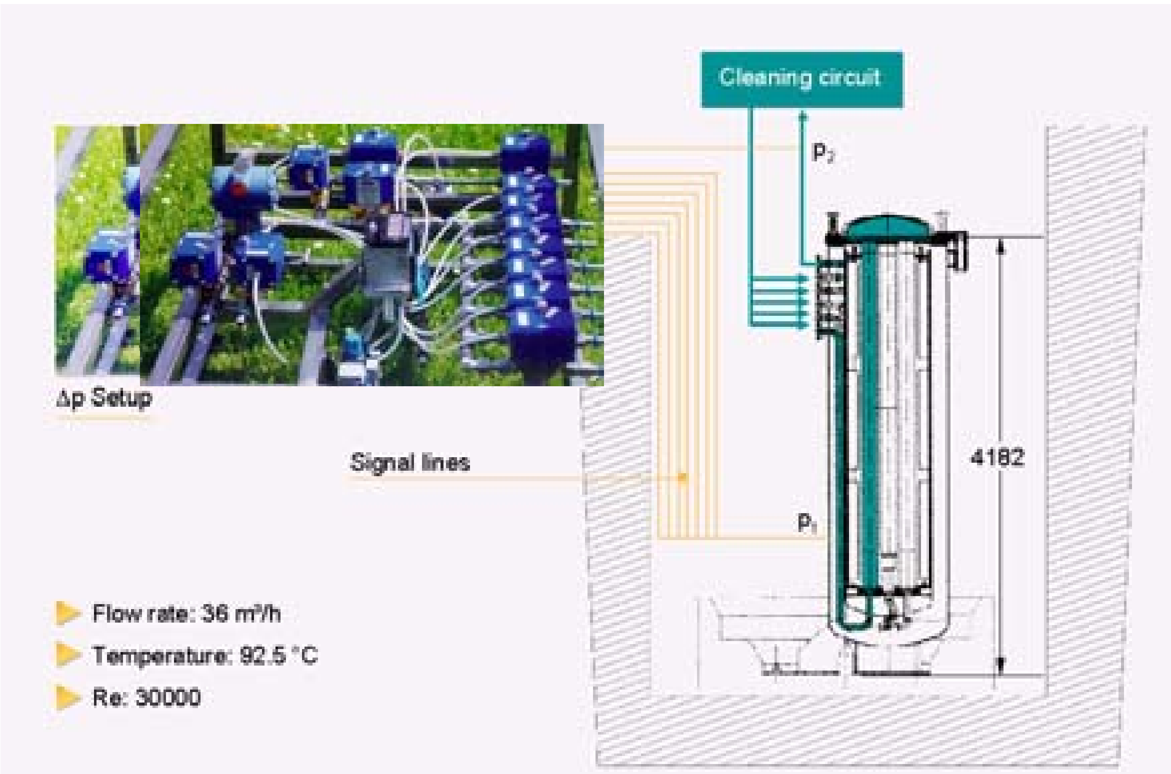


FIG. 10. Δp measuring arrangement.

The degree of cleaning was determined for 170 FAs in context with the chemical cleaning process by utilizing the FANP Δp measuring setup and the degree of blocking was determined 1998 by utilization of the Paflo device. The Paflo device is a setup for exclusive measurement of the differential pressure of one FA without a cleaning function. NPP Paks has used the Paflo device for selection of FAs to be cleaned and for establishment of hydraulic limits.

3.4.3. Establishment the cleanness criterion

The information indicated in the upper part of Fig. 11 and the limit value for hydraulic acceptance were considered for establishment the cleanness criterion. The cleanness criterion was determined during the test cleaning of 7 FAs with the result, that the cleanness of 7 FAs was proven by FANP and confirmed by NPP Paks.

The cleanness criterion represents a correlation of measured Δp ratios obtained from two different measuring setups, the FANP and the Paflo measuring arrangement. Therefore the requirement for Framatome was to ensure the same measuring accuracy than the Paflo device even under the special boundary conditions. The realization of this is shown in Fig. 10.

The cleanness criterion is presented graphically in Fig. 12 indicating the comparison of the PAFLO and FANP results regarding the relative hydraulic resistances of the cleaned FAs. The green marked points indicate the FANP results in relation to the defined cleanness criterion. The cleaning effect is approved when the FANP values of cleaned FAs are located above the cleanness criterion. Or in other words: The ratio $(\Delta p_{blocked}/\Delta p_{cleaned})_{FANP}$ has to be greater than $(\Delta p_{blocked}/\Delta p_{new})_{1998,Paflo}/k_{limit}$ to achieve the hydraulic acceptance criterion. The table beside the graph contains the ratios indicated in the diagram.

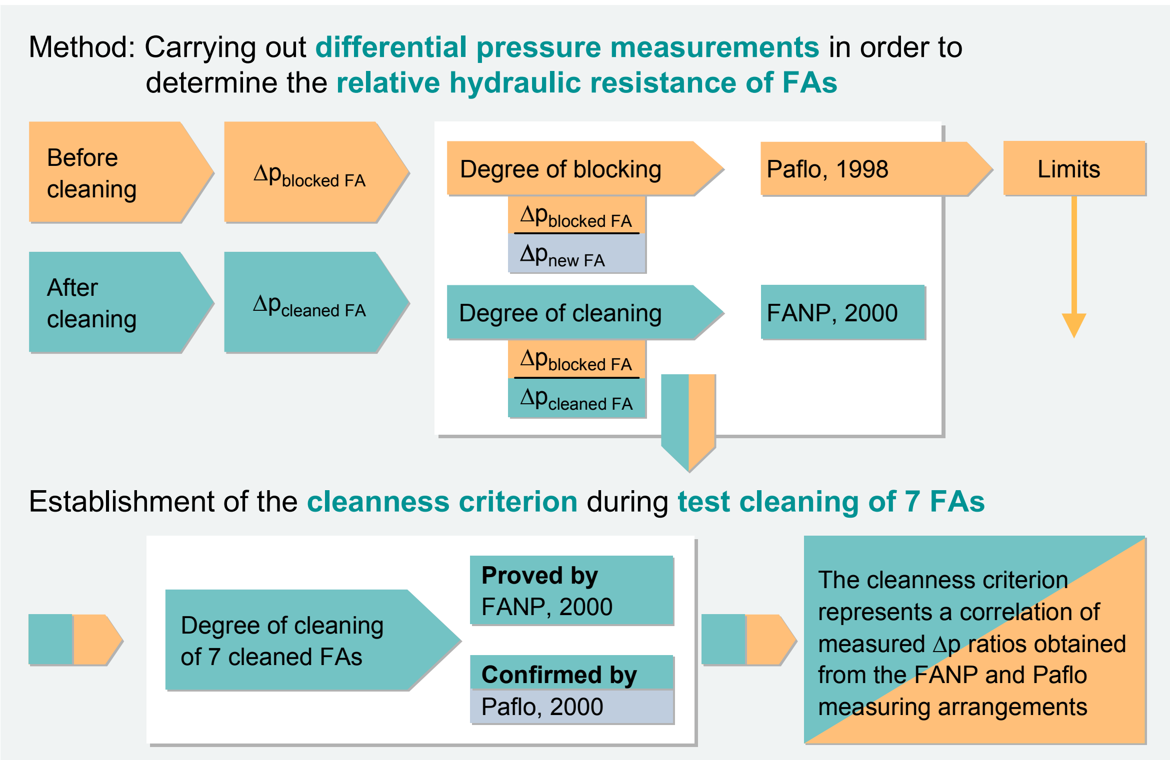
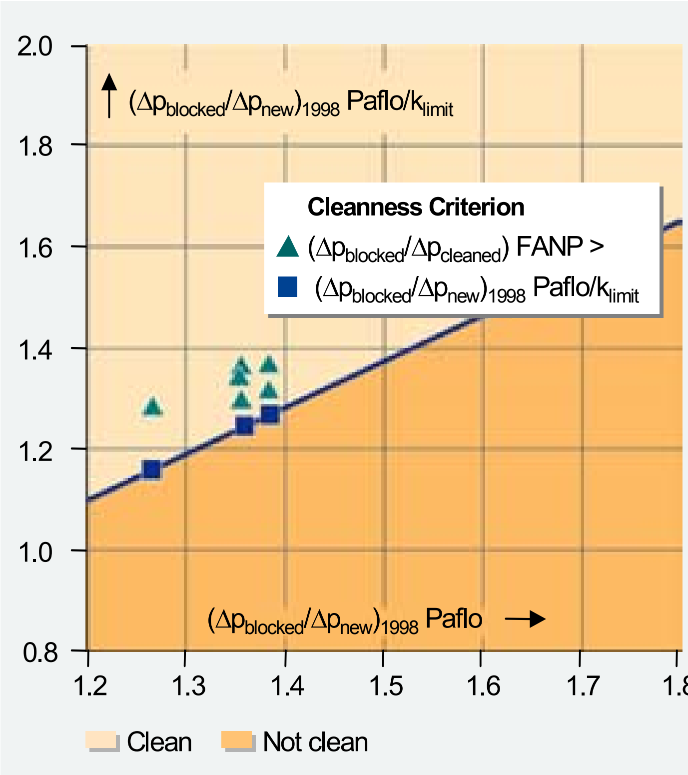


Fig. 11. Establishment the cleanness criterion to confirm the fuel assembly cleanness.



Measured Δp-ratios

Pos. FA	No. FA	$\Delta p_{blocked}/\Delta p_{cleaned}$ FANP	$(\Delta p_{blocked}/\Delta p_{new})/k_{limit}$ Paflo
2	50665	1.240	1.161
3	53292	1.295	1.266
4	50220	1.343	1.254
5	53343	1.356	1.268
6	49555	1.313	1.253
7	48613	1.291	1.252

Fig. 12. Assessment of the fuel assembly cleanness.

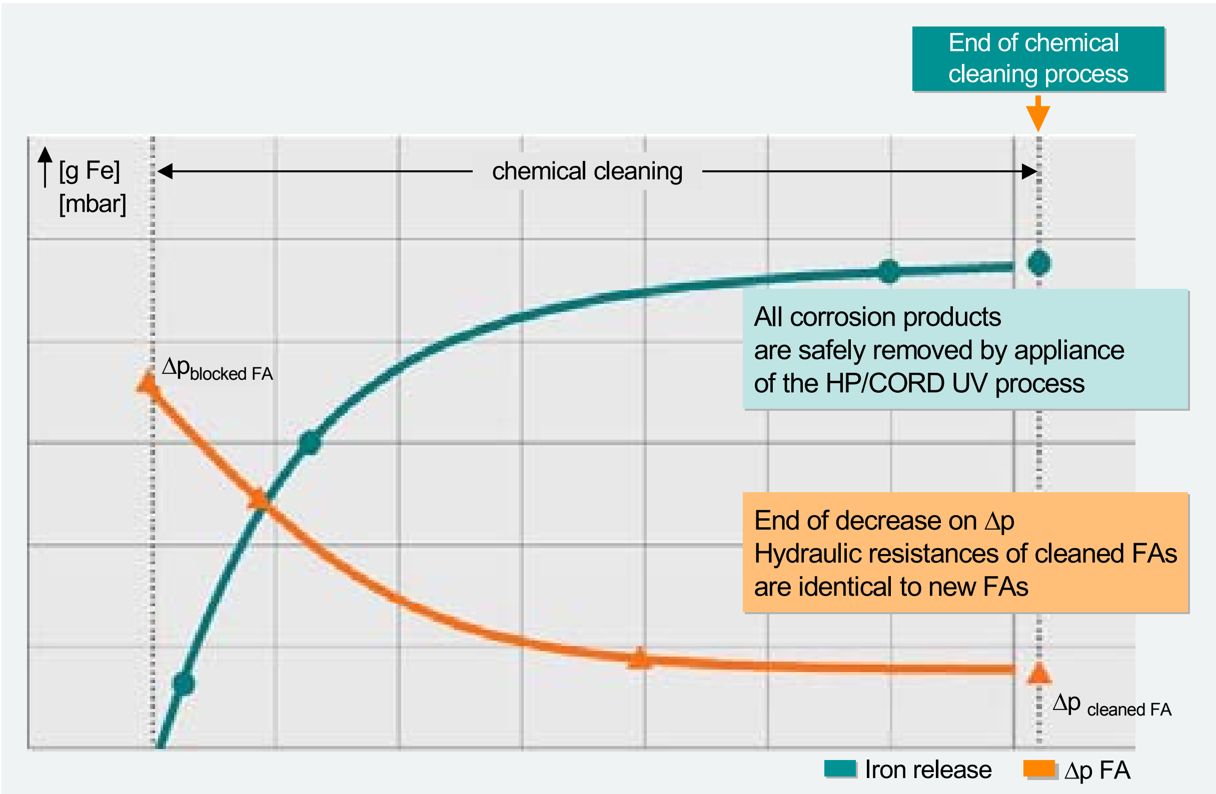


FIG. 13. Indication of the end of the chemical cleaning process.

4. SUMMARY

4.1. Chemical cleaning and cleaning effectiveness

Fig. 13 contains two typical curves indicating the end of the chemical cleaning process. Firstly, the iron removal curve shows all corrosion products are safely removed by application of the HP/CORD® UV process. Secondly, the drop of differential pressure indicates the end of decrease of Δp and also that the hydraulic resistances of cleaned FAs are identical to new FAs. It is illustrated that after a certain cleaning time the cleaning process has been finished, the important parameter don't change anymore. Fig. 14 shows again the assessment of the FA cleanness from the hydraulic point of view.

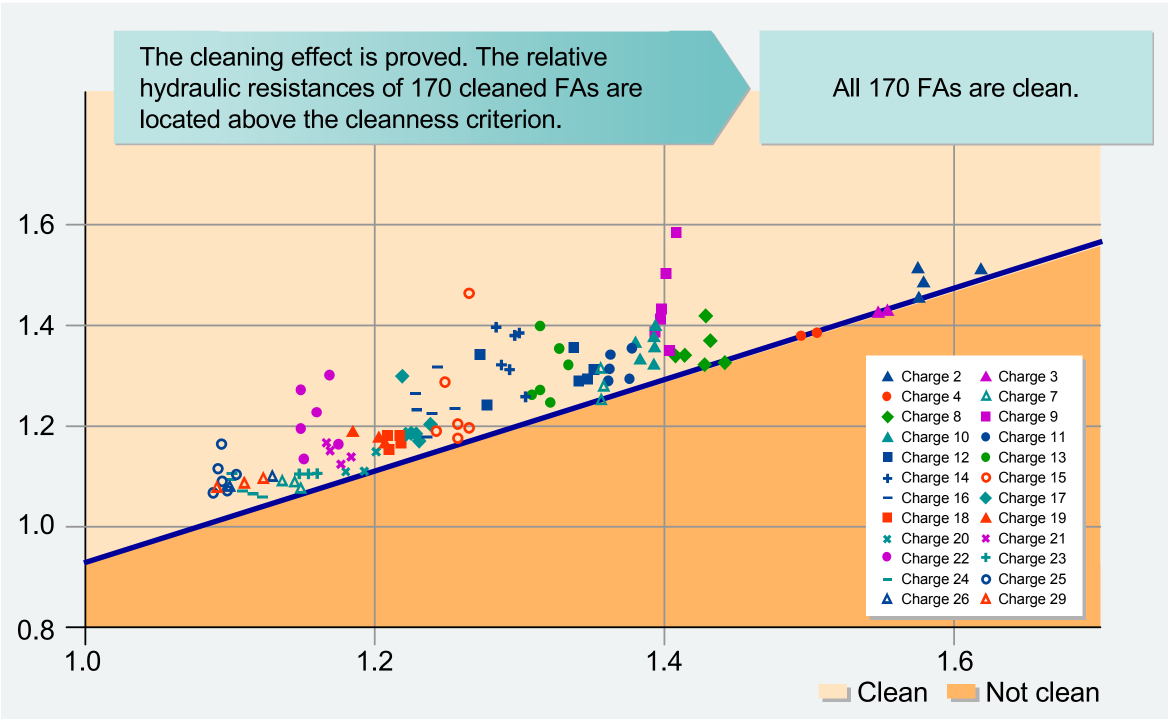


FIG. 14.

4.2. Facts of particular interest

Further facts are of particular interest and may be summarized as follows:

Integrity

Continuous Kr 85 monitoring demonstrated that that the CORD UV process did not impact the tightness of FAs. **No damages** occurred on the 170 FAs.

Waste

All corrosion products (17000 g Fe-oxide) and the activity dissolved during the cleaning process were transferred onto iron exchange resins.

Only **914 liters of resins** were needed and has to be stored as radioactive waste in the spent resin tank.

No flushing water was necessary.

Time frame

170 FAs were cleaned within approx. **10 weeks.**

Approval

Based on the results obtained from the chemical cleaning and the qualification measures all cleaned **170 FAs are accepted and approved** by the MSZ and the Hungarian Authority **for reuse in the reactor.**

Up to now 68 FAs have been successfully reused in the core since March 2001.

The results of the chemical cleaning of 170 FAs at NPP Paks showed that by appliance of the overall concept consisting of the HP/CORD® UV cleaning process and the FANP measuring arrangements for FA qualification, FAs have been safely cleaned and the cleaning success is safely proved.

ACKNOWLEDGEMENTS

The successful accomplishment of this project is based on the outstanding teamwork between Framatome ANP GmbH and the Nuclear Power Plant Paks. Therefore the authors wish to express thanks to all involved staff members of NPP Paks for their excellent cooperation during the entire project. Especially the authors like to thank Mr. Tilky and Mr. Dr. Schunk for their helpful and fruitful discussions, Mr. Dr. Miko for his contribution with regard to the qualification of FAs from the hydraulic point of view and Mr. Skach for his permanent support.

KNPP PRACTICES IN ASSURANCE OF RELIABLE FUEL OPERATION FOR WWER-440 REACTORS

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Abstract

This paper makes an overview of fuel integrity control performed at KNPP. Some of the possible causes for fuel failure are discussed. The practices in improving of fuel performance and assurance of reliable fuel operation are represented.

1. INTRODUCTION

Fuel cladding represents the second and the most important barrier against releases of the fission products beyond the nuclear plant. That is why an effective and reliable cladding integrity control is essential for the safe operation of the plant. Fuel integrity control at Kozloduy NPP is realized in two ways:

(a) During reactor operation – through radiochemical control of the primary circuit coolant;
(b) During reactor reloading (outages) – individual assembly control.

KNPP operates 6 WWER type reactors. Units 1-4 are WWER-440/B230 with a “double unit” configuration. They were put into operation in 1974, 1975, 1981 and 1982 respectively; units 5 and 6 are WWER – 1000 type and have been operating since 1987 and 1991.

This report reviews the results of the fuel cladding integrity control performed on WWER 440 fuel during KNPP outages and summarizes our practices in improvement of fuel reliability.

2. METHOD OF FUEL CLADDING INTEGRITY CONTROL

The individual assembly control is performed by “wet canister” method. The method consists in isolating every tested assembly in a can situated in the spent fuel pool, its washing, followed by forced “pumping-out” of fission products from leaking rods by changing in system pressure. At the end of this procedure a water sample from the system is taken. Isotopes ^{131}I, ^{134}Cs and ^{137}Cs are used as indicators for leak. They are measured directly in the water sample. Two criteria are used:

(a) Fixed criterion for ^{131}I activity in the water sample: 3.7E6 Bq/dm^3
(b) Statistical criterion: $A_{average} + 3\sigma$ (σ - standard deviation)

Fuel assembly (FA) with ^{131}I specific activity in the water sample exceeding the fixed criterion is considered a failed FA. It is not allowed to put failed fuel assemblies into reactor core for further use. Assemblies with specific activity of the indicators in the water sample greater than statistical criterion are considered to be leaking. These FAs might be used again only upon additional assessment and for each specific case a decision has to be made, depending on the radiation situation.

A new more effective sipping test system within a frame of the IAEA Technical assistance Project BUL/4/006 is intended for implementation from the next reloading.

3. SUMMARY OF THE RESULTS

A total 78 fuel cycles (FC) passed on units 1÷4, but the number of fuel tests are a little lower. The scope of the control is shown on Table 1.

The number of the tested fuel assemblies (TFAs) depending on working time into reactor core (in fuel cycles) is summarized in the Fig. 1.

Table 1

	Unit 1	Unit 2	Unit 3	Unit 4	Total
Number of FC	22	23	17	16	78
Number of FA's cycles	7318	7667	5537	5584	26106
Number of Tested FAs	3141	3590	2133	2196	11060

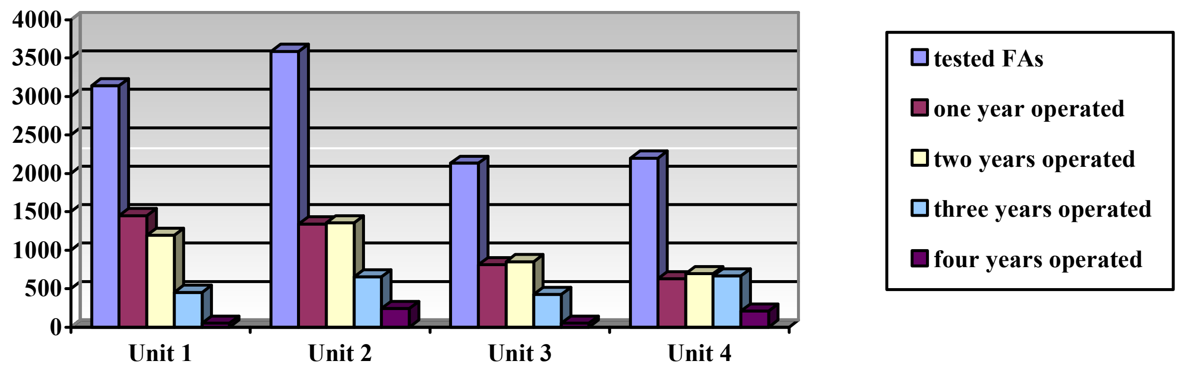

FIG.1. Distribution of TFAs depending on years of operating.

As it can be seen most of the FAs are tested after their first and second fuel cycle – this is an important fact, which has to be taken into account when drawing conclusions about their reliability.
The scope of control is shown on the next figure (Fig. 2) and the working time of the reactor units is divided into two periods:

(a) From the start up of every reactor unit to the year 1993/4;
(b) Since 1993/4 to now.

Further this work will focus on the last 26 FC for all reactors. One of the reasons for this is obvious – for the last 26 fuel cycles the scope of control grew more than twice. There are also some technical considerations – changes in the fuel design, improvements in fuel testing procedure ensuring better quality of the results etc.

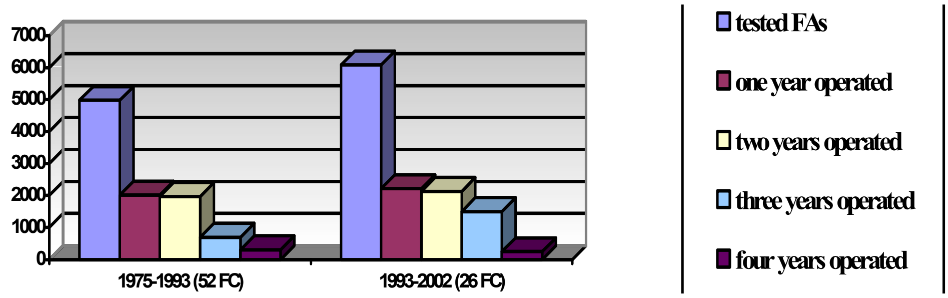

FIG.2. Scope of the control for the different number of cycles.

As it was mentioned above the number of tested FAs differs depending on the working time in the reactor core. If we draw the distribution of leaking fuel assemblies (LFAs) depending on working time without taking into account this fact, the conclusions might be wrong. For example, we may conclude that the most of the leaking FAs are one and two years old.

Having in mind the above mentioned regarding the number of the TFAs we made a normalization of a LFAs to the number of TFAs from the same type and the results are shown on the FIG. 3.

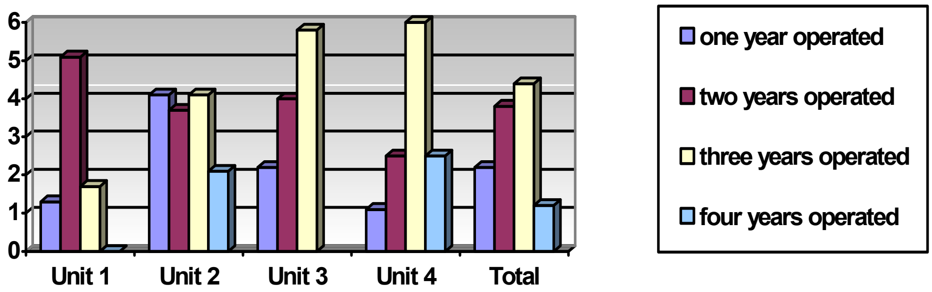

FIG.3. Distribution of LFAs/TFAs depending on years of operating

Several conclusions could be made from this statistics:

(a) A specific distribution is observed for every reactor unit: the distribution for Units 3 and 4 shows that the biggest share of leaking assemblies could be attributed to three years operated FAs. For Unit 1 the two years operated FAs have shown the greatest number of LFAs, while for Unit 2 there is nearly uniform distribution of LFAs between 1, 2 and 3 years operated FAs.

(b) The three years operated FAs have the greatest leaking rate in the total distribution. The leaking rate of four years operated FAs is the smallest for all distributions except for Unit 4.

4. ANALYSIS OF SOME POSSIBLE LEAK CAUSES

Generally the causes for fuel failures could be divided in two groups:

(a) external – depending on fuel manufacturing process
(b) internal – reasons related to the fuel operation conditions

Although the external causes cannot be neglected they are not of interest for this presentation. Here has to be mentioned only that there is a big potential in the wide implementation of quality assurance program covering the entire fuel manufacturing process.

We investigated some of the possibilities within the internal reasons using only "passive" instrumentations – i.e. without hot cell or direct examinations. (These examinations were planned to be done at Dimitrovgrad but this issue is still pending due to various reasons).

4.1. Problems with reactor internals

After the first implementation of statistical observation we found separate cases of repeated fuel failures in one core position. After a careful examination the root cause was identified (deformed flow restrictor) and eliminated.

Now a new data base for collection, analysis and archiving of every kind of data covering all fuel "life cycle" is under implementation.

4.2. Implementation of dummy assemblies

The dummy assemblies have been implemented on Unit 1 (1988, 13FC), Unit 2 (1989, 14 FC) and Unit 3 (1987, 7 FC). The average power was increased from 3.94 to 4.39 MW and the average linear power of fuel rod was increased from 129 to 143 W/cm (of course all this new values are within the allowed limits). Although at the beginning of the investigation this fact seemed to be a very likely reason for failures, now it is rejected as a stand alone reason. May be it has an impact in combination with other factors. This fact is supported by the same failure rate for Unit 4 (the dummy assemblies are not installed in this core) as for Unit 1 and Unit 3.

4.3. Position influence

LFA position analysis makes an attempt to find out some of the possible reasons causing the assembly failure. Particularly, the influence of the following assembly positions was investigated, considered to be "risky (dangerous)"positions:

(a) around the control assemblies (group VI region)
(b) In the periphery of the reactor core (with or without dummy assemblies (DA). This position is interesting with respect to origin of additional hydraulic loading.

Using a methodology proposed by the WNIINM [1] with introducing of minor modifications [2] we made (in 1999) an attempt to distinguish non typical distribution for every reactor unit in order to find out if there is a relation between the location of the FA in the reactor core and the probability for leaking. The analysis of the influence of the "risk position" over a leak rate shows:

- only for Unit 2 was observed a different from normal distribution with a more LFAs in the periphery region;
- There is no influence of the working control assembly group on the frequency of leaks.

The first of these conclusions is in a good agreement with the assumption for vibration related damages.

5. MEASURES TO IMPROVE FUEL RELIABILITY

In order to improve the fuel reliability the following measures are undertaken:

- Elaborating of actions for quality assurance covering “life cycle” of the fuel. A new detailed quality assurance program was elaborated. This program includes actions for quality control during manufacturing, transportation and fuel storage. The complete history of each assembly is followed from its first year insertion into the core to its last year of performance in the core.
- A Bulgarian-Russian task force was established with participation of representatives of AO “TVEL”, OAO “MSZ”, VNIINM, OKB “GP” and RNC “KI”. This group is working on finding of the root causes and elaborating measures to reduce leaking of the nuclear fuel.
- Hot cell examination are planned to provide further information on both location and mechanism of leaking;
- Operation of new fuel with enhanced vibration resistance. The first results are encouraging: for totally 234 working cycles of vibration resistant FAs in the reactor core of Unit 2 one FA is leaking.
- Enhanced measures during reloading for preventing from loose parts in primary circuit;
- Purification of the coolant with special filtering device during maintenance.
- A system for neutron noise vibrodiagnostic of the reactor internals is in the process of development and collecting initial data for its implementation. During each refueling outages additional inspections of the reactor internals are carried out aimed at reducing their vibration.
- At the end of each fuel cycle, after carrying out sipping tests a detailed calculation model of the cycle operation mode is performed and additionally the neutron-physical characteristics of the LFAs are analyzed.

6. CONCLUSION

- The growth of the scope of control for KNPP fuel during the last years is considerable.
- Implementation of dummy assemblies is rejected as a stand alone reason for fuel failures.
- It is not observed an influence of the working control assembly group on the increased frequency of leaks.
- Vibration related fretting damages are very likely reason for the bigger failure rate for reactor Unit 2
- Implementation of assemblies with enhanced vibration resistance gives encouraging results.

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UKRAINIAN WWER-TYPE NPP UNITS. RESULTS OF CLADDING TIGHTNESS INSPECTION

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Abstract

In the report the generalized results of cladding leak tightness inspection are reviewed for the Ukrainian WWER-type NPP units. All observed defect types are described. Influence of reused untight fuel assemblies on coolant activity level is being discussed briefly. Current state of fuel performance and joint work with Russian fuel supplier JSC «TVEL» on cladding tightness inspection instruction are considered. Some results of postreactor fuel examination are included.

1. INTRODUCTION

NAEC "Energoatom" is Ukrainian nuclear utility. There are 4 nuclear power plants (NPP) in Ukraine. Today 13 units are in operation. 11 of them have WWER-1000 reactor another 2 are WWER-440. 2 WWER-1000 units are under construction. The eldest unit has more than 20 years operation experience. During units operation large fuel depressurization statistics (results of cladding leak tightness inspection (CLTI) during refuelling time) was accumulated. Part of that statistics is presented on Fig.1. For the commercial reasons relative values were used.

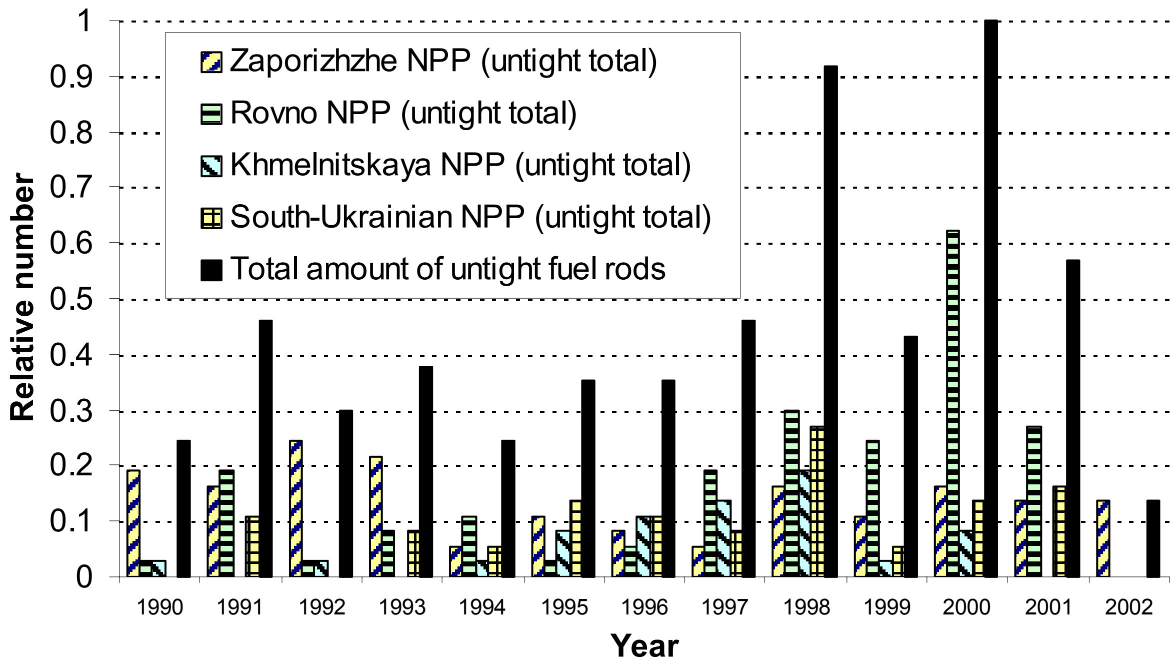

FIG. 1. Results of WWER-1000 CLTI for the period 1990-2002, values are relative.

Year 2002 data are not indicative due to the fact that only part of NPP units have finished its outage. It is necessary to keep in mind that showed data describe NPP with different number of units. That fact makes input of Rovno NPP unit 3 dominant in the whole statistics. Zaporizhzhe NPP units are characterized by the smallest values. Khmelnitsk and South-Ukrainian NPP have middle values of depressurization per one unit. From Fig.1 it is visible, that the growth of depressurization parameters of fuel was observed last years. The same quantity of depressurized fuel is predicted from the analysis of coolant activity for 2002 as it was in 2001.

From the reasons of fuel depressurization analysis it is interesting to take a look at the dependence of untight fuel assemblies (FA) quantity from the year of operation (see Fig.2).

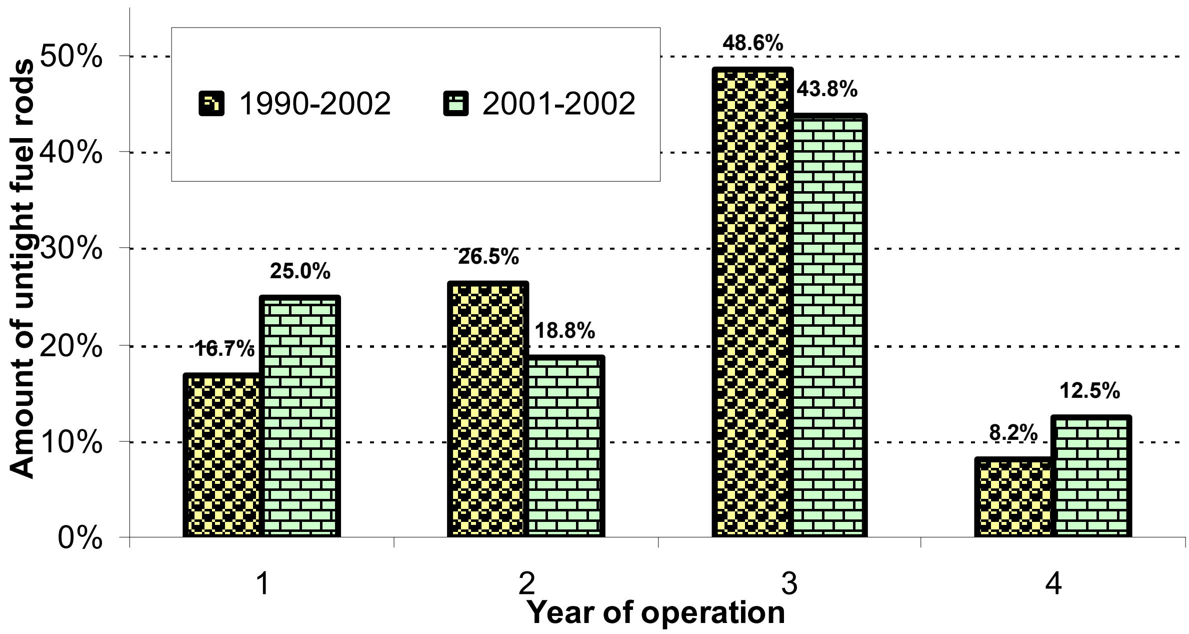

FIG. 2. Dependence of untight fuel assemblies (FA) quantity from the year of operation.

It is necessary to take into account difference in quantity of used FAs in reactor core depending from the year of operation. If you take into account this difference, the quantity of untight 4-th year operation FAs per one FA of 4-th year will grow 3 times. It is necessary to note, that depressurization parameters for Ukrainian WWER-1000 units are a little bit higher than for the same Russian ones.

The growth of the first year operation FA depressurization, as it figured on histogram (Fig. 2), causes anxiety of the utility. Unloaded FAs with a small burnup cause great expenses for the utility, so the main accent in the cooperation with the fuel supplier JSC «TVEL» is made on that problem.

For the reasons of formation of fuel loading some untight FA were used again. Secondary CLTI of that fuel were performed after operation. Results of such practical experience confirm the value of untight FA criterion ($A_{lim}{}^1$) that is given in fuel supplier CLTI instruction. In particular:

— Almost all statistically untight FAs (activity level (A_i) is between statistical criterion $\langle A\rangle+3\sigma$ and $A_{lim}{}^1$) had the same level of activity after the secondary usage.
— Level of activity of almost all untight FAs (activity level (A_i) greater than $A_{lim}{}^1$) was greater after the secondary usage. It seems to be due to the defect growth.

So, it was recommended for all Ukrainian NPPs to avoid secondary usage of untight FA. Secondary usage of untight FA leads to the defect evolution and to the radiological consequences due to the large pollution of surrounding FA fuel cladding by the fuel composition and fission products.

Depressurization statistics for the Ukrainian WWER-440 reactors are a little bit less than that in Russia and other countries.

It is necessary to note, that the picture of fuel failures based only on depressurization statistics is not complete. It is also necessary to take into account fuel failures due to the mechanical reasons. That fuel can be tight yet, but changes of its mechanical integrity or features will lead to its depressurization during further operation.

2. THE MAIN MECHANISMS OF FUEL FAILURE

Water chemistry regimes in WWER primary circuit coolant are much softer than the ones in PWR. It allows us to believe that cladding corrosion in primary circuit coolant is insufficient. In fact, according to the results of onsite visual inspection and postreactor fuel examination in RIAR (Dimitrovgrad, Russia) cladding corrosion was never observed. Moreover, deposition of corrosion products on the cladding and FA elements was also never observed. That allows us to intend that thermo-hydraulic characteristics of FA are not changing during the operation. It was frequently mentioned in papers VNIINM (Bochvar institute) that oxide layer thickness in the cladding is less than 15 microns. Besides it was repeatedly marked, that the alloy E110, used by the fuel supplier JSC «TVEL» for fuel cladding manufacturing, has a significant reserve of corrosion resistance. That allows us to use E110 alloy for increasing fuel burnup.

Fuel rod overheating connected with excess of energy release over design limit was also never observed for the Ukrainian NPP in case of maintenance of design requirements during operation.

As a rule the following defect types are observed during the reactor operation:

2.1. Fretting-corrosion

It has to be mentioned that FA was designed very successfully. First of all, it is a spacer grid construction that softens interaction in pair spacer grid - fuel rod and helps us to avoid fretting-corrosion. But in some last cases of postreactor fuel examination in RIAR fretting was assumed as one of the fuel failure reasons. It is difficult to prove the fretting nature of defect without postreactor fuel examination.

2.2. Failure of FA mechanical integrity

As a rule failure of FA mechanical integrity or FA geometry change happens due to two mechanisms:

— malfunction during the fuel transportation and reload;
— demonstration of defects that laid in design or manufacturing during FA operation.

First kind of failure is easy to detect by continuous control of fuel transport operations. It is required from the utility to inform fuel supplier about cases of mechanical integrity failure immediately. Then joint commission of utility, fuel supplier JSC «TVEL», scientific organizations and designer will investigate causes of fuel failure. As a result commission gives the recommendations how to avoid fuel failure next time. Those can be recommendations how to change operation process or how to change FA or equipment design. Then utility has to implement all the recommendations of fuel supplier, scientific organizations and reactor designer. From the viewpoint of geometrical and mechanical stability covered FAs is preferable, e.g. the one used in WWER-440. But usage of covered FAs in WWER-1000 is impossible from many reasons.

2.3. Damage of fuel cladding by the extrinsic subjects (debris) during the reactor operation

One of the main features of WWER FA is a direct-flow spacer grid. Until recent times it was considered that little amount of debris in coolant can not damage cladding. Till 2000 there was not found any WWER-1000 fuel rod with debris damage. But deterioration of the primary circuit and main equipment leads to the growth of maintenance activities on the opened surfaces of primary circuit. Obviously it rises quantity of debris that can damage fuel. Unfortunately technology of debris extraction from primary circuit is not developed. Operational experience indicates strong necessity of developing such technology. May be designing the new FA with debris-catching grids will be partial solution of the problem.

2.4. Fuel depressurization with undefined reasons

As it was mentioned above, results of fabrication plants inspection carried out by NAEC “Energoatom”, state that fuel quality is reliable. Depressurization statistics has sufficiently risen for the last few years. In addition the reason of depressurization was not defined both by on-site joint commission of utility and fuel supplier and by the results of postreactor fuel examination in RIAR. Unfortunately quantity of examined FAs in RIAR is very low and it can not be raised from many reasons. Depressurization mechanism seems to be as follows:

- Primary defect formation. In many cases primary defect is in the bottom part of the fuel rod. Primary defect type identification is strongly difficult to make due to the secondary effects (oxidation and hydrogenizing at the defect zone);
- Opening of primary defect due to cladding cracking up to the direct coolant-pellet interaction;
- Formation of secondary defects due to cladding hydrogenizing. As a rule it happens above the primary defect area. Consequences are up to the top plug separation;
- Fuel pellet structure is changing first of all in the area nearby defect. That changes result in intensifying fission products leakage into the coolant. In case the defect opening at the beginning of fuel cycle it would be possible to observe less activity of CLTI probe then the one for the “gas leaking” defect.

3. ONCE MORE ABOUT CLTI RESULTS

Each type of reactor has Cladding Leak Tightness Inspection (CLTI) instructions that were recommended by nuclear fuel supplier JSC "TVEL". As you know CLTI method is based on measurements of reference isotopes activity. Results of CLTI measurements of coolant activity in operating reactor define quantity of fuel assemblies that have to be tested during refuelling outage in Cask for Cladding Leak Test. So quantity of FAs tested during refuelling outage can differ from 0 up to whole number of FAs loaded in reactor core. Due to the large operational experience the separate lacks of the working CLTI instruction were revealed. For example, the spike-effect during operation specified presence of depressurized FA in the fuel loading of not tight fuel. But sometimes depressurized FA was not found during refuelling among tested FA.

Now fuel supplier JSC «TVEL» revises its CLTI instructions to comply them with operational experience. NAEC "Energoatom" is waiting for the development of the new criteria such as:

— Clarifying quantity of FA tested during refuelling. For example, ratio of Cs-134/137 activities during spike-effect can be used for definition of operational time of depressurized FA;
— Classifying depressurized FA according to the defect type based on the CLTI results. For example, value of solid fission products activity can be used.

Now NAEC "Energoatom" is considering possibility of equipping Ukrainian NPP units with on-line CLTI testing system in the refuelling machine mast (so called sipping-control system). It will allow us to carry out 100% CLTI tests during every refuelling. We hope that it will help us to exclude reload of depressurized FA.

Current WWER-1000/440 CLTI instructions divide FA according to the activity of CLTI probe (A_i) on 4 categories:

— Leak-tight FA ($A_i < \langle A \rangle + 3 \times \sigma$);
— Statistically untight FA ($\langle A \rangle + 3 \times \sigma < A_i < A_{lim}^1$);
— Depressurized FA ($A_{lim}^1 < A_i < A_{lim}^2$) and
— Failed FA ($A_{lim}^2 < A_i$).

Standards allow only tight FA and FA with defect of "Gas Leak" type for the shipping to the storage and recycling facilities. FA with "direct contact" defect is prohibited for shipping. We have to note that definition of "direct contact" defect is too fuzzy.

During the outage of unit #3 at Rovno NPP in 2000/2001 two FA were visually detected as untight with depressurized peripheral fuel rods. Direct contact pellet-coolant was visually detected. But both FA had level of activity lower then that one for the failed FA (A_{lim}^2) and were classified according to the CLTI instruction as a depressurized one. From the viewpoint of operation it is the indicative case as far as the utility has to decide how to use such FA lacking reliable information. Probability of visual detection of big defect is too small (it is proportional to the ratio of visible surface of peripheral fuel rods to total surface of all fuel rods in FA).

It is evident that development of new criteria of large defect detection by CLTI is imminent. For that purposes widened analysis of CLTI probes was provided. Big amount of solid fission products activity (barium-140, niobium-95, rubidium-103, cerium-141, 144, etc.) was detected in the all failed FAs and in some of depressurized FAs. The information was sent to the fuel supplier JSC TVEL and to all Ukrainian NPPs. From that moment all depressurized FAs were tested using widened CLTI probe. Some more depressurized FAs with high activity of solid fission products were detected at Zaporizhzhe NPP. Moreover, activity of that probe was tested twice: before and after mechanical filtration. Activity of probe has decreased by order of magnitude after filtration. That means that from the point of view of the utility probe has activity mainly in a form of fuel grains. Size of grains is about 10-50 micron.

That work was partially carried out together with the fuel supplier JSC «TVEL» and «Kurchatov Institute». Now a great deal of work on using solid fission products to detect big defect by CLTI is planned together with JSC TVEL. Additional postreactor fuel examination in RIAR will be carried out with assistance of the fuel supplier JSC «TVEL» within the framework of this work. Expected completion date - end of 2002.

It is not possible to carry out full-scale researches directly in Ukraine, because of the following reasons:

— Ukrainian NPPs have no FA inspection stand;
— Both reactor and storage pool are placed in the same containment, thus on-site research time is limited by outage even if we had a FA inspection stand;
— There is no chance to carry out postreactor examination directly in Ukraine due to the absence of big hot cells and necessary equipment.

4. EXCLUSION OF FUEL FAILURES. MITIGATION OF FUEL DEPRESSURIZATION CONSEQUENCES

We can exclude fuel failures or mitigate its consequences in two ways:

— By optimising fuel design;
— By optimising reactor operation and maintenance.

Fuel design optimization is not the subject of this report so only reactor operation optimization is considered.

Arising fuel failure due to debris damage requires first of all debris extraction from FA and primary circuit. As during the last 3 years debris have been visually detected in the fuel at the unit #3 Rovno NPP and there is no technology for its extraction, it was decided to wash FA in CLTI cask additionally. Visual control after washing did not show previously observed debris. Unfortunately one cannot guarantee the absence of debris in FA after washing because of impossibility to observe inner volume of FA. In this connection debris extraction still remains undecided.

On the basis of the obtained information weak points of existing procedures and technologies were analysed. Main attention was focused on repair works. Some additional equipment was developed. Requirements for the preventing ingress of contamination of primary circuit were strengthened. That changes were extended to all Ukrainian NPP.

Now Ukrainian NPP units are not equipped with the remote continual control systems of coolant activity. Work is underway on equipping NPPs. Current CLTI instructions stipulate shortage of interval between activity measurements. As a rule, measurements' interval is shorter then it is recommended in instructions. So it allows operating personnel to react on coolant activity changes more adequately.

In case of coolant activity growth following activities were tested (but they are still not in reactor operation documentation):

4.1. Reduction of main equipment loading cycles

Loading cycles are the controlled fuel operation parameters and their amount is limited for the reasons of fuel integrity. So if we have depressurized fuel in the reactor core we have to minimize amount of loading cycles to avoid further evolution of defect. For example, for the multiunit NPP it means restriction on changing unit power. Other units will make all changes of total NPP power output.

4.2. Strengthening of the demands to the transients

The aim of such activity is to relieve the stress of cladding due to pellet-cladding interaction. For that purpose power changing rate reduces up to the lowest reasonable level (less then existing requirements). Moreover, we try to avoid power distribution changes. For example, movement range of cluster working group is seriously shortened. Changing of boron concentration in the coolant provides power maintenance. All main parameters of reactor (pressure, inlet temperature etc.) are maintained at the stable level.

4.3. Strengthening of the demands to the water chemistry

WWER water chemistry regimes do not provide special measures for binding fission products (for example, iodine). During the operation no influence of water chemistry on coolant activity behaviour was observed. But we try to keep stable chemical conditions that prevent corrosion.

4.4. Unit power reduction

In some cases small power reduction (2-4%) on the unit with high coolant activity results in sufficient activity drop (2-5 times). Coolant activity changes slowly. Stabilization time is about a week. Such step in operation is justified in case of coolant activity close to operational limit.

5. CONCLUSION

For the reliable fuel operation permanent contact on real fuel behaviour in reactor core with fuel supplier is needed. Feedback of operating experience can ensure prevention of weak points in design and their elimination. It is better to implement scientific and technical support with assistance of the fuel supplier. JSC «TVEL» as a fuel supplier has a big deal of operational information from all consumers and one can analyse and compare it.

In the nearest future it is desirable to equip Ukrainian NPP units with sipping-control systems. It is evident that units under construction should be equipped with sipping-control system before starting their operation. Other units can be equipped step-by-step.

It is desirable to pay more attention to the deterioration of the primary circuit and main equipment. At the same time development of debris extracting technology is urgent. It is also desirable to modify FA design using grids catching debris.

Existing and developed fuel failure detection criteria should be modified to accurately separate defects by type. Operational experience should be analysed to select the most effective way of fuel failure consequences mitigation.

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MITIGATION OF FAILURES BY DESIGN AND MANUFACTURING
(Session 2)

MECHANICAL DESIGN OF THE TRIPLEWAVE DEBRIS FILTER

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Abstract

This paper presents the design of the TripleWave debris filter. The function of the debris filter is to reduce the amount of debris that can enter the active fuel region and thereby reduce the risk for damage of the fuel cladding by fretting. The design is aimed at catching long and thin debris as this has proved to constitute the largest fretting risk. The trapping efficiency tests demonstrate that the TripleWave debris filter reduces the risk for harmful debris to enter the fuel assembly significantly compared with the current filter design. Fuel assemblies equipped with the TripleWave filter are thermal hydraulic compatible with assemblies equipped with current filter design even with large amounts of debris trapped in the filter. A full-scale endurance test showed no signs of wear between the components in the lower part of the test assembly. The design is robust and redundant, it is subjected to very small stresses and a very well known and proven material is used and reactor operation without mechanical problems can be expected.

1. INTRODUCTION

Fuel reliability is basic to a safe and economical operation of nuclear plants. Westinghouse Atom has developed a more efficient debris filter, the TripleWave debris filter. The function of the debris filter is to reduce the amount of debris that can enter the active fuel region and thereby reduce the risk for damage of the fuel cladding by fretting. The design is aimed at catching long and thin debris as this has proved to constitute the largest fretting risk.

The TripleWave debris filter is placed in the inlet section of the fuel assembly, below the bottom tie plates of the subbundles.

The design requirements for the TripleWave debris filter are:

- The debris separating efficiency shall be considerably improved and at least 90% for harmful debris.
- Debris considered not to be harmful should pass the filter, not to risk clogging.
- The pressure drop shall match the pressure drop for current fuel assemblies.
- The design shall be robust and redundant; i.e. shall not contain any loose parts and shall survive considerable fretting from trapped debris.
- Manufacturing of the filter shall be cost efficient and meet appropriate quality requirements.

2. DESCRIPTION

2.1. Mechanical and Functional Design

The SVEA-96 Optima2 fuel assembly consists of:
- Fuel Bundle
- Fuel Channel
- Handle with Spring

The fuel bundle consists of four 5x5-1 subbundles. The subbundles are separated by a cruciform internal structure (water cross) in the channel. The subbundles are inserted into the fuel channel. They are supported at the bottom end by the bottom support and transition piece (inlet piece), which is bolted to the channel The fuel assembly is lifted in a handle, connected to the top end of the channel, and supported in the core module by a double leaf spring. The transition piece of the fuel channel fits into the core support plate.

The bottom support is machined from stainless steel bar material and is on its inlet side equipped with four TripleWave filter units. The bottom support is designed with grooves for fitting the TripleWave filter units below each subbundle. The bottom support, supports the filter units in four directions and the filter units are secured to the bottom support by lock welds on four sides of the filter unit. The filter mounted in the fuel assembly inlet is shown in Figure 1.

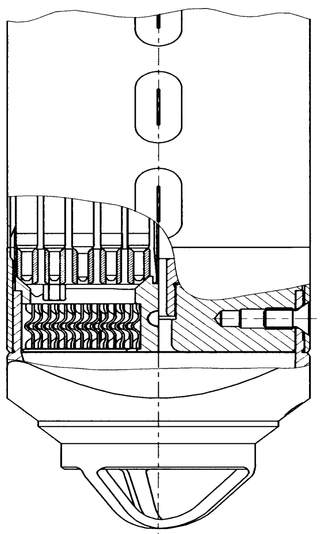

FIG. 1. Fuel Assembly Inlet Part with TripleWave Debris Filter.

The TripleWave design is aimed at catching long and slender debris as this is regarded to pose the largest risk for fretting on the fuel rods. Simultaneously small rounded objects like blasting grit shall be let through not to risk clogging the filter.

The filter is built from small corrugated or wavy plates, formed from 1 mm stainless steel sheet metal. The nominal pitch for the plates is 3,8 mm and the minimum nominal width of the flow path is 1,6 mm. The plates have a wavy shape across the flow in the inlet, a wavy shape along the flow in the centre and then again a wavy shape across the flow in the outlet, hence the name TripleWave.

The wavy shape of the inlet and the outlet edge serves several purposes. It functions as support points where the plates can be welded together. It also forms a grid that provides a first filter for large objects. In the inlet it reorients medium size objects parallel to the flow that are subsequently trapped at their head-on entrance into the filter. Finally, the design of the plates forms a flow path with a smooth and vertical outward flow from the filter outlet.

The plates are assembled into filter units. Each filter unit consists of 14 plates welded together at 10-12 points to the next plate (5-6 points on each edge). The filter units are mounted in the bottom support, see Figure 2. The filter unit is shown in Figure 3.
The units are 28,5 mm in height.

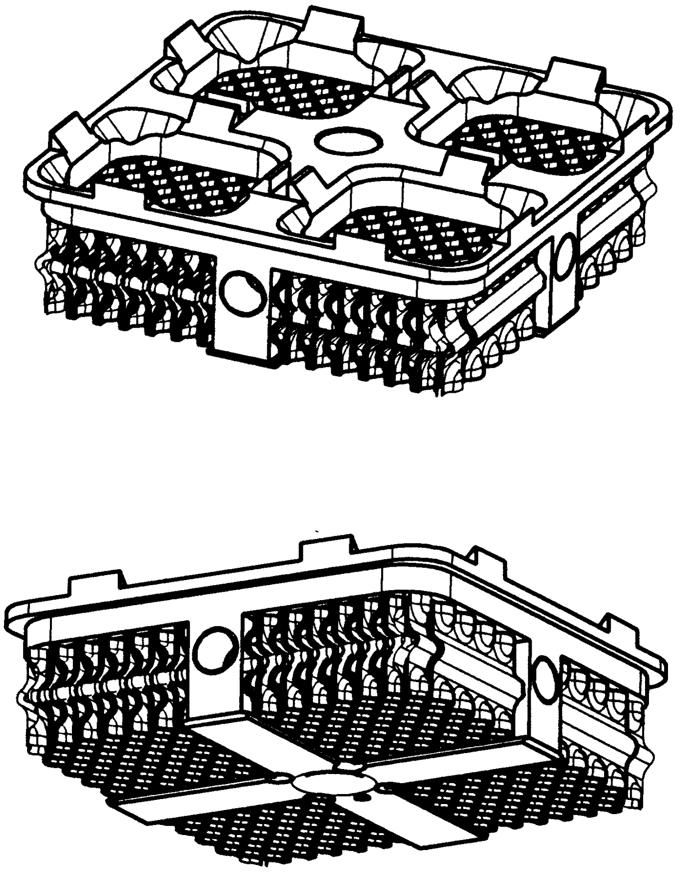

FIG. 2. Bottom Support with TripleWave Debris Filter.

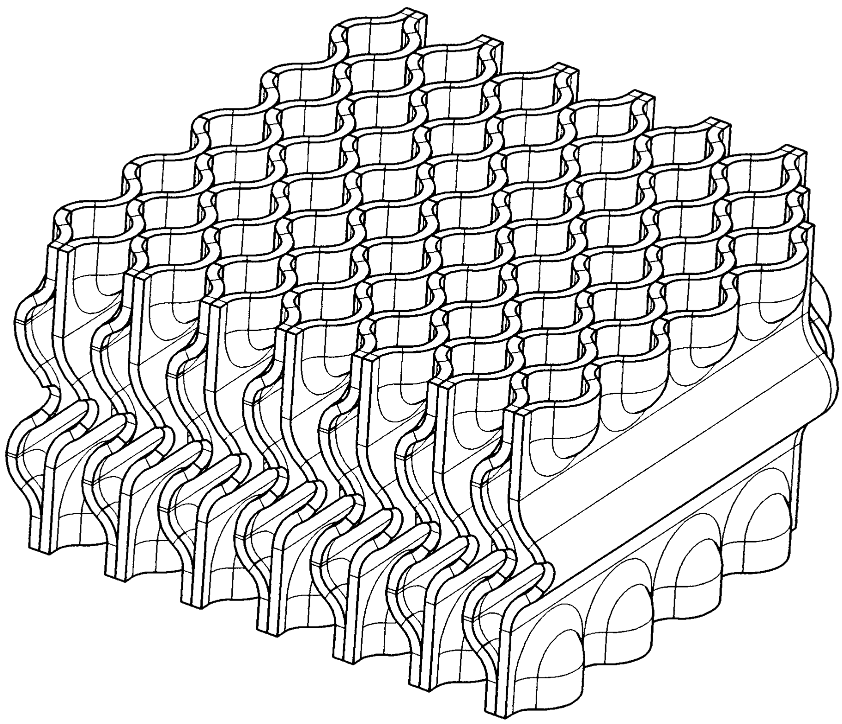

FIG. 3. TripleWave Debris Filter Unit.

2.2. Material

The material in the filter plates is stainless steel type AISI 316L. It has the same composition as the Westinghouse Atom BWR control rod material that has been used in several years and also the same requirement on low Cobalt content (<200ppm). Initial deliveries will have a Co content <400ppm.

2.3. Manufacturing

The first step is to cut and stamp individual plates from 1 mm thick sheet metal. Four slightly different plates are used in each filter unit.

The formed plates are mounted in a welding fixture. All edge contact points between the plates at the inlet and outlet edges is electron beam welded.

The finished filter units are heat-treated and then finally attached to the bottom support by welding at a number of spots, both at the inlet and the outlet sides.

3. PERFORMANCE EVALUATIONS

3.1. Test Facilities

The performance is tested in the Westinghouse Atom test loops FRODE and BURE. Most of the tests are done at atmospheric pressure in the temperature range 20-80°C in the FRODE loop. The inlet is correctly modelled but only a short mock-up test assembly is used.

The endurance test to prove that the new filter does not negatively influence the fuel were done in the BURE loop at operating data (about 7 MPa and 270°C) and one-phase flow. A full-scale test assembly was used.

3.2. Debris Trapping Efficiency

Based on experience from fuel inspections and support from some theoretical considerations a set of test debris has been composed for the efficiency tests. The amount of debris passing the new filter is reduced significantly compared with the current filter design.

The TripleWave filter is almost only passed by the shortest and thinnest wire pieces. If pieces <10 mm are disregarded from the test and only 15 mm or longer is counted, the filter is 99% effective.

3.3. Pressure Drop

The pressure drop tests were partly run separately and partly combined with the efficiency tests. In the separate tests Re numbers up to 90000 were obtained. Extrapolation to normal operating conditions (Re number 150000) is straightforward.

The pressure loss is about 24% higher than for the current filter design. The total core pressure drop is increased 1 to 1,5%, which in most cases is acceptable and does not imply any thermal hydraulic compatibility concerns. Core stability is slightly improved through the higher single-phase pressure drop.

The increase in the pressure drop loss coefficient with debris trapped in the filter is about 15% for the full set of test debris, most of which should have been stuck in the filter during the pressure drop registration.

4. ENDURANCE

A full-scale endurance test was performed in April - May 2001 (700 h), on a SVEA 96 Optima2 fuel assembly in the BURE loop. The inspection after the test showed no signs of wear between the components in the lower part of the test assembly. No negative impact on filter or filter components has been found, and the conclusion is that the bottom support with TripleWave debris filter has shown satisfactory co-operation with the rest of the test assembly and that reactor operation without mechanical problems can be expected.

5. CONCLUSIONS

The new TripleWave debris filter design meets or exceeds all the specified design requirements.

The trapping efficiency tests demonstrate that the TripleWave debris filter reduces the risk for harmful debris to enter the fuel assembly significantly compared with the current filter design.

Fuel assemblies equipped with the TripleWave filter are thermal hydraulic compatible with assemblies equipped with current filter design even with large amounts of debris trapped in the filter.

A full-scale endurance test showed no signs of wear between the components in the lower part of the test assembly.

The design is robust and redundant, it is subjected to very small stresses and a very well known and proven material is used and reactor operation without mechanical problems can be expected.

FUEL FAILURES AT ANGRA 1: CAUSE AND MITIGATION

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Abstract

Angra 1 Nuclear Power Plant, a Westinghouse-designed 657 MWe PWR, was prematurely shut down in cycle 4 due to high activity in the reactor coolant system. Inspections revealed failures in one-sixth of the fuel assemblies (FAs). According to the fuel vendors the grid-to-rod fretting failures were caused by grid spring force losses. In order to prevent the recurrence of the fretting failures, a new spring design was developed. A new reload batch using the new spring design was loaded in the core for cycle 5. In cycle 6, eight of these FAs failed, showing friction marks at outside strips of spacer grids, due to excessive amplitude of FA vibrations. Flow tests in laboratory were performed using a full-scale fuel assembly, and peaks of resonance due to flow induced vibration were observed. Analysis and modeling of the fuel assembly mechanical behavior can explain the resonance observed. All these analyses supported the need for a new fuel assembly design. In cycle 7, the reactor core was loaded with FAs of the same design inserted in cycle 5. Two leaking assemblies were detected during the fuel inspections conducted at the end of cycle 7. A new core was purchased from Angra 1 former fuel supplier (cycle 1), to load cycle 8. The use of a proven fuel design got rid of the fuel failures. The reload batch for the cycle 9 came also from the same origin as cycle 8. The zero-defect aim was also attained in cycle 9. For cycle 10 it was decided to use twelve once burned assemblies, which had composed cycle 7, with a damping system that, according to flow tests, could reduce the FAs vibration. Cycle 10 is presently under operation with no indication of fuel failures. This paper discusses Angra 1 fuel performance with emphasis to the fuel failures, the inspections and out-of-pile tests performed, the mechanical modeling, the corrective actions proposed by the fuel vendors and the failures mitigation.

1. INTRODUCTION

Angra 1 is a Westinghouse-designed 657 MWe PWR, 130 km from Rio de Janeiro, Brazil. The plant is operated by ELETRONUCLEAR, a government utility. Angra 1 began commercial operation in 1985. On 31 May 2002, the plant reached about 90% of the planned burn-up for cycle 10. The reactor core comprises 121 (16x16 array) fuel assemblies (FAs). Each FA contains 235 Zircaloy-4 fuel rods (FRs) supported at intervals along their length by eight Inconel-718 spacer grids. The first core (batches A, B and C) was supplied by Westinghouse. Siemens and Indústrias Nucleares do Brasil (INB) have provided the reloads for cycles 2 to 7. Due to the failures occurred in cycles 4, 6 and 7 a new core with 121 original Angra 1 FAs (standard project) was purchased from Westinghouse to load cycle 8. Later on INB has implemented a contract of technological transfer with Westinghouse in order to produce the standard project. New reloads of Angra 1 (started with the 10^{th} cycle) use this FA fabricated by INB with Westinghouse design. Table I show the core configuration and the failures occurred during the ten operation cycles of the plant.

Table I. Angra 1 Fuel Assembly Failures

Cycle	Period	Fuel Assemblies				FA Leaking Indication	Inspection Test	Comments
		Batch	Vendor/ Design	Enrich. (wt %)	Number of F.A			
1	01/85 to 01/86	A B C	W/W W/W W/W	2.1 2.6 3.1	41 36 44	No No No	-	No indication of fuel failure.
2	10/86 to 10/89	A B C D	 INB/KWU	 3.3	1 36 44 40	No No 1 No	Sipping can	One failed fuel (estimation of 1 fuel rod). Mechanism and root cause not determined.
3	01/90 to 08/91	C D E	 INB/KWU	 3.4	41 40 40	1 No No	Sipping can	One failed fuel (estimation of 1 fuel rod). Mechanism and root cause not determined.
4	05/92 to 03/93	C D E F	 INB/KWU	 3.4	1 40 40 40	No 17 4 1	In-mast sipping; Visual; Ultrasonic test.	Main mechanism rod-to-grid fretting; secondary damage; loose fuel rods; some fuel rods slipped down onto end fitting. One FA of batch F was damaged by handling.
5	12/94 to 03/96	A B C F G	 INB/KWU	 3.4	36 8 1 36 40	No No No No No		Batch G has a new design with a higher spacer grid spring force. (No flow test was performed and no design change was done to the spacer grid mixing vanes).
6	06/96 to 09/97	G H J	 INB/KWU INB/KWU	 3.2 1.9	40 40 41	8 1 No	In-mast sipping; Visual; Sipping can	Debris or handling failure in batch H. Loose fuel rods in batch G, rod-to grid fretting wears in batch G. Secondary damage observed. Grid-to-grid fretting (west and east faces, higher in FA middle position) for all FA.
7	12/97 to 10/98	F H J L	 INB/KWU	 3.3	4 36 41 40	1 1 No No	In-mast sipping; sipping can; Visual	Failure mechanism was not determined. May be the same as cycle 4 and 6. Batch L grid-to-grid fretting (west and east faces, higher in FA middle position). Some FA from batch L showed handling damage.
8	12/98 to 03/00	M N P	W/W W/W W/W	2.1 2.6 3.3	41 40 40	No No No	In-mast sipping	No indication of fuel failure.
9	03/00 to 04/01	M N P Q[*1] R	 INB/S[*1] W/W	 3.4 3.4	21 40 40 4 16	No No No No No		No indication of fuel failure.
10	04/01 to now	M N P Q[*1] R L[*2] S[*3]	 INB/W	 3.4	1 8 40 4 16 12 40			90% of the cycle completed, with no indication of failure.

(*1) – New Siemens F.A design with split mixing vanes in the spacer-grids, manufactured by INB.
(*2) – F.A with dampers inside the guide-thimbles.
(*3) – Westinghouse design, manufactured by INB.

2. FUEL FAILURES: CYCLES 1 TO 5

During its first three cycles fuel performance at Angra 1 was very good. The fuel inspections carried out showed: no failures in cycle 1;one failed rod in cycle 2 (undefined cause); one failed rod in cycle 3 (undefined cause). In cycle 4, the reactor was loaded with 120 FAs fabricated by Siemens/INB (batches D, E and F) and one assembly that remained from the initial core. During this fourth cycle fuel failures led to increasing activity levels in the reactor coolant system (RCS) and the reactor was prematurely shut down on 5 March 1993, as shown in Fig. 1 and discussed in [1].

Ultrasonic testing performed during outage found 64 leaking FRs in 17 FAs of batch D, that had been loaded in cycle 2 (Figure 2).

The visual inspections identified grid-to-rod-fretting as the main failure mechanism of the leaking assemblies. Seventy-six "loose" and/or fretted FRs were observed in the failed 17 FAs of batch D (see Fig. 3). The RCS trend plots recorded during cycle 4 showed that the FA containing the first failed FR had an average burn-up ranging from 19 to 24 MWD/kgU.

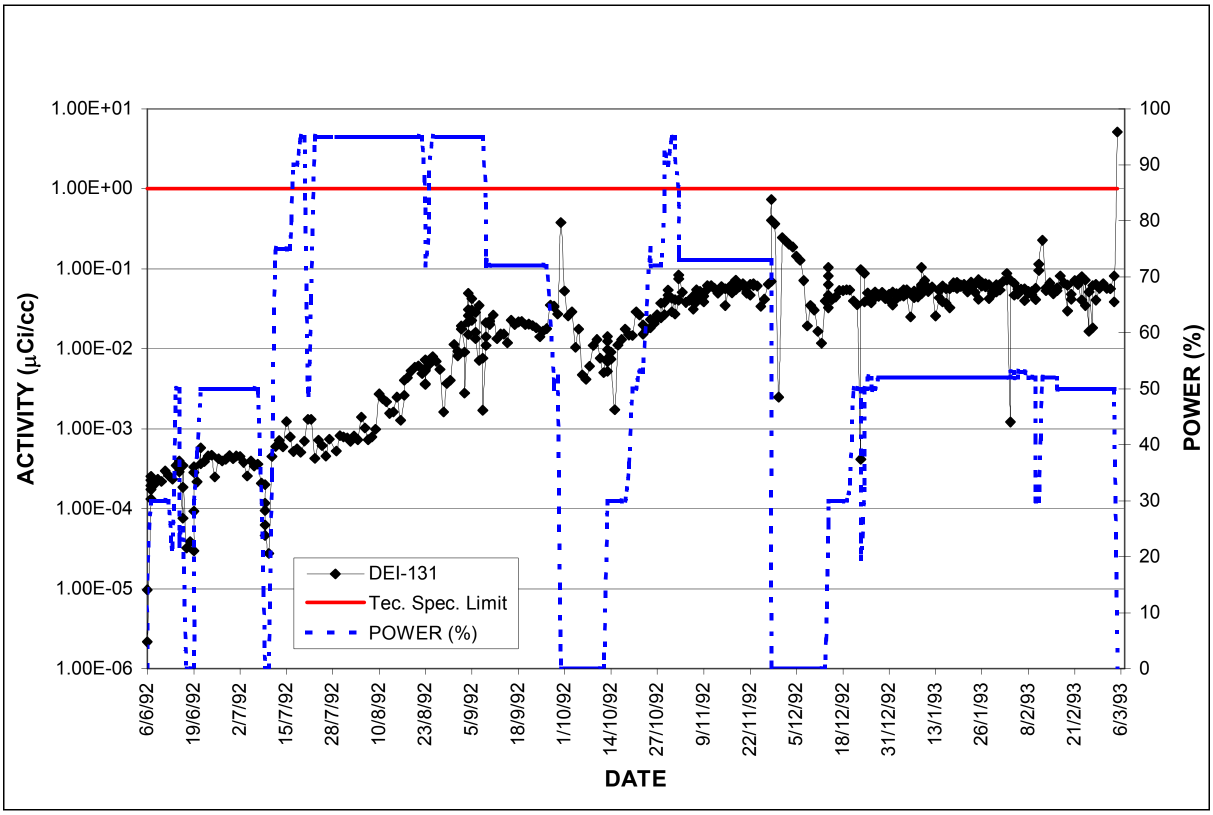


FIG. 1. Angra1 cycle 4: dose equivalent I-131.

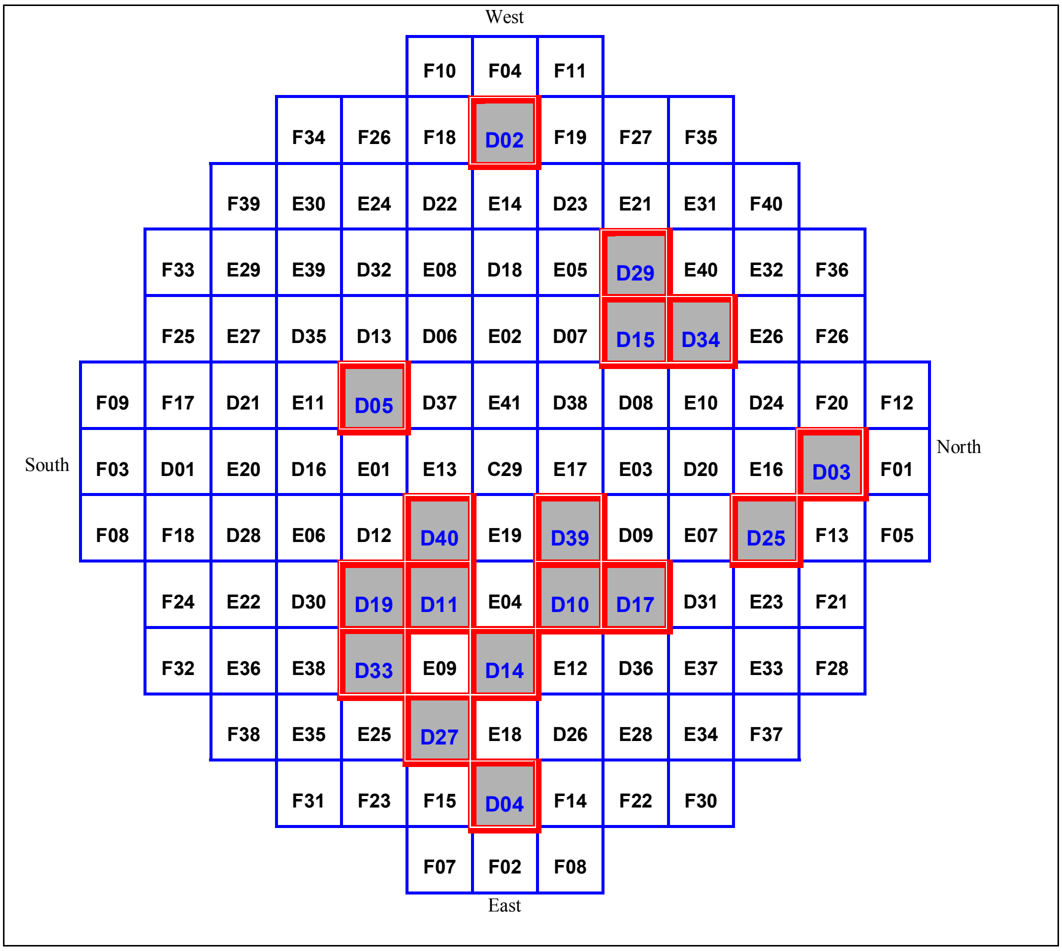


FIG. 2. Angra 1 cycle 4: Failures location.

In February 1994, Siemens issued a report, Ref. [2], evaluating the root-cause of the failures. According to Siemens they were caused by grid spring force losses occurred when the fuel rods were inserted in the skeleton, possibly in combination with loads sustained during transport to the site. In order to prevent the recurrence of the fretting failures, Siemens developed a new spring design whose main characteristics were a new shape and a higher initial force.

The cycle 5 core loading came from three sources:

- A new reload batch G (40 FAs), manufactured by Siemens/INB using the new spring design;
- Thirty-six FAs from batch F used in cycle 4 (average discharge burn-up of 5 MWD/kgU). The batch F assemblies have the same spring as batch D, so the burn-up of batch F was conservatively limited to 18.5 MWD/kgU in cycle 5. Assemblies from batch E could not be used because they have an average discharge burn-up of 15 MWD/kgU, and they would soon reach the limit of 18.5 MWD/kgU. This would risk fretting failures similar to those found in batch D;

- Forty-fiv Westinghouse FAs that had been stored in the spent fuel pool since they were unloaded after the three first cycles.

No failure was detected during cycle 5 operation.

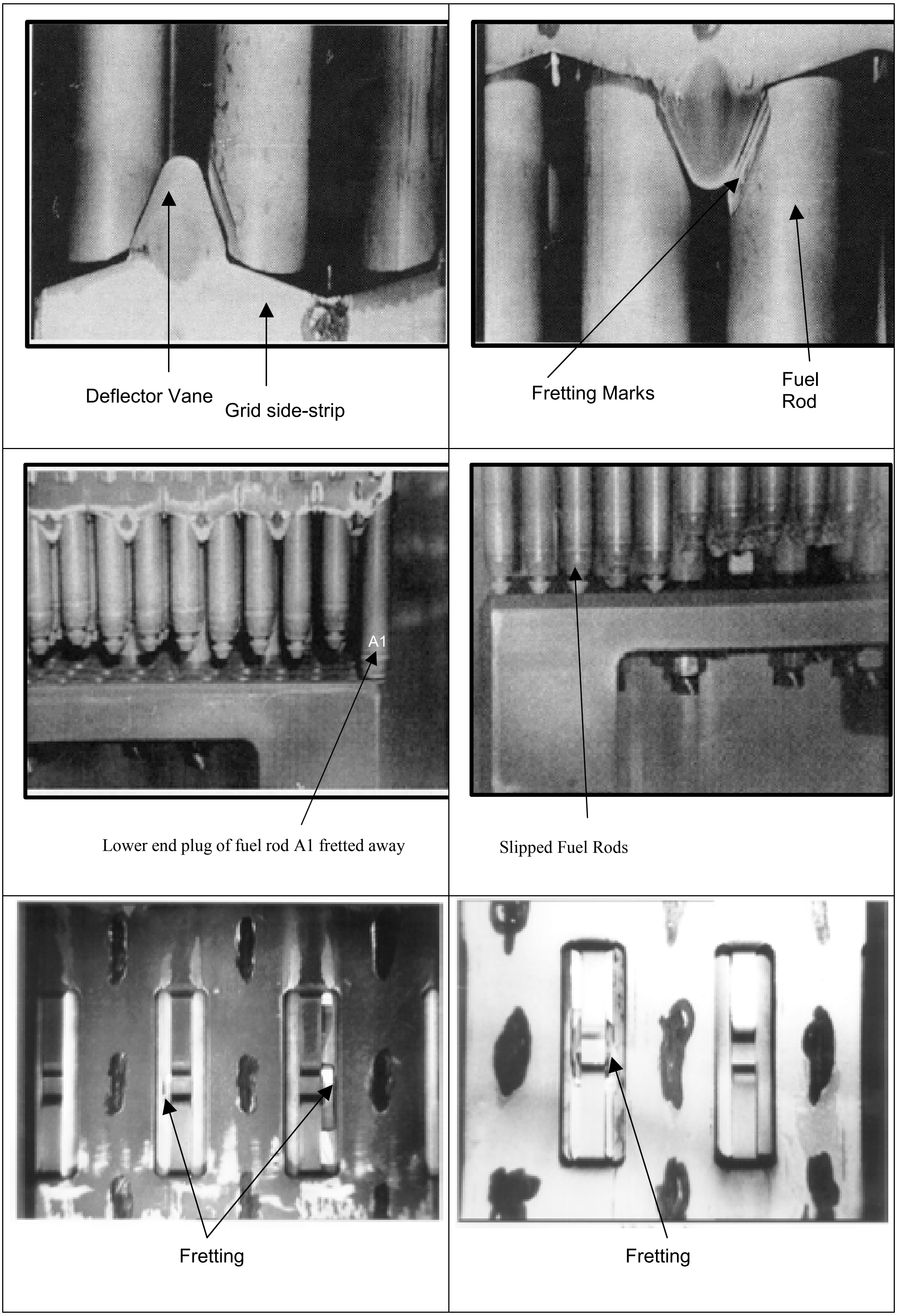


FIG. 3. Angra 1 cycle 4: Fretting on batch D FAs.

3. FUEL FAILURES IN CYCLE 6

In cycle 6 Angra 1 core was loaded with 121 Siemens/INB FAs discriminated as follows:

- Forty G FAs used in cycle 5;
- Forty new FAs enriched to 3.2% (batch H);
- Forty-one new FAs enriched to 1.9% (batch J). This additional reload batch had been purchased to be used exclusively in cycle 6 because of the fuel failures in cycle 4.

Cycle 6 started on 8 August 1996 with a low RCS activity level. On 28 August, with the reactor at 93% of nominal power, the dose equivalent iodine-131 (DEI-131) was 1.3×10^{-3} µCi/g, and the sum of noble gases activities was 7.7×10^{-3} µCi/g. However, on 30 August the RCS activities increased very significantly. On 4 September, DEI-131 reached 0.1 µCi/g and the sum of gases 1.6 µCi/g. Following Angra 1 Fuel Failures Action Plan the chemical and volume control system letdown rate had to be increased from 220 to 440 lpm and the primary system periodically degassed. DEI-131 activity decreased continuously up to mid-January 1997 (see Fig. 4).

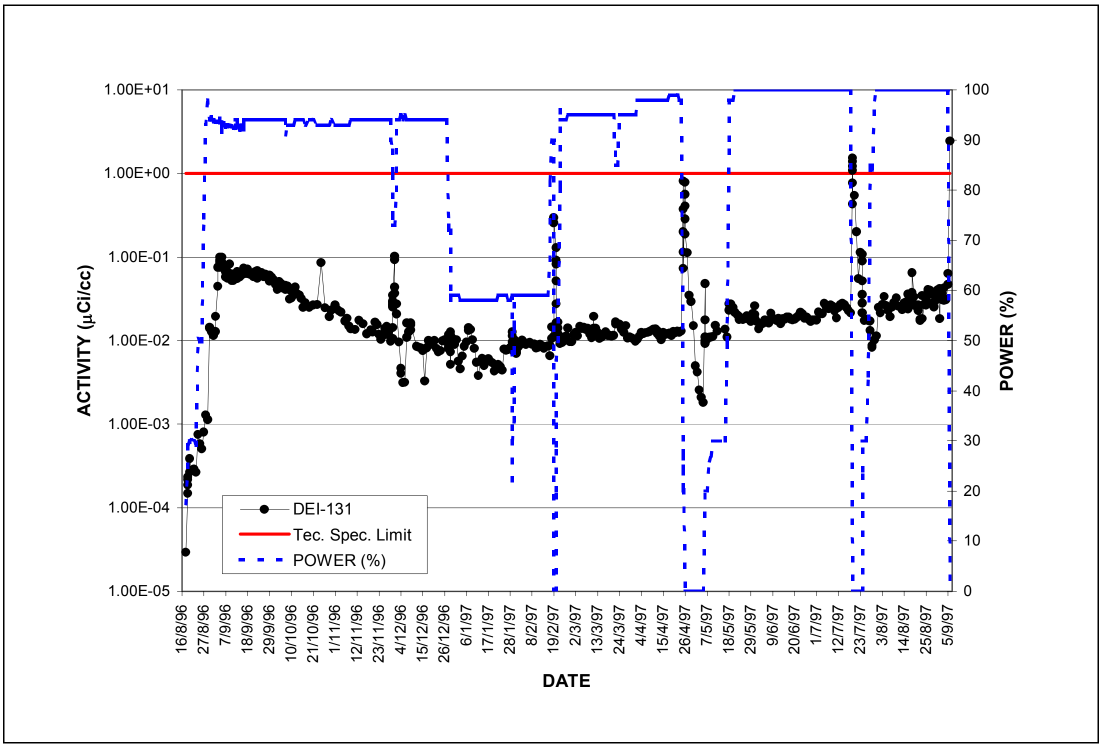


FIG. 4. Angra 1 cycle 6: dose equivalent I-131.

From mid-January, DEI-131 started to increase smoothly up to about the end of July. However, from August to the end of cycle (EOC) the rate of activity increase was significantly higher than in the previous 6 months. On 05 September 1997, the last DEI-131 measured before the scheduled plant shut down was about 0.05 µCi/g. This value is twice as high that one detected one month before, but approximately the half of the maximum DEI-131 value measured during cycle 6 (0.1 µCi/g, on 4 September 1996). Therefore, it should be emphasized that dose-equivalent I-131 remained significantly below the Technical Specifications limit (1µCi/g) during cycle 6.

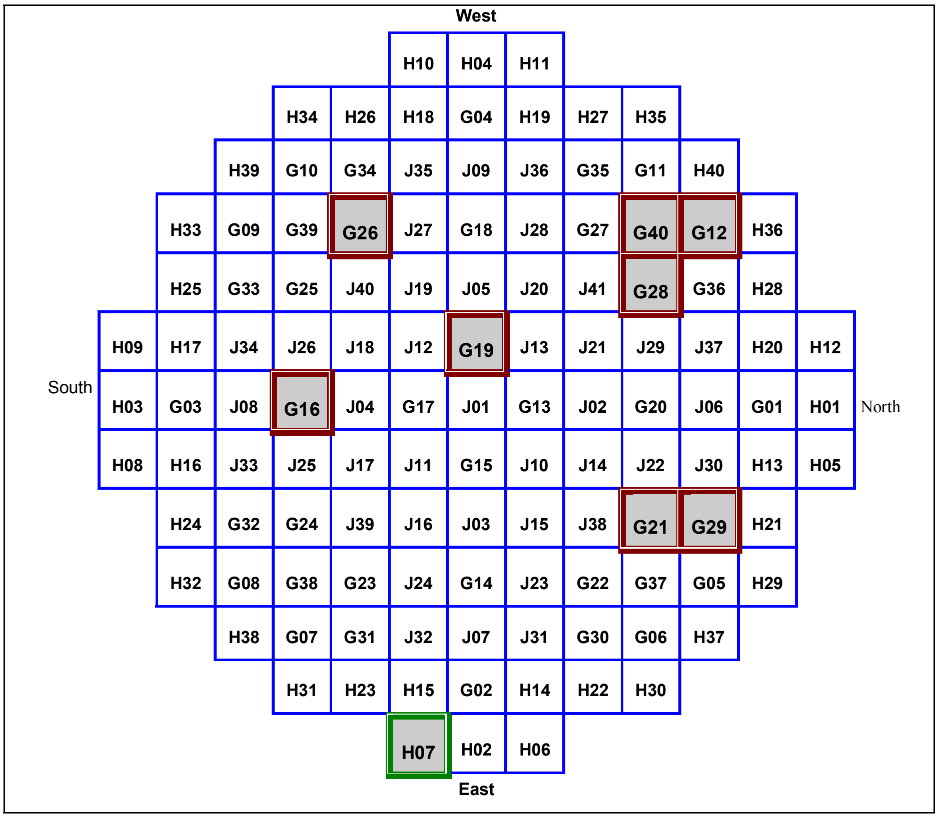


FIG. 5. Angra 1 cycle 6: Failures location.

A number of different inspections were carried out: in-mast sipping (IMS), wet sipping and visual inspection, Ref. [3,4,5,6,7,8]. The tests identified the following:

- Eight failed FAs in batch G [G12, G16, G19, G21, G26, G28, G29 and G40] and one in batch H [H07] (see Fig. 5);
- Fretting wear on spacer grid side-strips of several FAs.

The main observations on grid side-strip fretting wear were:

- Fretting limited to west and east faces; and hammered out areas on south and north faces;
- Fretting most severe on grids 4 and 5, less on grids 3 and 6.
- Fretting marks observed in batches G, H and J (see Fig. 6).

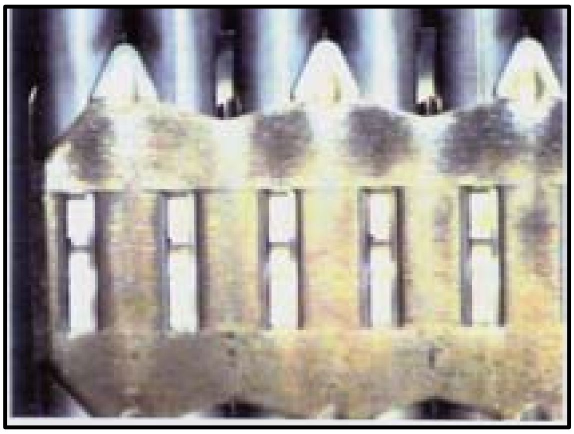
The key observations on the fuel failures are shown below:

Batch G

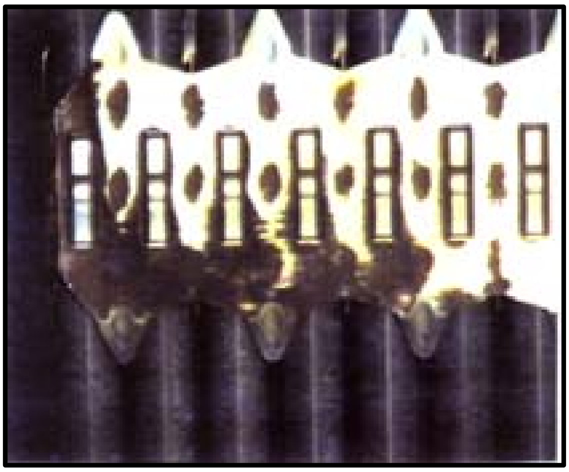
- At least 5 “loose” rods (rod movement): 2 rods on east face of G28 and one corner rod each in G16, G19, and G21.
- Rod fretting wear under contact spring: G21 south left corner; G19 east corner (both “loose” rods).
- Hydride blister: G28 east; G21 south.
- No apparent primary failure locations.
- All eight FAs appeared to maintain good structural integrity; no degradation except the “loose” rods.

Batch H:

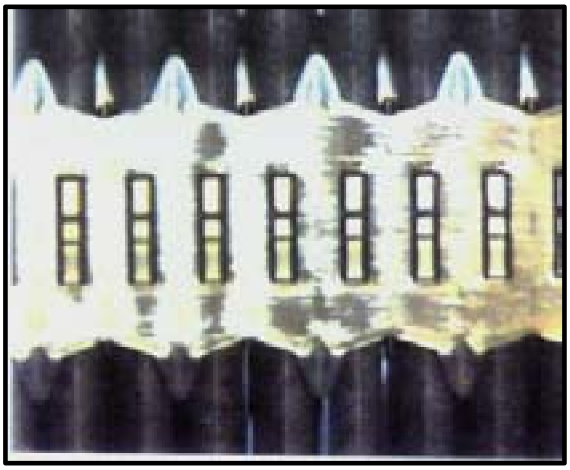
- Severe damage on FA H-07 at FR 14, at face north below grid 2. Possible debris mark below grid 8 (bottom) on same rod. A metallic debris found at face north, above grid 8, between rods 6 and 7 (see Fig. 7)



Fuel Assembly G04, grid 4, face west.



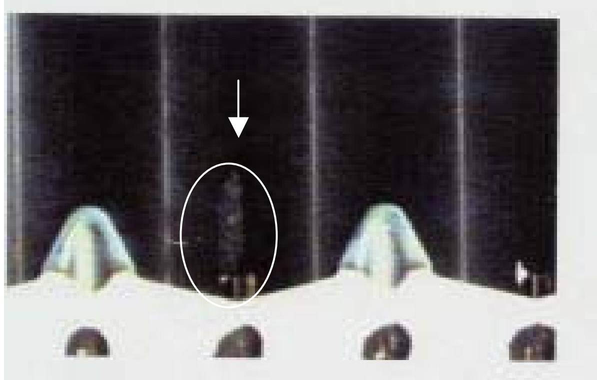
Fuel Assembly H19, grid 6, face west.



Fuel Assembly J01, grid 3, face east.

FIG. 6. Angra 1 cycle 6: wear marks on spacer grids side strips.

Face north, fuel rod 14, between grids 2and 3.



Face north, between rods 6 and 7, grid 8: debris.

FIG. 7. Angra 1, cycle 6, fuel assembly H07.

3.1 Root cause evaluation

3.1.1. Fuel assembly H07

H07 is most likely to have failed by debris or by damage during core loading. The FR 14 probably degraded by secondary hydriding below grid 2. This large failure could explain the activity behavior in cycle 6. Our interpretation is that the high I-131 activity at the beginning of cycle has been caused by FR 14 failure early in cycle. It should be still emphasized that the high RCS activity due to H07 failure masked batch G failures.

3.1.2. Batch G

The root-cause analysis presented below is based on the information available at the end of 1997,Ref. [9]. The visual inspections could neither show the primary failure locations of batch G nor determine conclusively the failure mechanism. However, the failures and the fuel damages detected in cycle 4 can help us interpreting the event observed in cycle 6. As stated in item 2, grid-to-rod fretting has been identified as the failure mechanism of batch D in Angra 1 cycle 4. Comparisons between the predominant characteristics registered by the inspections carried out during the outages of Angra 1 cycles 4 and 6 are shown in Table II. The analysis of this table shows that there are significant differences between the visual observations in cycles 4 and 6. However, the wear marks on some batch G fuel rods, together with “loose” rods on leaking FAs, are strong evidences that grid-to-rod fretting was the main failure mechanism of batch G in cycle 6.

Table II. Comparison between Angra 1 cycles 4 and 6 fuel failures

	CYCLE 4 (BATCH D)	CYCLE 6 (BATCH G)
FRETTING MARKS ON THE FUEL RODS	*MANY*	*6*
SLIPPED FUEL RODS DOWN ONTO THE FUEL ASSEMBLY LOWER END FITTING	*MANY*	*NONE*
“LOOSE” RODS IN THE GRID CELLS	*MANY (grids #1 to #8)*	*5 (mid-grids/corner rods)*
GRID SIDE-STRIP FRETTING WEAR	*RARE*	*MANY (mid-grids/east-west direction)*
AVERAGE DISCHARGE BURN-UP (MWD/kgU)	*26*	*22.8*
SECONDARY DAMAGE (HYDRIDE BLISTERS)	*YES*	*YES*

The number of leaking FRs in cycle 4 was much higher than in the sixth cycle; no sliding of FRs through the grid cells was observed in cycle 6. Therefore, apparently the rod/spring grid contact of batch G was better than that one of batch D. Nevertheless, as can be seen from Table II, the average discharge burn-up of batch D in cycle 4 was 3.2 MWD/kgU higher than the average discharge burn-up of batch G in cycle 6. A simple calculation shows that, in order to reach the same discharge burn-up as batch D, fuel assemblies G would have to remain more 103 EFPD in the core. Probably, the damages caused by fretting during an additional period of about 3.5 effective full power months would not be negligible. However, as we have no means to predict the future fuel deterioration, we were constrained to evaluate the failure root-cause in function of the evidences arising from the inspections. These evidences pointed out to a better rod/spring grid contact of batch G, when compared to batch D. Nevertheless, in spite of the higher grid spring force of batch G, there has been a recurrence of fretting in cycle 6. Undoubtedly, the friction observed on spacer grids side-strips of several FAs is the key for understanding this phenomenon.

Rare in cycle 4, the wear marks were the dominant characteristic of cycle 6 event, indicating that occurred a unidirectional contact (east-west) between neighbour FAs during operation. Abnormal vibrations of the FAs probably caused the grid-to-grid contact. The unexpected vibrations were most likely induced by the reactor coolant flow. In other words, flow-induced vibration (FIV) would have been again the root-cause of the fretting failures in Angra 1. The new spring design introduced in batch G, having as main characteristics a new shape and a higher initial force, did not correct the problem, because FIV is fundamentally related to the mixing vane design (shape/orientation), which was not changed from batch D to batch G. The reinforced springs led to a better rod/spring grid contact, resulting in less “loose” rods in cycle 6. On the other hand, the higher spring forces changed the vibrational characteristics (amplitudes and frequencies of the different modes) of the FAs. Due to mechanisms not clearly understood, under the action of the coolant flow the vibrations of the reinforced FAs were intensified, leading to the strong grid-to-grid interactions showed by the visual inspections in cycle 6. Although these hypotheses were reasonable - the fuel vendor agreed in general with them, Ref. [5] - they had to be confirmed by experimental tests (as it will be shown in item 6).

4. FUEL LOADING CYCLE 7

The cycle 6 fuel failures had some important effects:

- The Regulatory Authority has imposed tougher licensing requirements.
- During cycle 7, the RCS radiochemistry analyses were performed at least once a day.
- Batch G was discarded.
- Cycle 7 startup delayed approximately 2 months.
- Cycle 7 was very short (223 EFPD).
- Batches H (twice burned) was discarded at the end of cycle 7.
- An entire new core with 121 original Angra 1 fuel assemblies has been purchased from Westinghouse to load cycle 8.

Cycle 7 core loading came from four sources:

- A new reload batch L (40 FAs), manufactured by Siemens/INB using the same design as batches G, H and J.
- Forty-one assemblies from batch J used in cycle 6.
- Thirty-six assemblies from batch H used in cycle 6.
- Four F assemblies that had been used in cycles 4 and 5.

Batch G was scheduled to return to the core in cycle 7. However, due to the systematic failures detected, ELETRONUCLEAR decided to discard the 40 G FAs. They were replaced by J batch that had been originally purchased to be used exclusively in cycle 6. Besides, H07 and its three symmetric FAs were replaced by four F assemblies, which had been used in cycles 4 and 5. As the residual reactivity of batch J was very low, cycle 7 length was the shortest one of Angra 1 operation history: 223 EFPD.

The Brazilian Regulatory Authority (CNEN) main requirements to license cycle 7 operation were related to the estimation of the fuel failures propagation in cycle 7 as a function of burn-up, and the evaluation of the mechanical integrity of the 81 FAs that would return to the core in cycle 7. For the first item, ELETRONUCLEAR estimated the limiting burn-up as a function of cycles 4 to 6 experience. For the second item, Siemens, INB and the brazilian research institute IPEN/CNEN-SP analyzed the influence of the friction at the outer surface of the spacer grids on the operational behavior of the FAs, and the influence of impacts due to vibrations on the integrity of guide thimbles. The results arising from experimental tests, Ref. [10], and theoretical calculations pointed out to the conclusion that the FAs would remain functional during cycle 7. Therefore, safety related control rod insertion malfunctions and loss of fuel assembly cooling geometry were not expected to occur. Based on these analyses, CNEN allowed ELETRONUCLEAR to operate cycle 7 initially during only 80 EFPD, and could grant an extension depending upon the RCS activity level detected at the end of this licensed period.

Cycle 7 started on 04 December 1997. Next to the end of the licensed operation period (80 EFPD), theoretical calculations based on the RCS activities led to a number of about 2 leaking fuel rods in the core. However, due to the relatively low activity levels detected, which were well below the Technical Specifications limit (see Fig. 8), CNEN authorized the continuation of the cycle operation. Cycle 7 ended on 17 October 1998. Inspections performed during outage showed 2 failed assemblies: H36 and F29 (see Fig. 9). It should be stressed that the single failure of batch H was on H36, the lowest burnt FA of this region (18.1 MWD/kgU).

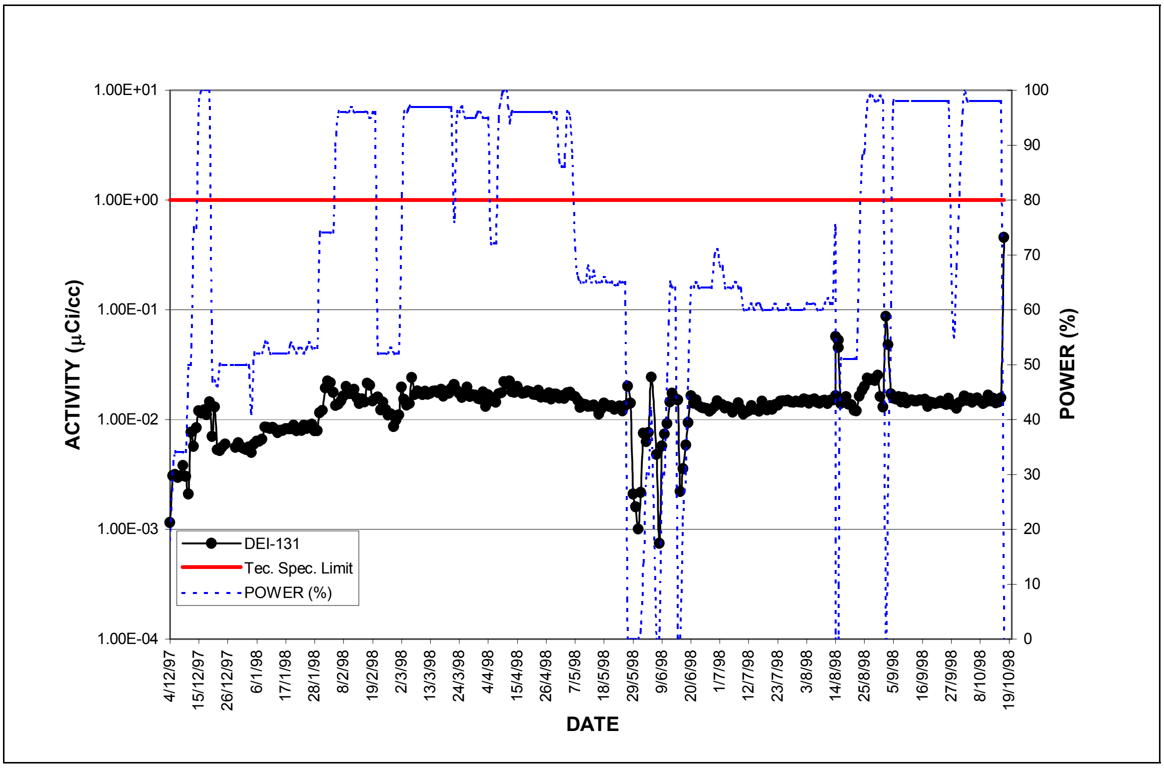


FIG. 8. Angra1 cycle 7: dose equivalent I-131.

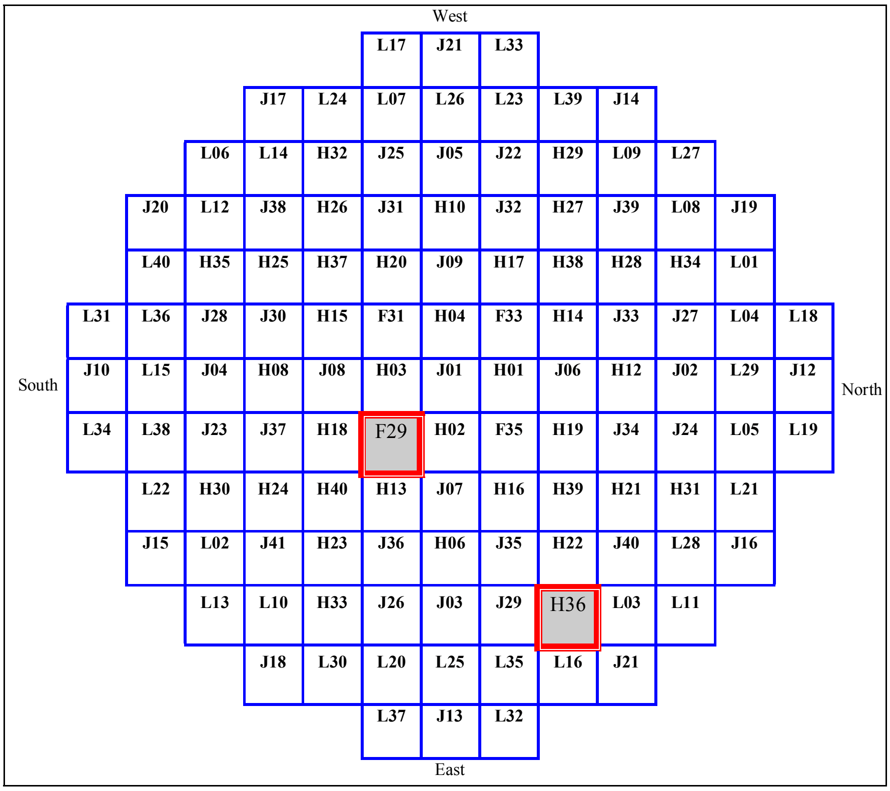


FIG. 9. Angra 1 cycle 7: Failures location

5. CYCLE 8 FUEL PERFORMANCE: THE FAILURES MITIGATION

Perhaps the most important consequence of the failures in cycle 6 was that an entire new core with 121 original Angra 1 fuel assemblies was purchased from Westinghouse to load cycle 8. This *corrective action*, i.e., the use of a proven fuel design, was expected to eliminate the fretting failures experienced by Angra 1. Additionally, as the FAs supplied by the plant designer have built-in anti-debris filters it was also expected that the recurrence of failures by debris would be minimized.

Three batches composed the cycle 8 core:

- M (41 FAs enriched to 2.1%)
- N (40 FAs enriched to 2.6%)
- P (40 FAs enriched to 3.245%)

Cycle 8 length was 394 EFPD. It started on 11 December 1998 and ended on 19 April 2000. The fuel performance was very good. No failures were observed.

6. FLOW EXPERIMENTS OF ANGRA 1 FUEL ASSEMBLIES

Along 1998, Siemens/INB performed laboratory flow vibration tests in order to assess the susceptibility of G fuel assemblies to flow-induced vibration, Refs [11,12,13,14]. The tests were done in a low-pressure loop (< 110°C, < 6,5 m/s, <10 bar). At select locations, the flow velocity, and the vibration of the rods and that of the fuel assembly were measured. Inductive displacement measurement devices were applied to determine the distance between the spacer grids and the inner test channel wall. Several windows, in the test rig wall, allowed measurement of the flow velocity.

The main observations were:

- The vibration was only significant perpendicular to the cross flow direction, determined by the unidirectional mixing vane pattern. It was assumed that this self-induced excitation was directly related to cross flow. There was a sharp resonance, for the fuel assembly vibration perpendicular to the mixing vane direction, in the range of 25-27 Hz.
- The amplitude of vibration was dependent on the coolant flow velocity. For flow velocities from 5.2 to 5.6 m/s the amplitude increased and for flow velocity from 5.7 to 6.1 m/s the amplitude decreased. The maximum FA vibration amplitude was on the range of 100 microns.
- The resonance seemed only to occur for sufficient strong grid spring forces, related to the beginning of life (BOL) condition. The resonance was not observed for the simulation of end of life (EOL) condition, with very low force at the grid spring.
- For BOL condition the fuel assembly, including the cage and fuel rods, was vibrating in a highly synchronized manner. The neighbour spacer grids were vibrating with a phase difference of 180^o, and the motion of the rods in each span was fully determined by the corresponding spacers.
- The resonance was considerably reduced by the insertion of a damper inside the fuel assembly (rods inside the guide tubes).

Based on the test observations, Siemens/INB recommended the reinsertion of the partially burned FAs (Batch L) in the core, but using dampers inside them.

Based also on the test information, it was developed a new mixing vane design, using split pattern. Tests carried out on a modified fuel assembly (PS4) using the new spacer grid design showed that the resonance was not any more a problem and no restriction was found to the usage of this new design related to the fuel rod fretting behavior.

7. CYCLE 9 FUEL PERFORMANCE

Since the end of cycle 7 there were more than 150 FAs prematurely stored in Angra 1 spent fuel pit, due to propensity for fretting failures. As the remaining reactivity of batches E, F, G, H (twice burned) and L (once burned) is highly significant, ELETRONUCLEAR analyzed the viability of starting a fuel assembly reconstitution program. Nevertheless, there was an obstacle for accomplishing that goal: the lack of a qualified Siemens/INB skeleton. However, the flow test results presented in item 6 indicated that, under laboratory condition, the new FA designed by Siemens/INB (PS4) had no susceptibility to flow-induced vibration. Eletronuclear decided then to launch a qualification program for the PS4 design. Four lead test assemblies (batch Q - 3.4% enriched) were then inserted in the Angra 1 core in cycle 9. The other FAs came from the following sources:

- A new reload batch R (16 FAs - 3.4% enriched), manufactured by Westinghouse (the same design as batches M, N and P).
- Twenty-one assemblies from batch M used in cycle 8.
- Forty assemblies from batch N used in cycle 8.
- Forty assemblies from batch P used in cycle 8.

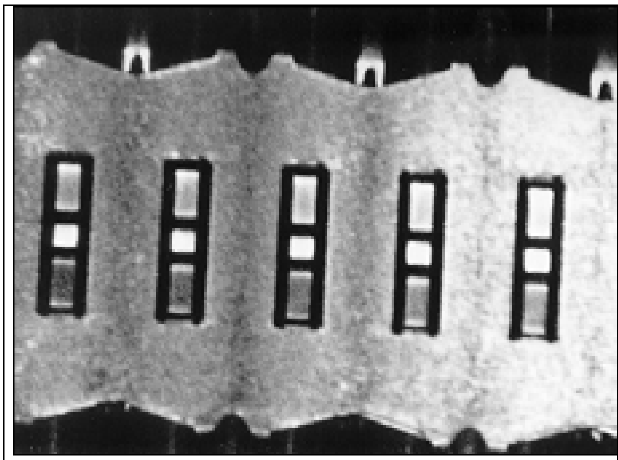
Cycle 9 started on 11 July 2000 and ended on 07 April 2001. The goal of zero fuel failures was achieved again.

8. CYCLE 10: REINSERTION OF PARTIALLY BURNED FUEL ASSEMBLIES WITH DAMPING DEVICES

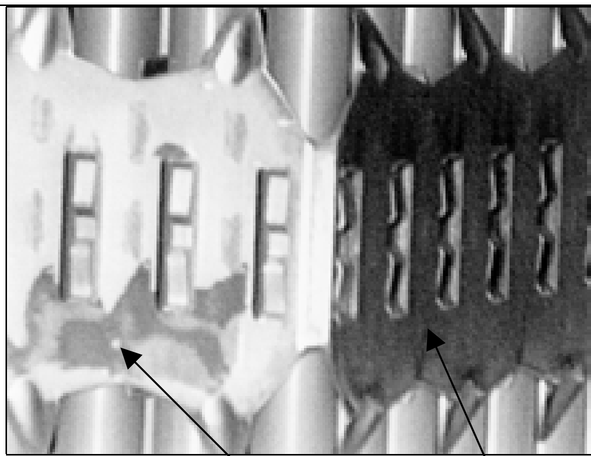
At the end of cycle 7, the burn-up of batch L (Siemens/INB design) ranged from 5 to 11 MWD/kgU. As a result of the flow experiments discussed in item 6, the fuel vendor recommended the reinsertion of partially burned FAs of batch L, with damping device, into Angra 1 core. Due to the high remaining reactivity of those elements, ELETRONUCLEAR decided then to use 12 L FAs to compose cycle 10 core.

Before the use of the batch L fuel assemblies all them (40 FAs) were visually inspected, Ref. [15]. The main observations of the inspection were:

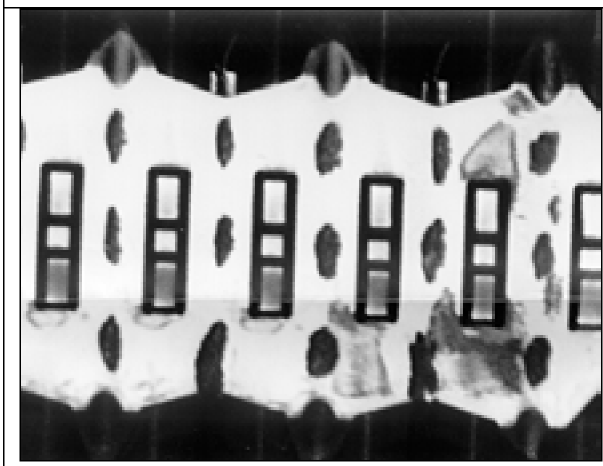
- All FAs showed wear marks at the side strip of spacer grids, fretting limited to west and east faces; and hammered out areas on south and north faces (see Fig. 10);



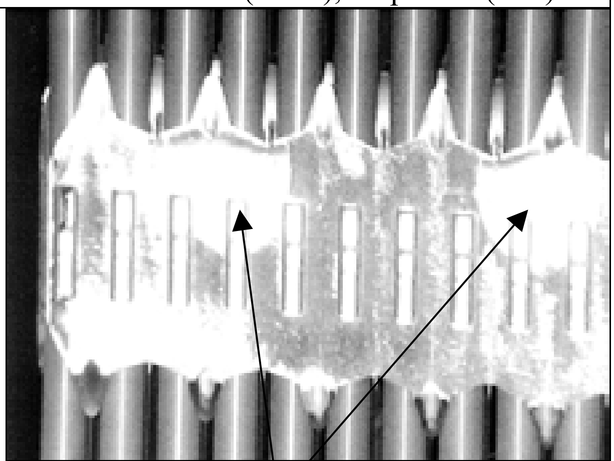
FA L21 – face east– grid 3 – side strip wear



FA L21 – corner south/east– grid 3 – side, hammered out (south), strip wear (east)



FA L21 – face south – grid 4 – side strip hammered out



FA L27 – face west – grid 4 – side strip wear (baffle position)

FIG. 10. Angra 1, batch L, wear marks at spacer grids side strips.

- FA L04 had one of the grids damaged by excessive wear (see Fig. 11);
- 25% of FAs showed also wear at the grid spring of the side strip, which can cause loss of the FR fixing force and can lead to fretting (see Fig. 12);
- FAs L08 and L01 showed damage in spacer grids due to improper handling during core loading or unloading (see Fig. 13).

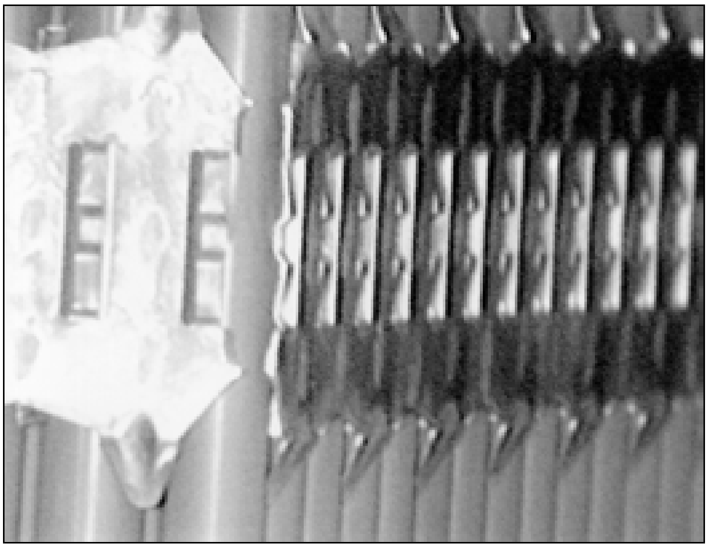
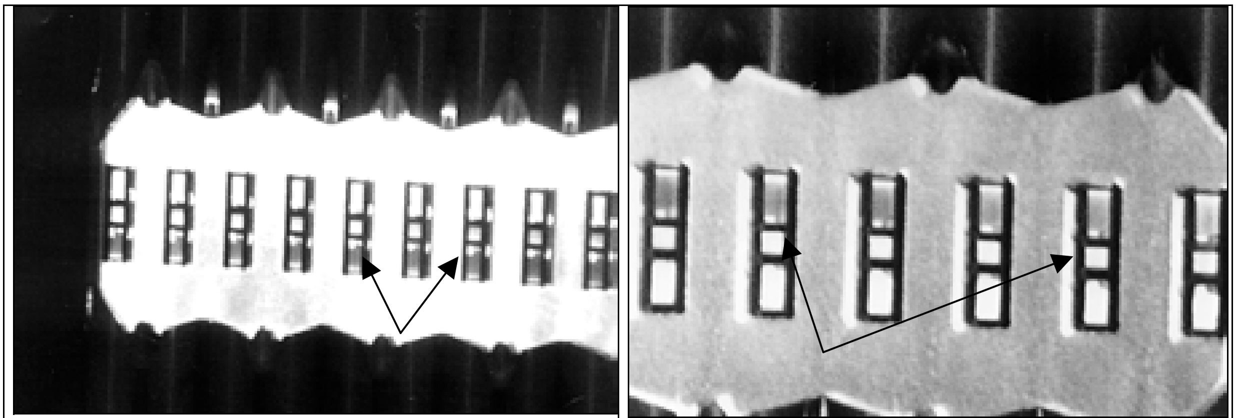
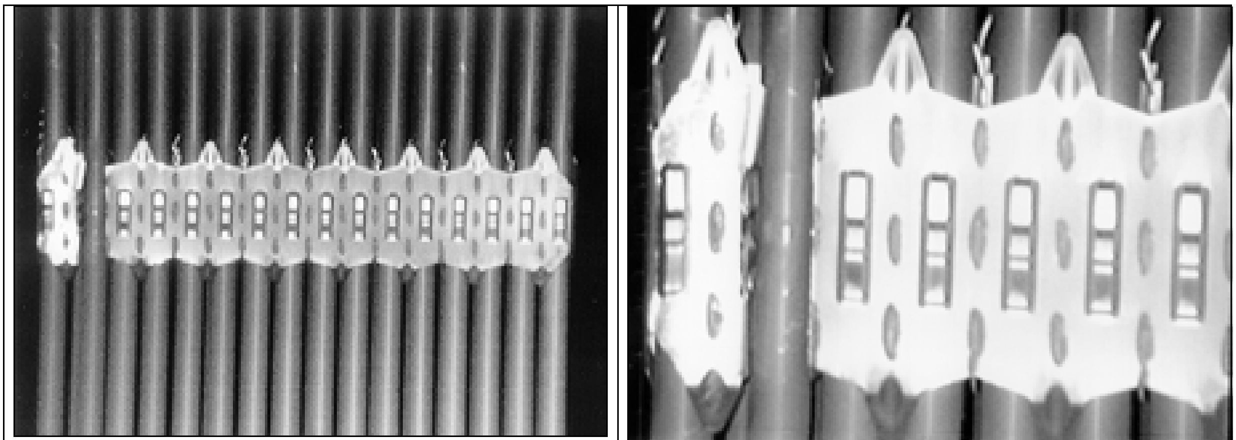


FIG. 11. Angra 1 batch L - FA L04, face south, grid 4, damaged strip by wear.



FA L27 – face east – grid 5 – spring wear	FA L18 – face east – grid 4 – spring wear

FIG. 12. Angra 1 batch L spacer grid strip and spring wear.



FA L08 – face north – grid 2	FA L08 – face north – grid 2

FIG. 13. Angra 1 batch L, grid damaged by handling.

The pattern of wear verified in batch L, which has a low average burn-up, matches well with the flow induced vibration for beginning of life conditions simulated in flow test at laboratory as shown in item 6. Nevertheless the amplitude of vibration observed were much bigger (more than 10 times) in reactor than in laboratory rig.

Based on the visual inspection results, 12 batch L fuel assemblies were chosen for being used with dampers. Besides these 12 FAs, Angra 1 cycle 10 core was loaded with:

- a new reload batch S (40 FAs - 3.4% enriched), manufactured by INB using Westinghouse technology;
- one assembly from batch M used in cycle 8;
- eight assemblies from batch N used in cycle 9;
- forty assemblies from batch P used in cycle 9;
- four lead test assemblies (batch Q) used in cycle 9;
- sixteen assemblies from batch R used in cycle 9.

Cycle 10 started on the 6 June 2001 and will finish at 20 of July 2002. No failures had been detected up to the middle of June 2002. Visual inspections of cycle 10 FAs are scheduled for being done during the next outage, when the mechanical behavior for flow induced vibration will be verified.

9. FAILURE ANALYSIS - DISCUSSION

The structure of a PWR fuel assembly is formed by a support structure (skeleton) composed by end nozzles, guide tubes and spacer grids. The guide tubes are rigidly fixed to the end nozzles and the spacer grids. The spacer grids support the fuel rods through fixing devices that allows the differential movement between the fuel rod and the spacer grids (or the fuel rod and the skeleton). The fuel assembly mechanical stiffness depends mainly on the stiffness of the fuel rod-spacer grid fixing device. This fixing device stiffness (in each of the three directions of displacement and the three directions of rotation) depends on the spacer grid spring and dimple stiffness, and on the geometry characteristics of the pair spring-dimple in each spacer grid cell. It is important to realize that the forces acting on a fuel rod "*flow*" from the rod to the spacer grid, then to the guide tubes and then to the end nozzles which receives the reaction of the core support plate structure. The forces acting on the fuel rod are the hydraulic ones and those from differential displacements due to thermal gradients and to irradiation growth. The fuel assembly displacements and rotations result from all forces acting in all fuel rods. An important force acting on the fuel rod is that due to the coolant flow. The force acts mainly perpendicular to the axis of the fuel rod and it is due to the parallel flow of the water and due to cross-flow originated from mixing vanes in the spacer spring or from thermal gradients and flow gradients along the fuel assembly. It is well known among fuel designers that the fuel rod is mainly excited in its lower mode of vibration by the coolant flow, see [16]. This is an important point in the behavior of FAs in the reactor and to identify fuel rod failure by fretting.

Perrotta, Ref. [17], developed, using a matrix method, the computer code ELCOM for PWR fuel assembly static structural analysis. The method description helps the understanding of the fuel assembly structural behavior. The main assumption of the method is that the spacer grid behaves like a rigid body, so each of its cells node displacement or rotation can be related to the displacement and rotation of the grid center of gravity. The method takes into account all fuel rods, guide tubes, spacer grids and end nozzles and constructs an equivalent beam to the fuel assembly where the beam nodes are at the center of gravity of each spacer grid and end nozzles. Each fuel rod or guide tube is considered as a local system and the equivalent beam as the global system in the matrix method. The method considers up to three displacement and three rotation directions (three-dimensional problem). The method considers linear behavior for small displacements.

The fuel rod local system can be described by:

$$\{R_{vi}\}+\{R_{Mi}\}= [K_{vi}]\{r_{vi}\} \qquad (1)$$

$\{R_{vi}\}$ – vector of the equivalent nodal external forces acting on the ith fuel rod (this could be from thermal gradients, mechanical forces, irradiation growth, and hydraulic forces);
$\{R_{Mi}\}$ – vector of the reaction forces from the fixing device in the spacer grid (due to the spring-dimple system in the grid cell);
$[K_{vi}]$ – stiffness matrix of the i^{th} fuel rod;
$\{r_{vi}\}$ – vector of the nodal displacement of the i^{th} fuel rod.

The equation (1) can be set for all fuel rods taking the subscript *i* out, and the size of the vectors will be 6*m*, being *m* the number of fuel rods.

The guide tube local system can be described by:

$$\{R_{tj}\}+\{R_{sj}\}= [K_{tj}]\{r_{tj}\} \qquad (2)$$

$\{R_{tj}\}$– vector of the equivalent nodal external forces acting on the j^{th} guide tube;
{Rsj} – vector of the reaction forces from the fixing device in the spacer grid (considered to be a rigid joint);
$[K_{tj}]$ – stiffness matrix of the j^{th} guide tube;
$\{r_{tj}\}$ – vector of the nodal displacement of the j^{th} guide tube.

The equation (2) can be set for all guide tubes taking the subscript *j* out, and the size of the vectors will be 6*n*, being *n* the number of guide tubes.

The displacement vector of the local system of the fuel rod - grid joint (spring or dimple) is given by:

$$\{r_{Mi}\} = \{r_{gi}\} – \{r_{vi}\} \qquad (3)$$

$\{r_{gi}\}$ –displacement vector of the ith node at the spacer-grid cell (external node);

The reacting force in the fuel rod-spacer grid-fixing device is given by:

$$\{ R_{Mi}\} = [K_M]\{r_{Mi}\} = [K_M](\{r_{gi}\} – \{r_{vi}\}) \quad \text{(before fuel rod sliding)} \qquad (4)$$

$$\{ R_{Mi}\} = \{R_o\} \quad \text{(after fuel rod sliding)} \qquad (4a)$$

$[K_M]$ is the stiffness matrix (assumed linear) of the joint device (spring-dimple).
$\{R_o\}$ is the vector of the limiting forces for fuel rod sliding in the spacer grid.

The equation (4) can be set for all fuel rods – spacer grid system taking the subscript *i* out. The size of the vectors will be 6*m*, being *m* the number of fuel rods.

Assuming rigid body movement for the spacer grid, the following relations are obtained:

$$\{r_g\} = [A_M]\{r_G\} \tag{5}$$

$$\{r_t\} = [A_t]\{r_G\} \tag{6}$$

$\{r_G\}$ –displacement vector for the grid center of gravity;
$[A_M]$ – transformation matrix for the fuel rods;
$[A_t]$ – transformation matrix for the guide tubes.

The equilibrium equation for the spacer grids and end nozzles (global system) is given by:

$$\{R_G\} = [A_M]^T\{R_M\} + [A_t]^T\{R_s\} + [A_f]^T\{R_f\} \tag{7}$$

$\{R_G\}$ – vector of the external forces acting in the grids and nozzles;
$[A_f]^T.\{R_f\}$ – this term represents the vector of reacting forces of the fuel assembly fixing spring at the top nozzle, and this term can be written as:

$$[A_f]^T\{R_f\} = [A_f]^T(-[K_f]\{r_{f0}\} - [K_f][A_f]\{r_G\}) \tag{8}$$

$[A_f]$ – transformation matrix for the fuel assembly fixing spring;
$[K_f]$ – stiffness matrix (assumed linear) of the fuel assembly fixing spring;
$\{r_{f0}\}$ – initial displacement vector of the fuel assembly fixing spring;

The following system of equations can be written:

$$\{R_v\}+[K_M]([A_M]\{r_G\}- \{r_v\}) - [K_v]\{r_v\} = \{0\} \tag{9}$$

$$\{R_G\}-[A_M]^T[K_M]([A_M]\{r_G\}-\{r_v\})+[A_t]^T\{R_t\}-$$
$$[A_t]^T[K_t][A_t]\{r_G\}+[A_f]^T[K_f]\{r_{f0}\}+[A_f]^T[K_f][A_f]\{r_G\}) = \{0\} \tag{10}$$

Taking the value of $\{r_v\}$ in (9) and replacing it in (10) gives the equilibrium equation for the equivalent beam to the fuel assembly:

$$\{R_G\}+[A_t]^T\{R_t\}+[A_f]^T[K_f]\{r_{f0}\}+[A_M]^T[K_M]([K_M]+[K_v])^{-1}\{R_v\} =$$
$$([A_M]^T[K_M][A_M]-[A_M]^T[K_M]([K_M]+[K_v])^{-1}[K_M][A_M] +[A_t]^T[K_t][A_t]+ [A_f]^T[K_f].[A_f]).\{r_G\} \tag{11}$$

Equation (11) can be written as:

$$\{F_G\} = [K_G]\{r_G\} \tag{12}$$

Where:

$$\{F_G\} = \{R_G\} + [A_t]^T\{R_t\}+[A_f]^T[K_f]\{r_{f0}\} + [A_M]^T[K_M]([K_M]+[K_v])^{-1}\{R_v\} \tag{13}$$

$$[K_G] = [A_M]^T[K_M][A_M]-[A_M]^T[K_M]([K_M]+[K_v])^{-1}[K_M][A_M]+[A_t]^T.[K_t][A_t]+[A_f]^T[K_f][A_f] \tag{14}$$

It can be seen from equation (14) that the stiffness of the equivalent beam of the fuel assembly takes into account the stiffness of all components: fuel rods, guide tubes, fuel rod-spacer grid fixing device, fuel assembly fixing spring. As the value of the stiffness of the fuel rod fixing device increases, the stiffness of the fuel assembly increases. The opposite is true, that means, as the stiffness (or the fixing force) of the fuel rod in the spacer grid decreases, the stiffness of the fuel assembly decreases. As the fixing force of the fuel rod decreases along burn-up (mainly by creep), the stiffness of the fuel rod (local system) and the stiffness of fuel assembly (global system) decrease along burn-up either.

It can be seen from equation (13) that the forces acting in the equivalent beam of the fuel assembly come from forces acting directly on the spacer grids, the compression of the fuel assembly fixing spring, the forces acting on the guides tubes and the forces acting on the fuel rods. These last ones are transmitted to the fuel assembly by the fuel rod – spacer grid fixing device. As the fixing force of the fuel rod decreases along burn-up (mainly by creep), the forces transmitted from the fuel rod (local system) to the fuel assembly not necessarily decreases, but the differential displacement between the fuel rod and the grid (fuel rod sliding in the grid) certainly increases, which may increase fretting in the fuel rod along irradiation.

It is interesting to compare the values of stiffness for the fuel rod – spacer grid fixing device and the fuel rod stiffness (terms of $[K_M]$ and $[K_v]$). Table III presents these values. From this table is seen that the shear stiffness of the fixing device is much bigger than the fuel rod value. This means that for the term $([K_M] + [K_v])$ $[K_M]$ is the most important for beginning of live and also for end of live. For rotation there is a higher importance of the term $[K_M]$ for beginning of live, but $[K_v]$ gets important for end of live.

Table III. Comparison between stiffness values

Spacer Grid Device Stiffness Value for Beginning of Live		***Fuel Rod***		
		Stiffness Value (*)	***Between 1st and 2nd spacer grid (620.5mm)***	***Other segments (522 mm)***
Shear (Spring)	~60 N/mm	$12\ EJ/l^3$	0.69 N/mm	1.16 N/mm
Shear (Dimple)	~ 600 N/mm	$12\ EJ/l^3$	0.69 N/mm	1.16 N/mm
Rotation	~ 400000 Nmm/rad	4EJ/l	88345 Nmm/rad	105016 Nmm/rad

(*) $E = 7{,}8x10^4\ N/mm^2$; $J = 175{,}7\ mm^4$

From the static equilibrium system shown before, one can estimate the natural frequencies of the local system (fuel rod) and the fuel assembly equivalent beam. The complete time dependent equation may be written:

$$[m_{vi}]\{\ddot{r}_{vi}\}+[c]\{\dot{r}_{vi}\}+([K_M]+[K_v])\{r_{vi}\}=\{R_{vi}(t)\} \qquad (15)$$

$$[M]\{r_G\}+[C]\{r_G\}+[K_G]\{r_G\}=\{F_G(t)\} \qquad (16)$$

$[m_{vi}]$, $[M]$, $[c]$, and $[C]$ are the mass and damper matrix at the local system (fuel rod) and fuel assembly equivalent beam. $\{R_v(t)\}$, and $\{F_G(t)\}$ are the forces, same definition as Eq.(13), but time dependent. The damping behavior for the fuel rods depends on the structural damping, the pellet cladding interaction (burn-up dependent), friction between the fuel rod and the grid spring and the viscosity damping (temperature dependant). The damping behavior of the fuel assembly depends on the damping of all fuel rods plus the interaction of the guide tubes with the grids and the rods inserted (control rods or burnable poison rods).

As in the case of the system stiffness, the fuel rod and the fuel assembly natural frequencies will increase or decrease as a function of the stiffness (or fixing force) of the fuel rod fixing device. As the fixing force of the fuel rod decreases along burn-up (mainly by creep), the natural frequencies of the fuel rod (local system) and of the fuel assembly decrease. Fig. 14 shows the modes of vibration of the fuel assembly and fuel rod in the plane along the axial direction (dry condition, without added mass). Table IV presents the variation of the natural frequencies with the stiffness values of the spacer grid spring device.

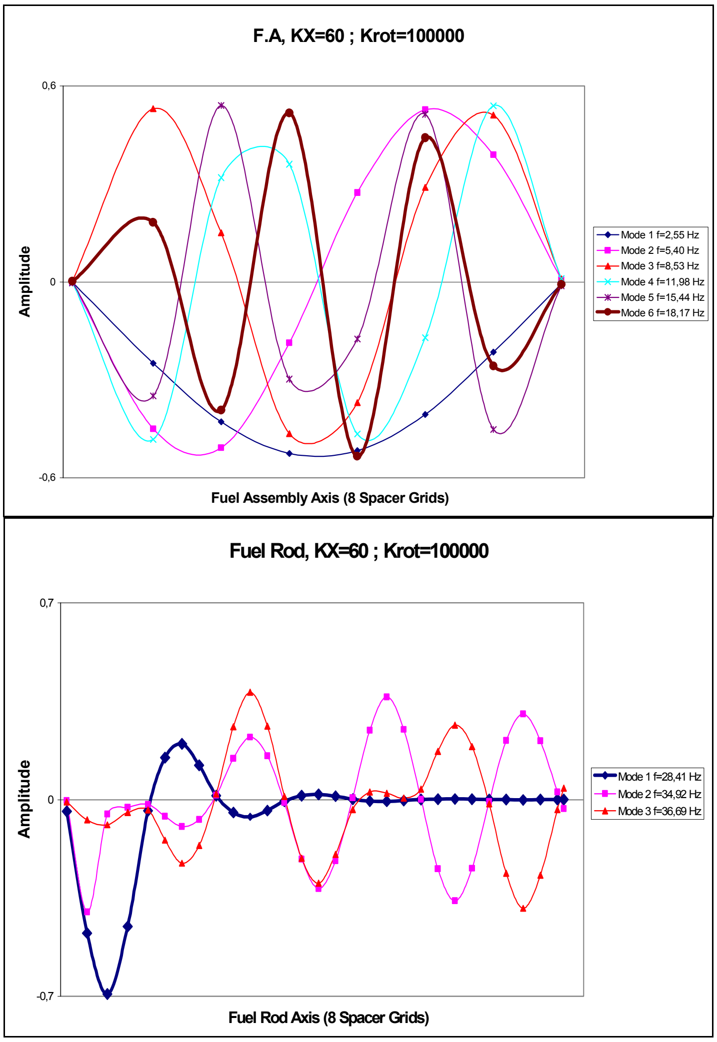


FIG. 14. Fuel Assembly and Fuel Rod Natural Frequencies and Modes of Vibration (BOL).

The main cause of Angra 1 fuel failure in cycles 4,6 and 7 was related to rod to grid fretting due to fuel rod and fuel assembly vibration. Cycle 4 (batch D) fuel failures showed a pattern related to fuel rod vibration. Fuel assembly burn-up were higher than 20 MWD/kgU, and were related to the third cycle of the fuel assembly inside the core. Ultrasonic tests performed showed that the majority of failed fuel rods were located in the peripheral rows of the failed fuel assemblies and there were a higher concentration of failed rods at the north and south sides, which gave an indication of preferential direction of flow induced forces acting on the fuel rods (perpendicular to the mixing vane direction). The grid position where fretting and loosed rods were observed in visual inspection can be seen in Fig. 15. The axial position of fretting in the fuel rod matches very well with the highest amplitude of the first natural mode of vibration of the fuel rod. The grids 3 and 4 (from bottom to top of the F.A) would be the ones where spring relaxation along burn-up would be the highest. Although for some assemblies (D14 is a good example, see Fig. 16) fretting was observed also in the first grid (F.A bottom). These fuel assemblies should have had problems during assembling (procedure or equipment) and the first grid spring forces should be very small. Again, in these cases, there is a good match of the highest amplitude of vibration for the first natural mode of the fuel rod with the fretting position. The root cause of the cycle 4 (batch D) failure is related to the cross-flow generated by the mixing vane pattern that leads to a force acting on the fuel rod much higher than that assumed by design.

Table IV– Comparative values of frequencies for Angra 1 fuel assembly

Fixing Device Shear Stiffness (N/mm)	*Fixing Device Rotation Stiffness (Nmm/rad)*	*FA Flexure Equivalent Stiffness (N/mm)*	*FA First Natural Frequency (Hz)*	*FR First Natural Frequency (Hz)*
60	10^7	5621/22.12=254,1	3.72	38.9
60	10^5	5621/47=119.6	2.55	28.4
60	0	5621/112=50.2	1.66	21.7
30	10^7	5621/22,2=253.2	3.71	35.4
30	10^5	5621/47=119.6	2.55	27.4
30	0	5621/112=50.2	1.66	21.2
5	10^7	5621/23,6=238.2	3.57	19.0
5	10^5	5621/47,8=117.6	2.52	18.3
5	0	5621/112=50.2	1.65	16.4
1	10^7	5621/29.19=192.6	3.12	8.9
1	10^5	5621/50.87=110.5	2.41	8.8
1	0	5621/112=50.2	1.65	8.6

After cycle 4 failure, the fuel designer issued a report, Ref. [2], evaluating the root-cause of the failures. According to it the failures were caused by grid spring force losses occurred when the fuel rods were inserted in the skeleton, possibly in combination with the loads sustained during transport to the site. The designer developed then a new grid spring design with a new shape and higher initial fuel rod fixing force (batch G fuel assembly). Neither flow experiments in hydraulic loops nor any change in the grid mixing vane were done, this means, the exciting force in fuel rods coming from the water flow through the fuel assembly was not evaluated or changed. Looking to the equations (9) to (16) one can conclude that the designer increased the stiffness of the grid spring $[K_M]$ and the fuel rod sliding force $\{R_o\}$, and consequently increased the fuel assembly equivalent beam stiffness $[K_G]$. However there was no change in the forces acting on fuel rods $\{R_v\}$ and, consequently, in the forces acting on the fuel assembly equivalent beam $\{F_G\}$. Then the problem of excessive fuel rod vibration or fuel rod fretting might not be solved.

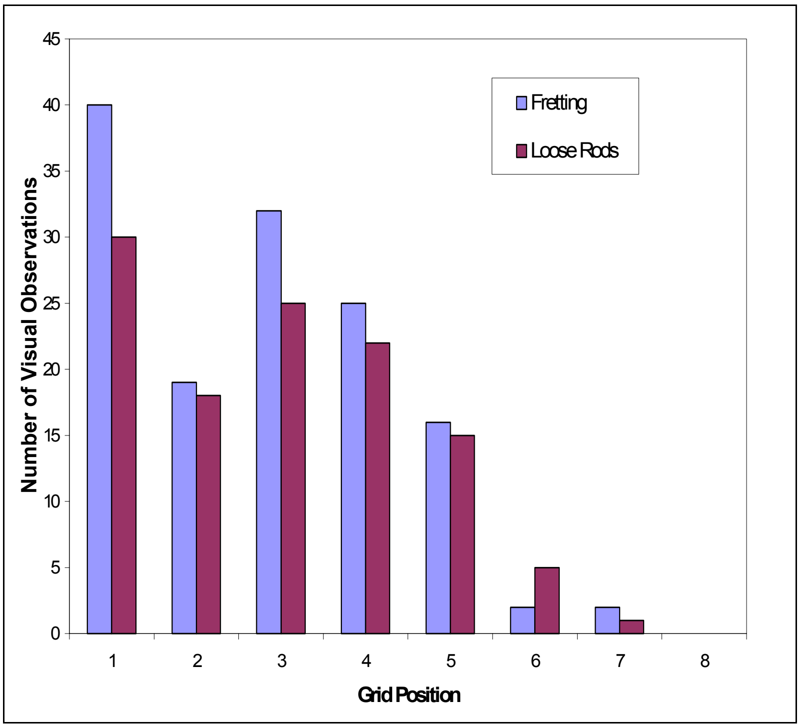


FIG. 15. Angra 1 – cycle 4: fretting and loose FRs observed during visual inspection in all failed FAs.

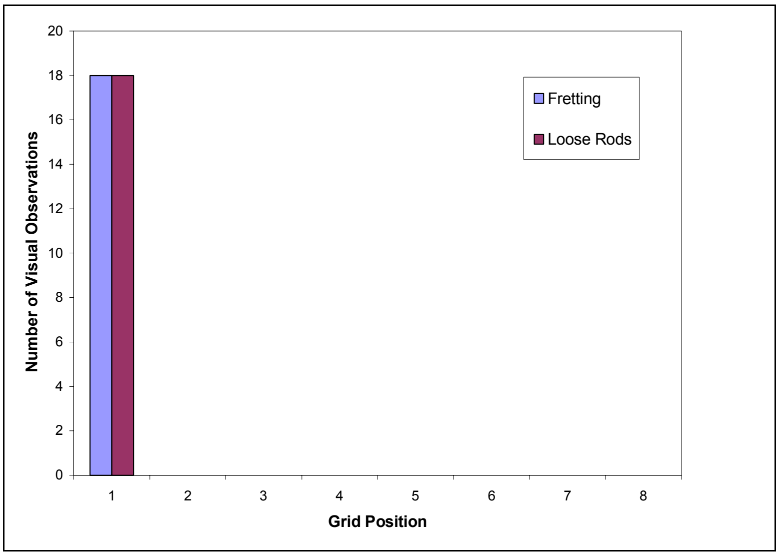


FIG. 16. Angra 1 – cycle 4: fretting and loose FRs observed during visual inspection in FA D14.

Cycle 6 (batch G) showed a pattern related to fuel assembly vibration. Grid wear (east-west side), due to fretting between adjacent fuel assemblies, was located at fuel assembly position where the highest amplitude of higher ($7^{th}/8^{th}$) modes of fuel assembly vibration would appear. Visual inspection of batch L (same design as batch G) used in cycle 7, but just once burned, showed the same pattern of grid wears as batch G. Fig. 17 presents the numbers of observation of wears in the fuel assemblies grids, and shows similarity to the amplitude of vibration of higher modes of the fuel assembly equivalent beam. As discussed in item 6, the designer performed flow experiments in a hydraulic loop with the batch G fuel assembly design. It was verified that the fuel assembly vibrates perpendicular to the spacer grid mixing vane direction in a sharp resonance at the range of 25-27 Hz for beginning of life mechanical condition. This resonance was not observed for end of life mechanical condition. It was concluded that the spacer grid mixing vane pattern design was responsible for the excitation mechanism. Looking to equation (15), one can assume that the fuel rod excitation force $\{R_{vi}(t)\}$ is a consequence of the coolant flow and cross-flow generated by the mixing vane. As has been observed that the first mode is predominant for the fuel rod vibration under flow condition, it can be also assumed that the frequency of this excitation force is in the range of the first natural frequency of the fuel rod. So, looking to equation (16), one can assume that the fuel assembly equivalent beam will also receive an excitation in the range of the first natural frequency of the fuel rods (taking into account the phase angle among them). The resonance observed in the laboratory flow experiment, similar that one observed inside the reactor, is explained by this assumption presented before. 25 Hz represents the first natural frequency of the fuel rod (excitation) for BOL condition, and resonance occurs at the 25 Hz frequency of the higher mode of vibration of the fuel assembly equivalent beam. For EOL condition there is a decrease of the natural frequency (smaller stiffness) and an increase in the damping factor for the fuel rod and also to the fuel assembly equivalent beam, which may decrease in a sharp way the amplitude of vibration for a resonance condition.

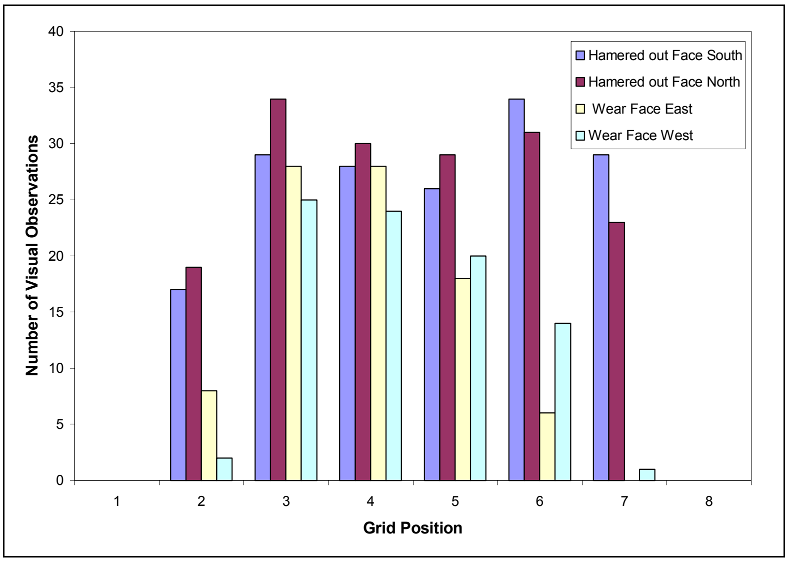


FIG. 17. Angra 1, batch L, number of visual observations - wear marks at spacer grids side strips.

Laboratory flow tests with batch G fuel assembly design showed that the use of dampers would lower the amplitude of vibration observed at the resonance. The use of dampers (zircaloy rods inside guide tubes) was decided to be used to the batch L, low burn-up fuel assembly. Looking to equation (16) one can verify that the damping factor [C] will increase due to change in the damping factor of the guide tubes. This damper device does not change the damping factor of the fuel rods. Assuming the previous experience of cycle 4, 6, and 7, rod fretting may occur after 19 MWD/kgU. This is assumed because the forces due to the mixing vane are still acting on the fuel rod and are higher than that predicted by design, Ref. [18]. Twelve fuel assemblies of batch L are being used (with dampers) in cycle 10 with a maximum planned discharge burn-up of 25 MWD/kgU.

INB received from the designer an alternative fuel assembly design that uses split mixing vanes at the spacer grids. Four fuel assemblies (batch Q) were manufactured by INB with this new design and used (in an experimental basis) since cycle 9. This design with split mixing vane would certainly mitigate the fuel assembly vibration and the fuel rod fretting due to the smaller forces acting on the fuel rods compared to that of directional mixing vane. Visual inspections will be carried out on this fuel, after each reactor cycle, in order to verify its performance.

ACKNOWLEDGEMENTS

The authors would like to thank the IAEA for supporting the presentation of this work at this TCM in Bratislava, Slovakia, and particularly to thank Dr Vladimir Onoufriev of IAEA Division of Nuclear Fuel Cycle and Waste Technology.

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OPERATION EXPERIENCE OF WWER-440 FUEL ASSEMBLIES AND MEASURES TO INCREASE FUEL RELIABILITY

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Abstract

The paper presents technical data for the fuel cycles used in 14 WWER-440 reactors of B-213 type situated outside CIS-territory on the basis of the 2001 operational results. The paper reflects the dynamics of average and maximum fuel burnup as well as information on the annual rate of the leaking fuel rods for the above reactor group identified during the 1997-2001 discharge period. As an example of work performed by RIAR in 2001 the paper brings forth the PIE-results of a leaking WWER-440 fuel assemblies (FAs). It is reported that the reason behind the leaking and failed fuel rods of the FA was interaction with a foreign object being in the coolant flow. The paper describes the measures taken by the NPPs together with the Supplier (JSC TVEL) and Manufacturer (JSC MSZ) to enhance the fuel operational safety.

1. INTRODUCTION

At present JSC MSZ is a major manufacturer of the nuclear fuel. The fuel is supplied and operated at 58 units of 11 European and Asian countries, half of these NPPs having WWER-440 reactors (28 units). JSC MSZ and JSC TVEL have a data acquisition system being used by all NPPs to collect operational information about FAs discharged from the reactors. This information allows to follow the dynamics of the main reliability indicators for the WWER-440 fuel rods and assemblies as well as to take necessary actions to improve FA manufacture for the purpose of enhancing fuel operational safety.

The analysis of the WWER-440 fuel operation is carried out on the basis of the official data received from the NPPs.

2. ANALYSIS OF WWER-440 FA'S OPERATION

The results of the nuclear fuel operation were evaluated both from the point of view of effectiveness of the fuel cycles being used and from the point of view of operational reliability that should ensure a reactor safety during its service life. According to the results of the 2001 reloads the fuel cycles were characterized which are used at present in 14 reactors WWER-440 (B-213 type) of the second generation located outside CIS countries. Fig. 1 demonstrates that 8 reactors of those 14 ones are in the state of transfer from 3-years cycle to a 4-years cycle with maximum enrichment 3,82%.

Fig. 1 demonstrates the dynamics of the average and maximum burnup of the fuel in the 14 reactors WWER-440 during the period of unloads 1997-2001.

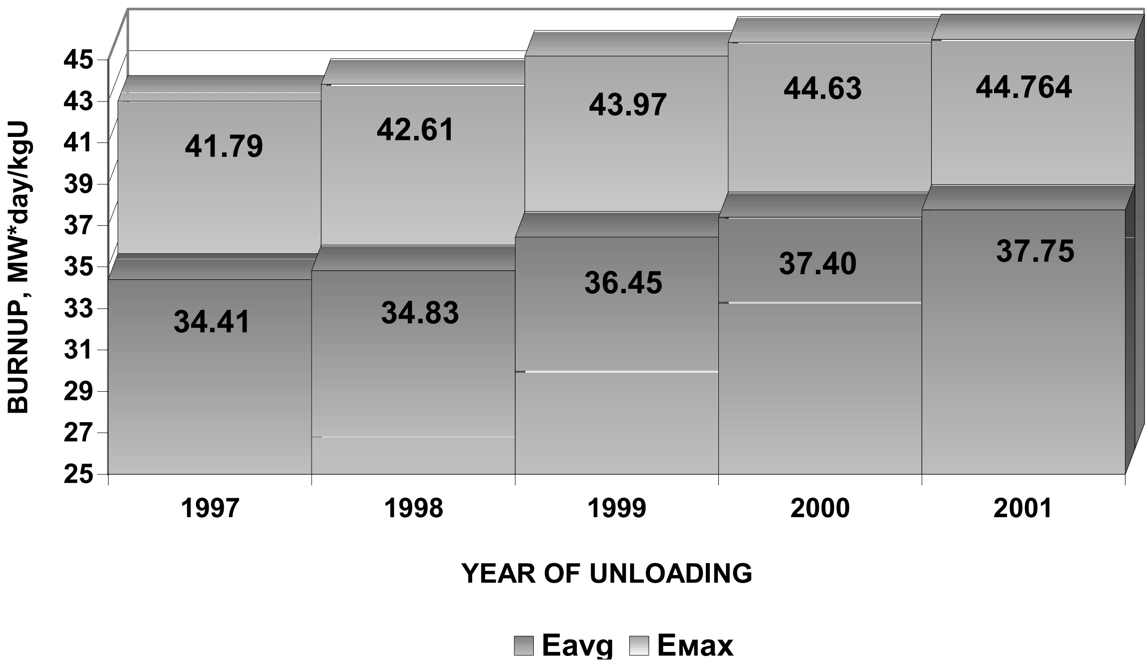

FIG. 1. Dynamics of average and maximum burnup of WWER-440 (type V-213) in 1997-2001 (14 units).

The highest values of average and maximum burnup in 2001 for the above group of the reactors were achieved at the NPP "Dukovany" (40,41(unit 4) and 44,76 MWd/kg U (unit 2), respectively), the NPP being in a transfer state to a 4-year cycle. The highest values of the of average and maximum burnup in 2001 among all the WWER-440 reactors were achieved at the Kola-3 NPP (46,7 and 51,04 MWd/kgU respectively), the unit operating on the 5-year fuel cycle with maximum enrichment 4,4%.

One of the most important indicators characterizing the performance reliability is the fuel leakage rate.

Fig. 2 shows the annual leakage rate of the fuel rods over the 1997-2001 discharge period for the 14 reactors of the second generation, located outside the CIS countries, which use the fuel manufactured by JSC MSZ. It should be noted that in 2001 not a single leaking fuel assembly was unloaded. The fuel rod leakage rate for the last 5 years was 1,5*10-6.

The average leakage rate of the fuel rods in the 14 above WWER-440 reactors (B-213 type) within the unloading period of 1997-2001 is shown in Fig. 3. It demonstrates that within the unloading period 1997-2001 the average annual rate of leaking fuel rods in reactors 1-2 of the Loviisa NPP, in reactor 4 of the Bohunice NPP and in reactor 4 of the Dukovany NPP made up 4,5·10-6 and there were no failed fuel assemblies. As for the rest 10 reactors presented in Fig. 3 not a single unloaded fuel assembly happened to be leaking or failed.

3. POST-IRRADIATION EXAMINATION

Figures 4 and 5 give the results of the post-irradiation examinations of one leaking WWER-440 fuel assembly as an example of work carried out by RRC RIAR in 2001.

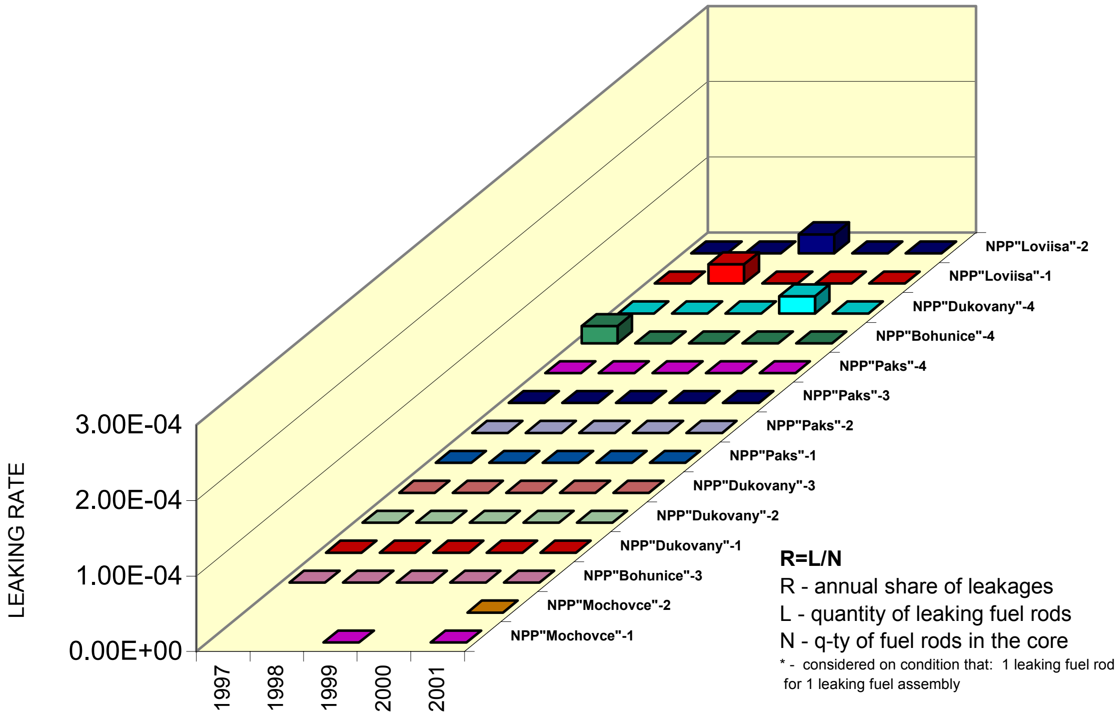

FIG. 2. Annual rate of WWER-440 (type V-213) fuel rod leakages during the period of 1997-2001 (14 units). Average value during 5 years-1.5E-06. Average value during 2001-0.

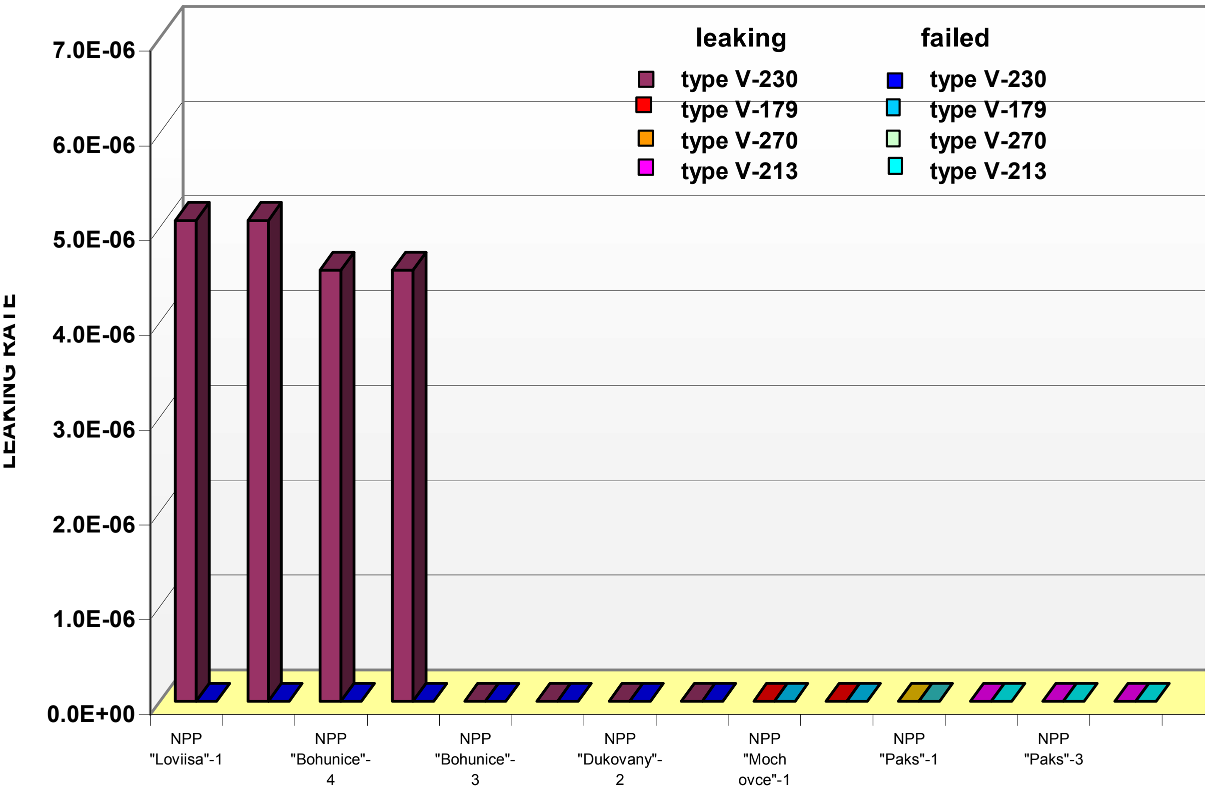

FIG. 3. Average rate of WWER-440 leaking and failed fuel rods (1997-2001 period of fuel discharge).

Fig. 4 shows that the FA contained 5 leaking fuel rods as well as a piece of a metal wire, stuck in the lower lattice. Comparing the appearance and geometry parameters of debris-damages of the fuel rods as well as considering the orientation of these damages in the bundle (see Fig. 5), there are good reasons to assume that the leakage of the above fuel rods could be caused by interaction with a similar wire. The analysis of the wire chemical composition showed that the wire was made of carbon or low-alloy steel, and the deposits found in the place of leakage of one of the five fuel rods of the bundle are the corrosion products of that material. It can be assumed that the wire fragment causing leakage of the four fuel rods has corroded by the time of the FA discharge from the reactor (650 EFPD) and was not discovered during PIE.

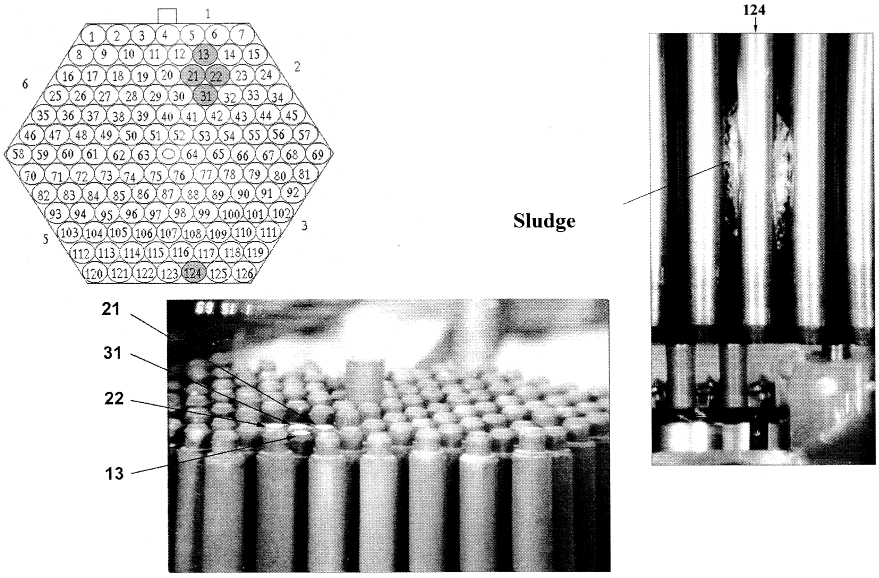

FIG. 4. Leaking fuel rod locations in the FA.

Thus, the interaction with the foreign objects of the coolant flow caused leakage and failure of the fuel rods.

More detailed information on failure mechanism is given in paper presented by Mr. A.Smirnov at this meeting.

4. MEASURES AIMED AT ENHANCEMENT OF FUEL SAFETY

4.1. Implemented measures

Some modifications have been recently made in the FA design (FA of the 2-d generation) to improve the economic indices of the fuel operation which are expressed in higher burnup, increase of effective operation and reduction of possible fuel failures.

4.1.1. Working fuel assembly:

- increased fuel load due to the increase of the active fuel length;
- dismantling design that enables to dismount the working fuel assembly, withdraw leaking fuel rods from the bundle and replace them with the dummies;
- engineering solutions (optimization of spacer grid positioning along the bundle height, backlash-free securing of the fuel rods in the lower lattice, a system of supporting ribs) aimed at the improvement of the working fuel assembly stiffness and vibrostability.

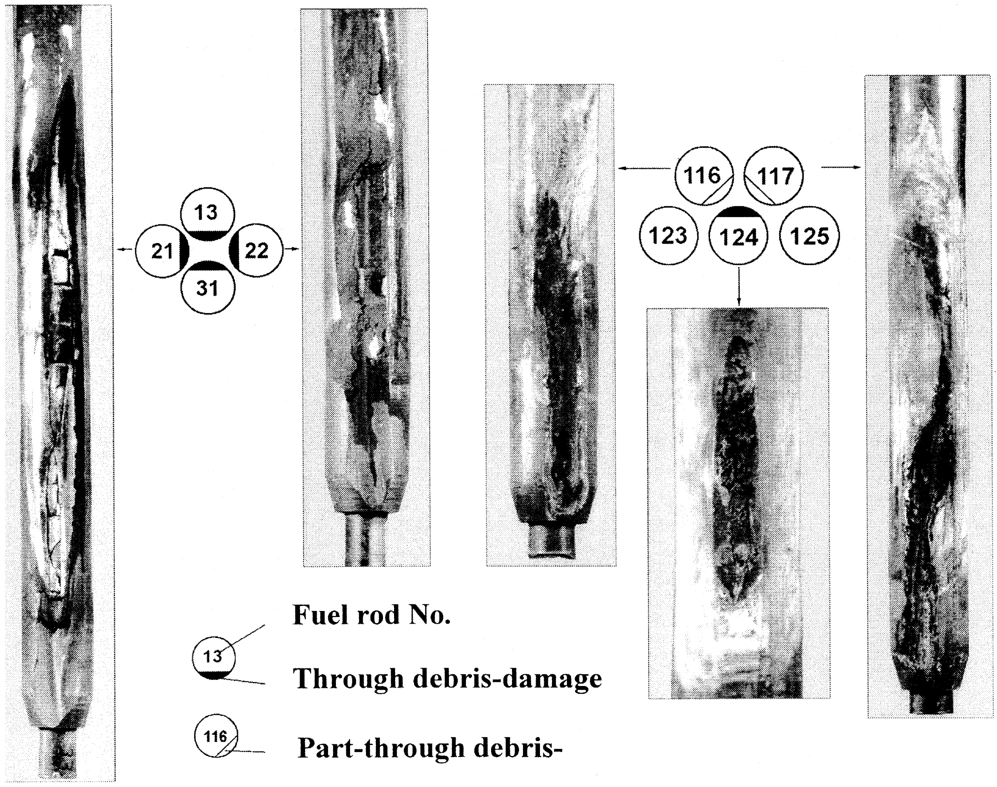

FIG. 5. Appearance and orientation of fuel rod debris damages in the FA cross-section.

4.1.2. Fuel follower of control rod:

- An engineering solution has been implemented that eliminates power density ramping in the working fuel assemblies surrounding the fuel follower of control rod, thus reducing possible failure of the fuel rods of the working fuel assembly during higher burnup fuel cycles.

4.2. Measures under development

To prevent ingress of the debris-particles into the fuel assembly an anti-debris device is supposed to be used. The device might enable to reduce a potential interaction of the debris-particles with the fuel rods and FA bundle components and will decrease possible FA leakage respectively. Such activities are currently carried out by JSC MSZ in association with OKB Gidropress.

5. CONCLUSIONS

The NPPs in association with the Supplier (JSC TVEL) and Manufacturer (JSC MSZ) are steadily improving the effectiveness of the fuel cycles being used and fuel operational reliability.

The actions to be taken for the enhancement of the fuel safety could be divided into two groups. One group should combine the activities aimed at the FA improvement. JSC TVEL together with the Manufacturer are currently carrying out such work, an implementation of vibro-stable fuel assemblies and those with an anti-debris lattice included.

The other group of activities should be carried out by NPPs in the framework of the quality assurance of the fuel performance. It will require a system of measures and techniques to be developed by the NPPs for cleaning the primary coolant from the corrosion products and foreign objects. The latter group of activities is becoming of essential importance because of the work carried out for the purpose of extension of the service life of the WWER-440 first generation reactors.

Table 1. Brief description of fuel cycle

Unit/Units	Brief cycle description
Bohunice 3 and 4	Mixed 3-years fuel cycle, a part of FAs is left for the 4th year; max. enrichment-3.6%
Dukovany 1-4	Transitional from a mixed 3-years fuel cycle to a 4-years cycle; max. enrichment-3.82%
Loviisa 1 and 2	Transitional from a 3-years fuel cycle (max. enrichment-3.6%)to a 4-years cycle with max. enrichment-3.82%
Mochovce 1	Transient cycle from an unsteady 3-years fuel cycle with max. enrichment-3.6% to a 4-years cycle with max. enrichment-3.82%
Mochovce 2	Unsteady 3-yeares fuel cycle with max. enrichment-3.6%
Paks 1, 2 and 4	Mixed 3-years fuel cycle, a part of FAs is left for the 4th year; max. enrichment-3.6%
Paks 3	Transient cycle from an unsteady 3-years fuel cycle with max. enrichment-3.6% to a 4-years cycle with max. enrichment-3.82%

FUEL FAILURE MITIGATION AT THE RINGHALS PLANT

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Abstract

Based on the various types of fuel failures that have occurred at Ringhals, the paper describes the efforts made at Ringhals and by the fuel suppliers to reduce the risk of fuel failures. The defense against secondary degradation at Ringhals 1 includes 10x10 liner fuel with debris filters, a conservative management of PCI margins as well as measures to prevent debris from entering the coolant system. For the PWRs the risk for fuel rod fretting remains a concern. The design margins against fuel rod fretting need to be improved on new fuel types.

1. INTRODUCTION

During the past decade 13 new fuel types have been introduced at the Ringhals plants (the number depends somewhat on how a new fuel type is defined). All of these fuel types have been or will be delivered in reload quantities. In addition, within a given fuel type, there are usually a number of varieties: different material compositions, different dashpot designs etc. With the exception for Ringhals 2, the Ringhals cores are therefore composed of a mixture of different fuel types and varieties of a given fuel type.

The reason for this diversity is both technical and commercial. To maintain competition, new DEMO/LFA fuel types are introduced. Fuel contracts are usually signed for 4-year periods, which is less than the period required to fully replace the fuel types of a given core.

Assembly bow has required a number of design changes within a given fuel type - reinforced dashpot, relaxed hold-down forces and thicker guide thimbles. Further modifications have been introduced in order to limit assembly growth and to improve corrosion properties. Fragema, Siemens, Westinghouse and ABB-Atom have supplied the fuel.

The introduction of new fuel types requires an assessment of both performance and reliability. New fuel types generally have better thermal performance than their predecessors, which gives an economic incentive to introduce new fuel types. New fuel types are also expected to be more reliable than their predecessors. On the other hand there's the risk that the new fuel types have unknown reliability problems that will show up at high burnups.

One difficulty in the reliability assessment is to account for the variability in the fuel's operating environment: surrounding fuel types, core locations as well as power dependent flow conditions. The manufacturers flow testing hasn't been able to account for all these variables.

Fuel types that operate without problems in Ringhals have been subject to flow induced fretting failures in other plants. The fact that it has been possible to verify failure mechanisms in test loops after the delivery of the fuel to Ringhals or to verify that the Ringhals reactor in question was not subject to a particular fretting mechanism shows that loop testing can be made more efficient.

The management of the PCI margins gained by the introduction of 10x10 liner fuel requires a trade-off between the demand for improved fuel economy and operating flexibility and the requirement to maintain sufficient margins to PCI.

The decision to introduce a new fuel type is also influenced by the marginal costs. If improved thermal or mechanical properties come at little or no extra costs it is likely that that such features will be introduced irrespective of if they are needed or not from a licensing point of view - resulting in better operating margins if the boundary conditions remain the same. If the cost is too high, much needed hardware or methodology improvements might not be made.

Considering all aspects of fuel reliability the total costs associated with failures and poor performance can be quite substantial. There are costs for fuel repair, fuel inspection programs, safety analysis, restrictions on loading and handling of the fuel, increased core tilts, increased uncertainties in the peaking factors, increased background activity levels, reduced power peaking factors, prolonged outages, PIE, restrictions in core design, burnup restrictions etc.

2. THE RINGHALS PLANTS

Ringhals is Sweden’s largest power plant. Last year the production from the four units equaled 18% of the Swedish electricity consumption. The thermal ratings of the Ringhals plants are the following:

Ringhals 1: Asea-Atom BWR, external pumps, 2500 MW_{th}
Ringhals 2: Westinghouse PWR, 15x15 lattice, 2652 MW_{th}
Ringhals 3: Westinghouse PWR, 17x17 lattice, 2775 MW_{th}
Ringhals 3: Westinghouse PWR, 17x17 lattice, 2775 MW_{th}

The plants are operated on 12-month cycles. Refueling outages are in the period May to September. Extensive periods of coast down operation are used in order to improve the fuel economy and to adjust production to the reduced demand in the summer period.

The backend costs in Sweden are at present very low. There's therefore little incentive to go to very high burnups. Presently the most economical discharge burnup is in the 45–50 MWD/KgU range.

3. FUEL TYPES AND FUEL SUPPLIERS

Siemens replaced ABB-Atom as fuel supplier to Ringhals 1 in 1999. The present replacement fuel to Ringhals 1 is ATRIUM 10B. For the period 2003–2006, Framatome ANP will deliver replacement fuel to Ringhals 1 (ATRIUM 10B). The core composition of Ringhals 1 is shown in fig. 3.

Fuel supplier to Ringhals 2 during the period 1995–2002 is Westinghouse. Delivered fuel type is Performance + with IFMs. Cladding and grid material is ZIRLO. Framatome ANP is contracted as the fuel supplier for the period 2003–2006. Fuel type to be delivered is AFA 3G with M5 cladding and grid material.

Fragema has delivered fuel to Ringhals 3 and 4 during the period. In 1999, AFA 3G replaced AFA 2G as replacement fuel. For the period 2003–2006 Framatome ANP will supply HTP as replacement fuel.

4. NUMBER OF FUEL FAILURES

The total number of fuel failures (leakers) at Ringhals between 1989–2001 were 38. Failure mechanisms have been PCI (Ringhals 1), debris fretting, fuel rod fretting wear and mechanisms related to manufacturing. For 9 of the failures, the failure causes are still unknown. The division of the failures among the units is shown in table. 1. Most failures have appeared at Ringhals 3.

Prior to 1989, a large number of fuel failures on Ringhals 2 were caused by baffle jetting. In 1989, the barrel coolant flow in Ringhals 2 was converted from "down-flow" to "up-flow" to prevent baffle-jetting failures.

Starting in 1993, debris filters have been introduced in the Ringhals plants. All of the fuel assemblies in the PWR cores are equipped with debris filters. The fraction of fuel assemblies with debris filter on Ringhals 1 is about 30%.

Table 1: Fuel failures at Ringhals units between 1989 and 2001

Failure mechanism	**Ringhals 1**	**Ringhals 2**	**Ringhals 3**	**Ringhals 4**
Debris	3	1	3	5
PCI	6			
Rod fretting			3	
Corrosion			1	
Primary hydriding			5	
Manufacturing			1	1
Unknown		1	7	1
Total	**9**	**2**	**20**	**7**

5. STRATEGIES TO IMPROVE FUEL RELIABILITY

Thermo-mechanical spare margins are included in the design and licensing of the fuel. The fuel is presently licensed to a burnup of 60 MWD/kgU despite the fact that the average batch discharge burnup is in the range 45–50 MWD/kgU.

The introduction of IFMs in PWR gives additional DNB-margins that have not been licensed as higher peaking factors. Parts of the design margins have been used to account for assembly bow and thimble plug removal.

The design margins also simplify burnup extensions for lead assemblies within a given batch. Examples of margin recoveries are the introduction of IFMs on the PWRs and liner 10x10 fuel on the BWR.

An extensive reliability assessment done for all offered fuel types is made prior to contract. The assessment includes fuel economy, thermal performance, mechanical design and operating experiences. Important parameters are debris filter efficiencies, assembly bow properties and fretting resistance.

New fuel types of commercial or technical interest are usually introduced as LUAs. The objective with the LUA program is generally a complete licensing of the fuel type covering possible future reloads.

A long term and standardized fuel assembly inspection program is used to verify performance and for early detection of problems. The inspection program includes oxide thickness measurements (5-10), assembly growth measurements (during unloading), visual inspection and assembly/channel bow measurements. CRUD sampling are made every two years as a part of the water chemistry follow-up program.

Ringhals along with the other Swedish plants participates in a long-term PIE program with Studsvik. The main purpose of the program is to examine the burnup properties of various types of cladding materials as well as primary failure mechanisms. A few fuel rods are sent each year to the Studsvik facility for inspection. A typical PIE includes measurements of oxide thickness, hydriding, creep and growth.

6. PERFORMANCE OF MODERN FUEL TYPES

To what extent have the fuel suppliers been able to make the fuel products more robust and reliable given the lessons learned from older designs? To what extent have the operators been able to introduce reliable fuel types and prevent debris from entering the coolant systems? Do the most recent designs perform better then the previous designs?

Our expectations for modern fuel products is an average failure rate of one leaking fuel assembly per 10 reactor cycles, corresponding to a failure rate of $3x10^{-6}$.

In general terms there has been an improvement of the leakage situation with the deliveries of more advanced fuel types. This can be seen in fig.1. With the exception for Ringhals 2, the cores are still a mixture of older and advanced fuel types. It is therefore too early to tell if the goal of less than one failure per 10 reactor cycles can be met.

IFMs are being introduced on all PWRs for better DNB performance. During the transition period there's a pressure difference between fuel with and without IFMs. Rod fretting wear caused by this pressure difference has not been observed.

Fuel rod fretting wear hasn’t caused any failures in recent years but remains a latent threat. Variations in the flow environment of the fuel as well as variations in the grids mechanical properties related to burnup and manufacturing might create conditions that leads to fretting wear. The design margins against fretting needs to be improved on new fuel types.

No debris failures have been found on fuel with debris filters so far. Introduction of more efficient filters will further reduce the risks. All of the assemblies in the PWR cores have debris filters.

7. FUEL FAILURES AND REMEDIES

7.1. Ringhals 1

Ringhals has adopted a **defense in depth strategy** (Section 8) to prevent secondary degradation at Ringhals 1. A range of precautionary measures has been taken in order to **prevent debris from entering the coolant system** during repair works and refueling. The introduction of efficient **debris filters** on the fuel prevents debris induced fretting. The **cyclones** installed in the feedwater pipes further separate debris from the coolant.

Activity monitoring with the FLEA code enables an early detection of failures. If a leaker is detected, **operating restrictions** are imposed in order to delay the secondary degradation. If the activity level reaches a threshold level, the plant is shutdown for replacement of the leaking assembly.

If the plant is shutdown for removal of a leaking assembly, **full core sipping** is performed in order to prevent reinsertion of undetected leaking assemblies.

The defense in depth strategy includes **PIE** at Studsvik for better understanding of the mechanisms behind the degradation. In the **technical assessment of new fuel types**, the cladding material's secondary degradation properties are evaluated. Finally Vattenfall participates in various **R&D** activities aimed at finding cladding materials with better protection against secondary degradation. The **water chemistry specifications** ensure that the cladding corrosion is not adversely affected by the water chemistry. The defense in depth strategy is illustrated in table 2 and figure 1.

Table 2: Defense in depth matrix: For each main core component, two defenses are introduced. The full defense includes 18 areas

Level of defense	**General design criteria**	**Failure mechanisms**	**First defense**	**Second defense**
1	Core design [a)]	Corrosion	Chemistry restrictions	Fuel inspection program [c)]
2	Assembly design [b)]	Debris fretting	Debris prevention	Debris filters
3	Grid design	Fuel rod fretting	Flow loop testing [d)]	Design margins
4	Fuel rod design	PCI	Power ramp restrictions	Liner/Conditioning
5	Fuel pellet design	Impurities/cracks	Quality control	Manufacturing procedures
6	Pellet matrix composition	Secondary degradation	Activity monitoring	Operating strategies
	Fission products			

a) Includes burnup limitations and thermal margins.
b) Material composition, debris filters efficiencies, hold-down forces.
c) Includes leak testing and repair.
d) Includes operating experience.

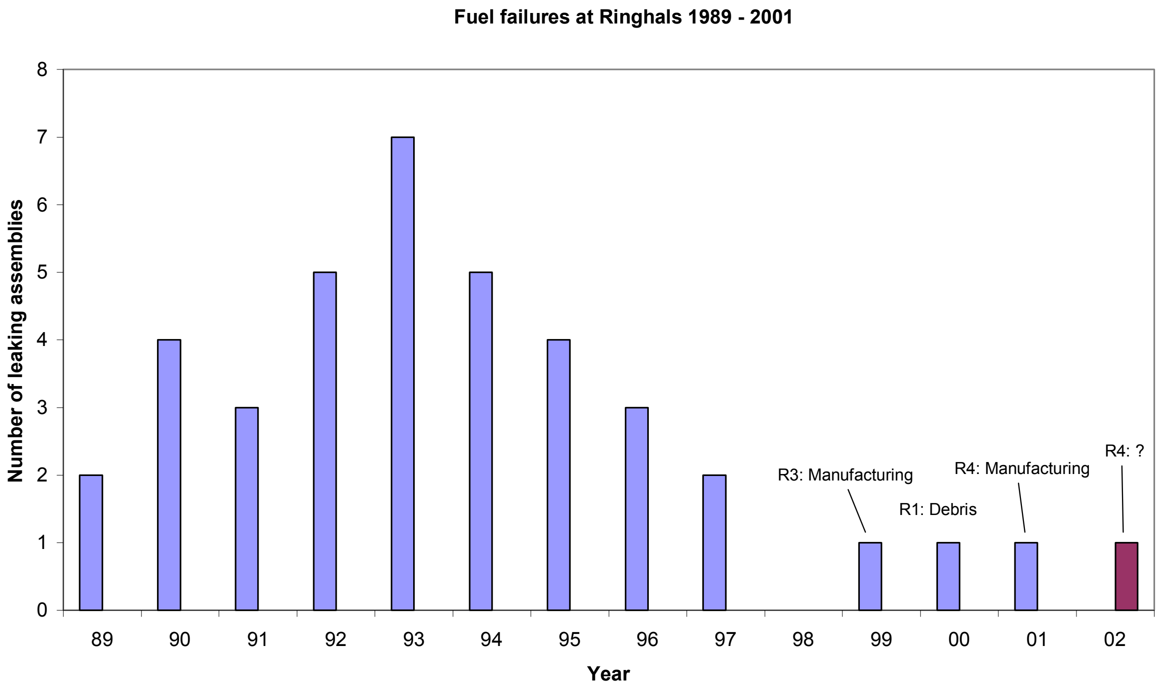

FIG. 1. Fuel failure development at Ringhals.

The 6 PCI failures on Ringhals 1, fig. 2, were the results of unconservative PCI thresholds in control cells. More conservative PCI limits were therefore introduced in 1995. The introduction of 10x10 liner fuel in 1997 has further improved the PCI margins. About 60% of the fuel assemblies of the core have liner fuel.

Less than half of the fuel assemblies in Ringhals 1 have debris filter. Presently only 1/6th of the core is replaced at each refueling. The small reload batches means that there's a considerable time lag before improvements are fully in effect.

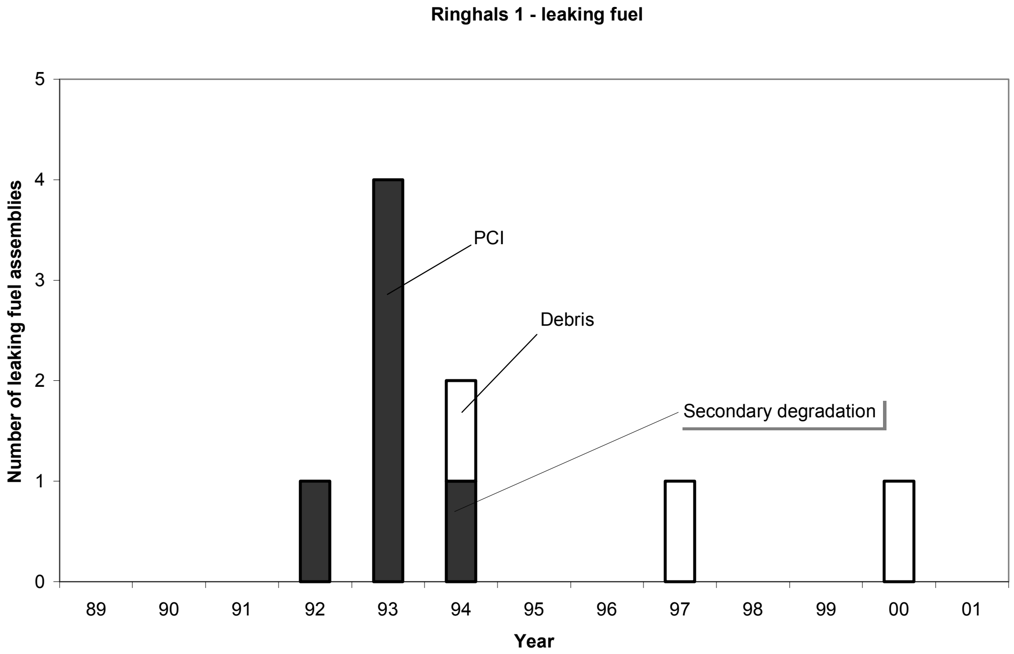

FIG. 2. Fuel failures at Ringhals 1.

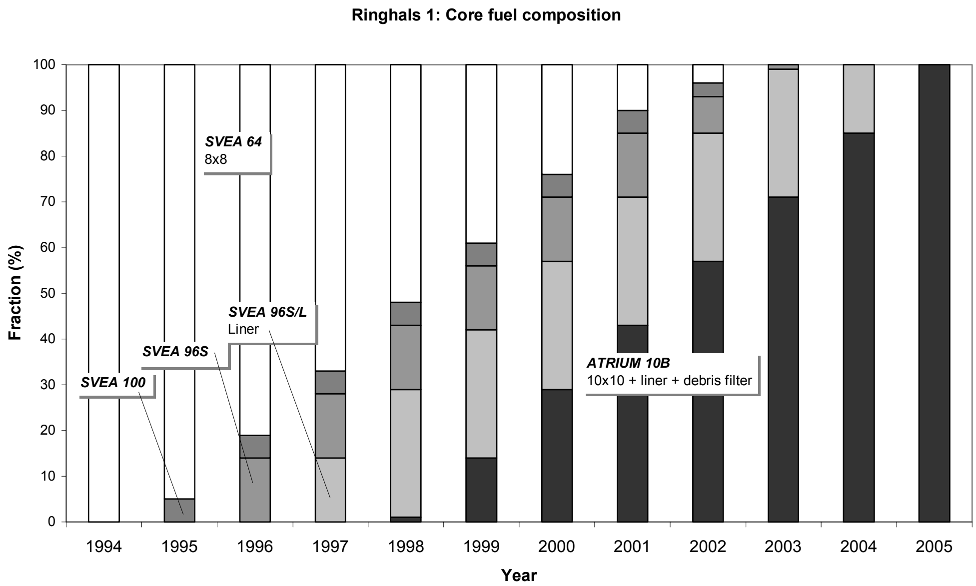

FIG. 3. Composition of the Ringhals 1 core.

The introduction of debris filter was delayed when rig testing showed that the filter in question was subject to fretting wear at critical flow conditions.

The PCI failures in 1994 and the associated secondary degradation lead to a large increase in the background activity levels. The present policy is to shut the plant down and replace the leaking fuel if secondary degradation develops.

7.2. Ringhals 2

Only two failures have occurred in the period 1989–2001–both failures on fuel types without debris filters. The Performance+ 15x15 fuel has not been subject to the kind of rod fretting failures that have affected similar 17x17 Westinghouse fuel.

The risk of fretting wear caused by pressure drop between assemblies with and without IFMs was a reliability concern during the introduction of IFMs at Ringhals 2. However, no fretting failures have occurred on Ringhals 2. All assemblies of the core now have IFMs.

Assembly bow has emerged as a problem on Ringhals 2 during the last cycles. Bow measurements made in the early 90-ties showed no indications of bow. Measurements made during the most recent refuelings show the same bow magnitudes similar to those found at Ringhals 3 and 4. The countermeasure has been to introduce top nozzles with a lower hold-down force. The fuel supplier and Vattenfall have extensively analyzed the impact of the bow on the safety analysis.

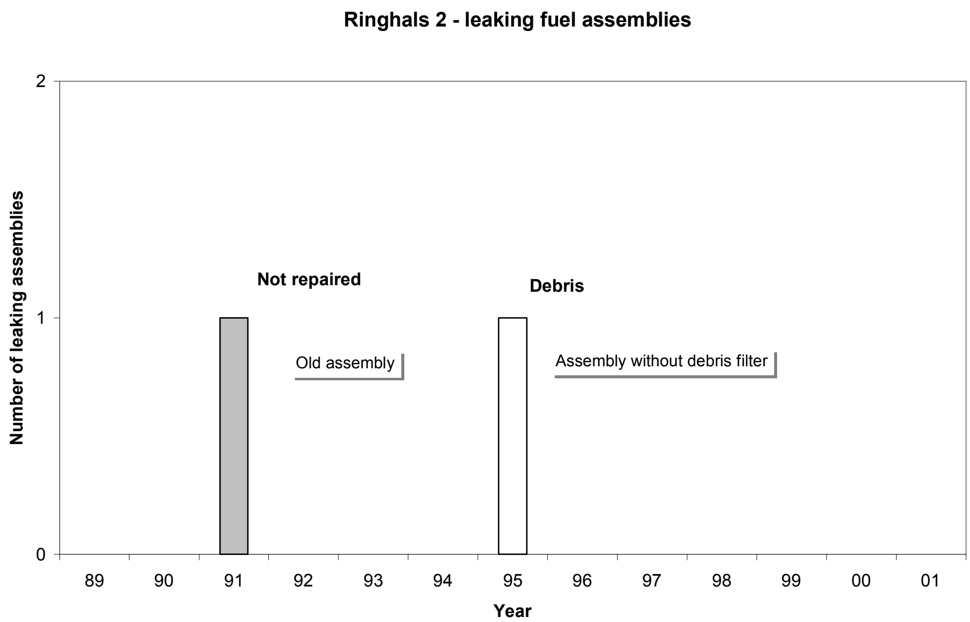

FIG. 4. Fuel failures at Ringhals 2.

7.3. Ringhals 3 and 4

More than half of all fuel failures in Ringhals during the period 1989–2001 has occurred in Ringhals 3, or 20 out of 38 in total. The result is puzzling considering that the same fuel types were delivered to Ringhals 4, manufactured in the same facilities. The core operating conditions are also very similar.

Ringhals 3 and 4 have the same mix of fuel types as in Cattenom 3 (AFA 2G and AFA 3G). Fuel inspection of individual rods in one AFA 2G assembly shows no indications of fretting wear (12 ft cores as opposed to 14 ft core).

In addition to debris, also primary hydriding related to manufacturing has caused failures at Ringhals 3. Grids with insufficient spring forces caused the fretting damages at Ringhals 3. The problem was related to two reloads. A failed end plug weld caused one of the failures.

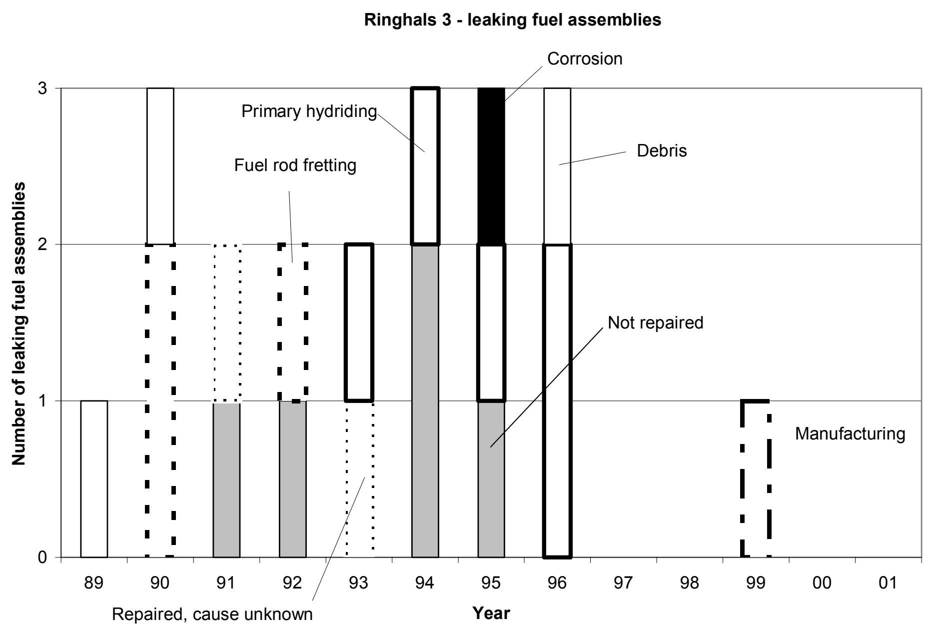

FIG. 5. Fuel Failures at Ringhals 3.

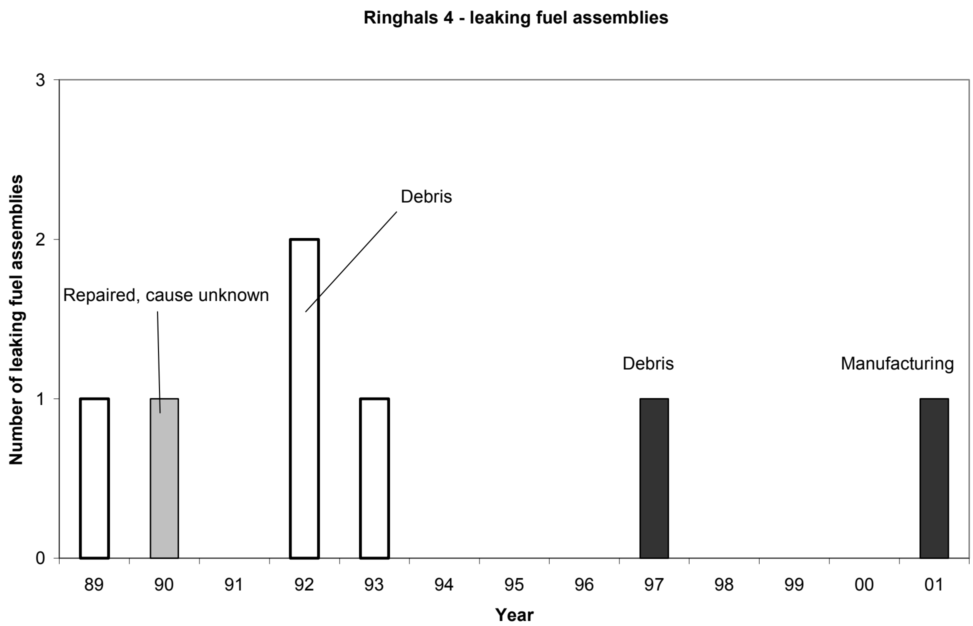

FIG. 6. Fuel failures at Ringhals 4.

8. DEFENCE IN DEPTH

The core region might be partitioned in logical units in the following way. The **core** is composed of **fuel assemblies** that in turn comprises a **skeleton/grid** structure with suspended **fuel rods**. The fuel rods encapsulate **fuel pellets**. The pellet's **matrix** finally encapsulates the fission products. Qualitatively, therefore, six different shells surround the fission products. To prevent the fission products from escaping, the integrity of all six shells must be maintained. This is illustrated in figure 7. The integrity requirements are specified as general design criteria.

Fuel failure mechanisms tend to appear at the interfaces between the shells. Main failure mechanisms are corrosion, debris fretting, fuel rod fretting, PCI and primary hydriding. The probabilistic nature of these mechanisms makes it difficult to establish simple design criteria.

Table 2 describes the defense in depth strategy to prevent failures. Fuel failures are prevented by the general design criteria applicable to all six layers of defense. In addition, two defensive measures are introduced for each failure mechanism at the interface between the main core components - in all 18 areas of fuel failure defense.

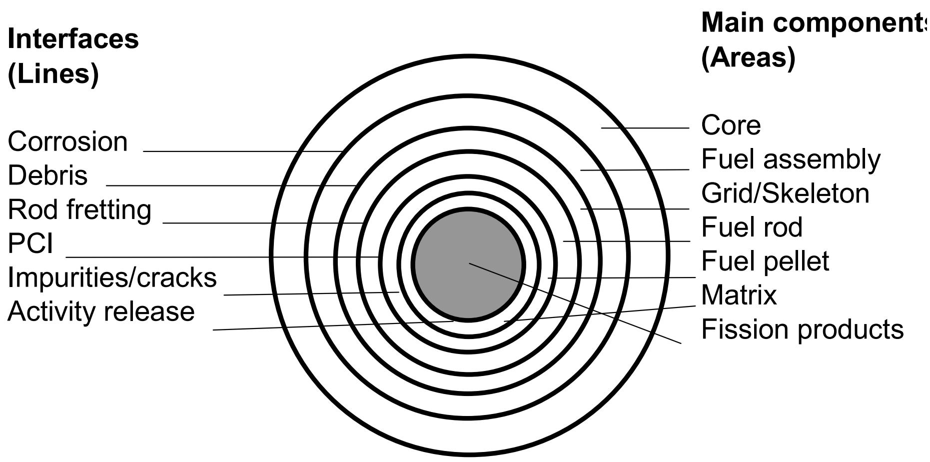

FIG. 7. The defense in depth strategy requires that attention is paid to the interfaces between the main core components where failures tend to appear. The fuel cladding is exposed to all main components.

9. HOT CELL EXAMINATIONS AT STUDSVIK

Fuel rods are regularly sent to the hot cells at Studsvik for root cause examination of failed rods and for more general examination within the long term R&D program.

Hot cell examinations at Studsvik verified the PCI failures at Ringhals 1 as well as the primary hydriding failures at Ringhals 3. Hot cell examinations have also been used to investigate the corrosion properties of advanced alloys such as M5 and ZIRLO as well as creep and growth rates.

The secondary degradation failure mechanisms have been investigated thoroughly at Studsvik.

Valuable results have also been gained by ramp testing of various fuel rods at Studsvik.

10. ASSEMBLY BOW

The most demanding problem related to fuel reliability has been the assembly bow problem in the PWRs. All PWRs are affected by assembly bow.

Fuel types with thicker guide thimbles, reinforced dashpot and reduced hold down spring forces have been introduced in order to reduce the bow. In addition, materials with a lower growth rate and better creep properties have been introduced. As a result of the mechanical modifications, we see little impact of the bow on control rod friction and drop time of the bow.

On Ringhals 3 and 4 we have seen a gradual reduction of the bow. Last year, however, we saw an increased bow on Ringhals 3. It remains to bee seen how effective the countermeasures are in the long term.

Extensive studies performed by our fuel supplier and by Vattenfall have shown that the assembly bow causes additional uncertainties in the peaking factors of around 10%. These additional uncertainties have been accounted for in the transient analysis. Some of the previously mentioned spare design margins have been utilized to account for assembly bow.

To verify the safety limits, the assembly bow is measured on 10-20 assemblies during each refueling. The maximum bow is verified by the measurements to be less than the largest bow assumed in the safety studies.

In addition to the mechanical forces giving rise to bow, there are measurements as well as calculations showing that the coolant flow through the core gives rise to bow forces. Various forms of bow management or reshuffling strategies are discussed to counteract the bow.

11. PRODUCT LIFE CYCLES

New fuel types generally have better thermal mechanical properties than their predecessors. Improved thermal performance translates to better safety margins if properly managed. The dilemma is to avoid "children's diseases" i.e. undetected reliability problems. The dilemma is illustrated in figure 8.

12. COOLANT ACTIVITY ANALYSIS AND LEAK TESTING

On-line leak testing of the whole core is generally performed irrespective of the activity levels prior to shutdown. On-line sipping is reliable, efficient and does not impact critical path during outage. Wet sipping is used to verify fuel integrity after fuel repair. Full core leak testing eliminates the risk of reloading leaking fuel.

Activity analysis is a sensitive instrument for detection of small failures, the outset of secondary degradation and for estimation of the amount of tramp uranium in the system.

Secondary degradation in the BWR in most cases requires shutdown and replacement of the leaking fuel.

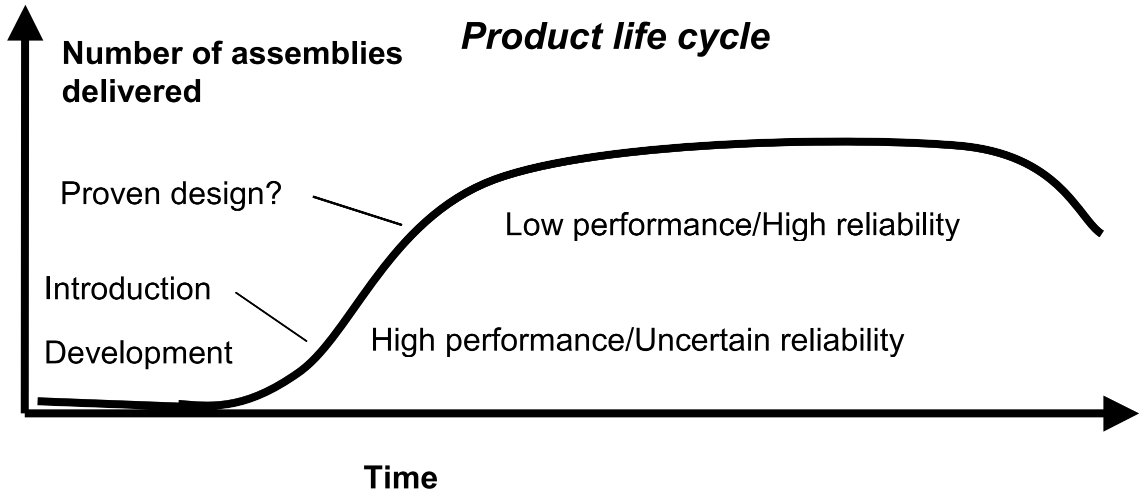

FIG. 8. Product life cycle.

13. CONCLUSIONS

- Prevention of secondary degradation requires a defense-in-depth strategy.
- Flow testing of fuel prototypes can be made more efficient.
- The design margins to fuel rod fretting needs to be improved on new fuel types.
- Fuel failures related to manufacturing indicate a need to further improve the quality control during manufacturing.
- The mechanisms behind assembly bow must be better understood. Bowing adds to the peaking factor and fuel duty uncertainties and hence might impact fuel reliability.
- A long-term fuel inspection program combined with PIE makes it possible to detect deviations from expected performance. PIE is essential for root cause investigation.
- A conservative management of thermal and mechanical margins is required in order to improve both reliability and fuel economy.
- Efficient activity analysis is necessary in order to detect small leakers and the outset of secondary degradation.
- Full core leak testing eliminates the risk of reloading leaking fuel

EXPERIMENTAL STUDIES OF FUEL FAILURE AND DEGRADATION MECHANISMS
(Session 3)

PWR FUEL FAILURE ANALYSIS DUE TO HYDRIDING BASED ON PIE DATA

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Abstract

Recently failures of nuclear fuel rods in Korean nuclear power plants were reported and their failure causes have been investigated by using PIE techniques. Destructive and physico-chemical examinations reveal that the clad hydriding phenomena had caused the rod failures primarily and secondarily in each case. In this study, the basic mechanisms of the primary and the secondary hydriding failures are reviewed, PIE data such as cladding inner and outer surface oxide thickness and the restructuring of the fuel pellets are analyzed, and they are compared with the predicted behaviors by a fuel performance code. In addition, post-defected fuel behaviors are reviewed and qualitatively analyzed. The results strongly support that the hydriding processes, primary and secondary, played critical roles in the respective fuel rods failures and the secondary hydriding failure can take place even in the fuel rod with low linear heat generation rate.

1. INTRODUCTION

One of the frequent fuel failures in light water reactors is hydrided or hydriding-related failure [1,2]. When a reactor fuel element is defective during operation, cladding no longer provides a barrier between the fission-induced burning fuel element and the coolant. The presence of this defective fuel may present economic penalties to the power utility.

Sometimes severe secondary damage can follow the primary failure. Existing leak path can let the coolant enter the element, the coolant flashes into steam, and then complicated processes such as steam oxidation of UO_2, oxidation and hydriding of the clad inner surface, restructuring of UO_2, are developed, ending up with catastrophic failure [3]. Recent investigations have provided a better understanding of the failure processes, physical and chemical [4,5].

Fuel rod failure were recently reported in the two different Korean nuclear power plants. Two rods were failed during cycle 1 start-up operation in the prototype Korean standard nuclear power plant and one rod was failed during cycle 7 reload core operation in the other plant. Through the post irradiation examination it was revealed that the causes of the rod failures are ascribed to the primary and secondary hydriding. These findings are supported by the mechanistic review of the hydriding failures, fuel behaviors predicted by a fuel performance code, and qualitative post-defected fuel behaviour analysis.

2. POST-IRRADIATION EXAMINATION

Failed fuel rods were transported from the storage pool in the plant sites to the PIE facility in KAERI. The examinations, non-destructive, destructive, and chemical, were carried out in the hot cells. Whole rods were thoroughly inspected using telescope, gamma-scanned, and the diametral change of the cladding was measured in the non-destructive test. Then the oxide thicknesses of clad inner and outer surface and micro-structural change of UO_2 pellet were examined. In addition, axial hydrogen content distribution of the clad was determined by using LECO analyzer.

Through PIE, it is found that D103-K2 rod was failed due to the random internal hydring while in B208-R8 rod through-wall defect was developed by the debris caught in the bottom grid and the defect gradually induces the breach of the cladding due to the secondary massive hydriding. These two rods were failed during cycle 1 start-up operation in the prototype Korean standard nuclear power plant. With the design calculation in detail and the gamma scanning measurements, the burn-ups and the ratios of $q'/q'_{core\ avg}$ for both rods were determined: 1875 MWD/MTU and 1.27 for D103-K2 rod, and 2013 MWD/MTU and 1.38 for B208-R8, respectively.

Figure 1 shows the visual inspection results of the damaged D103-K2 rod due to hydride blister and hydride-induced crack. Figure 2 renders the cross sectional view of the microstructural change of UO_2 and the breached cladding of the rod. As shown in the middle micrograph of Figure 2, bright spots in the fuel pellet are observed in an annular band at around r = 0.6R. Extensive investigation reveals that the annular band of the bright spots are observed at about the elevation (1200~2000mm) with high linear power and possibly under the highly oxidizing environment. Thus the spots were closely examined and the micro-hardness was measured in the radial direction (Figure 3). It is found that extra-ordinary columnar grain growth took place in the annular band.

Figure 4 shows the visual inspection results of the secondarily damaged B208-R8 rod due to massive hydride formation. Figure 5 provides the cross sectional view of microstructural change of the fuel pellet and the breached cladding of B208-R8. In this rod the annular band of bright spots were also observed at the high linear power elevation even though they are not as clear as those of D103-K2 rod

The third fuel rod for which PIE was carried out is the J09-L01 rod burned up during cycle 7 reload core operation in another plant. The PIE results are shown in Figure 6. It is obviously seen that this rod was also damaged secondarily owing to the massive hydriding. However, this result is very unexpected because $q'/q'_{core\ avg}$ and dischge burnup of the rod are found to be only 0.66 and 11806 MWD/MTU, respectively, based on the design calculation in detail and the gamma scanning measurements. In fact, the rod was located outmost periphery of the reactor core during the operarion, facing the baffle of the reactor vessel.

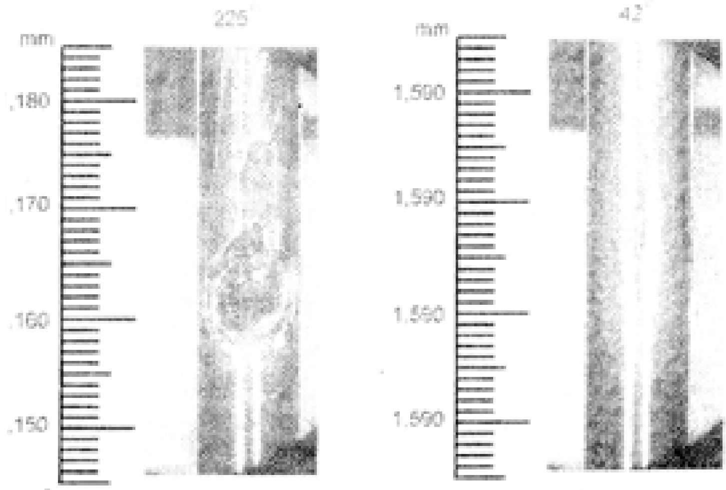

FIG. 1. Visual Inspection of D103-K2 Fuel Rod.

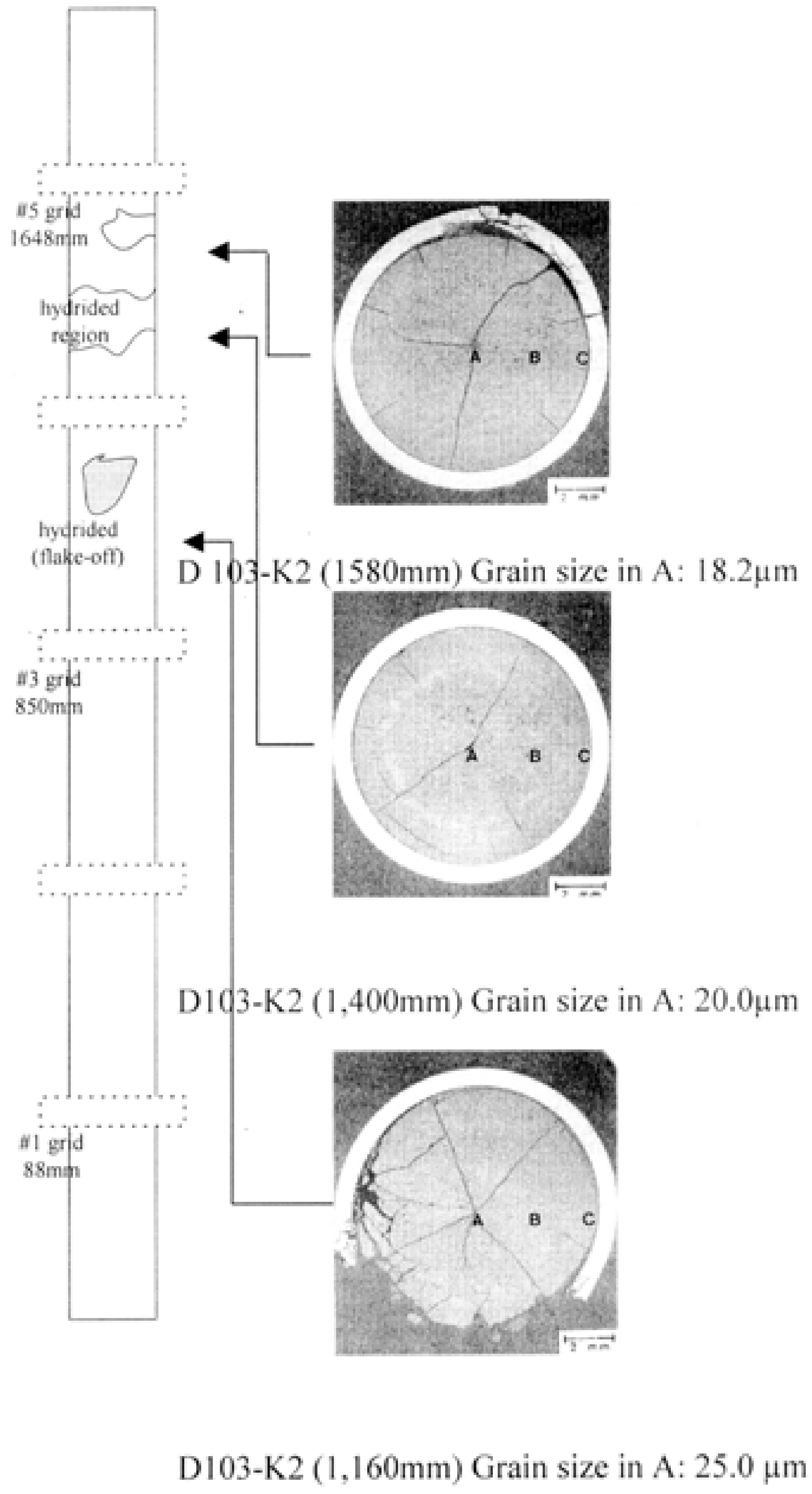

FIG. 2. Micro-structural Change of UO_2.

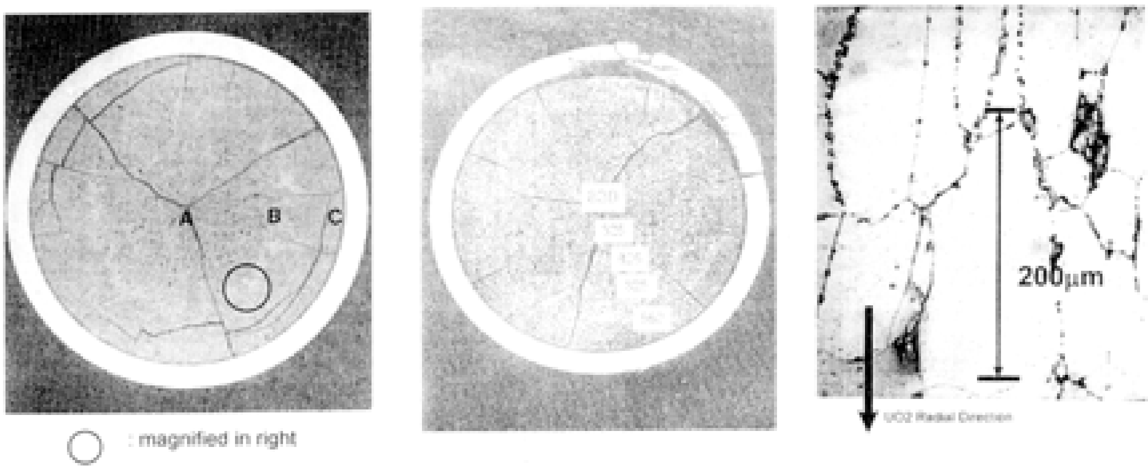

FIG. 3. Columnar Grain Growth in Steam Oxidation Environment.

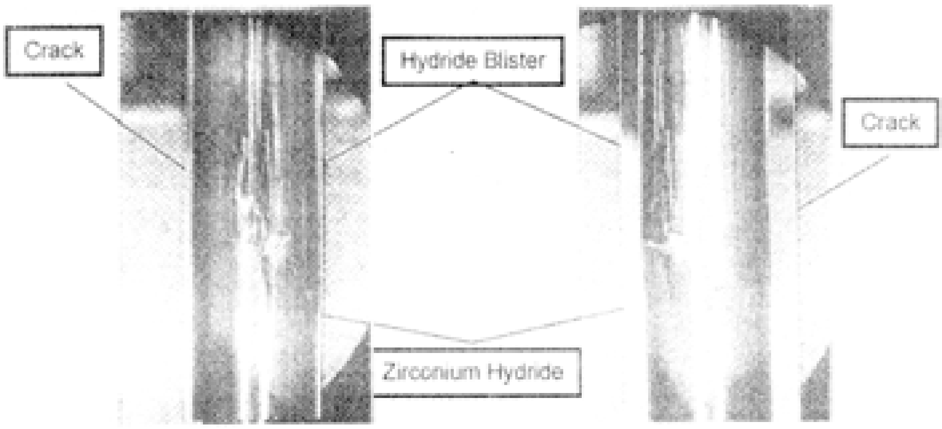

FIG. 4. A View of Hydride Failure on B208-R8 Cladding Tube (2,660mm).

3. RESULTS AND DISCUSSION

3.1. Internal Hydriding Failure of D103-K2

Hydrogenous impurities inside a fuel rod will ultimately hydride the Zircaloy cladding, regardless of their initial chemical state. Massive localized hydriding leads to the hydride blisters where the volume change is visually evident on the outside of the fuel rod, to the serious deterioration of the mechanical properties of the clad so that splits can easily develop, and eventually to the perforation of the clad after local breakthrough [3]. The main source of the hydrogen in the typical hydriding rod failures is the residual moisture in the UO_2 fuel pellets [1].

In the primary hydriding process, the residual moisture (steam) oxidizes the inner surface of the clad. Thus the thickness of oxide, over the whole length of the rod, generally follows the axial temperature profile since the Zircaloy oxidation is an activated process thus the thickness depends on the temperature. Figure 7a) and 7b) show that in the D103-K2 rod the inner and outer oxide thickness profiles are generally in good agreement with the temperature profiles predicted by the best estimate fuel performance code, ESCORE [6]. The code simulation is based on the nuclear design report for the fuel cycle of the unit [7].

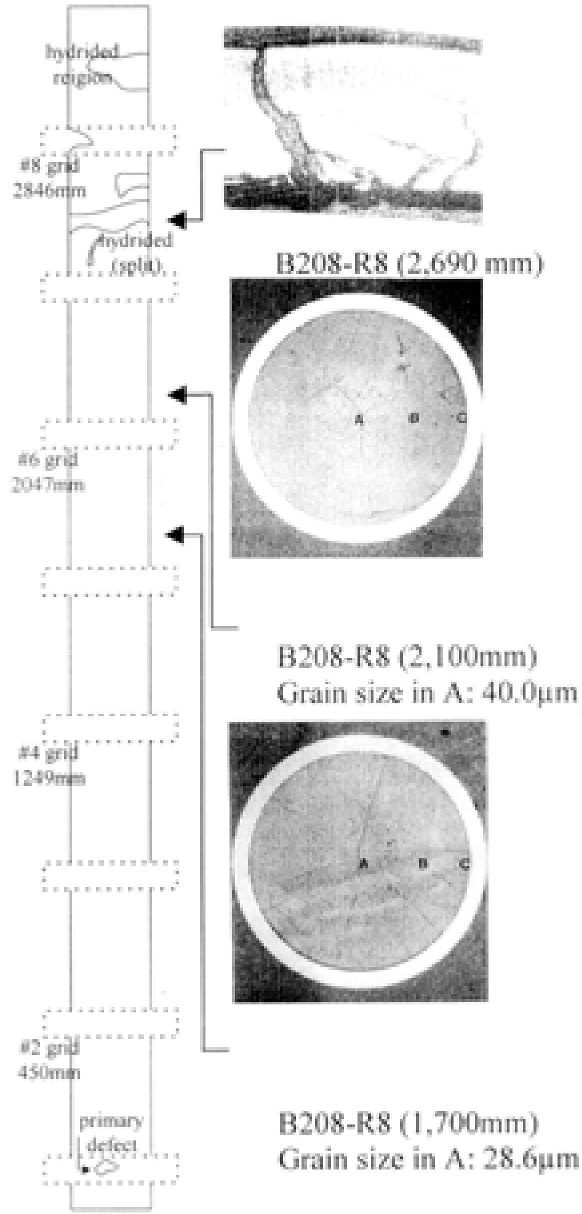

FIG. 5. Micro-structural Change of UO_2.

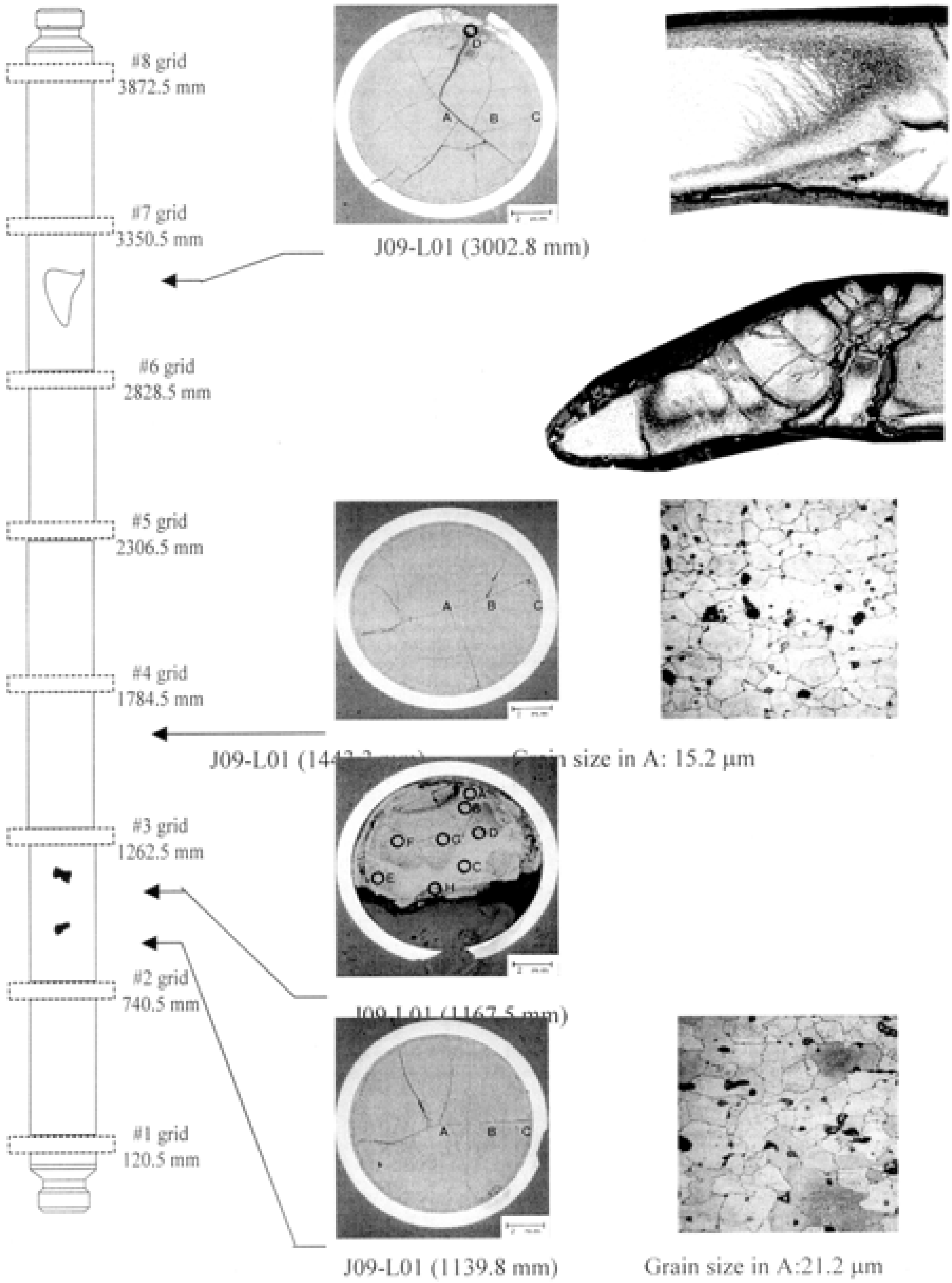

FIG. 6. PIE Results of J09-L01 Rod.

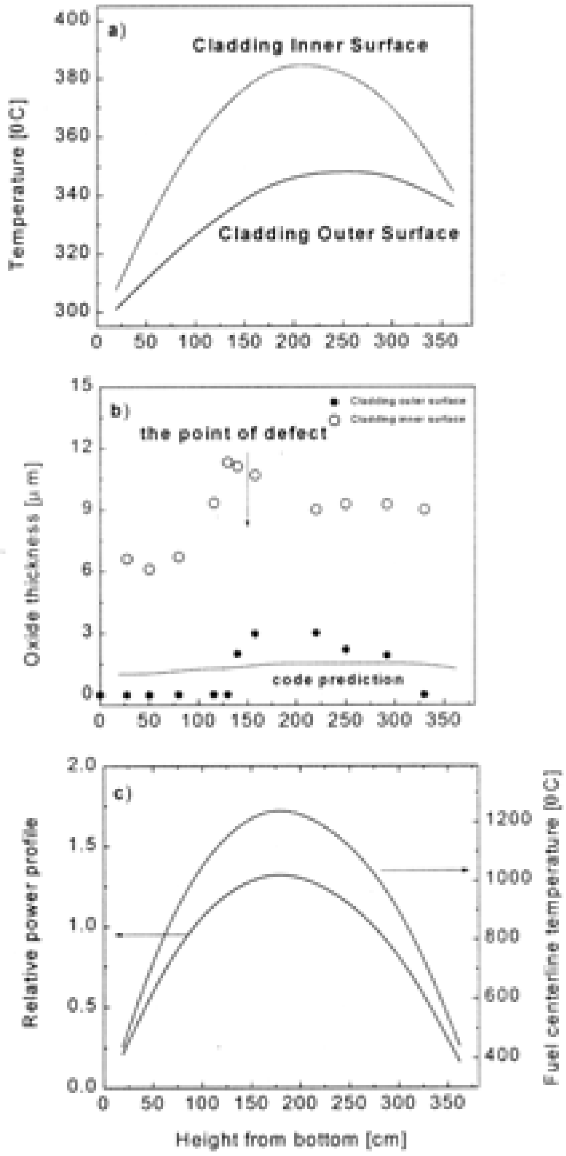

FIG. 7. Comparison Between the PIE Results and Code Prediction (D103-K2 Rod).

High temperature inside the fuel pellet, especially during start-up operation, and oxidizing environment in the moisture-rich environment induce fast restructuring of the pellet microstructure, i.e., grain growth and stoichiometry changes. Through PIE these phenomena are observed at the elevation of high linear heat generation rate of the rod, which is consistent with the fuel centerline temperature profile predicted by the code (Figure 7c). The extraordinary columnar grain growth is discussed in Section 3.3.

3.2. Secondary Hydriding Failure of B208-R8 and J09-L01 Rod

Occasionally, small primary defect leads to heavy secondary hydride formation [8,9]. During the initiation stage, coolant enters the fuel rod through the defect and flashes into steam. Once the internal and external pressures have equalized, steam oxidizes the internal Zircaloy cladding surface into ZrO_2, resulting in the release of hydrogen. As the gases react in the fuel-cladding gap, the concentration of hydrogen continuously increases while steam is depleted. Thus, the ratio of H_2/H_2O increases rapidly in the gap. In the secondary hydriding process, molecular diffusion of the steam in the gap is required, that is, the steam must be supplied from the primary defect site along the rod axis for the continues oxidation. If the ratio of H_2/H_2O exceeds a certain critical value at a certain elevation and the conditions for the accelerated hydriding is achieved in the region, such as damaged protective oxide and/or flawed surface, then the massive hydriding can instantaneously take place thus breaches the mechanically-degraded cladding [4,5].

In general, therefore, it is believed that the inner surface oxide is thickest at the primary defect site and gradually decreases when it gets far from the primary defect point. However, it is not true if temperature is varying along the rod since the temperature plays the most significant role in the Zircaloy oxidation kinetics. In the failed fuel rod B208-R8, fuel performance code predicts the cladding inner surface temperature profile which has a maximum about mid-point (Figure 8a). Therefore, instead of gradual decrease from the primary defect site, the thickness of the inner surface oxide is increasing, following the temperature profile (Figure 8b). The oxide profile and the results of the restructured pellet examined in the PIE are consistent with the predicted axial power profile of the rod.

As previously discussed, it is known, in general, that the hydriding failure, primary or secondary, takes place in the fuel rod with high linear heat generation rate since rapid oxidation and hydriding reaction of Zircaloy cladding are required for abundant hydrogen production in the gap between cladding and pellet. However, the PIE results of J09-L01 rod reveal that this rod was damaged secondarily due to the massive hydriding even though its linear power was only 66% of the core average. This implies that in the defective fuel the ambient environment is also very crucial as well as the fuel temperature.

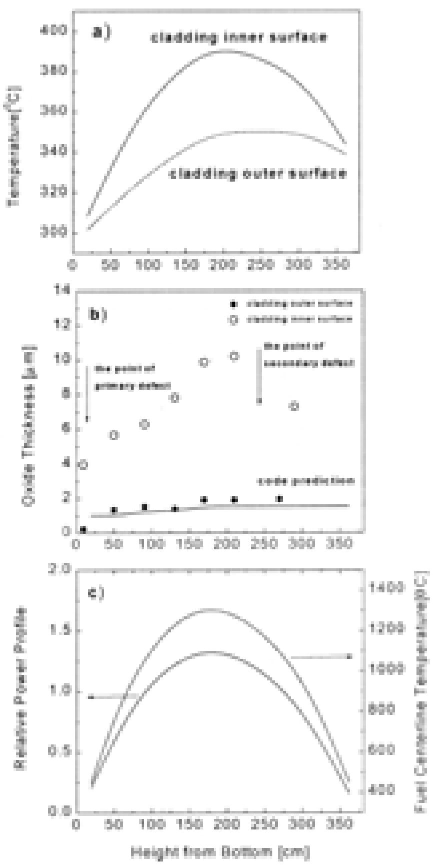

FIG. 8. Comparison Between the PIE Results and Code Prediction (B208-R8 Rod).

3.3. Enhanced Oxidation and Extra-ordinary Columnar Grain Growth

In Figures 7 and 8 it is seen that the inner surface oxide thicknesses of the two rods are generally in good agreement with the temperature of the surface predicted by the ESCORE code. However, close and careful examination reveals that the oxide in the neighborhood of the defect is thicker than that in other regions. The cause of the oxidation enhancement is believed to be due to the hydride precipitation in the Zircaloy matrix during the oxidation reaction.

Now it is well-known that the zirconium alloy oxidation kinetics is enhanced when the hydride precipitation takes place in the metal-oxide interface [10,11]. Recently it is demonstrated that the oxidation can be accelerated under the hydrogenous environment with high H_2/H_2O ratio even without the hydride precipitation in the interface, if the oxidation and hydriding reactions take place simultaneously (Figure 9) [12]. Thus, the unusual oxide thickness near the defect seems be ascribed to the simultaneous reactions in the gap between the cladding and the pellet during the internal hydriding process.

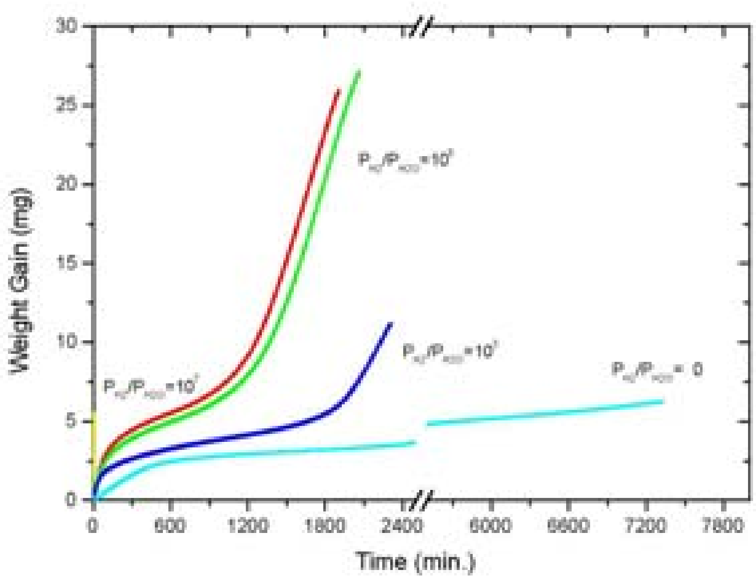

FIG. 9. Zircaloy Oxidation Enhancement in Hydriding Environment.

It is understood that, as the reactions goes on in the gap, steam is depleted and hydrogen builds up rapidly at the high temperature spot because the oxidation kinetics is faster at the high temperature spot than that at low temperature region. This leads to further oxidation, more hydrogen production, and further oxidation again, ending up with the enhanced oxidation and massive hydride formation and the eventual breach of the spot.

The observed remarkable columnar grain growth in the pellet is very unusual because UO_2 grain growth kinetics is not fast enough to grow the grain size up to 200 μm under the temperature predicted by the code. However, when the thermo-physical property changes of defected fuel and thermal conductivity degradation of the gap are taken into consideration, the extraordinary growth is understandable.

During the internal or secondary hydriding process, water entering the fuel rod turns into steam and thus the fuel pellet-to-cladding gap is quickly filled with steam. The thermal conductivity of steam is less than 10% of the helium filling gas, thus, the temperature difference in the gap drastically increases, leading to the abrupt fuel temperature increase. In addition, it has been reported that the thermal conductivity of UO_2 is degraded in the hyperstoichiometric transition during the steam oxidation 13,14]. Therefore, the increase of the fuel pellet temperature may be at least a few hundreds degree, which makes the grain growth fast.

In the mean time, steam oxidation of UO_2 start to take place from the pellet outer surface and the oxygen begins diffusing towards the center, which induces the oxygen concentration gradient. A few reports are available that under the oxidative environment, stoichiometry of the UO_2 increases and this induces the rapid grain growth [15,16].

Thus, in the defective fuel, once the gap is filled with steam fuel temperature goes up thus the centerline temperature of the fuel pellet also increases. This temperature is believed not to be high enough for the grains of the stoichiometric pellet to grow fast, however, high enough for the grain growth of the oxidizing fuel pellet in transition to hyperstoichiometry.

Therefore, on summarizing these property changes, we can explain the extra-ordinary columnar grain growth. If in a certain circumferential region the fuel temperature is high enough for the grain growth of a hyperstoichiometric UO_2 which can be achieved by the steam oxidation in the gap, the remarkable grain growth can take place in an annular band. If the temperature of the pellet is too low or the gap does not provide the oxidizing environment, any noticeable grain growth never occurs even in the local spot.

4. CONCLUSIONS

PIE data analyses reveal that fuel rod D103-K2 was failed by internal hydriding and the fuel rods, B208-R8 and J09-L01 with non-hydriding primary defects, were additionally damaged by secondary hydriding. These results are generally in good agreement with fuel performance code predictions.

Careful observation of the oxide thicknesses of clad inner surface and microstructural change of the fuel pellet raises some questions on the oxidation enhancement and extra-ordinary large columnar grain growth in the pellet. The unusual oxide thickness near the defect is ascribed to the hydride precipitation in the zirconium matrix, which accelerate the oxidation.

The extra-ordinary large columnar grain growth can be explained in terms of the fuel pellet temperature increase, the local stoichiometry change of the pellet, and the fast grain growth of the UO_2 in transition to hyperstoichiometry.

It is reported that secondary failure can take place in the rod even with low linear power because the oxidizing environment in a defected rod plays the most significant role in the thermo-physical properties changes. Finally, PIE data support the proposed mechanism for the defective fuel behaviors such as hydriding-enhanced corrosion and the secondary hydriding.

ACKNOWLEDGEMENT

This project was supported by Ministry of Science and Technology (MOST) as a part of the nuclear R&D programs.

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OUTSIDE-IN FAILURE OF BWR SEGMENT RODS DURING POWER RAMP TESTS

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Abstract

BWR 8x8 fuel assemblies with segmented rods (Step II LUAs) were irradiated up to 5 cycles in Fukushima Daini Nuclear Power Station No. 2 Unit. The ramp tests of 25 segments were conducted in Japanese Material Test Reactor. One segment rod irradiated for 3 cycles (43 GWd/t) failed by a single step ramp test after 9 minutes at terminal ramp power of 614 W/cm with a pinhole due to PCI/SCC. One segment irradiated for 4 cycles (56 GWd/t) failed by a single step ramp test after 149 minutes at 551 W/cm with an outer side axial crack. Among 9 segments irradiated for 5 cycles (61 GWd/t) two of them failed by a single step ramp test after 100 and 68 minutes at 421 and 428 W/cm, respectively and one of them failed at 446 W/cm by a stair ramp test with outer side axial crack. The decrease of the failure threshold for higher burnup segment rods is clear and the failure mode of higher burnup segment rods is different from that of low burnup (less than 43 GWd/t) segment rods. This new failure mode is caused by the combination of the high stress and the radially oriented hydride precipitated during ramp test. Through detailed PIEs before and after ramp tests, following characteristics on failed segment rods were observed. Hydrogen contents in the cladding tubes increased with burnup and exceeded solubility limits during base irradiation of segment rods. Radial hydrides were observed at the outer rim of the cladding tubes irradiated for 4 and 5 cycles and ramp-tested rods, while few radial hydrides were observed before ramp tests. Crack started from the outer surface of the cladding tube and propagated to inside and axial directions and finally failed by ductile fracture. The fracture surface showed brittle features developed by the combination of the stress due to PCMI and hydride in the cladding.

1. VERIFICATION TEST ON BWR HIGH BURNUP FUEL

Under the sponsorship of the Ministry of Economy, Trade and Industry (METI), Nuclear Power Engineering Corporation (NUPEC) has completed the fuel irradiation test program “The Verification Test on BWR High Burnup Fuel". The objectives of this program were to verify fuel integrity and to study fuel behaviors of high burnup 8x8 fuel (BWR Step II Fuel) at high burnup [1], [2], [3], [8].

Step II fuel has an 8×8 lattice configuration of 60 fuel rods in which Zr-lined and high corrosion resistant cladding tubes are used. Its design maximum bundle burnup is 50GWd/t. The bundle design has a central large diameter water rod to soften the neutron spectrum. It

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uses ferrule-type spacers in place of the earlier grid-type ones to improve thermal hydraulic performance. Furthermore, some design improvements to reduce fission gas release, such as an increased initial He pressure, increased pellet density and decreased gap width between pellet and cladding, are incorporated. The design parameters and fuel rod arrangement in the fuel assembly are shown in Figure 1 [1], [8].

Item	Unit	Value
Rod He Pressure	MPa	0.5
Pellet Clad Gap	mm	0.20
Clad Outer Diam	mm	12.3
Clad Thickness	mm	0.86
Pellet Diameter	mm	10.4
Pellet Density	%TD	97

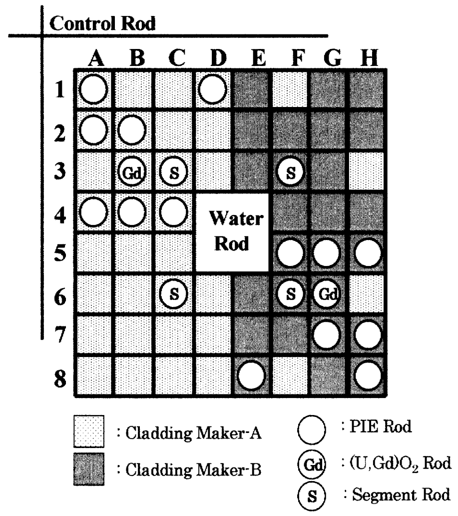

Figure 1. Design Parameters and Rods Arrangement.

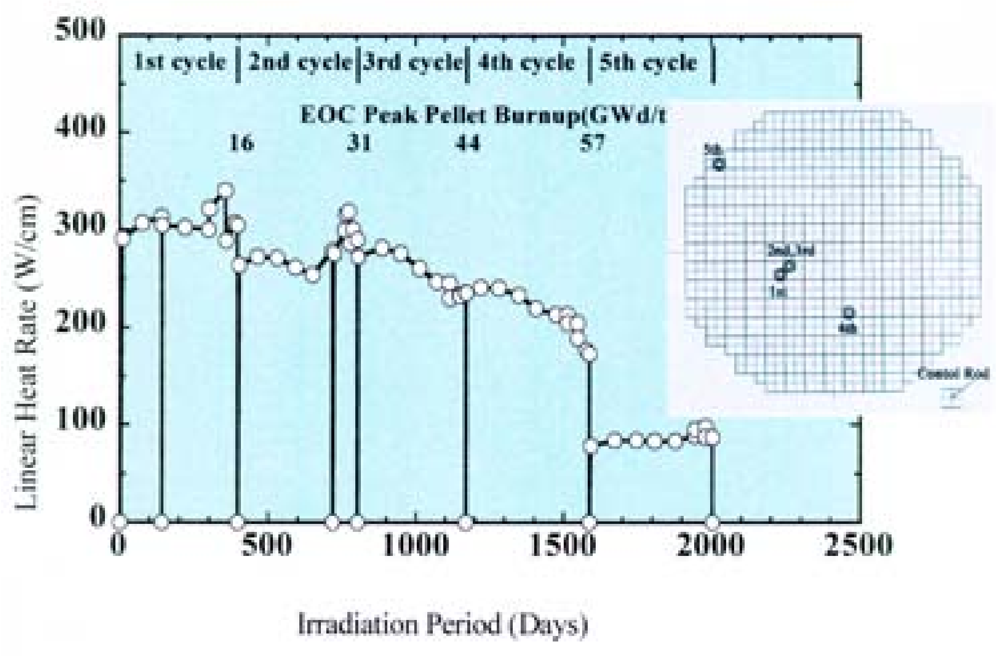

Figure 2. Irradiation History of Step II luas.

Eight lead use assemblies (LUAs) equipped with segment rods were loaded in a relatively high power and symmetric position in the core of Fukushima Daini Nuclear Power Station No. 2 Unit, operated by Tokyo Electric Power Co. They were irradiated up to 5 irradiation cycles under normal BWR conditions and one assembly was discharged after each irradiation cycle for detailed PIEs. The reactor was operated at almost full power through the five cycles. Irradiation history of LUAs is shown in Figure 2. Maximum linear heat rates of LUAs were kept above 300 W/cm in the first cycle, above 250 W/cm in the second and third cycles and decreased to 200 W/cm in the fourth cycle and 80 W/cm in the fifth cycle. The integrity of high burnup 8x8 fuel was confirmed up to the bundle burnup of 48 GWd/t after 5 cycles of irradiation.

After each of the reactor cycles, LUAs were subjected to basic non-destructive examinations in the spent fuel pool, including visual examination with an underwater TV camera. Five LUAs have been examined in the hot laboratory of Nippon Nuclear Fuel Development Co., Ltd. (NFD) and detailed PIE data were systematically obtained. The burnups of these five LUAs were about 13, 24, 35, 44 and 48GWd/t, respectively. At each burnup stage, non-destructive tests (NDT) were executed on principally 16 rods and destructive tests (DT) were performed on 2 or 3 rods, including a $(U,Gd)O_2$ rod.

In order to confirm both the integrity of fuel rods during normal/off-normal operational transients and the margin of ramp performance, power ramp test series were executed using 25 segment rods irradiated for three to five cycles (burnup range: from 43 to 61GWd/t).

The ramp sequences are a stair case power ramp (Ramp sequence A), a single step power ramp (Ramp sequence B), and a power cycling test (Ramp sequence C). Ramp sequence A is used to calibrate heat generation by He-3 gas pressure and to determine power to failure. There are three types of ramp sequence B, that is, Bs, Bt and Br. Ramp sequence Bs is employed to confirm the integrity of fuel during normal operational transients as is ramp sequence C also. In this case, ramp terminal power (RTP) corresponds to the designed maximum power level. Ramp sequence Bs is also employed to investigate the margin of ramp performance by increasing RTP up to about 150% of the designed maximum power level. The integrity of fuel rods under abnormal operational transients, such as control rod withdrawal, is studied by ramp sequence Bt in which RTP and the holding time are selected as about 125% of the designed maximum power level and 10 minutes, respectively [2]. RTP and the cumulative holding time of ramp sequence Br are the same as those of ramp sequence C, so it is possible to obtain the characteristics of fuel behavior during cyclic operation by comparing with ramp sequence Br. Four typical power ramp sequences (two Bs, Bt and C) are shown in Figure 3.

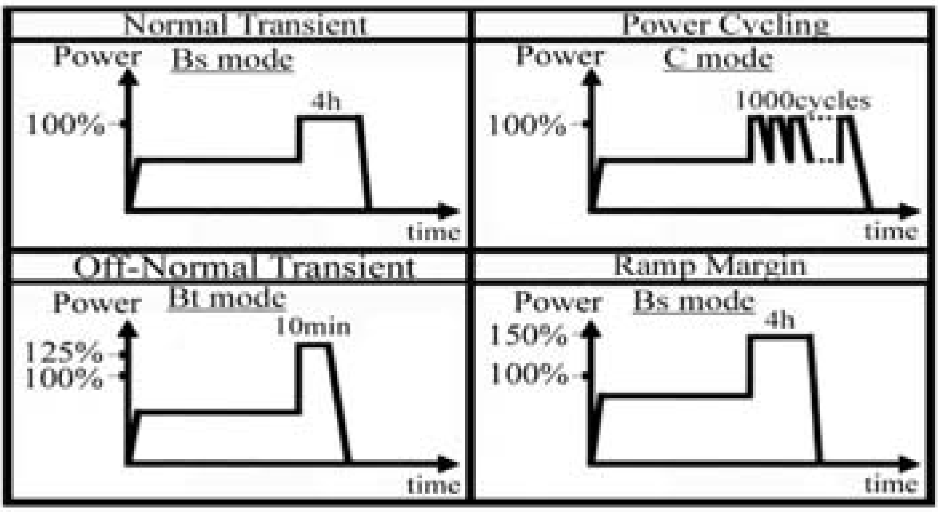

Figure 3. Power Ramp Sequence.

2. CHARACTERISTICS CHANGES OF HIGH BURNUP FUELS

Cladding water-side corrosion, resulting hydrogen uptake is considered to be a key factor for PCMI failures at high burnup. Corrosion and hydrogen uptake behaviors are examined in detail.

Visual appearances and cross-sectional micrographs of the cladding are shown in Photo 1. Two types of corrosion were observed. Nodular corrosion was clearly observed at the gas plenum region, although the oxide layer of fuel stack region was uniform and the thickness was 10 - 20 micrometers in the transverse sections of each fuel rod, as shown in Photo 1.

Burnup dependence of maximum oxide thickness in fuel stack regions of Step II LUAs is shown in Figure 4, which also includes those of 8x8 fuels and Step I LUAs [3], [4], [8]. In this figure, oxide thickness obtained by metallography is plotted as a function of specimen burnup. Maximum oxide thickness of Step II LUAs was less than one half of 8x8 fuels throughout their irradiation cycles.

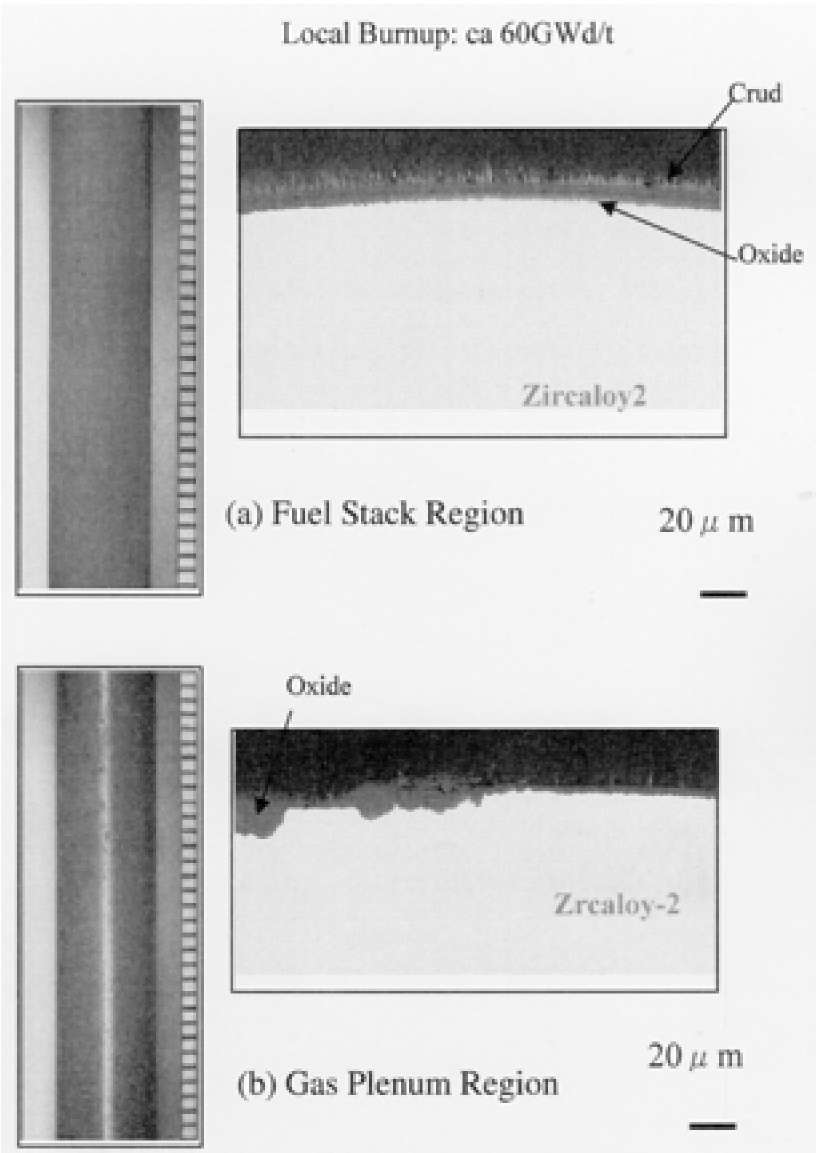

Photo 1. Visual Appearance and Cross Sections of Oxides on Fuel Rods.

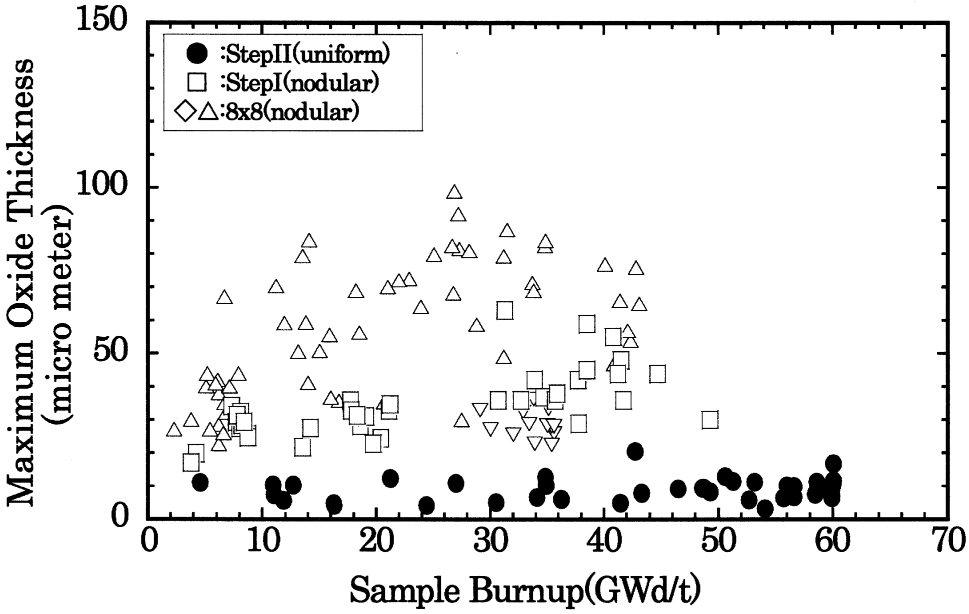

Figure 4. Burnup Dependence of Maximum Oxide Thickness.

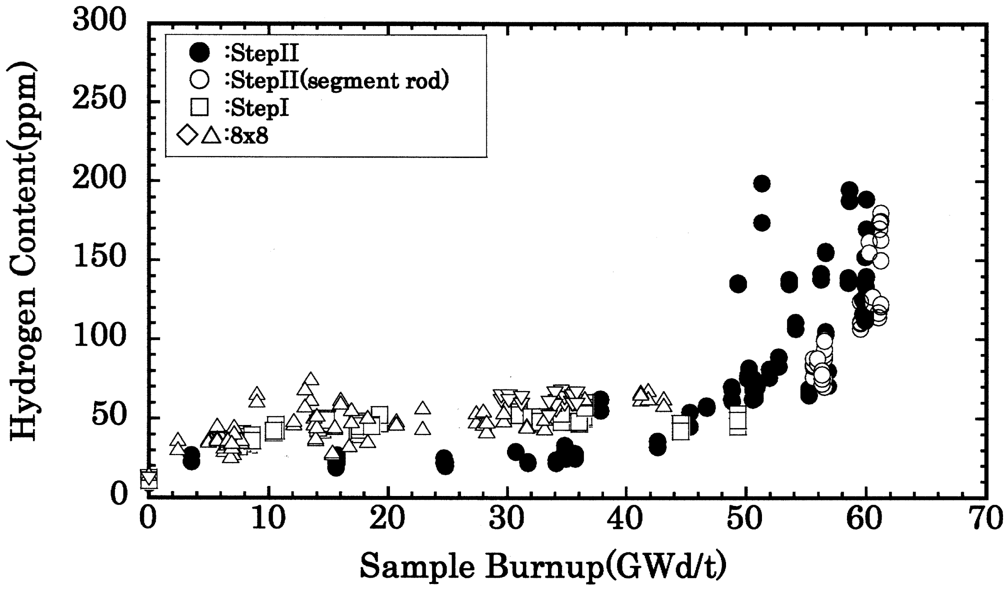

Figure 5. Hydrogen Content in Cladding v.s. Irradiation Period.

The content of hydrogen in cladding was measured by an inert gas fusion method. Hydrogen content of BWR fuel cladding is shown in Figure 5 as a function of irradiation period. Maximum hydrogen content of Step II LUA cladding was less than that of 8x8 fuels up to three irradiation cycles, but it tended to increase with an increase in irradiation period thereafter, and after five irradiation cycles, the content was almost the same as those of the reported.

Hydride morphologies in the cross-sectional metallography of the cladding are shown in Photo 2. Hydrides in the cladding irradiated for 3 cycles are small and distribute uniformly. Hydrides increase with burnup after 4 cycles of irradiation and the number densities of hydride in outer peripheral and Zr-liner are high [8].

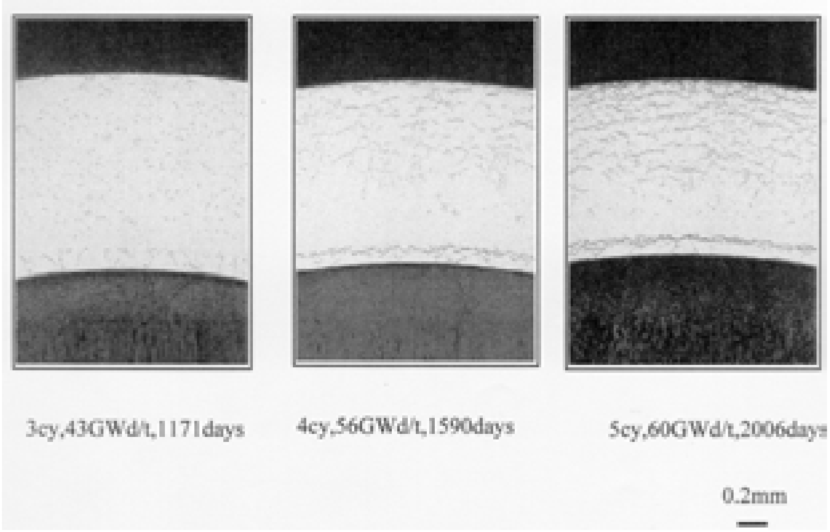

Photo 2. Hydride Morphology of Step II Fuel Cladding.

With increasing burnup, pellet-cladding gap in fuel rods tends to close due to pellet swelling and cladding creep-down, and eventually a bonding layer is formed between the pellet and cladding under contact conditions at high burnups. The bonding layer was observed for Step II LUAs after three cycles of irradiation. The thickness of the layer is about 10 – 20 μm and remained almost constant in spite of the increase in burnup. Considering EPMA measurement results at the pellet-cladding interface and published information [5], this layer may be produced by subsequent mutual diffusion of UO_2 and ZrO_2 [8].

3. FUEL RODS FAILURE DURING POWER RAMP TESTS

A series of ramp tests for 25 segment rods of burnup ranging from 43 to 61 GWd/t was carried out in JMTR using the Boiling Water Capsule under simulated BWR temperature and pressure conditions. The ramp test results are shown in Figure 6 with published data [2], [3], [4], [8].

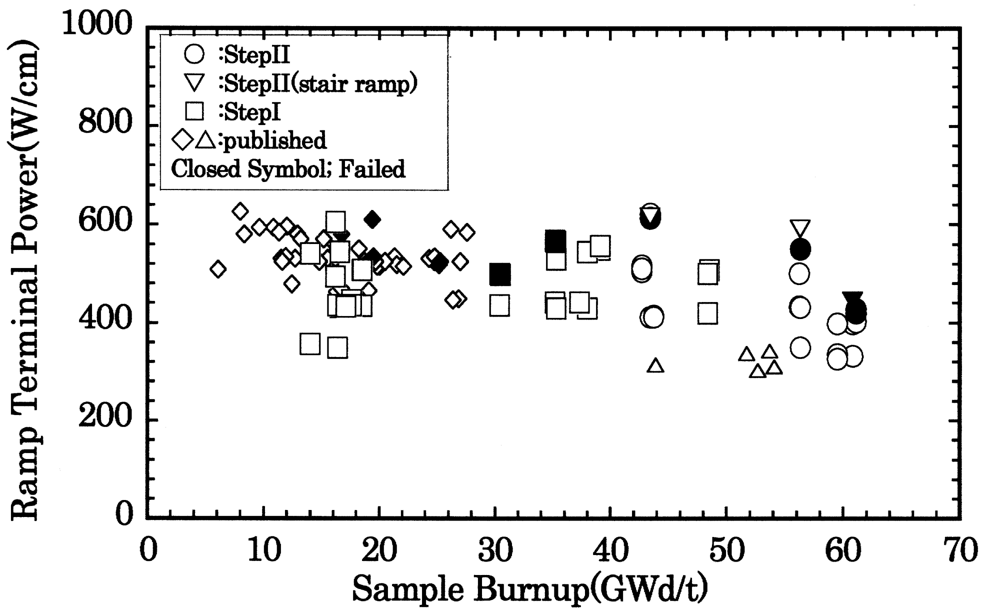

Figure 6. Power Ramp Test Results for Zr-lined Fuel Rods.

Five segment rods failed during the power ramp tests. The appearances of failed segment rods are shown in Photo 3. One segment rod irradiated for 3 cycles failed at RTP of 610 W/cm by PCI/SCC mechanism as shown in Photo 4, while four segment rods irradiated for 4 and 5 cycles failed at about 550 and 420 W/cm, respectively and had axial cracks starting from the outer surface of cladding tubes.

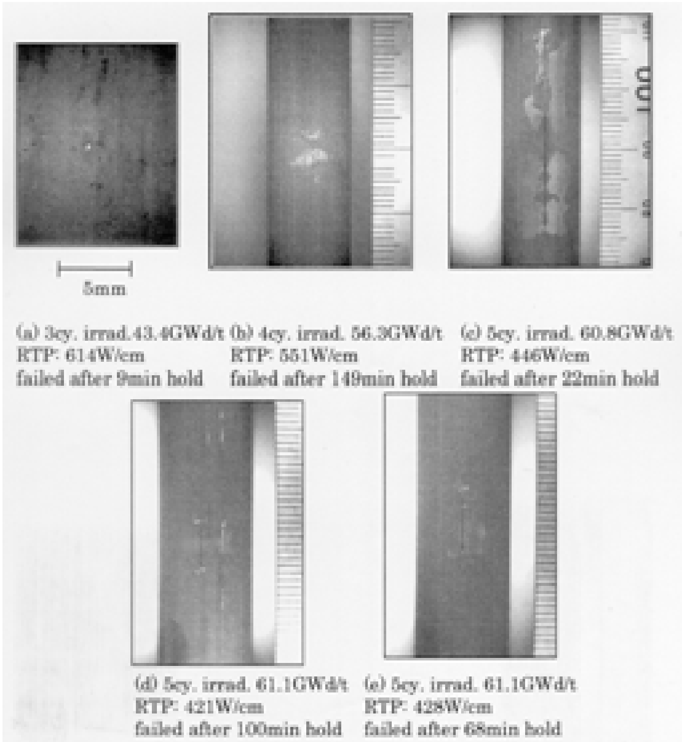

Photo 3. Visual Appearance of Failed Segment Rods.

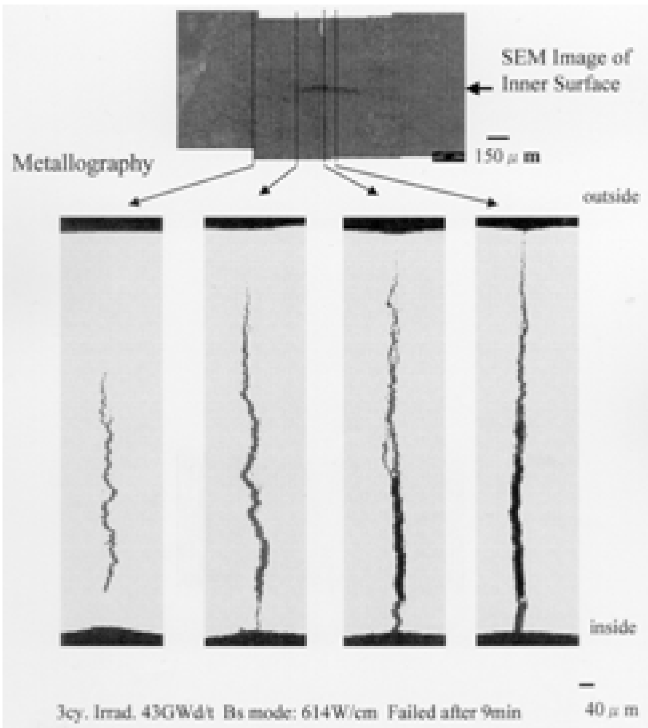

Photo 4. SEM Image and Cross-sectional Metallography of Failed Segment (3cy. Irradiated).

Following characteristics on failed segment rods were observed through detailed PIEs before and after ramp tests.

The cross-sectional micrographs at four elevations of the through wall crack of the segment rod irradiated for 5 cycles are shown in Photo5. The shape of cross section near the center of the axial crack is perpendicular to the outer surface of the cladding tube and the failure mode is strongly brittle. At the upper part of the crack, there is a small brittle portion in the outer rim of the cladding tube and a large ductile area in the middle part [8].

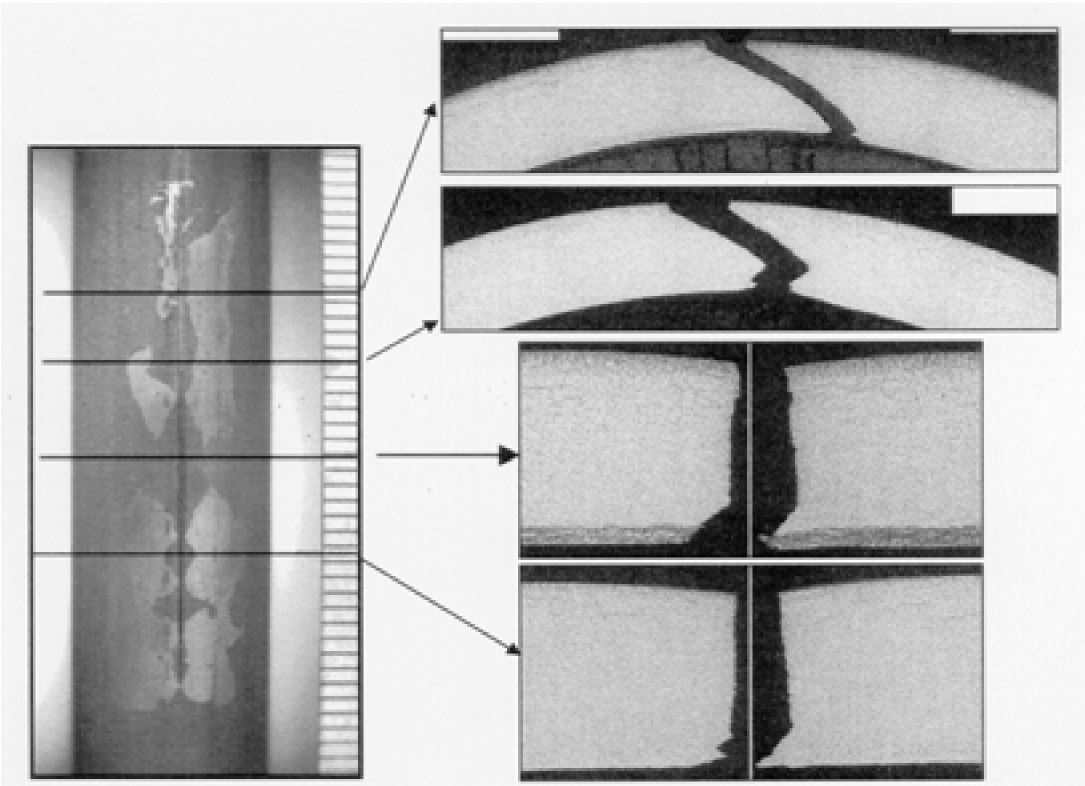

Photo 5. Visual Appearance and Cross Sectional Metallograpy of Failed Segment Rod.

Radial hydrides are observed at the outer rim of the cladding tubes in Photo 6 of which was taken from the failed segment rod during power ramp test after 5 cycles of irradiation. The length of radial hydrides is about 70 micrometers. These radially-oriented hydrides were observed on all rods failed by ramp-tests after 4 and 5 cycle irradiations, and the length of the hydrides depended on RTP. No radially-oriented hydrides were found on rods before ramp tests [8].

Photo 7 shows the SEM images of fracture surface. The fracture surface is composed of three regions. The region near outer surface is strongly brittle, but that of the near inner surface is ductile while the middle region is macroscopically brittle but microscopically the mixture of brittle and ductile features. The width of strongly brittle region is about 50 to 80 micrometers and that of macroscopically brittle crack is about 700 micrometers from outer surface. The width of strongly brittle region is close to the length of radial hydrides at the outer rim of the cladding tubes. In Photo 8 two pairs of the two opposing fracture surface are shown. It is evident that there is a good fit between the two sides of the crack, i.e., a “hill” on one fracture surface corresponds to a “valley” on the matching fracture surface [6], [8].

Photo 6. Hydride Distribution in Cladding of Failed Segment Rod.

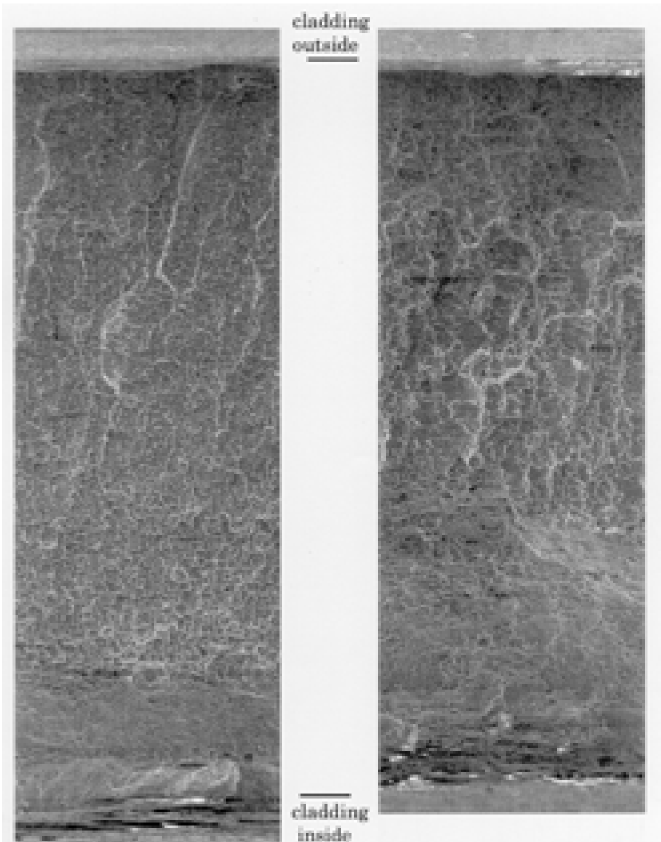

Photo 7. SEM Images of Fracture Surfaces.

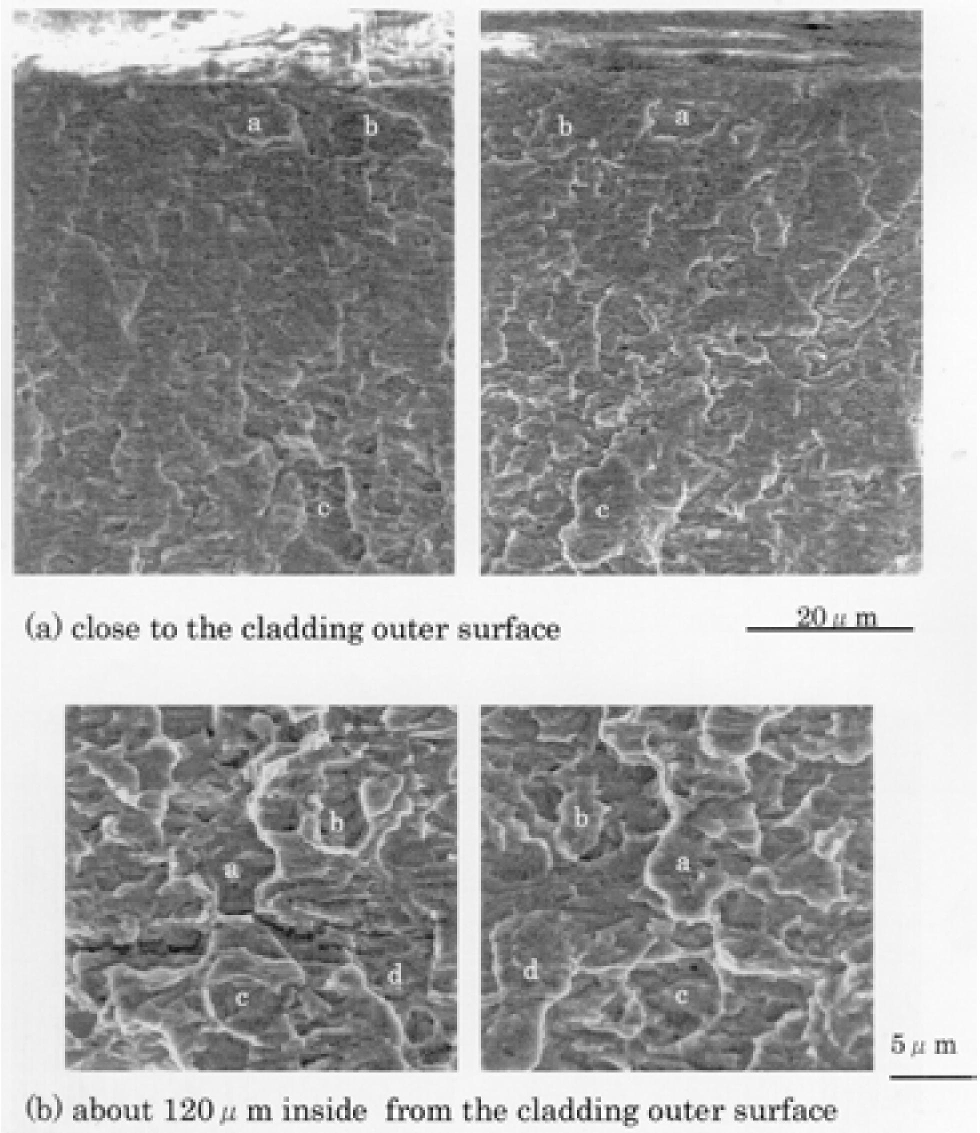

Photo 8. SEM Images of two Matching Fracture Surfaces.

The cross-sectional micrographs of another crack sample from the segment rod irradiated for 5 cycles are shown in Photo 9. There are many hydrides perpendicular to fracture surface. These hydrides are also seen in Photo 6.

In order to get the root cause of the crack from the outer surface of the cladding tube, many metallurgical observations were done, from three different directions, i.e. axial, radial and circumferential directions. Photo 10 shows the view from axial direction at non-penetrated cracks. There are many hydrides perpendicular to the crack on both sides and some hydrides are gathering at the crack tips. There is a small hydride just at the crack tip as shown in Photo 11. View from radial direction close to the cladding outer surface at the axial ends of crack indicates a nest of hydrides at the small crack tip as shown in Photo 12. These photos tell us the crack propagation is strongly connected with hydrides gathering at the crack tip.

4. FAILURE MECHANISM [5], [6], [7], [8]

By the detail PIE and the analysis we estimated the failure mechanism by axial crack would be as follows.

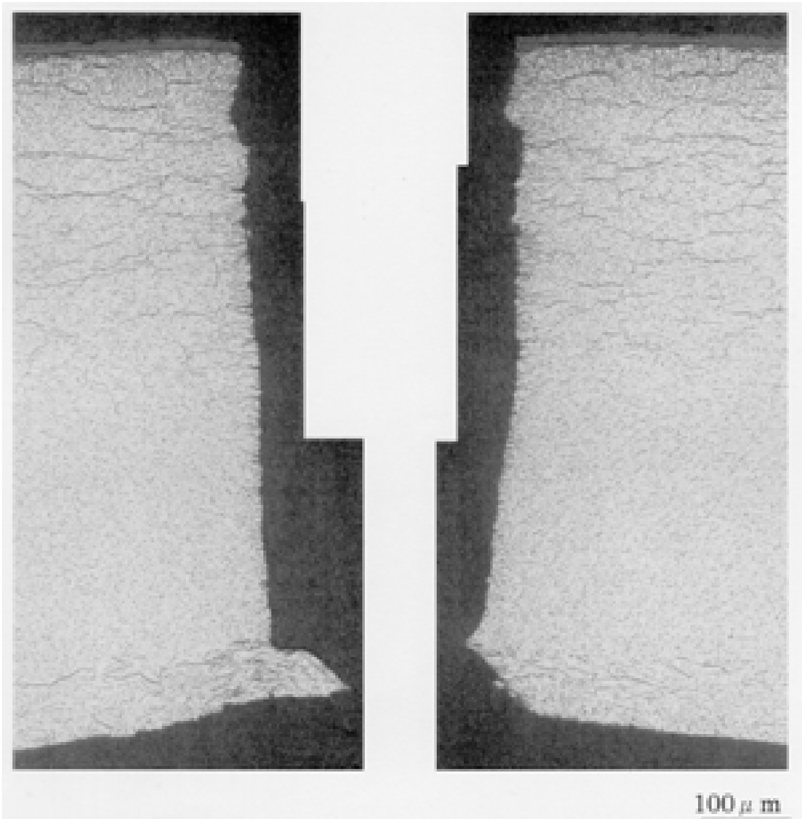

Photo 9. Hydride Metallography at the Crack.

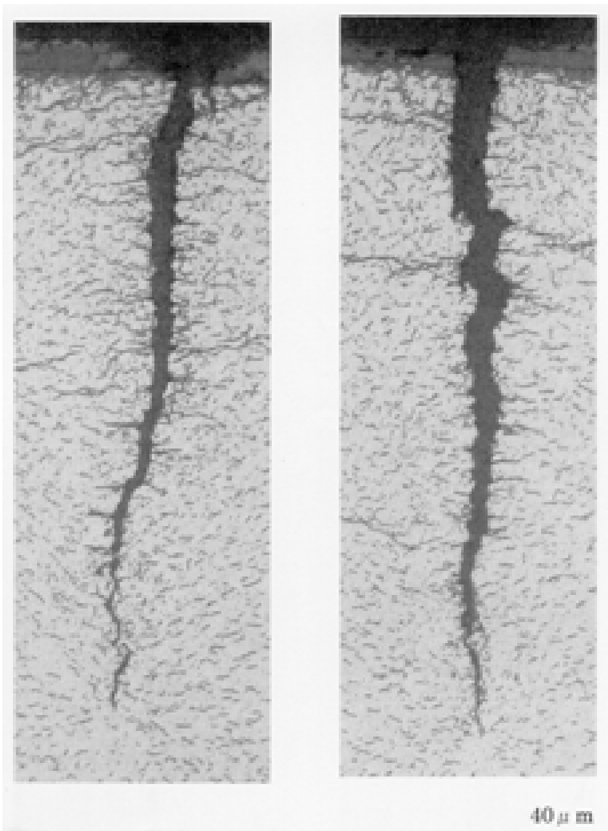

Photo 10. Hydride Distribution at Non Penetrated Cracks.

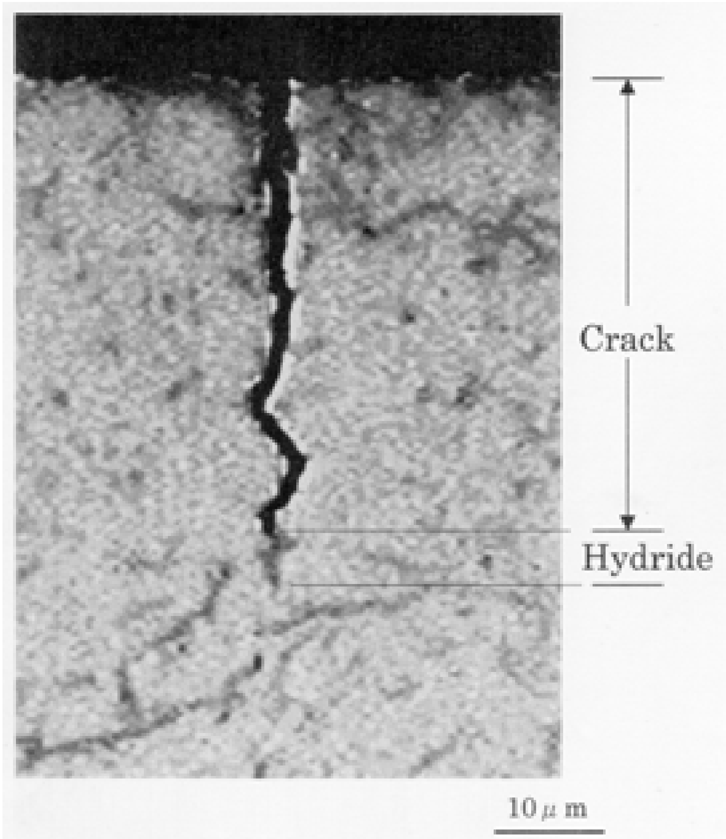

Photo 11. Hydride Observed at Crack Tip (Radial Cross Section).

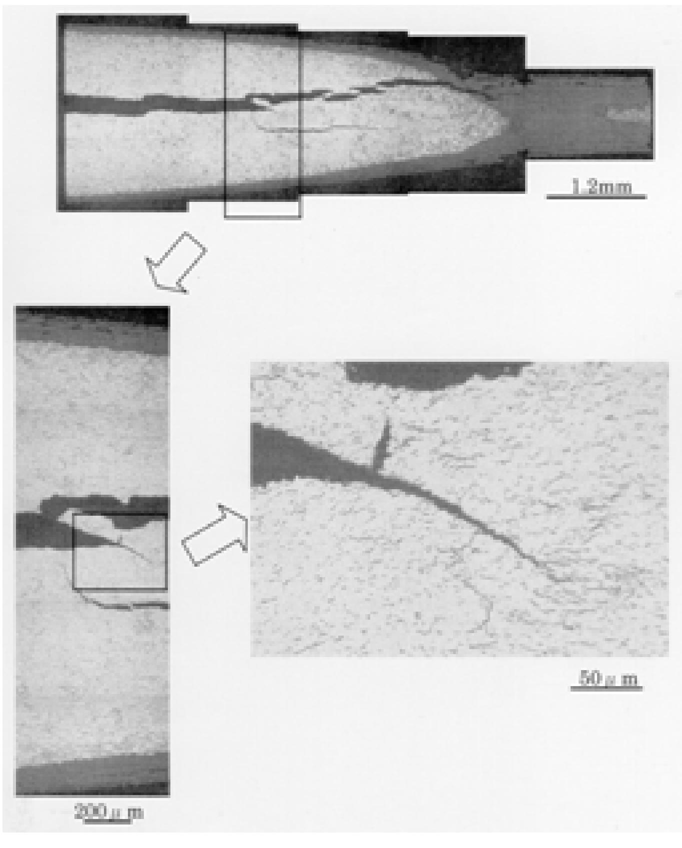

Photo 12. Hydride Observed at Crack Tip (Tangential Cross Section.

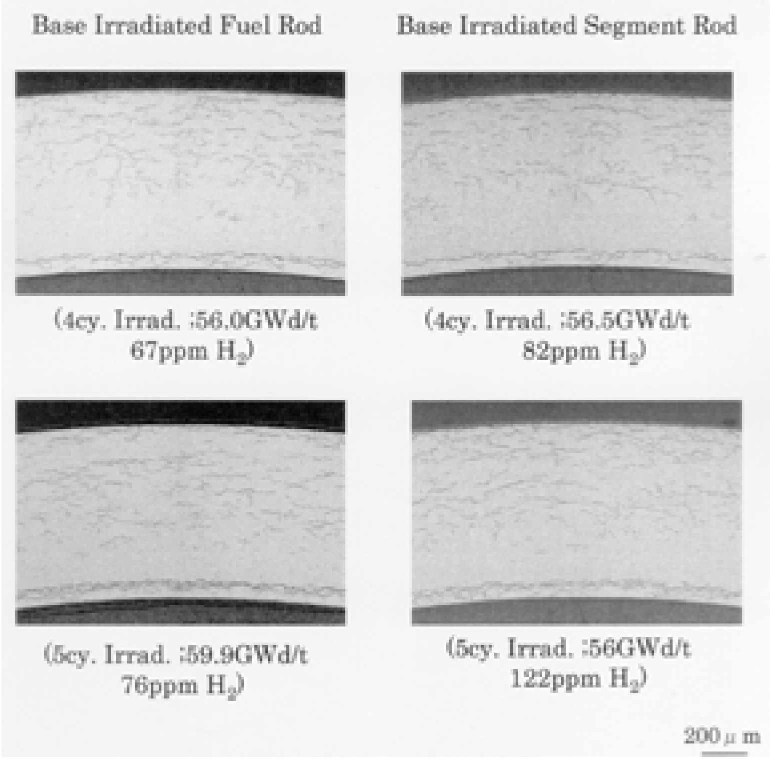

Photo 13. Hydride Distribution in Cladding of Base Irradiated Fuel Rod and Segment Rod.

Hydrogen contents in the cladding tubes increased with burnup as shown in Figure 5 and exceeded solubility limits of cladding material after 4 and 5 cycle of base irradiations. Photo 13 shows the distribution of hydride in the cladding tubes irradiated 4 and 5 cycles. In the outer region of the cladding tube there are many circumferentially oriented hydrides. Inner portion of the cladding tube, Zr liner has more hydrides due to lower solubility than Zircaloy-2. Estimated relative hydrogen content distributions are shown in Figure 7. These hydrogen contents were evaluated using the densities of hydrides in the micro photos.

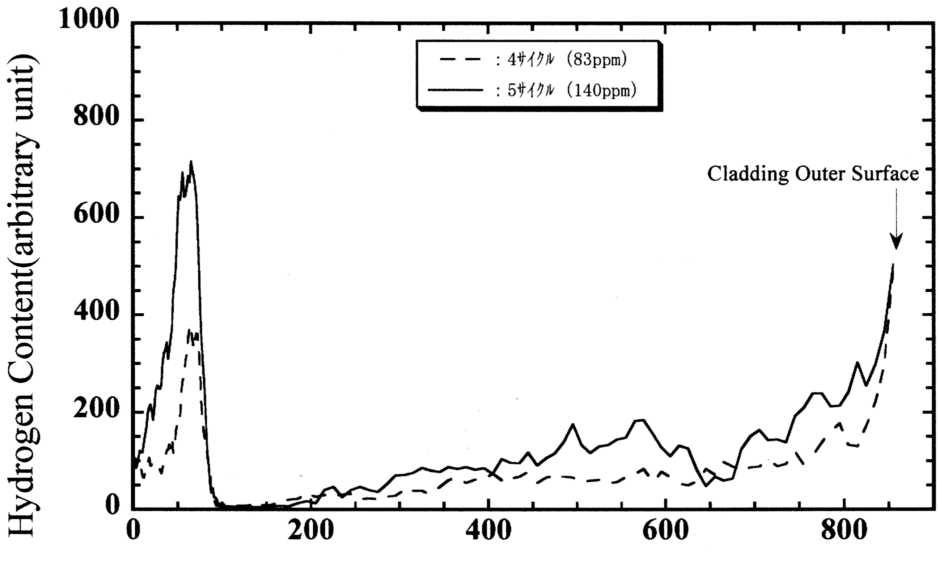

Figure 7. Hydride Distribution in Cladding after Base Irradiation.

During power ramp tests, some hydrogen diffused from inner to outer region in the cladding tube due to temperature gradient. The distribution of hydrogen solubility limit in the cladding tube is so steep as shown in Figure 8 under power ramp conditions. Diffused hydrogen precipitated radially at the outer region due to the combined effects of relatively lower solubility and high tensile stress in the cladding caused by PCMI.

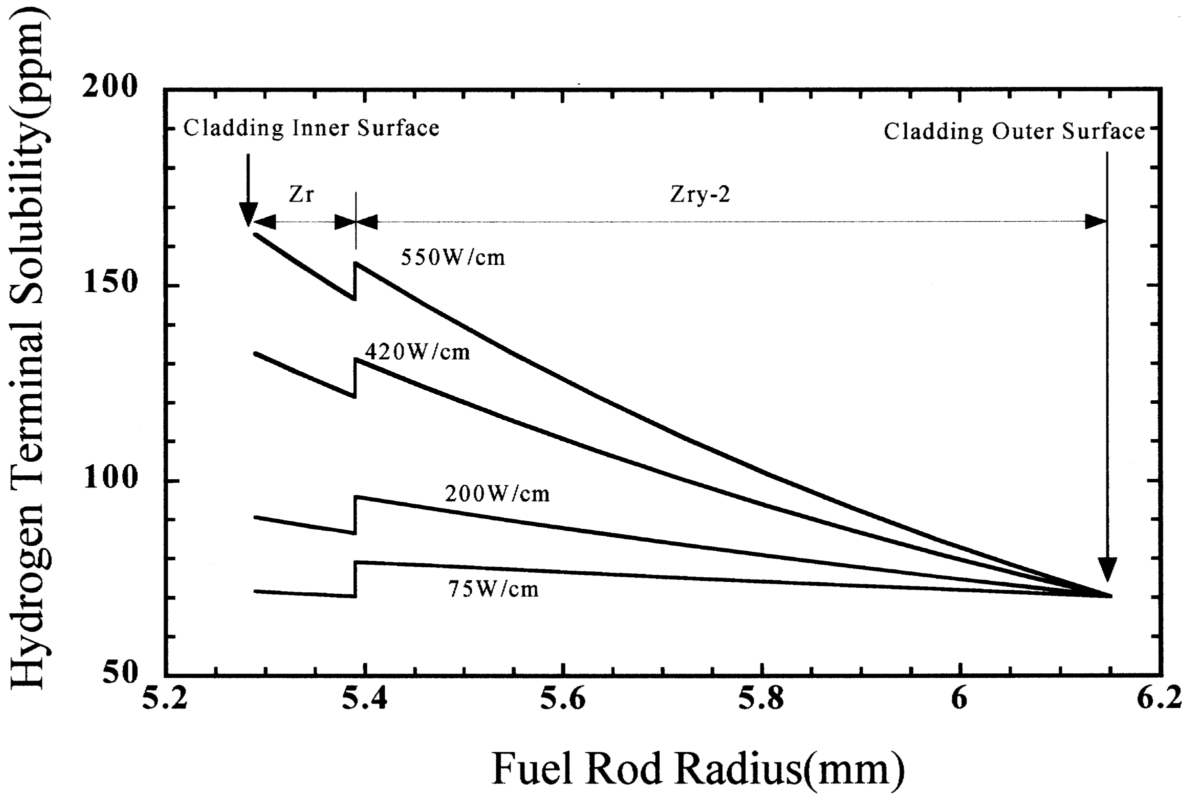

Figure 8. Hydrogen Solubility Limits in Cladding.

Radial hydride was cracked by high stress or strain by PCMI.

This is the initiation of the failure due to crack and the schematic diagram is shown in Figure 9(1).

When initial crack formed during the power ramp test, hydrogen concentrated at the crack tip due to stress concentration and small hydride was formed as shown in Photo 10 and Photo 12. These small hydrides at the crack tip cracked and formed “hill” and corresponding “valley” on matching fracture surface as shown in Photo 8 by the stress due to PCMI. Then fresh crack tip is formed as shown in Photo 11.

These stress concentration, hydrogen diffusion, hydrogen precipitation and the cracking of hydride repeated and the crack propagated. This is the propagation mechanism of the crack and the schematic diagram is shown in Figure 9(2).

When stress exceeded the yield strength of residual cladding thickness, ductile failure appeared.

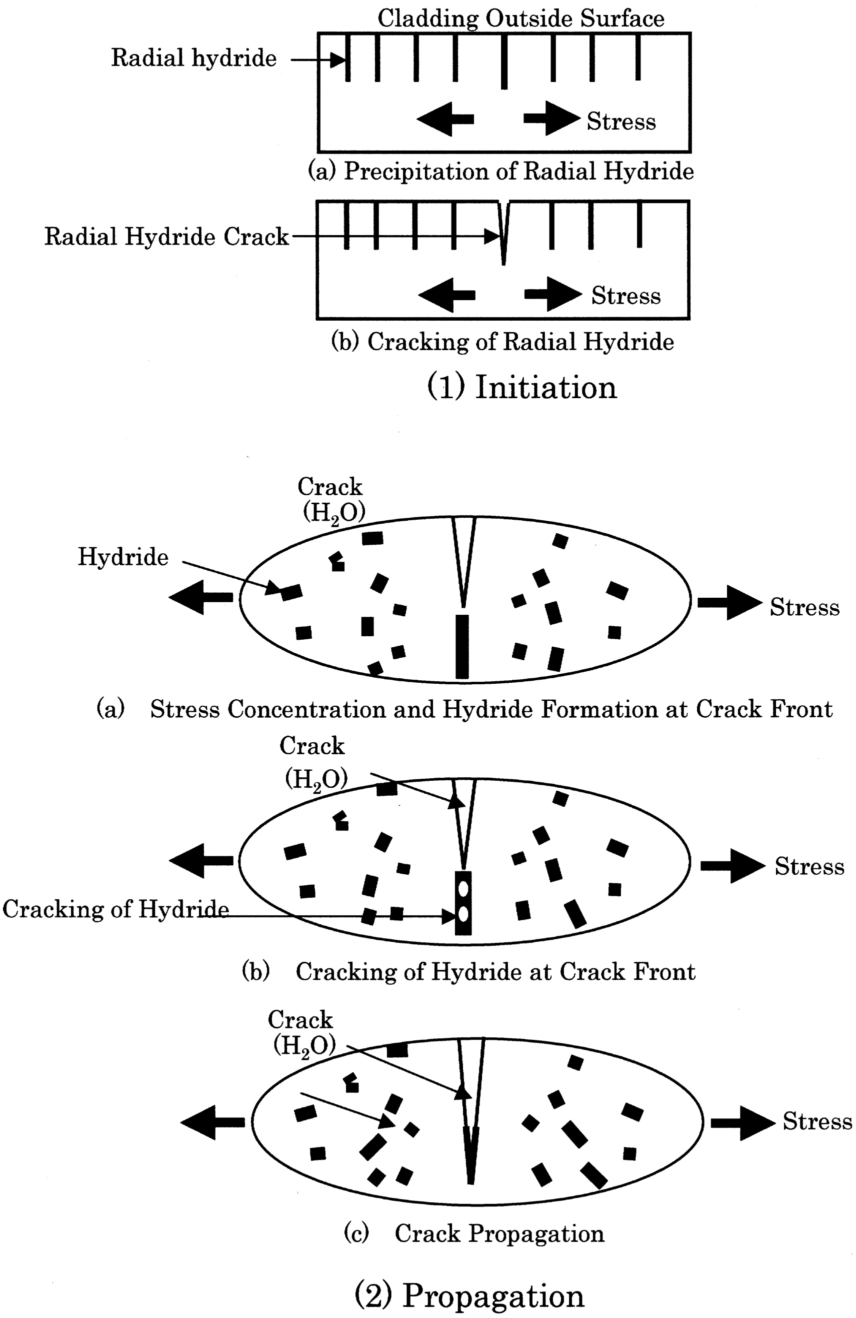

Figure 9. Schematic Diagram of Crack Initiation and Propagation.

5. SUMMARY

In this verification test program, well-characterized LUAs of high burnup 8x8 fuel were irradiated in a typical commercial BWR in Japan, Fukushima Daini Nuclear Power Station No. 2 Unit, under normal operational conditions. After each irradiation cycle, detailed PIEs at a hot laboratory and power ramp tests on segment rods, which were installed in LUAs, were carried out.

As the results of these examinations, some segment rods irradiated for 4 and 5 cycles failed at about 550 and 420 W/cm, respectively, during power ramp tests and showed axial cracks initiated at outer surface of cladding tubes.

Estimated failure mechanism by axial crack can be summarized as follows:

1) Hydrogen contents in the cladding tubes increased with burnup and exceeded solubility limits during base irradiation of segment rods.
2) During power ramp tests, some hydrogen diffused from inner to outer region in the cladding tube due to temperature gradient. Diffused hydrogen precipitated radially at the outer region due to coupled effects of high hydrogen content and high tensile stress by PCMI.
3) Radial hydride cracked by high stress and/or strain.
4) New, small hydride precipitated at the crack tip due to stress concentration and cracked repeatedly.
5) When stress exceeded the yield strength of residual cladding thickness, it failed ductilely.

ACKNOWLEDGEMENTS

This study was carried out mainly under the direction of committees of NUPEC. The authors wish to express their gratitude to the members of the "Committee of Verification Tests on High Burnup Fuel", especially Dr. T. Okubo and Dr. M. Yamawaki for their discussions and evaluations with respect to this program. The authors also gratefully acknowledge METI for the sponsorship on this project.

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