IAEA-TECDOC-997

Water channel reactor fuels and fuel channels: Design, performance, research and development

Proceedings of a Technical Committee meeting held in Vienna, 16–19 December 1996

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

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

WATER CHANNEL REACTOR FUELS AND FUEL CHANNELS:
DESIGN, PERFORMANCE, RESEARCH AND DEVELOPMENT
IAEA, VIENNA, 1998
IAEA-TECDOC-997
ISSN 1011-4289

Printed by the IAEA in Austria
January 1998

FOREWORD

The issues of BWR, PWR and WWER fuel technology and performance, and fuel thermal and mechanical designs providing sufficient safety margins under high burnup conditions have been a focus of IAEA activities during the last several years. CANDU fuel and pressure tube performance and reliability have been also considered, especially taking into account the perspective of transfer from natural uranium to slightly enriched fuel. However, fuel and pressure tube/fuel channel performance and design aspects of other types of water cooled reactors including Atucha-I and II (Argentina), FUGEN (Japan) and RBMK (16 units in Lithuania, Russian Federation and Ukraine) have been considered to a rather lesser extent.

With this in mind, the International Working Group on Water Reactor Fuel Performance and Technology (IWGFPT) recommended holding a Technical Committee Meeting on Water Channel Reactor Fuel including into this category fuels and pressure tubes / fuel channels for Atucha-I and II, BWR, CANDU, FUGEN and RBMK reactors. The IWGFPT considered that even if the characteristics of Atucha, CANDUs, BWRs, FUGEN and RBMKs differ considerably, there are also common features.

These features include materials aspects, as well as core, fuel assembly and fuel rod design, and some safety issues. There is also some similarity in fuel power history and operating conditions (Atucha-I and II, FUGEN and RBMK).

Experts from 11 countries participated at the meeting and presented papers on technology, performance, safety and design, and materials aspects of fuels and pressure tubes / fuel channels for the above types of water channel reactors.

The IAEA wishes to thank all the participants for their contributions to this publication, which summarizes experience to date with fuel and pressure tube/fuel channel for water channel reactors, and ongoing research and development in participating countries. The IAEA officer responsible for this TECDOC was V. Onoufriev of the Division of Nuclear Power and the Fuel Cycle.

CONTENTS

SUMMARY

The International Working Group on Water Reactor Fuel Performance and Technology (IWGFPT) recommended to the IAEA to hold a Technical Committee Meeting (TCM) on Water Channel Reactor Fuel. This proposal was elaborated at 12th (May 1994) and 13th (September 1995) IWGFPT plenary meetings and discussed in detail at the consultants meeting in April 1996. It was suggested to consider both coolant channels in BWRs and PHWRs (which support only the pressure difference between inlet and outlet) and pressure tubes in CANDUs, RBMKs and FUGEN reactor as water channels due to the close thermohydraulic conditions (coolant temperature, pressure and flowrate). Pressure tube behaviour under irradiation and evaluation of their lifetime were recognized as subjects of high importance because retubing (if necessary) is a very costly and radiation hazardous operation. With respect to fuel (assemblies, rods, claddings and fuel itself) it was recommended to evaluate the impact of reload type (on-power or during outages) and coolant condition (boiling, not boiling) on fuel performance and safety margins.

Subsequent analysis confirmed that, inspite of significant differences in design, burnup targets and performance requirements for the above mentioned reactor types, many common features were identified. Therefore, a comparison of fuel and pressure tube designs, performance experience and trends in technical development supported cross fertilization of ideas between countries and groupswith different reactor types.

Major features of reactors which fuel and/or pressure tubes considered at the TCM are given in Table 1. Table 2 presents fuel rod and cladding operational parameters and some post-irradiation examination data (fission gas release - FGR), and Table 3 - the information on material and operational conditions of the pressure tubes.

As can be seen from Table 1, CANDU, FUGEN and RBMK reactors have pressure tubes which withstand coolant pressure and serve as a pressure vessel. In BWRs and PHWRs, e.g. Atucha-1 (Argentina) and PHWR-500 (Indian design which is under development), the channels serve to distribute and adjust the coolant flow and withstand only the pressure difference between inlet and outlet. Coolant pressure is very close for all reactors considered, nevertheless coolant boiling is supressed in PHWRs. With regard to fuel utilization, it is very similar in FUGEN and RBMKs (burnups ~ 20 MWd/kg HM, enrichments ~ 2-2.5% fissile) and in CANDU and PHWRs (6-8 MWd/kg HM and natural U), while in BWRs both burnup and enrichment are significantly higher. However, there is a strong tendency for fuel utilization in all above-mentioned reactors, driven by economics, to increase fuel burnup.

Fuel operate at CANDUs, FUGEN, PHWRs and RBMKs at rather high linear heat generation rates (LHGRs, Table2) that results in rather high fission gas release (FGR). Lower LHGR in combination with longer fuel residence time in BWRs could also result in high FGR. Power transients during fuel assembly reload (in CANDUs, PHWRs and RBMKs) and power transients during control rod manoeuvring (in BWRs and FUGEN) could result in high FGR and pellet-cladding interaction (PCI). Cladding water-side corrosion, especially nodular corrosion, is a serious issue for reactors with boiling coolant.

Only one Zr-based alloy, namely Zr-2.5%Nb, was selected in Canada, Japan and Russia as material for the pressure tubes (Table 3). These long (several meters) tubes (with wall thickness of about 4 mm and outer diameter of about 100 mm) pass through the center of the reactor core and therefore are subjected to neutron irradiation, high temperatures and stresses, hydrogen pick-up and corrosion. The challenge is to operate these tubes during the whole service lifetime (app 30-40 years). These tubes also have joints of Zr-2.5%Nb alloy with stainless steel extensions, either of weld (RBMK)or mechanical-rolled (CANDU and FUGEN) type. It also causes additional problems to guarantee their integrity. Pressure tubes should have minimal changes in geaometry in order to avoid contact either with calandria tube (CANDU and FUGEN) or graphite stack (RBMK).

Thus, the list of priorities for the information exchange at the TCM was defined. Papers on fuel for BWRs, CANDUs, FUGEN, PHWRs and RBMKs and pressure tubes for CANDUs, FUGEN and

Table 1. Major features of reactors which fuel and/or pressure tubes discussed at the TCM

Reactor	Major features	Reload type	Coolant condition, pressure	Fuel	Burnup, batch average, MWd/kg HM
BWR	LWR, FAs in coolant channels (support only the pressure difference between inlet and outlet)	During outage (every 12-18 months)	Boiling (~10-13% of steam at outlet), 7 MPa	UO_2 or MOX (3 0-4 0 %)	32-42
CANDU	HWR, FAs in pressure tubes	On- power	Boiling (~4% of steam at outlet), 10 0 MPa	Nat UO_2 (SEU or MOX are under consideration)	7 2-8 3
FUGEN	HW moderated, LW cooled, FAs in pressure tubes	During outage (every 6 months)	Boiling, 7 0-7 2 MPa	MOX (up to 2 % of fissile Pu)	20
PHWR (Atucha-1, PHWR-500)	HW moderated and cooled, FAs in coolant channels (support only the pressure difference between inlet and outlet)	On-power	Not boiling, 11 3 MPa	Nat UO_2 (SEU or MOX are under consideration)	~6
RBMK	LW cooled, graphite moderated, FAs in pressure tubes	On-power	Boiling (up to ~20% of steam at outlet), 7 0 MPa	UO_2 (2 0-2 6 %)	15-22

Table 2. Rod and cladding operational parameters

Reactor	Clad material	Clad operational mode	Max. LHGR, W/cm	Max. FGR, %	Fuel residence time
Atucha-1	Zry-4	free standing, pressurized, 1 7 MPa He	600 (design value for transients)	not applicable	195 EFPDs
BWR	Zry-2	free standing, pressurized with He	400-420	10-15	3-4 years
CANDU	Zry-4	collapsible, not pressurized	650 (design limit)	~10	average 320 EFPDs
FUGEN	Zry-2	free standing, pressurized, 0 1 MPa He	545	~10	app 3 years
RBMK	Zr-1%Nb	free standing, pressurized, 0 5 MPa He	485 (RBMK-1500)	~7	1100-1200 EFPDs

RBMKs were requested with emphasis on performance/operational experience, current design and trends, and R&D on on fuel and pressure tube materials. All areas were covered for the exception of BWR fuel performance and design Areas covered at the TCM are presented in Table 4.

Table 3 Pressure tubes - materials and operational conditions

Reactor	Material	Finished ready condition	Configuration and moderator	Outlet temp., °C	Hoop stress, MPa	Fast neutron flux, 10^{17}n/cm^{2}s
CANDU	Zr-2.5%Nb	Cold worked	Horizontal, D_2O	308	135	3.7
FUGEN	Zr-2.5%Nb	Solution heat treated	Vertical, D_2O	280-286	96	2.7
RBMK	Zr-2.5%Nb	Annealed	Vertical, Graphite	288	86	1.7

Table 4 Areas covered by the TCM participants

Reactor type Area	BWR FUEL	CANDU		FUGEN		PHWR	RBMK	
		FUEL	PR. TUBE	FUEL	PR. TUBE	FUEL	FUEL	PR. TUBE
Performance / Operational Experience	-	Manzer	-	*	-	Casario, Marken-deya	Aden, Afanasyev, Voroncov, Bek	Semenov Afanasyev, Petkevicius,
Design / Trends	-	Manzer	Calinescu	*	Takayama	Casario, Marken-deya	Aden, Voroncov, Bek, Koupalov, Troyanov, Bibilashvili	Semenov
Materials / R&D	Grounes McGrath	-	Calinescu	*	Takayama	Marken-deya	Aden, Afanasyev, Bek, Koupalov, Troyanov, Bibilashvili	Semenov Afanasyev, Petkevicius,

* Note. Mr. Takayama on behalf of Mr. Uematsu brought and distributed his paper entitled "Experience and Burnup Extension of MOX Fuel in ATR".

SESSION I - WATER CHANNEL REACTOR FUEL DESIGN AND PERFORMANCE

Most utilities that operate water channel reactors participate in fuel development and qualification programmes. Changes to the original fuel designs are being considered to meet new or changing requirements associated with:

- Improvements in fuel economics and performance (Atucha 1 & 2 in Argentina, RBMKs, BWRs)
- Widening of operating & safety margins (RBMKs, CANDUs, BWRs)
- Stockpile reduction of spent LWR fuel (FUGEN in Japan, CANDUs in Korea)
- Utilization of local natural resources of thorium (CANDUs in India).

All vendors report acceptable fuel element defect rates that generally range from 10^{-5} to 10^{-4} in recent years. Some of the fuel design features or solutions that have alleviated or eliminated defects that are due to known failure mechanisms are summarized below.

PCI (or SCC) Failures

Graphite coated cladding (CANLUB) introduced in the 1970s has increased the performance limits for CANDU fuel. Graphite coatings are being considered for slightly enriched uranium fuel in Atucha-1. Barrier clad fuel introduced in the 1970s has, along with a reduction in rod diameters have reduced PCI failures in BWRs. Different types of barrier clad fuel are still under development for BWRs. For RBMKs, fuel and cladding materials have been improved, implementation of annular pellets etc, but subsequent effect on PCI behaviour has not yet been fully investigated. The best remedy to avoid PCI in RBMKs is to carefully control the local power variations in the core.

Primary Hydride Sheath Failures

Strict adherence to design limits for the amount of excess hydrogen gas within fuel elements has eliminated the number of primary hydride failures observed for CANDUs. Pellet density has increased to about 10 6 g/cm^3 to reduce moisture content in RBMK rods.

Debris Fretting Failures

Most stations use filters in the primary circuit to avoid debris from entering the fuel. In BWRs, filters are available and can be installed on the bottom end fittings of the fuel assemblies.

Hydride Cracking at Endcap (or Endplug) Weld

Localized stress concentrations near the welds have occasionally caused circumferential cracking in CANDU and RBMK fuel cladding. Solutions include the use of end pellets that have non-standard dimensions (CANDU) or reduced enrichment (RBMK-1500) Also for RBMKs, a special endplug has been designed to resolve the cracking problems.

Water-Side Corrosion

RBMK fuel has experienced excessive corrosion in the past. The accumulated experience showed that the extent of corrosion can be reduced by 1) improving the quality of current cladding material and fabrication procedures already done, 2) replacing SS grid spacers with zirconium grid spacers in progress, and 3) developing new corrosion resistance cladding and spacer material (in progress). Specifications for maintaining coolant chemistry have been optimized to reduce fuel clad corrosion.

Waterside corrosion of BWR fuel cladding and structural material (Zry-2 and -4) could be reduced by introducing the annealing parameters to control the annealing processes after β-quenching CANDU fuel has not experienced corrosion problems.

In addition to the design changes that address the performance issues, other changes have been introduced to improve the safety margins, particularly for RBMK fuel. For example, fuel pellets are doped with erbium to improve reactivity characteristics, to increase enrichment and burnup.

Some operating characteristics of RBMK fuel are not completely understood. The thermal, corrosion and mechanical processes in the area of the contact between cladding and spacer grid need to be further investigated.

Future Work

Further work is needed to investigate:

1. Potential fuel features that could reduce the rate of secondary damage, as experienced in BWRs.
2. Ways to reduce the consumption of nuclear material for new fuel designs, i.e. natural uranium requirements per unit electrical energy.
3. Characteristics of secondary hydride damage to better understand post-defect deterioration behaviour.
4. The RBMK fuel behaviour under different RIA conditions and to develop the acceptance criteria for the various fuel designs.
5. Further improvements of calculation codes for predicting fuel behaviour under different operating conditions (e.g. higher burnup, load following, etc.)
6. Clarification of licensing criteria for fuel operating at higher powers and burnups.

SESSION II - R&D ACTIVITIES ON WATER CHANNEL REACTOR FUELS

Much experimental work is pursued on the properties and behaviour of different types of fuel rods. Two of the papers described the current plans for R and D work on BWR fuel within the Halden Project and Studsvik's international and bilateral projects. Comments on the relevance of these types of investigations for Atucha, FUGEN and RBMK type fuel have been added in the table 5.

SESSION III: R&D ACTIVITIES ON FUEL CHANNELS

Fuel channels (FCs) of Atucha I and Atucha II RBMK, CANDU and FUGEN are important parts of reactor as a whole. Reliable operation and safety of these reactors significantly depend on condition of FCs. Thus, it is important to control FC condition, and to predict its changes in the real time scale for their timely replacement..

Current Status

Experience of FC operation in Atucha I and Atucha II RBMK, CANDU, FUGEN reactors has shown up that the methodology of evaluation. of FC service lifetime and the methods of its increase are very similar. Main factors defining these characteristics are as following:

- decrease of resistance to the brittle fracture by increasing hydrogen content in FCs;
- formation of local corrosion nodules;
- change of geometrical parameters of FC under the impact of neutron flux, temperature and pressure;
- formation of defects in connection of Zr and SS tubes.

However, it is necessary to emphasize that at the present time there is no exact method of evaluation for the residual lifetime of FC.

Research and Development

Investigation of materials (test samples - Japan, irradiated FCs, including defective ones - Argentina, Canada, Lithuania, Romania, Russia, Ukraine)are being carried out in countries with channel type reactors. Nowadays, the "leak-before-break" concept is used in the analysis of RBMK FC reliability. It states that the complete failure will not occur after part-wall flaw growth in FC and turning in through-wall flow in process of operation, if through-wall length does not exceed critical value.

Processes of defect propagation in Zr alloys under hydriding conditions are studied as well as mechanism of hydrogen-induced delayed cracking (HIDC)

Work on investigation of the RBMK Zr-2.5% Nb tubes corrosion in the zones under spacer grids is carried out.

Table 5. Current plans for fuel R&D

SUBJECT	BWR		ATUCHA	FUGEN	RBMK
	HALDEN	STUDSVIK	ARGENT	JAPAN	RUSSIA
Separate Effects					
Thermal Conductivity Degradation	X_C	O	O	X_C, X_{NC}	CC
Fission Gas Release	X_C	O	X_{NC}	X_C, X_{NC}	X_C
PCI Studies (Ramp Tests, etc.)					
Comparison of UO_2 Fuel and MOX Fuel	X_{NC}	O	X_T	X_C, X_{NC}	O
Ramp Tests - Normal Operations	X_C	X_C, X_{NC}	X_C	O	X_C
Ramp Tests - Off-Normal Operation	O	X_{NC} X_T	O	X_C, X_{NC}	X_C
Support of RIA Studies (Design Based Accidents)	X_{NC}	O	X_C	O	X_C
Ramp Tests - Some Accident Conditions	O	X_{NC}, X_T	O	X_{CC}, X_C, X_{NC}	X_C
Effect of Non-Penetrating Cracks	O	X_C	O	O	X_{NC}
Effect of Post-DO/ DNB Operation	O	X_{NC}	O	O	CC
Fuel with Integrated Absorbers					
Thermal Conductivity Measurements	X_C	O	O	X_{NC}	X_C
Ramp Tests	O	X_C	O	X_{NC}	X_{NC}
Pellet Corosion in Coolant	O	O	O	O	X_C
Fission Gas Release	X_{NC}	X_C	O	X_{NC}	X_C
In-Pile Corrosion					
Clad Corrosion	X_C	X_{NC}	X_C	X_{CC} X_C, X_{NC}	X_C
Water Chemistry Studies	X_C	X_{NC}	X_C	O	X_C
Dry-Out Effects	X_C	O	O	O	CC
Lift-Off Studies	X_{NC}	X_C	O	O	X_C
Defect Fuel Degradation	X_{NC}	X_C, X_{NC}	X_{NC}	O	X_C
Stored Energy, Enthapy (for LOCA analysis)	X_T	X_C, X_{NC}	O	O	X_{NC}
Clad Creep Properties	X_C	X_{NC}	X_T	O	X_C
FR and FA Design Modernization	O	O	X_C	X_C	X_C

O = No activities
X = Activity
CC = Activities completed
C = Current
NC = Near Future
T = Trend

An estimate of the lifetime of RBMK-1500 FCs has been carried out in the team work of a project on safety analysis at the Ignalina NPP (SAR/ NPP). During this investigation results obtained by the Leningrad NPP FCs operation were used. The hydrogen content in replaced irradiated Zr-2.5% Nb tubes (after operation up to 18.5 years) has been systematically studied. As a result of the above-mentioned studies more severe requirements have been imposed on the fabrication process and methods of quality control.

Conclusions and Future Work

1. To improve the operational safety of RBMKs it is of special importance to develop a methodology for "step by step" replacing of FCs.
2. Important directions of the improvement of safety operation of all types of channel reactors are as the following:

- improvement of the methods of earlier defect detection, especially with regard to the defects which size does not exceed the critical one,
- development of methods and criteria to evaluate the residual FC lifetime after different periods of reactor operation,
- creation of data base on fabrication, operating parameters, results of post-irradiation investigations,
- investigation of fuel performance in pressure tubes that have changed dimensions due to irradiation.

PANEL DISCUSSION

Chaimen: **H.Weidinger** (Germany)
Yu.K. Bibilashvili (Russian Federation)

To assess the present status and future perspectives the following aspects have been suggested for discussion:

-RELIABILITY (Fuel and FCs):
Meeting performance targets, e.g. fuel operating availability,
Corrosion, hydriding, weld integrity
Fuel failures: types, impact on operation, frequency.

-EFFICIENCY:
Fuel utilization,
Burnup,
Operational flexibility.

-SAFETY MARGINS (Fuel and FCs)

-SPENT FUEL MANAGEMENT

-SPECIFIC ASPECTS, e.g. Pu burning, using domestic resources.

After the discussion moderated along the above-mentioned lines the meeting participants came to the following conclusions:

RELIABILITY

The overall fuel failure statistics is comparable for all considered fuel types in the range of 10^{-5} - 10^{-4}. Efforts are under way for all fuel types to undercut these values or at least to keep them in the lower range (10^{-5}).

The types of fuel failures are more important than average values of failure frequencies, e.g. their consequences on operation (activity releases and circuit contamination) or life time (pressure tubes). Some types of fuel and FC failures have been discussed in Sessions I and III respectively.

For CANDU and Atucha fuel:

Solutions have been found and proven by operating experience to avoid PCI, primary hydride defects and other fuel rod degradation by secondary hydriding.

For BWR fuel:

Primary hydriding is under secure control. PCI has been significantly decreased by barrier-type claddings. A serious problem is severe secondary degradation with long axial splits or transverse cracking. Satisfactory remedy is under development now. Fortunately, only single separated events have been observed. Presently several types of advanced barrier claddings are proposed by different vendors to avoid secondary degradation without loss of PCI-resistance.

For RBMK fuel:

PCI appears to be one of the generic failure causes. PCI-related fuel failure rate could be reduced by a combination of measures.

CORROSION

Corrosion can become service restricting for fuel rods, when nodular corrosion occurs to a lager extent, in particular:

BWR fuel - at present this effect is under good control by optimizing the properties of zircaloy.

RBMK fuel - this effect could be significantly reduced by replacing stainless steel (SS) spacer grids (SG) with Zr alloy SG, by improving the quality of the current Zr-1%Nb alloy and by implementing a new corrosion resistant Zr alloy (E-635).

Generally it should be stated that good control of the coolant chemistry is essential for reliable corrosion behaviour.

FC RELIABILITY

The reliability of the FCs has been shown to be very good for all relevant fuel types. Nevertheless, efforts are underway to achieve better mechanistic understanding and better life time control for FCs of all relevant fuel types.

EFFICIENCY

FUEL UTILIZATION - was not explicitly covered by the meeting.

BURNUP - is typically different for the different fuel types:

CANDU and Atucha	≈7 MWd/kg U,
FUGEN	≈16-20 MWd/kg HM,
BWR	≈35-45 MWd/kg HM,
RBMK	≈18-21 MWd/kg U

There is a continued tendency to increase burnup at all fuel types. In particular, for RBMK fuel strong efforts are successfully under way by combining increased enrichment with addition of erbium to the fuel as burnable absorber.

FUEL OPERATION FLEXIBILITY - was not explicitly covered, however there is an increased demand for more operational flexibility. So far broad and good experience exists with BWR fuel only.

SAFETY MARGINS

Continued efforts are made for all considered fuel types, e.g. improved thermal margins for BWR fuel.

In particular, strong and successful efforts have been made for RBMK fuel including:
- -measures to keep control on fast power density changes in the area of reload fuel,
- -new algorithms for power control, e.g. to reduce power ramp amplitudes at Ignalina NPP,
- -addition of erbium to the fuel as burnable absorber to improve the reactivity characteristics. In this case good agreement has been found between experimental and calculated results in Leningrad and Ignalina NPPs.

SPENT FUEL MANAGEMENT - - was not explicitly covered. However it was noticed that spent fuel management is a stringent issue, in particular for Atucha 1 and some BWRs and RBMKs. It is dealt with the shortage of onsite capacity rather than unsolved technical problems.

SPECIFIC ASPECTS

Burning of Pu became a general issue for all considered fuel types due to increasing volumes of Pu from thermal reactor operation and now also from ex-weapons Pu.

There is substantial experience to operate BWRs and FUGEN loaded with MOX fuel. Development and preparation is under way to use MOX fuel in CANDU cores.

India has decided to develop the thorium/plutonium fuel cycle for its CANDU NPPs in order to use domestic thorium resources.

FUTURE WORK

1. In general more emphasis should be given to improved fuel utilization and operational flexibility. Good experience already exists for BWR fuel.

2. Further R&D efforts are needed to obtain a better mechanistic understanding of:

   -severe (secondary) fuel rod degradation as observed with BWR fuel,
   -correlation between water chemistry and corrosion of fuel rod cladding and structural materials,
   -correlation between accelerated (nodular) corrosion and bimetallic contact, e.g. SS spacer grid and Zr alloy cladding,
   -service life limiting processes, e.g. crack propagation in FCs,
   -more realistic data base for RIA safety analysis at high burnups, e.g. pellet rim effect.

3. Further development efforts are also needed to:

   -improve methods for early defect detection at dimensions below critical size (RBMK FCs) For CANDU FCs this system has already been developed,
   -develop methods and criteria for the quantative evaluation of the service life time of core components, first of all FCs after various periods of reactor operation,
   -further improve calculation codes predicting fuel behaviour and clarify license criteria for fuel operated at high power and burnup.

4. Generally there is a need for a data base on:

   -fabrication characteristics,
   -operating parameters versus operating behaviour characteristics,
   -PIE results versus operating behaviour of the considered fuel types, in particular for RBMK fuel.

5. Specific R&D efforts should be supported that are necessary in order to get a safe data base for:

   -further increased burnup,
   -increased operational flexibility,
   -improved fuel utilization,
   -improved thermal efficiency of all considered fuel types (see list of planned R&D activities in the summary of Session II).

6. Methods and techniques should be evaluated and applied for more efficient exchange of knowledge, experience and opinions, in particular during international meetings, workshops, seminars etc.
   Modern technology like computer based presentation are basically available. However they have to be adjusted and then practically applied.

7. It has been recommended discuss periodically (on 2-3 years basis) on the international level issues related to water channel reactor fuel and pressure/coolant tubes.

WATER CHANNEL REACTOR FUEL DESIGN AND PERFORMANCE

(Session I)

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CANDU FUEL PERFORMANCE

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Abstract

The paper presents a review of CANDU fuel performance including a 28-element bundle for Pickering reactors, a 37-element bundle for the Bruce and Darlington reactors, and a 37-element bundle for the CANDU-6 reactors. Special emphasis is given to the analysis of fuel defect formation and propagation and definition of fuel element operating thresholds for normal operation and accident conditions.

1 OVERVIEW

One of the recognized strengths of the CANDU (CANada Deuterium Uranium) nuclear reactor is the excellent performance of its fuel. Of the more than 1 300 000 fuel bundles that have been irradiated to 1996 in Canada, less than 0.1% have developed defects[(1)]. Since most defective bundles have single-element failures, the cumulative fuel element defect rate is approaching 10^{-5}. The defect causes tend to fall into three categories: manufacturing flaws, operational and system design related fuel defects. In recent years, there have been very few system design related defects.

The low defect rate of CANDU fuel is attributed to the fuel element and bundle designs that meet all requirements, as demonstrated by qualification programs that include extensive out-reactor tests and in-reactor irradiations. Equally important factors are the specialized manufacturing processes and systems that have been developed to produce fuel within the design specifications and under good quality assurance.

When a fuel element becomes defective, it does not impose an immediate risk to station operation because

1. The systems and/or techniques for detecting and locating defective fuel do not normally require reactor shutdowns. At most stations, operators find the defects while they are small and usually before they contaminate the primary circuit.

2. The fuel handling systems have the capability for removing defective fuel using normal operating procedures at full power.

On the other hand, if there are several defective elements in the core resulting in high releases, there is a risk that regulatory limits will be exceeded, requiring a reactor shutdown.

There are 22 CANDU power reactors in Canada:

- 20 reactors in 5 four-unit stations (Pickering A and B, Bruce A and B, and Darlington, which are owned and operated by Ontario Hydro); and

- 2 reactors in single unit CANDU 6 stations (Point Lepreau and Gentilly 2, which are owned and operated by New Brunswick Power and Hydro Quebec, respectively).

Besides the power reactors, Canada had three prototype CANDU reactors that began operations in the 1960s and 1970s. These included the Nuclear Power Demonstration (NPD), Douglas Point, and

Gentilly 1 prototype reactors. All three have now been decommissioned. Except for the Gentilly 1 reactor, all CANDU reactor cores contain horizontal fuel channels, which are fuelled on a once-through natural uranium fuel cycle that requires heavy water moderation and cooling. The Gentilly 1 reactor used natural uranium fuel, contained vertical fuel channels, and had heavy water moderation and boiling light water coolant.

CANDU fuel bundles are small, simple assemblies that do not require structural components such as grid spacers, support rods or endfittings to contain the fuel rods or elements. The fuel elements and the thin endplates serve as structural components. The bundles are loaded and removed from fuel channels by fuel handling systems that operate at the full range of reactor power, pressure and temperature.

The fuel handling system has two fuelling machines that operate as a pair while refuelling a fuel channel. One machine loads new bundles into one end of a channel while the other unloads irradiated bundles from the other end. Each CANDU reactor has one of two types of fuelling machines, depending on the fuelling direction relative to the coolant flow direction:

- The "fuel-against-flow" machines load new bundles into the downstream end of the channel. Each channel contains latches to support the fuel against the drag force generated by the coolant, and the fuelling machine uses fuel carriers to transport bundles to and from the channel (Bruce and Darlington reactors); and

- The "fuel-with-flow" machines load new bundles into the upstream end of the channel. Each channel uses shield plugs to support the fuel against the coolant drag, and the fuelling machine uses sidestop separators to separate bundles from the fuel column during removal from the channel (Pickering and CANDU 6 reactors).

The Pickering and CANDU 6 designs were derived from Douglas Point prototype reactor that also permitted fuelling in the direction of coolant flow. The Bruce and Darlington designs were derived from NPD prototype reactor that permitted fuelling against the direction of coolant flow.

The pressure tubes used in fuel channels of all CANDU power reactors in Canada have a common inside diameter of 100 mm. This sets the diameter of all three fuel bundle designs built today in Canada. The current fuel designs include

- a 28-element bundle for the Pickering reactors,
- a 37-element bundle for the Bruce and Darlington reactors, and
- a 37-element bundle for the CANDU 6 reactors.

The fuel element subdivision was selected on the basis of the specific power requirements. The 37-element bundle has 13 mm diameter fuel elements that are smaller than those in the 28-element bundle. The Pickering bundle has 15 mm diameter elements that are the same size as those in the 19-element bundles previously used in the NPD and Douglas Point reactors. Since both sizes of fuel elements are qualified to operate at the same high linear powers of about 60 kW/m, the 37-element bundle delivers about 30% more power than that delivered by the 28-element bundle.

The two 37-element bundle designs are very similar, with the exceptions of the small differences in the endcap profiles and bearing pad positions along the fuel elements, as shown in Figures 1 and 2. These differences are needed to ensure compatibility with the different fuel handling systems and fuel channel configurations.

2 KEY DESIGN FEATURES

With a natural uranium fuel cycle, the excess reactivity of the core is quite small, and it is easily controlled with on-power fuelling. The average fuel bundle discharge burnup varies from about 7200 to 8300 MWd/MgU, depending on the reactor design, fuel type, heavy water isotopic purity, and fuel management strategies. The corresponding natural uranium consumption for CANDU 6 reactors is about 170 MgU_n per GWy electrical.

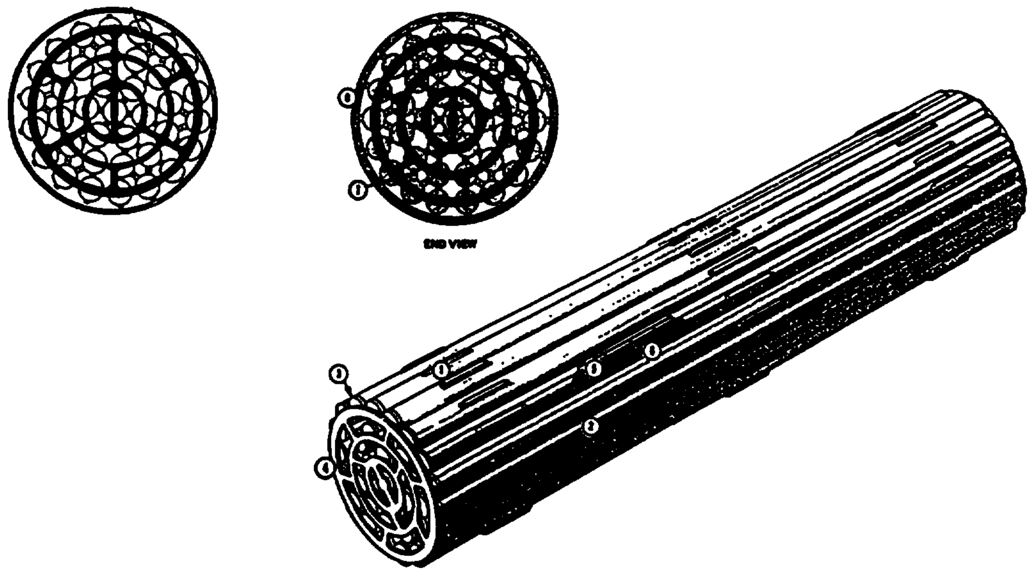


Figure 1: 37-Element CANDU 6 Fuel Bundle

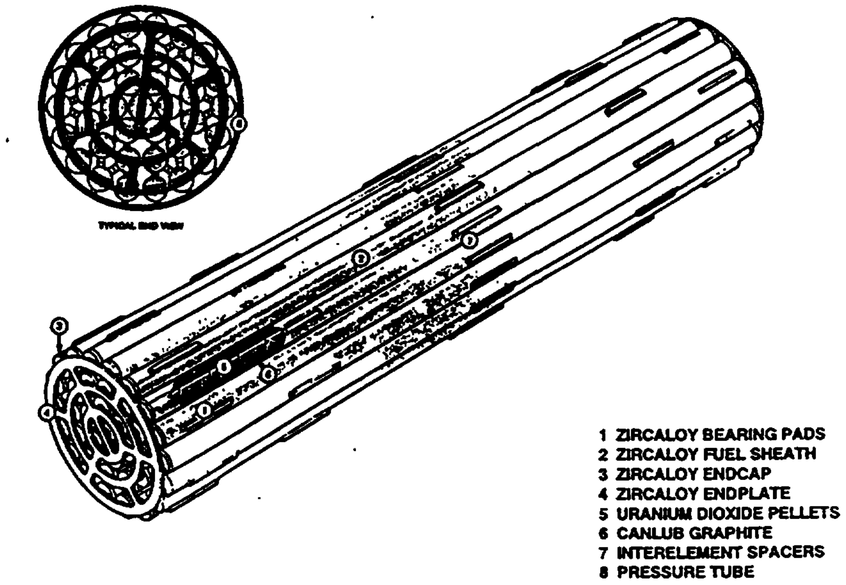


Figure 2: 37-Element Bruce/Darlington Fuel Bundle

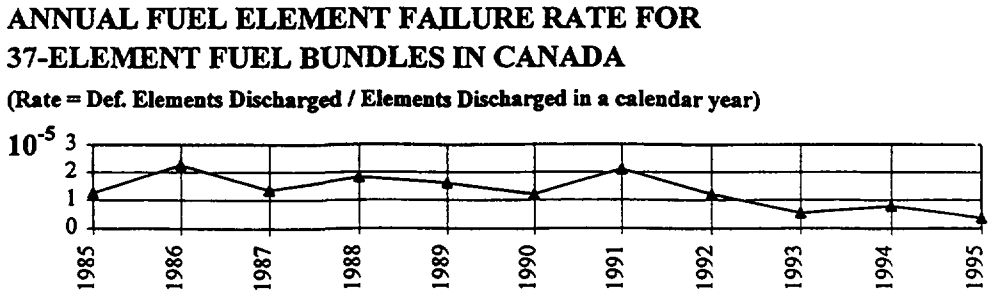


Figure 3: **Fuel Element Defect Rate for 14 Large CANDUs Operating in Canada (above 600MWe)**

The capability to replace fuel on-power has led to a unique fuel design that is quite different from those reactors that use enriched uranium and require off-power refuelling. The CANDU fuel design consists of 0.5 m long fuel elements assembled into a standard bundle configuration. The fuel bundles are inserted into specific reactor fuel channels as required by the fuel management system to control the excess core reactivity. Because of the low fuelling costs of less than about 1.5 mils/kWh, fuel bundle reconstitution is not economically attractive to replace defective elements, and is not practised by CANDU utilities.

2.1 FUEL ELEMENT COMPONENTS

The CANDU fuel element is comprised of UO_2 ceramic pellets, Zircaloy-4 cladding, graphite coatings on the inner sheath surface, and Zircaloy-4 endcaps.

The geometric parameters and material properties of the fuel pellet are chosen and controlled to

a. Maximize the amount of fissile material present in the fuel element,
b. Minimize the pellet volumetric changes during fuel in-reactor life,
c. Ensure that fission gas release is within acceptable limits,
d. Ensure that the pellet design meets the requirements imposed by production capability and economy, and
e. Minimize circumferential ridging of the sheath.

The UO_2 properties that bear the strongest influence on the pellet thermal behaviour are density and oxygen-to-uranium ratio. These characteristics determine the thermal conductivity of the oxide and are maintained within the specified ranges to ensure acceptable UO_2 temperatures and, hence, fission gas release.

Pellet ends are designed with spherical indentations, or dishes, to accommodate thermal volumetric expansion of the plastic core of the pellet and to accommodate fission product gases. The pellet ends are chamfered on the corners of the flat pellet surfaces to minimize pellet chipping during loading and during subsequent element handling. Chamfering also reduces sheath strain at pellet interfaces.

Zircaloy-4 is used in fuel sheath production because of its low neutron absorption. It also has good corrosion resistance and low hydrogen or deuterium pickup performance under CANDU coolant conditions. Material properties and heat treatments are specified so that the material will retain acceptable ductility at high irradiation levels.

Fuel element sheath diameter is locally reduced in the region of the sheath-to-endcap weld. To eliminate the possibility of sheath damage in this region from pellet-sheath interaction, a non-standard pellet is placed at the each end of the fuel stack.

As in all CANDU fuel designs, the sheathing is designed to collapse into contact with the UO_2 at reactor coolant conditions. The thin sheath provides fission product containment while ensuring minimum neutron absorption and resistance to heat transfer. The as-fabricated diametrical clearance between the UO_2 pellet stack and the sheath is chosen and controlled to be in the appropriate range to

a. Prevent the formation of longitudinal ridges in the sheath,
b. Facilitate pellet loading during fuel element manufacturing, and
c. Accommodate some of the pellet diametrical expansion and minimize sheath strain.

Before pellet loading, a thin layer of graphite (CANLUB) is applied to the inner surface of the fuel sheaths to reduce pellet-sheath interaction. The void within the fuel elements is filled (unpressurized) with a He/air or He/inert gas mixture prior to endcap welding. The presence of helium within the fuel element allows for leak detection during fabrication and provides some improvement in the pellet-to-sheath heat transfer.

Fuel element closure is provided by two endcaps that are resistance-welded to the ends of the sheath. The endcap material is specified and inspected to ensure adequate strength and lack of porosity, which is needed for fission product containment.

2.2 FUEL BUNDLE COMPONENTS

The fuel bundle components are made of Zircaloy-4 and include endplates, endcaps, interelement spacers, and bearing pads.

There are two endplate designs available for 37-element bundles and one for the 28-element bundle. All designs have been fully qualified for use in CANDU reactors. The endplates hold the fuel elements together in a bundle configuration. They have to be strong enough to maintain the bundle configuration and to allow axial loads to be distributed among many elements rather than being concentrated on a few. Simultaneously, they should be flexible enough to allow differential axial expansion among the elements and to permit bending and skewing of the bundle. The endplates should also be thin to minimize the quantity of neutron absorbing material and to minimize axial separation between the fuel pellets in adjacent bundles.

While the endplates maintain separation of the elements at the bundle extremities, interelement spacers maintain separation at the bundle midplane. The spacers are rectangular with an aspect ratio of about 3.5. They are mounted with their major axis slightly angled (skewed) with respect to the element axis, such that the spacers on any two adjacent elements are skewed in the opposite direction. This skewing increases the width of possible contact between spacer pairs and decreases the probability of spacer interlocking.

The bearing pads, brazed to the outer element sheaths near the element ends and at the midplane, support the bundle inside the fuel channel and fuel handling systems. They protect the fuel sheaths from any mechanical contact throughout the fuel bundle lifetime. The pads must be profiled to minimize pressure tube surface damage during the in-reactor residence time of a fuel bundle and during refuelling operations. The pads must also be designed to minimize local corrosion of the pressure tube.

Two basic endcap designs are available for CANDU bundles. Each has an external profile that is designed to interface with the fuel channel and fuel handling system components.

3 FUEL DEFECT EXPERIENCE

Most of the fuel defects found in CANDU reactors have occurred in small batches in situations termed as "defect excursions". CANDU reactors have had 12 defect excursions, as shown in Table 1;

Table 1: Fuel Defect Excursions* That Have Occurred In CANDU Reactors

YEAR	CANDU	FUEL DEFECT TYPE	Ref.
1969-72	Douglas Point	Stress Corrosion Cracking	7
1972-72	Pickering A Units1 and 2	Stress Corrosion Cracking	7
1976-77	Bruce A Unit 2	Incomplete welds, debris fretting	NA
1981	Douglas Point	Incomplete welds	NA
1984	Bruce A Unit 3	Endcap weld cracking	3, 4
1985	Pickering B Unit 5	Debris fretting due to strainer breakup	6
1984-85	Wolsong 1	Debris fretting	NA
1985	Embalse	Incomplete welds	9
1988	Pickering A Unit 1	Stress Corrosion Cracking	5
1990-92	Darlington 2	Endplate fatigue	8
1991-92	Point Lepreau	Excess Hydrogen	2, 4
1991	Wolsong 1	Debris fretting	NA

* A fuel defect excursion refers to a period of operation when many fuel elements fail because of a common cause.

most of these are described in References 2 to 10. Nine of the excursions occurred in Canada. Figure 3 shows the historical defect rate for 37-element fuel irradiated in Canada over a 10 year period. Although the fuel defect trend for the 28-element fuel is similar, detailed statistics are not available.

With the exceptions of the early power ramp fuel defects in the early 1970s and the mechanical fuel failures at the Darlington NGS in the 1990s, there have been no system design related defects. Most defects from all CANDU reactors are attributed to manufacturing and operations.

The manufacturing flaws that have caused fuel defects among CANDU power reactor fuel include

- incomplete endcap welds,
- porous endcap barstock,
- insufficient volume,
- excess hydrogen gas within the element, and
- fretting through the sheath from endcap weld flashings.

The causes of operational defects include:

- sheath fretting that is due to debris from the coolant,
- stress corrosion cracking (SCC) of the sheath that is due to power ramps associated with abnormal operations, and
- mechanical damage that is due to abnormal fuelling.

The causes of system design related defects include

- SCC of the sheath that is due to power ramps associated with normal operations, and
- bundle disassembly that is due to resonant vibration in acoustically active channels.

This section summarizes the characteristics of defective fuel as observed from previous defect excursions.

3.1 MANUFACTURING FLAWS

Of all defect causes that are due to manufacturing flaws, an incomplete endcap weld predominates over all the others, listed in the previous section Manufacturing flaws caused by fretting from weld flashing are extremely rare and tend to have the same characteristics as defects due to fretting by debris. Therefore, this section focuses on the first four of the five types of defects that are due to manufacturing flaws.

3.1.1 INCOMPLETE WELDS AND ENDCAP POROSITY

Flaws that are due to incomplete endcap welds or porosity in the endcaps have common characteristics and have similar performance in reactor The initial breach in the cladding begins as a small hole near the end of the fuel element. Because of the small size of the hole, the amounts of coolant entering the fuel element and of fission products released to the coolant are limited. With low fission product release, defects can remain undetected until the hole becomes larger. With coolant inside the element, a source of hydrogen (or deuterium) is available to attack the Zircaloy cladding and cause secondary hydriding (or deuteriding) damage. When this happens, local hydrided regions of the cladding become brittle and begin to crack open.

The incubation period for secondary damage depends on the initial size of the primary hole and the fuel temperature (or element linear power)[11] From previous experimental irradiations conducted at the Chalk River Laboratories, it has been shown that the defects with very small holes do not open up and release fission products in detectable quantities The release rate tends to increase after the fuel element achieves a burnup of about 40 MWh/kgU[11]. This burnup represents about one month of operation for a high powered element. Moreover, the small hole tends to remain stable at element linear powers below about 40 kW/m. Because of these power and burnup thresholds, defects associated with manufacturing flaws are not normally detected within the first few weeks after loading them into the core. Also, they tend to preferentially occur in high power regions of the core.

The deterioration rate of fuel elements that initially contain small holes is also influenced by other parameters, particularly

- element linear power (or fuel temperature), and
- inside sheath temperature.

Therefore, these types of fuel defects preferentially occur in these locations:

- the downstream half of the channel where the inside sheath temperature is the highest,
- the outer ring of fuel elements where the linear powers are the highest, and
- in high power channels near reactivity control devices.

The preferential distribution of defects was observed in 1981-1982 at the Douglas Point reactor during a defect excursion. At the time, the core contained many bundles with elements that had incomplete closure welds. The bundles were randomly distributed in the core, but the defects were preferentially located at high power positions towards the outlet end of the channel. (See Figure 4.)

In recent years, endcap porosity has been virtually eliminated by a change in the material specification that now requires 100% ultrasonic inspection of endcap barstock material.

3.1.2 EXCESS HYDROGEN GAS

In 1991-1992[(2)], at least 20 defective bundles, mostly from high power positions, were discharged from the Point Lepreau core. The investigation concluded that the cause was due to excess hydrogen gas, made available during irradiation. About 4 to 5 mg of excess hydrogen gas was measured inside new fuel elements built at the same time as the defective bundles. During this time, the CANLUB graphite coatings were insufficiently cured. This amount of hydrogen gas inside the fuel element exceeded the technical specification of 1 mg.

The defective elements failed at linear powers exceeding about 50 kW/m. At these high powers, a small hole is believed to have developed within a day or two after initial insertion of the bundles into the core. The ^{133}Xe activity concentrations in the coolant gave the first sign of defective fuel in the core. In some cases, noble gas escaped within a few days of the initial loading of the defective element. The ^{133}Xe concentration increased by about 3 MBq/kg per defective element to steady levels of about 150 to 250 MBq/kg when several defective elements were in the core. The release-to-birth rate ratio of ^{133}Xe was estimated to be about 20%[(2)]. The hole was large enough to allow the noble gases, particularly ^{133}Xe, to escape, but small enough to prevent coolant ingress and halogen release.

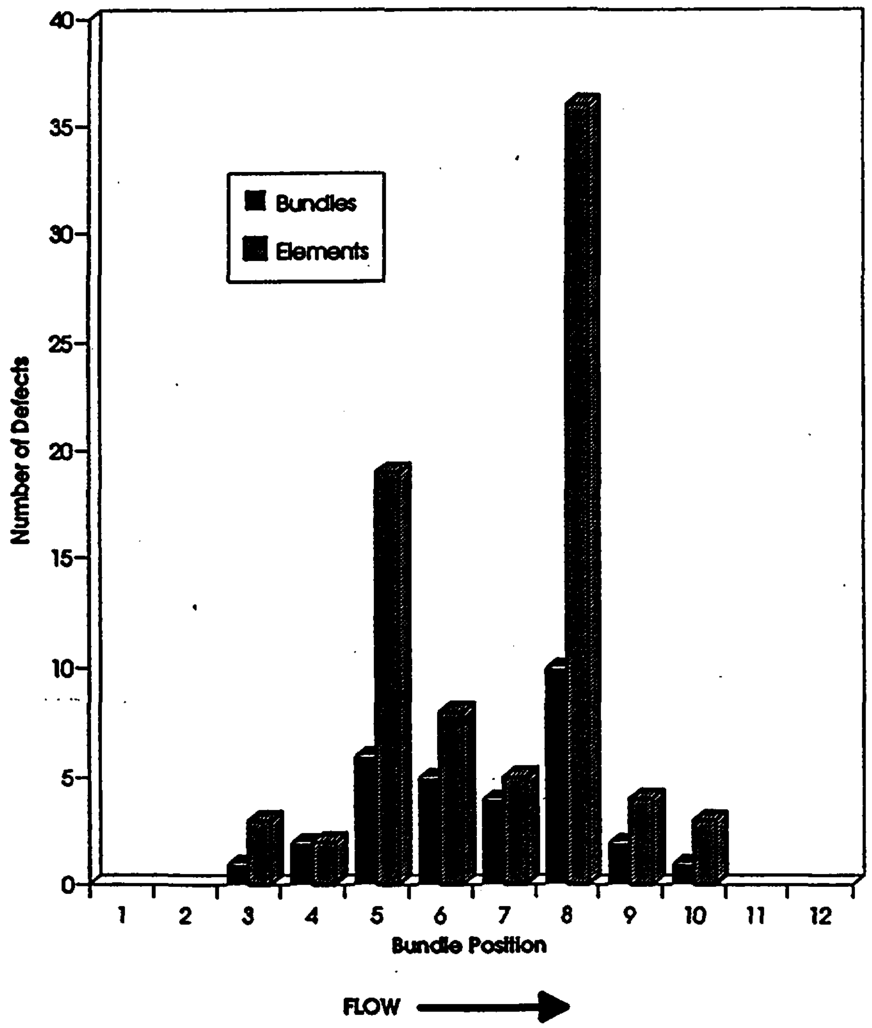


Figure 4: Axial Distribution Of Fuel Defects With Incomplete Welds At Douglas Point (1980-82)

Without halogen release, the defects could not be located with the failed fuel location system. The hole size was believed to have increased several weeks later, generally after the element achieved a burnup of about 50 MWh/kgU. When that happened, the iodines escaped in detectable quantities and the defects were found with the failed fuel location system.

Defective elements that are due to excessive hydrogen gas display the same characteristics as those that were discussed in the previous section for incomplete welds. The only difference is that the thresholds for secondary damage appear to be slightly higher than those that were observed for manufacturing flaws, that is, about 50 MWh/kgU and 50 kW/m, as shown in Figure 5. Because of the 50 kW/m threshold, defects are only expected among fuel elements located on the outer ring of the bundle. All other elements in the core will operate at linear powers below this threshold.

3.1.3 INSUFFICIENT VOLUME

In December 1983 and early 1984[3], a large defect excursion occurred at Bruce A Unit 3. Increasing levels of radioiodine in the coolant provided the first signs of fuel defects in the core. Within one week, the ^{131}I levels in the coolant rose by 10 times and remained high for several months at about 4 MBq/kg. The noble gas levels at Bruce are not normally reported. At least 43 defective bundles containing 140 defective fuel elements were located by the failed fuel location system. Dry sipping in the fuel transfer mechanism and visual inspections in the bay confirmed that the defects were discharged.

Most of the defective bundles (37 out of 43) had been made by the same manufacturer and had been irradiated in Unit 3 at Bruce A. They had been shifted from low power position #3 along the fuel channel to high power position #7 in the central region of the core. The defect appeared to be confined to a circumferential region in the endcap weld that is closest to the end bearing pad. In some cases, the cracking resulted in complete separation of the endcap from the element.

A review[4] of the manufacturing processes indicated that the affected fuel may not have conformed to the original AECL technical specifications that control the gap volumes within the elements. The suspected non-conformances were

1. The axial gap between the fuel stack and endcaps was below the minimum limit;
2. The diametral clearance between the pellets at the ends of the fuel stack and the sheath was below the minimum limit; and
3. The UO_2 density was above the maximum limit.

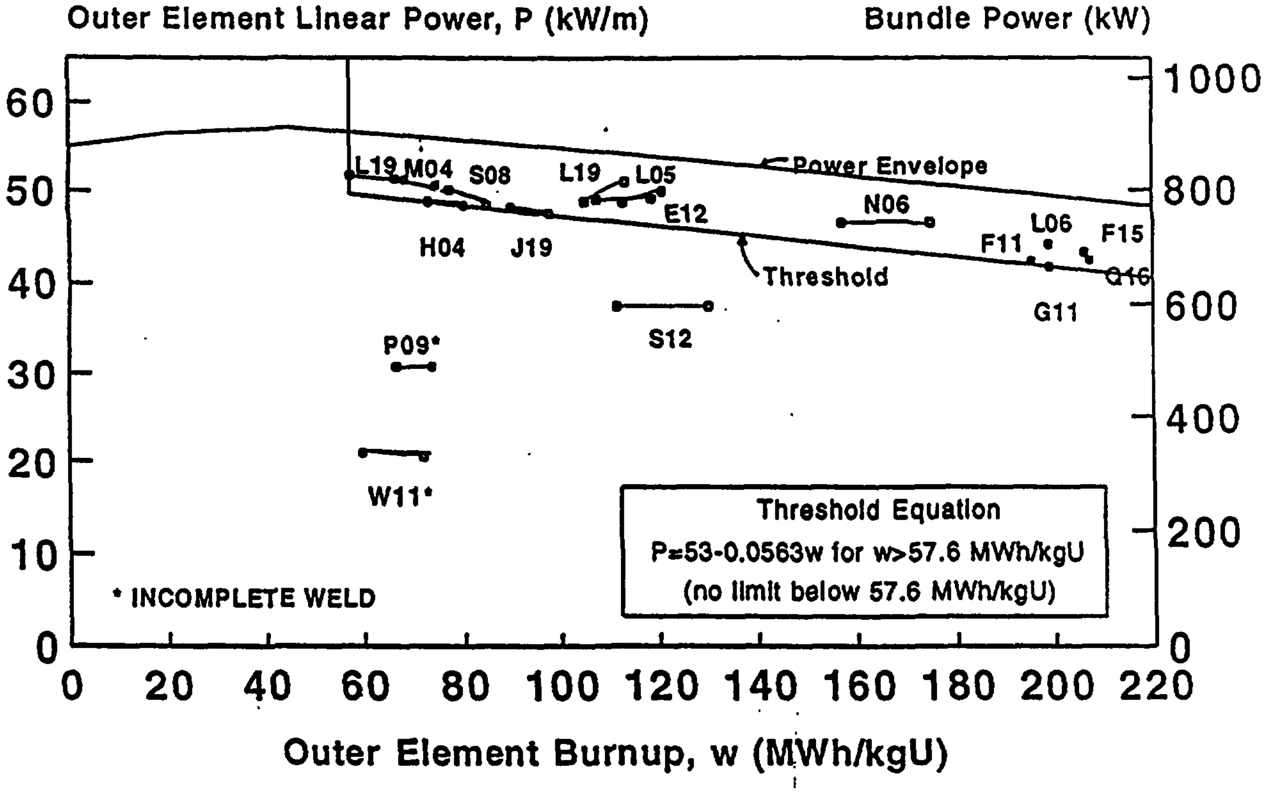
In addition, there is some evidence[4] that suggests that there was excess hydrogen gas within the fuel elements.

These non-conformances contributed to high stresses in the sheath, particularly at the ends of the elements. The excess hydrogen gas may have entered the sheath at the local regions of high stress where the diametral clearance was insufficient. The additional increase in the stresses during the refuelling power ramp combined with the presence of fission products in the gap likely led to SCC[4]. One explanation for the these failures having occurred in only one of the Bruce units may be due to slight differences in the power distributions among the Bruce units.

3.2 OPERATIONAL DEFECTS

3.2.1 STRESS CORROSION CRACKING DURING ABNORMAL OPERATION

One of the most common operational defects among CANDU fuel is SCC of the sheath because of power ramps. These types of failures have only occurred during defect excursions among the Pickering size fuel elements (15 mm diameter). High sheath stresses associated with a sudden increase in power combined with a high concentration of corrosive fission products within the pellet-to-sheath gap can lead to failure.



Note: Range indicates portion of bundle power history when DN signal began to increase.

Figure 5: Hydriding Cracking Threshold For Fuel Elements Having Excess Hydrogen Gas at Point Lepreau (1991-92)

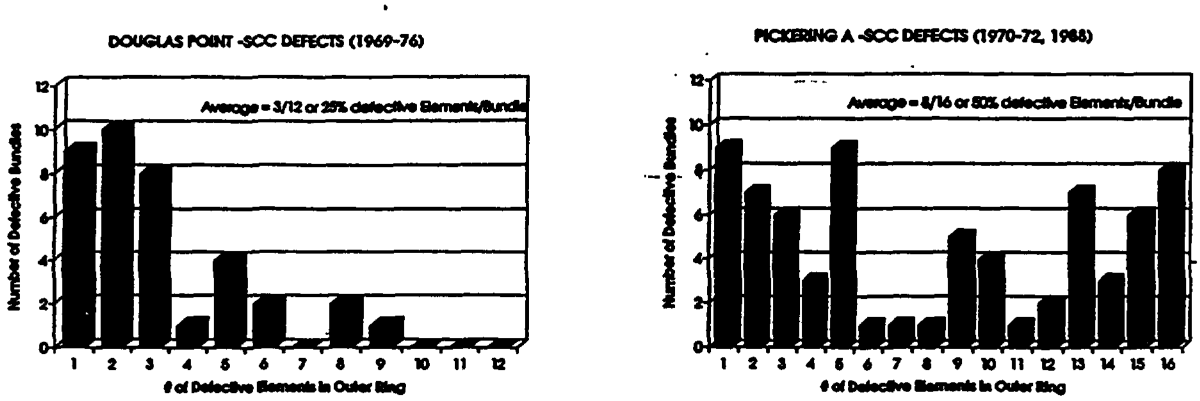


Figure 6: Stress Corrosion Cracking Fuel Failure Experience in CANDUs: Number of Defective Elements In The Outer Rings Of Bundles

In November 1988[5], a reactor trip occurred at Pickering A Unit 1. During trip recovery, all adjusters were withdrawn from the core and the reactor power was raised to 87% for 40 min. This was outside the range of normal operation; reactor power is normally limited to 65% in this situation. As a result of the transient, about 200 fuel bundles in 40 central channels sustained large power ramps. Following the transient, the radioiodine and noble gas levels in the coolant indicated the presence of many defective elements. Subsequently, fuel was discharged from these channels and inspected in the bays. Thirty-six defective bundles contained about 290 outer fuel element failures. Some of these experienced failures in all elements of the outer ring. No failures were observed among the non-outer elements that operate at significantly lower powers than the outer elements.

The defective fuel bundles caused by SCC usually have multiple fuel element failures in the outer ring which is the ring that experiences the highest linear powers. The defective elements usually display extensive damage such as: local swelling of the sheath because of UO_2 oxidation, uranium deposition downstream of defect sites, secondary deuteride damage at several sites, and irregular cracking patterns on the sheath. The primary defect site is not usually found during inspections in the bay nor in the hot cells. In previous controlled experimental irradiations, the rate of secondary degradation of the defective elements was rapid, generally within an hour of the power transient[11]. Although the initial degradation rate is high, it diminishes quickly with time as indicated by the condition of the defects. Defective bundles discharged shortly after the transient appeared to have the same extent of damage as did those that were discharged weeks later.

3.2.2 DEBRIS FRETTING

The other most common type of operational defect in CANDU reactors is sheath fretting by debris. Debris within the primary circuit can be circulated through the core by the coolant. Often new CANDU reactors experience several fuel defects because of debris being trapped within fuel bundles Depending on the size and location of the debris, the coolant velocity can cause it to vibrate and damage the fuel sheath. The main source of debris found within the primary circuit comes from the construction stage of new reactors. Other sources of debris can be made available to the primary circuit after startup when reactors are shut down for routine maintenance and inspection of primary circuit components.

For commissioning of Pickering B Unit 5, the core was loaded with first charge fuel and strainers were installed on the *pump discharge lines* to collect debris[6]. After the pumps were started, the strainers broke up and some of the pieces eventually became trapped in the fuel. The subsequent fretting led to fuel failures. To reduce the risk of having fretting defects during the initial startup, strainers are now normally installed in *specific channels* during commissioning to remove debris.

Unlike other defect types, fretting defects are not power related. Defective elements can appear anywhere in the bundle and anywhere along the fuel channel, depending on the size of the debris. If the fretted hole is small, the deterioration rate can be slow until the fuel element achieves a burnup threshold, similar to that identified for the fuel elements with manufacturing flaws[11].

Previous experience has shown that the defects caused by debris fretting are usually single element failures The defective elements may have: local areas of swelling because of UO_2 oxidation, uranium deposition downstream of the defect, and small amounts of secondary deuteride damage. The amount of secondary damage depends on the burnup or time duration in the core while defective. Primary defect sites are usually found during inpections in the bays: shiny surfaces indicate where the fretting has occurred.

The Wolsong 1 fuel defect experience in the mid 1980s appeared to be associated with debris fretting. The evidence seemed to suggest that the debris was small enough to circulate within the primary circuit and fall into inlet feeders located along the bottom of the headers near the pump discharge lines Channels with inlet feeders connected to the headers at these locations are more susceptible to debris fretting failures.

The primary circuits in CANDU reactors, which have a figure-of-eight configuration have two separate loop halves. This means that fuel defects tend to appear in the loop half that is downstream of the point where debris is introduced.

At the onset of a fuel defect excursion where many pieces of debris are caught among several bundles within the core, defects do not necessarily occur at the same time. Instead, they occur over a long time depending on the variability in fretting rates. This means that the fission product levels in the coolant would increase slowly with time.

3.2.3 MECHANICAL DAMAGE

The third operational defect is mechanical damage that is due to fuelling. The events when bundles have been mechanically damaged are extremely rare and have been attributed to human error or abnormal operations. Fuel bundles have been occasionally damaged during refuelling at the Bruce and Pickering reactors. Two examples are briefly described below:

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

2. Because of an interruption to refuelling operations at Pickering A, bundles resided for several days in the cross flow region near the liner holes of the inlet endfitting of the fuel channel. Under these conditions, the endplates broke causing a bundle to partially disassemble in the endfitting. Normal residence time in cross flow is about 2 min.

On these rare occasions, bundle damage can be extensive and requires a reactor shutdown to recover broken fuel components.

3.3 SYSTEM DESIGN RELATED DEFECTS

3.3.1 STRESS CORROSION CRACKING DURING NORMAL OPERATION

In 1971-1972[7], the early operation of Pickering A Unit 2 resulted in many defects from power ramping associated with adjuster rod movements and refuelling shifts. These failures occurred before the introduction of graphite CANLUB that provides protection against SCC failure. The event at Pickering was the largest fuel defect excursion in CANDU history The core contained over 400 defective elements.

Most of the fuel defects were discharged during the four month shutdown in 1972. The fuel bundles were moved while they were exposed to low coolant temperature and pressure. Although no fuel handling difficulties were reported, the defective bundles appeared to be badly damaged[11] The reasons for the severe damage have never been fully resolved. In particular, it is unclear whether or not the degradation took place in the reactor during full power operation or in the fuel transfer system at low temperature and pressure. However, several observations point to the latter, as suggested below:

1. Because of the high deuteride content of the cladding of the defective elements, the Zircaloy was likely brittle at the low temperatures when the fuel was removed from the core while the reactor was shutdown

2. Most of the secondary damage was preferentially located at the upstream end of bundle #9 (high power end); and downstream end of bundle #10 (low power end) The degradation at the bundle ends is likely attributed to the high deuteride content and to the fuel handling system that handles bundles in pairs. Since these two bundles were discharged together, the damage was located at the ends that were in contact with the fuelling machine rams and side stops during transfer to the bay.

This experience along with other events involving SCC defects at Douglas Point (early 1970s) and at Pickering (1988), have shown that power ramp failures are "systematic" The defective bundles were generally had more than one defective element among the outer ring of the bundle. (See Figure 6)

3.3.2 ENDPLATE FATIGUE

In November 1990[8], a routine fuelling operation on channel N12 of Darlington Unit 2 was aborted because of difficulties encountered while inserting a pair of fuel bundles recycled from another channel. The follow-up investigation later showed that the centre seven elements from the downstream bundle had broken loose and had interfered with normal refuelling operations. These elements had been carried past the fuel latch by the coolant flow through the channel and had obstructed other bundles from being completely inserted into the channel The bundle was extensively damaged during the attempted refuelling operation. The damage increased the activity levels for most fission products in the primary circuit. The increased activity levels were difficult to quantify at the time since the failed fuel detection system was not fully commissioned.

Inspections of other outlet bundles in the Darlington fuel bays showed the presence of endplate cracks. The post irradiation examinations of the damaged fuel revealed that the endplate cracks were the result of high cycle-low amplitude fatigue. Prior investigations demonstrated that the 5-vane impellers of the primary circuit pumps introduced pressure pulsations that were acoustically amplified within certain fuel channels. The pulsation frequency of 150 Hz coincided with the resonant frequency of the inner seven fuel elements of the 37 element bundle. With latch support of the fuel column, which is unique to the Darlington and Bruce reactors, the non-outer fuel elements are unrestrained and are free to vibrate in the axial direction. Axial vibration at the resonant frequency lead to endplate cracking.

To eliminate the acoustic amplification of the pressure pulsations in the fuel channels and to decouple the axial resonant response of the fuel, the 5-vane pump impellers were replaced with 7-vanes This change shifted the pressure pulsation frequency from 150 to 210 Hz and eliminated the endplate cracking problem at Darlington.

4 FUEL ELEMENT OPERATING THRESHOLDS

4.1 CENTRELINE MELTING

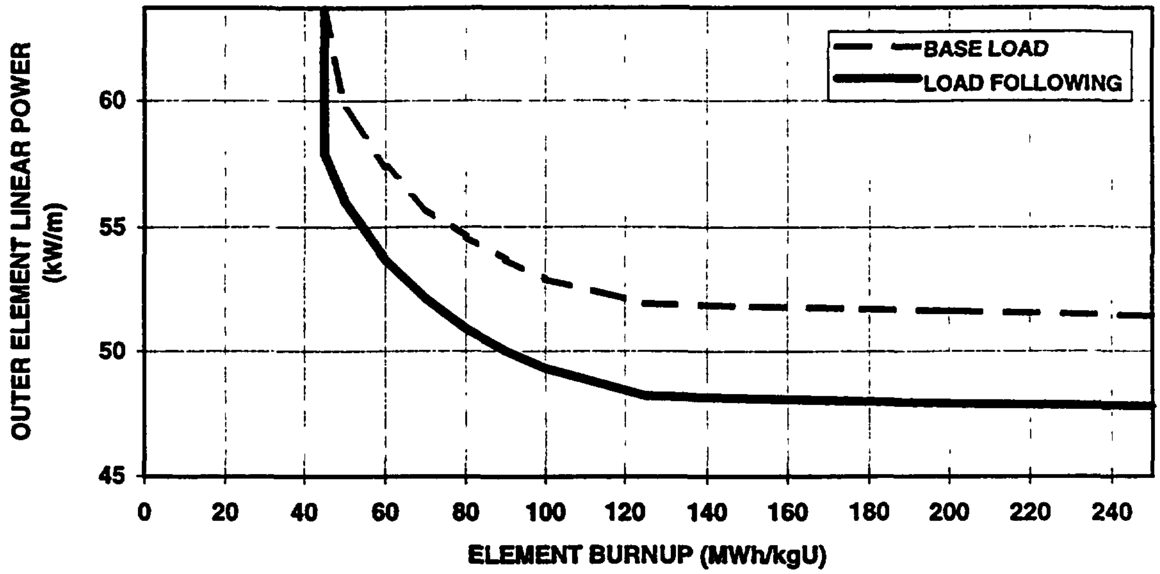
Experimental irradiations of CANDU type fuel elements indicate that centreline melting of the element occurs at element linear powers that exceed 70 kW/m. The fuel performance codes also predict centreline melting at these linear powers for fuel elements operating under normal heat transfer conditions of a CANDU reactor. The maximum fuel element linear powers corresponding to operating bundle power envelopes (and bundle power licensing limits) for all CANDU reactors, are less than 65 kW/m There has been no evidence of centreline melting in CANDU power reactor fuel

4.2 STRESS CORROSION CRACKING

The power histories of fuel defects caused by SCC of the sheath have been extensively studied. Empirically derived thresholds are used to limit the ramped power and change in power that bundles can undergo during refuelling and reactor power manouevres. These curves were empirically derived from the fuel defect experience at Pickering and Douglas Point[7] and from the experimental reactors in Chalk River[12]. The fuel defect database for the power reactors includes 15 mm diameter fuel elements only. With the possible exception of the high burnup Bruce bundles, there have been no SCC failures among the 13 mm diameter sheaths of the 37-element bundle. The risk of SCC fuel failure increases if the ramped power and power increase of a fuel element exceed BOTH thresholds.

The lower set of SCC thresholds used for CANDU 6 fuel is shown in Figure 7. These curves bound the power/burnup histories of 37-element fuel that has operated under load following conditions at

SCC Threshold for Ramped Power



SCC Threshold for Power Increase

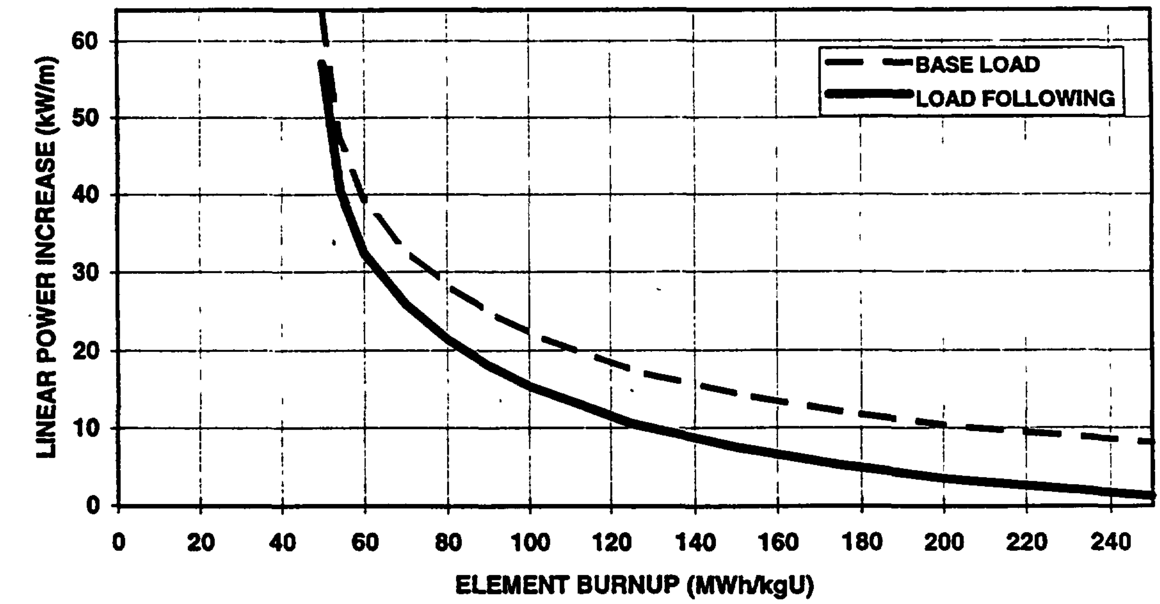


Figure 7: Thresholds For Stress Corrosion Cracking Used In CANDU 6 Stations

Bruce B, Embalse and Gentilly 2[13,14]. The upper set of thresholds in this figure bounds the power histories of bundles successfully irradiated under base load conditions at Bruce A. The difference in the two sets of thresholds reflect the differences in operating power envelopes among the CANDU reactors that experienced either base load or load following conditions. It is interesting that the recent SCC fuel failures at Pickering A Unit 1 in 1988[5] operated at powers and burnups that exceeded the upper set of SCC thresholds in Figure 7.

The thresholds used for Ontario Hydro are similar All normal operations at the CANDU stations are conducted in such a manner that the power conditions of all bundles in the core do not exceed these thresholds.

5 FUEL DESIGN BASIS FOR NORMAL AND ACCIDENT CONDITIONS

For normal operating conditions, CANDU fuel is so designed that systematic fuel failures do not normally occur. To avoid continued operation with many defects in the core, the nuclear regulator imposes reactor shutdown limits, based on fission product isotope concentrations in the coolant.

For many accidents, particularly those of moderate predicted frequency, fuel element sheath failures should not be systematic. In such transient conditions leading to high temperatures, the fuel element is considered to remain intact if the following criteria are satisfied:

a No fuel centerline melting. A fuel element will be assumed to have failed if its centreline temperature exceeds the UO_2 melting temperature of 2840°C;

b No excessive strain. A fuel element will be assumed to have failed if the uniform sheath strain exceeds 5% for sheath temperatures less than 1000°C or 2% for the temperatures higher than 1000°C; and

c No oxygen embrittlement. A fuel element will be assumed to have failed if its oxygen concentration exceeds 0.7 wt% over half the sheath thickness.

For other, less frequent accidents (for example, large LOCA, single channel events), some fuel element failures are permitted.

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DEVELOPMENT OF RBMK FUEL ASSEMBLIES. FEATURES OF DESIGN, RESEARCH FINDINGS

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Abstract

The report is an overview of the work done by RDIPE with participation of other enterprises on creation and improvement of characteristics of RBMK reactor fuel assemblies and subassemblies also including pre-irradiation and reactor in-pile experiments and post-irradiation research. A number of aspects concerning fuel element and assembly failures are considered.

Introduction

Graphite-uranium channel-type water-cooled reactors have been traditionally developed by RDIPE scientists and engineers (Table 1).

TABLE 1. GRAPHITE-URANIUM CHANNEL TYPE WATER COOLED REACTORS DEVELOPED BY RDIPE

Description	Electric power, MW	Year of commissioning
Production reactor "A"	-	1947
Production reactor "I-1"	-	1953
The first nuclear plant in Obninsk	5	1954
Siberia NPP	600	1958
Beloyarsk NPP (with steam superheating)	300	1964-1967
RBMK-1000	1000	
- Leningrad NPP 4 units		1973-1981
- Kursk NPP 4 units		1975-1985
- Chernobyl NPP 4 units		1977-1983
- Smolensk NPP 3 units		1983-1987
RBMK-1500	1500	
- Ignalina NPP 2 units		1983-1987

RBMK is a channel-type boiling-water graphite-uranium reactor with a single-circuit circulation scheme /1/. The distinction of the RBMK-type reactor is refuelling without reactor shutdown. This provides for a higher capacity factor and a possibility of a flexible fuel cycle but necessitates additional requirements in respect of fuel.

For a higher burnup, the fuel enrichment has been gradually altered as 1.8%, 2.0%, 2.4% and 2.6% (fuel with erbium addition) in the RBMK-1000 reactors; as 1.8%, 2.0% and 2.4% (with erbium addition) for the RBMK-1500 reactors /2/.

To ensure safety (after the Chernobyl accident), additional absorbers have been introduced into the core as a substitute to the assemblies to reduce the void reactivity effect. As the result, the fuel burnup has been considerably reduced down to 21 MW day/kg. A range of modifications to the core design has been developed including a larger fuel element diameter (from 9.1 to 14 mm) and a larger channel and fuel assembly diameter. It turned out to be practicable to restore the original characteristics in the latter case. Assemblies are currently implemented with no modifications made to their design and the channel diameter at the expense of using the fuel with addition of erbium. This provides for the required change in the void reactivity, effect and burnup characteristics. Simultaneously, the use of additional absorbers is excluded /3/.

The feature of the design of the existing assemblies is that they consist of two fuel subassemblies over the core length which leads to a contrary meeting-elongation of the fuel elements. Therefore, the gap between the fuel asemblies in the core centre should make up for their elongation due to thermal expansion, fuel-cladding interaction, irradiation growth and creep of the cladding material.

The assembly's carrier element is the centre rod or tube wherein different monitoring sensors may be installed /4/.

The cellular spacer grids are capable of an axial movement in relation to the centre tube to which they are attached.

The fuel is uranium dioxide produced out of natural or reprocessed uranium.

The basic problems set in creation and subsequent operation of the RBMK-type reactors were as follows:

- To validate the design characteristics by calculations and experiments (pre-reactor and in-pile tests, post-reactor research).
- To raise the RBMK-1000 reactor power by a factor of 1.5 (RBMK-1500)
- To raise the fuel burnup.
- To vary the physical characteristics to ensure the RBMK reactor safety (after of the Chernobyl accident);
- To analyse the fuel operational characteristics and behaviour in emergencies and the cladding failure criteria.

RBMK-1000 AND RBMK-1500 ASSEMBLIES

A traditional range of problems has been solved in the development of the RBMK-1000 assemblies including:

- to comply with specifications in respect of the reactor and its systems;

- to select and ground the structure materials with regard to their service in conditions of a neutral non-adjustable water chemistry;
- to ensure the required strength characteristics with regard to radiation and corrosion effects;
- to ensure the thermal-hydraulic reliability;
- to comply with the safety requirements.

A special attention was given to the problems of interaction between the assembly and the fuel channel (FC) tube and the avoidance of fretting corrosion. The design and technological approaches towards the fretting avoidance were to select the stretch in the spacer grid cell on the one hand and to ensure the admissible level of physical interaction with the FC tube as determined by the coolant's turbulent and acoustic pulsation on the other hand.

Accelerated vibration tests were carried out at fixed frequencies using an subassembly shortened model based on 5 10^6 cycles. In case of there was a natural vibration frequency within this range, the tests were carried out at a resonance frequency.

The vibration stress peak value in the fuel element cladding was 25 MPa The maximum cladding wear was 48 µm.

The aim of carried out works was not only to optimise the cell-fuel element connection but also to eliminate the weak points in the design.

To evaluate (to the utmost) the parameters of the FA-FC shock interaction, a theoretical research was carried out jointly by RDIPE and the Institute for Problems in Mechanics Russia Academy of Sciences (IPM RAS) /5/ to study the assembly's mathematical model represented as a double-mass oscillating system with the circular motion and side stops.

In the conditions of a thermal test rig, the pressure pulsation was measured in three sampling points throughout the FC height: in the area in the assembly's central section where the flow approaches the lower fuel element, at the upper and lower sub FA joint and at the site where the flow leaves the FA.

The experimental research into the FA and FC vibration characteristics was conducted using strain-gauge inserts fitted into the typical cross-sections of three fuel elements in the upper FSA and into one fuel element of the lower FSA. The FSA vibration was monitored using accelerometers.

To record potential contact interactions of the FA with the FC, electric prods were used which were fitted into four cross-sections throughout the FC height in electric isolation from the FC.

The research of the vibration and pressure pulsation that was conducted on a full-scale assembly model has shown that pulsation and vibration are random processes whose spectrum has the highest intensity in the low-frequency region.

The cladding vibration stress level in all rig-simulated modes was not high and did not exceed 1.5 MPa. This level is not dangerous in points of fatigue and does not cause, fatigue cracks and consequently cladding failures

The electric prods which protruded inside the FC by the value of 0.05 mm allowed to identify the FA-FC interaction in terms of frequency and time and the FA's spatial (azimuthal) orientation within the reactor cavity.

The analysis of the oscillographic recordings shows that FA may produce both coupled oscillations according to a single-mass system pattern and oscillations inherent to a double-mass system. The assembly may oscillate both in a single plane (flat oscillations) and with periodic generation over the internal FA surface (spatial oscillations). The most intense oscillations occur in the spectrum's low-frequency section within a range from fractions to   10 Hz.

The analysis of the FA vibration measurements has shown that channel oscillation processes have a broad spectrum with separate most distinct maximum values at 12.5, 17.0 and 25.0 Hz frequencies which are determined by natural oscillations of the FC and the rig as a whole.

Practically, the FA oscillations have no connection with the FC oscillations because of they have different frequencies /6/.

It was technically difficult to maintain the rig arrangement of sensors. In the in-pile experiment, therefore, they were installed on the assembly suspension outside the reactor core.

The pressure pulsation and suspension vibration were measured in the reactor conditions on three assemblies of the Ignalina NPP's unit 2 in various operating modes.

At a higher flow void fraction, an increase in the pressure pulsation intensity was observed which led to a higher FA vibrations level. As an example of the dependence of the pressure pulsation and vibration distribution on the unit's operating mode, the spectra of vibration and pulsation is shown in Fig.1. The maximum pulsation energy was generated in the low-frequency region. The pulsation peak value amounts to 0.84

Hence, the analysis of the in-pile test results allows to assume that the FA is in the auto-oscillation mode wherein natural oscillations caused by the shock interaction with the FC occur.

Further inspection of the spent FA and the FC internal surfaces demonstrated that the FC internal surfaces had circular traces of the FSA-FC interaction which was the evidence of the existence of the FA spatial oscillations in the circular gap.
The comparative characteristics of the coolant vibrations and pulsation are featured in the Table 2.

It follows from the table that the vibration action inside the core is much lower than in rig test conditions. Therefore, the initial rig tests are conservative and the FA serviceability is guaranteed in the event of positive results.

The service-life and reactor tests have allowed to establish that fretting problem is absent in the design conditions of the FA operation. In considering the assembly-FC interaction model, an estimation of the maximum contact shock interaction was provided. It was found that the force pulse peak value did not exceed 10 N. The dynamic force level obtained did not turn out to be high and it is not dangerous in terms of fatigue strength.

The complex of the pre-reactor mechanical tests also included the following:

- determination of the permissible FC curvature;
- research of the thermomechanical interaction between fuel elements and the FA frame;
- testing of spacer grid for strength;
- examination of the influence of hydration on serviceability of the structure carriers /7/.

The latter direction was connected with determining the possible assembly loading rate. At a loading rate of 10 m/min with an abrupt stop or the assembly fall from the height of 0.5 m (including with the use of the carrier materials exposed by artificial hydriding), the assembly was not damaged.

Besides the basic experiments on determination of the critical thermal loads, the pre-reactor thermotechnical experiments that were conducted to justify the RBMK-1000 FA design involved additional experiments including on the influence of the heating length, the axial power density irregularity and the comparison with different correlation of the experimental results.

It was established that the axial power density irregularity of K_z = 1.38 has no influence upon the critical loads but the heated length of 1.23 m produces a two-fold critical load value as compared to 7 m. A new correlation was proposed which has produced a satisfactory correspondence with the experiment /Fig. 2/.

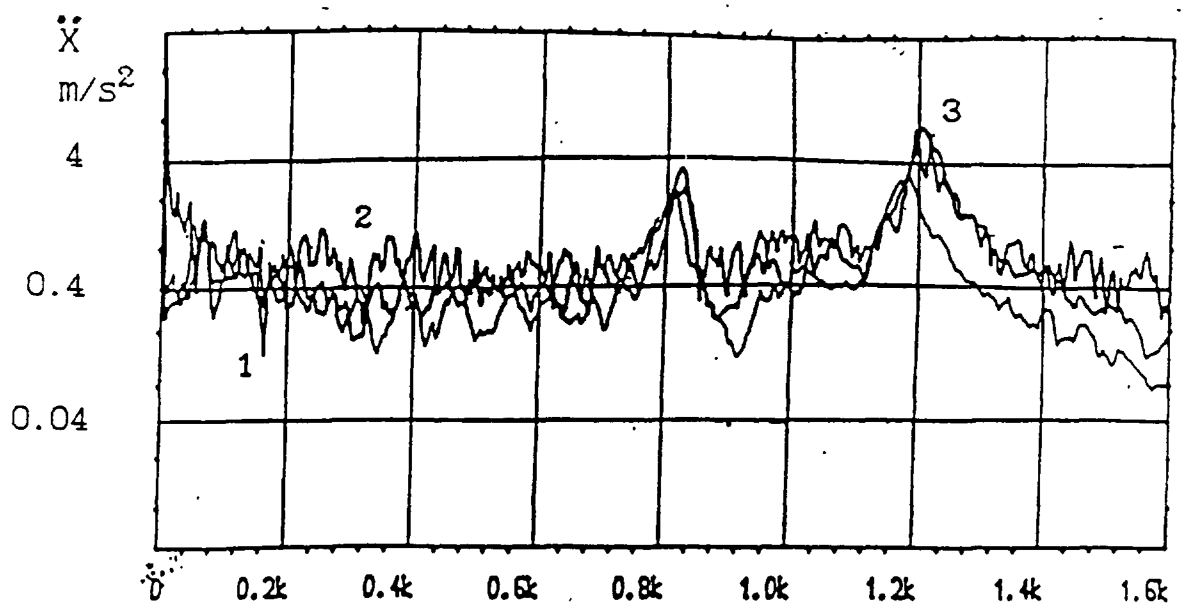


FIG. 1a. Spectra of vibration for the upper part of FA (FA suspension region) for thermal reactor power of 1000 MW (1), 2600 MW (2) and 4500 MW (3).

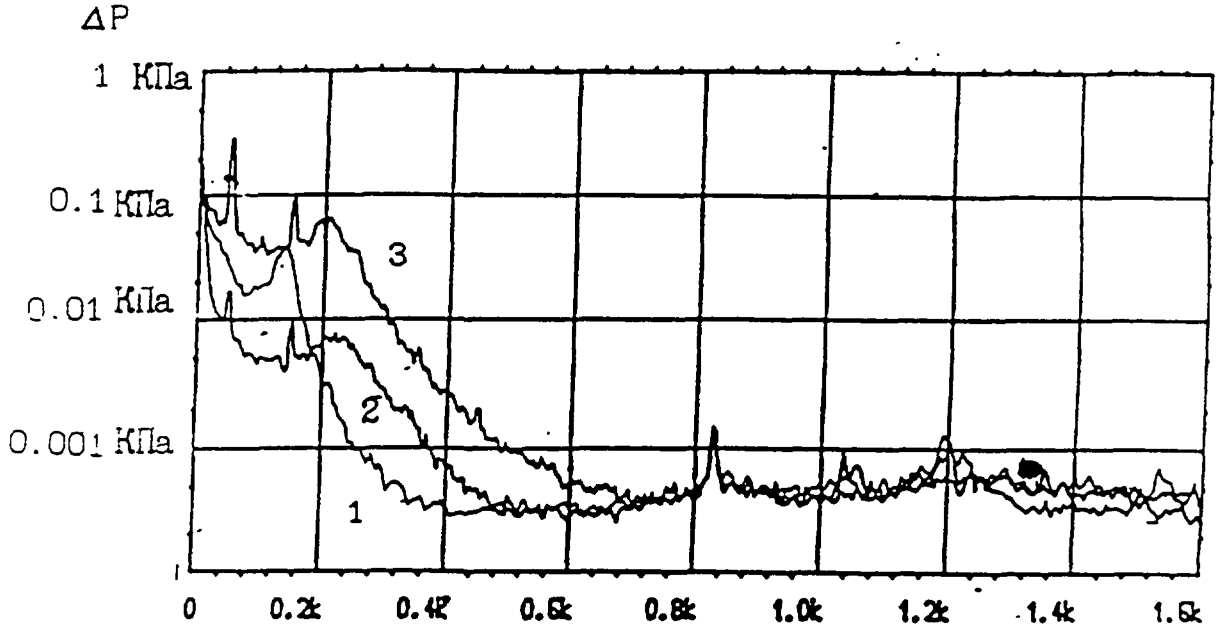


FIG. 1b. Spectra of coolant pressure fluctuation (region of steam-water lines to fuel channel connection) for thermal reactor power of 1000 MW (1), 2600 MW (2) and 4500 MW (3).

TABLE 2. CHARACTERISTICS OF THE COOLANT VIBRATION AND PULSATION

Vibration action	Duration	Coolant temperature, °C	Linear power, W/cm	Measured fuel cladding tension stress, MPa	Coolant pulsation, kPa	Fuel cladding damage depth, μm
Force action (shaker)	$5\ 10^6$ cycles	20	0	25	-	50
In steam-water mixture (rig)	10000 h	270-280	0	1.5-3.6	30-42	up to 40
With coolant boiling (reactor)	3-4	270-280	350-500		0.84	

The requirements to the RBMK-1500 assembly included a power increase by a factor of 1.5 without FC modifications and with a certain coolant flow reduction. Two solutions were possible:

– to reduce the fuel element diameter and enlarge the heat transfer surface;

– to intensify heat exchange.

The second way was selected.

Data on heat exchange intensification using 'twisted tapes' had been published by the time /8/ and such design was tested as applied to the RBMK assembly. Other intensifier types were considered and their test results are featured in the Table 3.

These results were obtained on direct electrical heating (DEH) rigs. Taking into consideration that no accurate simulation of power density distribution and coolant flow hydrodynamics was possible in such conditions (especially, at the gap between of two subassemblies within the FA) in-pile tests of the assemblies were additionally carried out and their results are also featured in Fig. 3, 4, 5. As the result of these experiments, the coolant vertical whirl variant was selected which had a simple configuration and an acceptable technology. Intensifying spacer grids with full sets of cells are placed within the upper subassembly with an interval of 360 mm as intensifying grids with incomplete cell sets are placed between them with a 120 mm spacing.

The pre-reactor thermotechnical experiments also involved a study of the influence of the following factors upon the critical heat flux

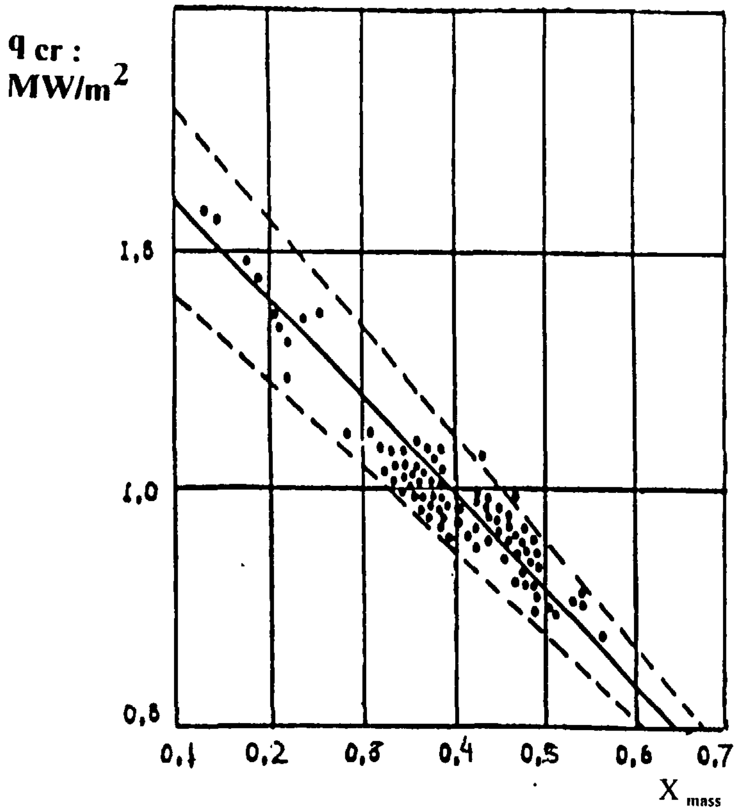
– a cross-section change by means of an increase in the FC diameter from 80 to 82 mm,

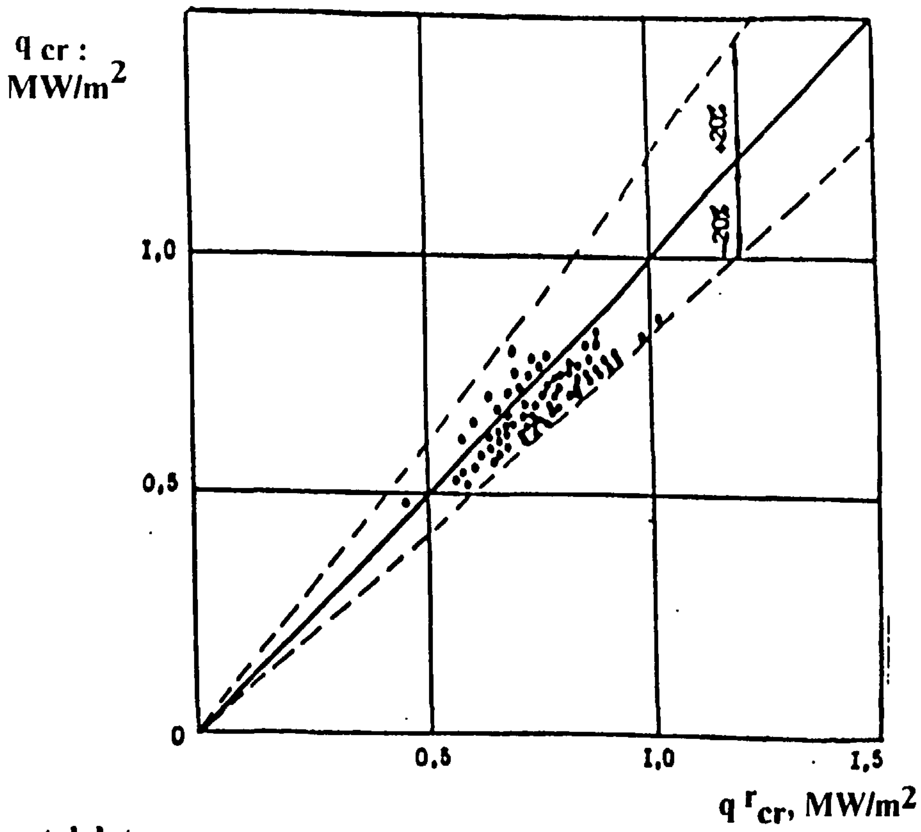
– the heated length;

– a change in the gap between the fuel elements from 1 to 2.3 mm;

– axial irregularity $K_z < 1.55$

– power density irregularities over the assembly cross-section $K_{cell} < 1.15$.





• - **experimental data,**

— - **calculated data,**

- - - - **maximum deviation from the calculated curve** **(± 3 σ).**

FIG. 2a. Experimental and calculated critical heat flux values versus steam quality for RBMK-1500 FA model (P=7.0-9.5 MPa, ρW=1000-3000 kg/m²s).

FIG. 2b. Comparison of experimental and calculated critical heat flux values for RBMK-1000 FA model (P=7.0-9.0 Mpa, ρW=2000-2500 kg/m²s).

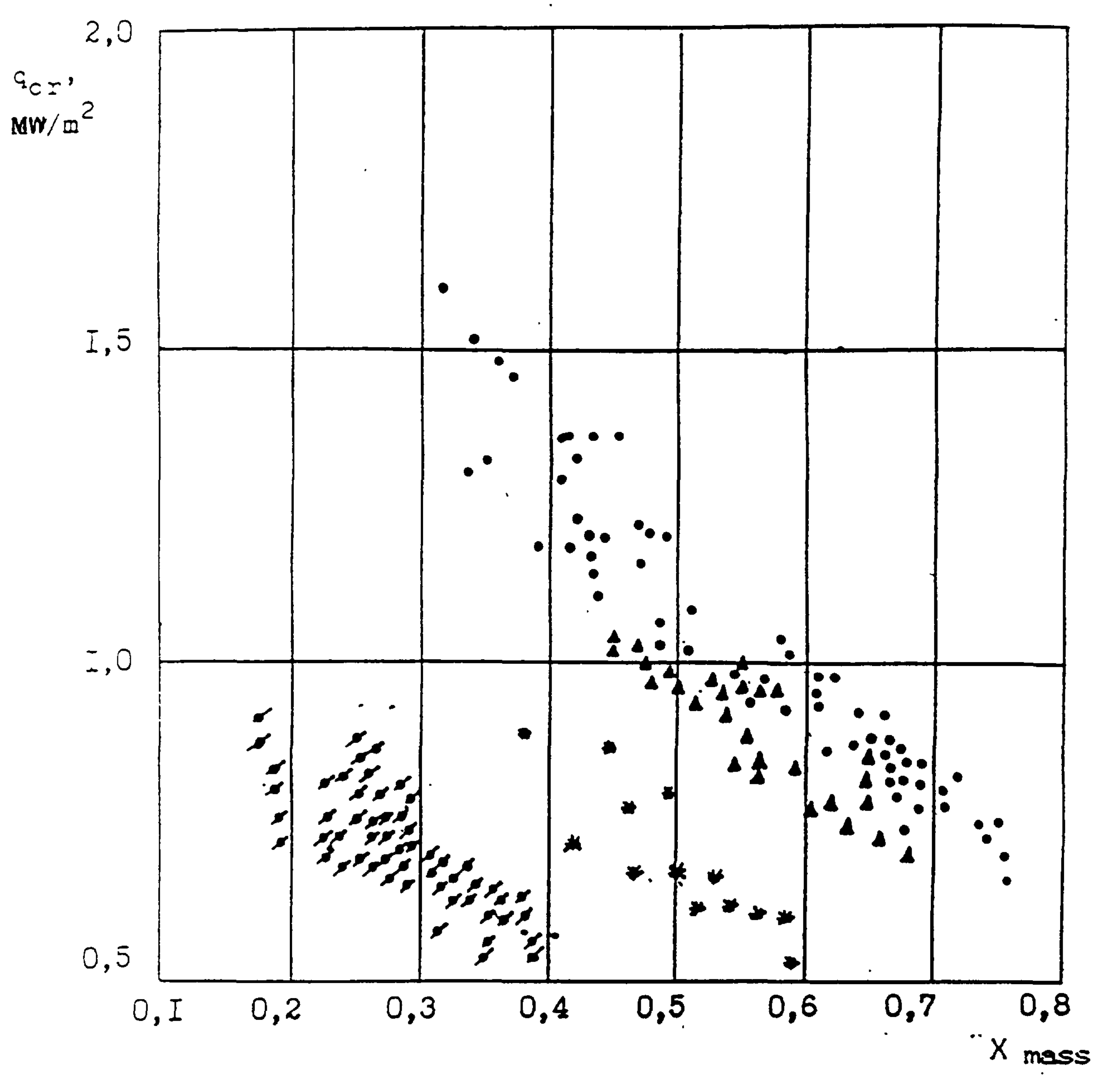


FIG. 3. Critical heat flux values versus steam quality in mock-up FA with different types of intensifier spacers [21].

- *- mock-up FA with standard RBMK-1000 spacers,*
- *- mock-up FA with standard RBMK-1500 intensifier spacers,*
- *- mock-up FA with screw spacers,*
- * *- mock-up FA with throttling intensifier spacers.*

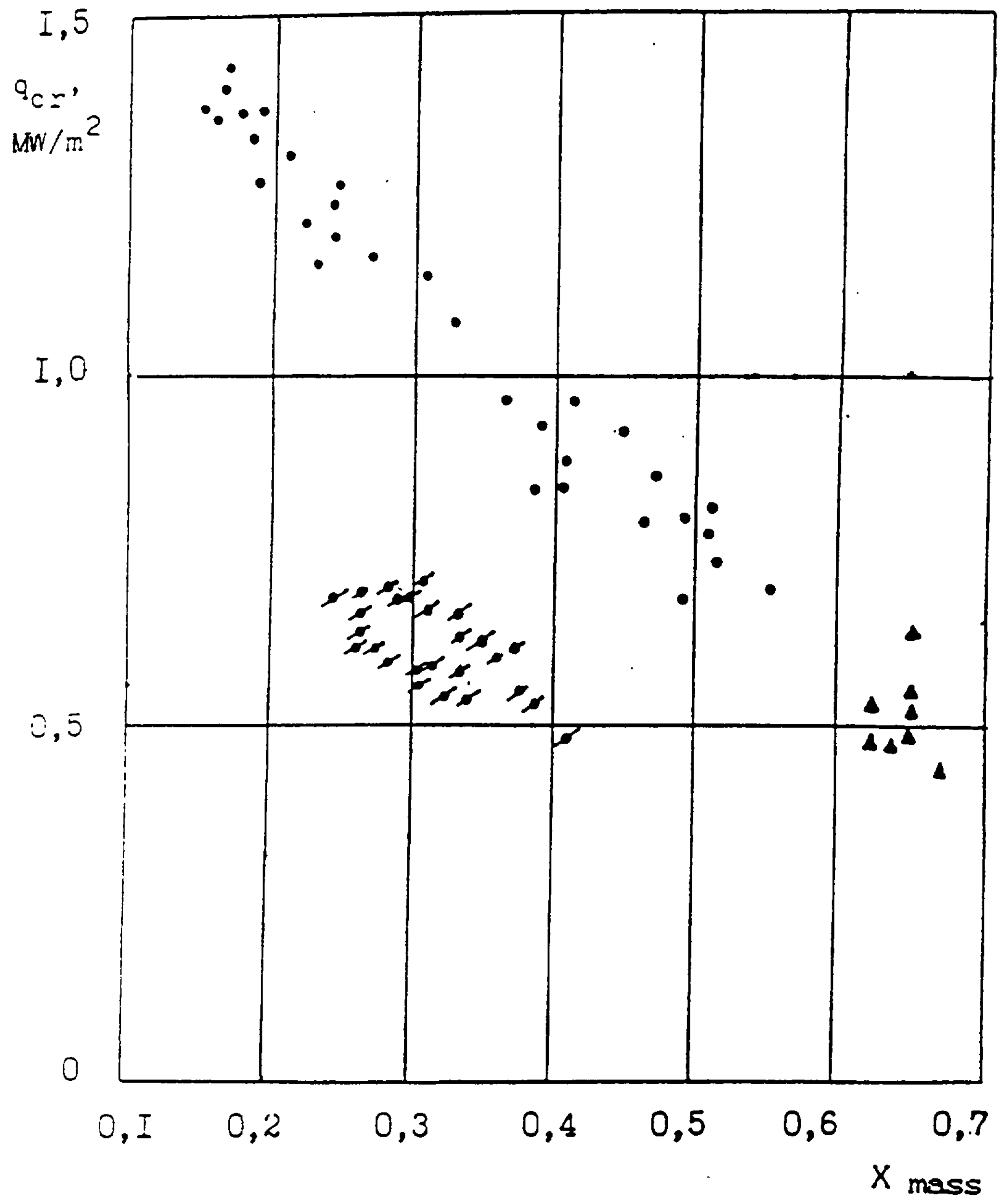


FIG. 4. Comparisonof experimental rig and in-pile data on dryout in mock-up RBMK-1000 FA [21].

- ▸ - *in-pile test, H_h=3.3 m,*
- • - *test data from freon rig, H_h=3.5 m,*
- • - *test data from water rig, H_h=7.0 m.*

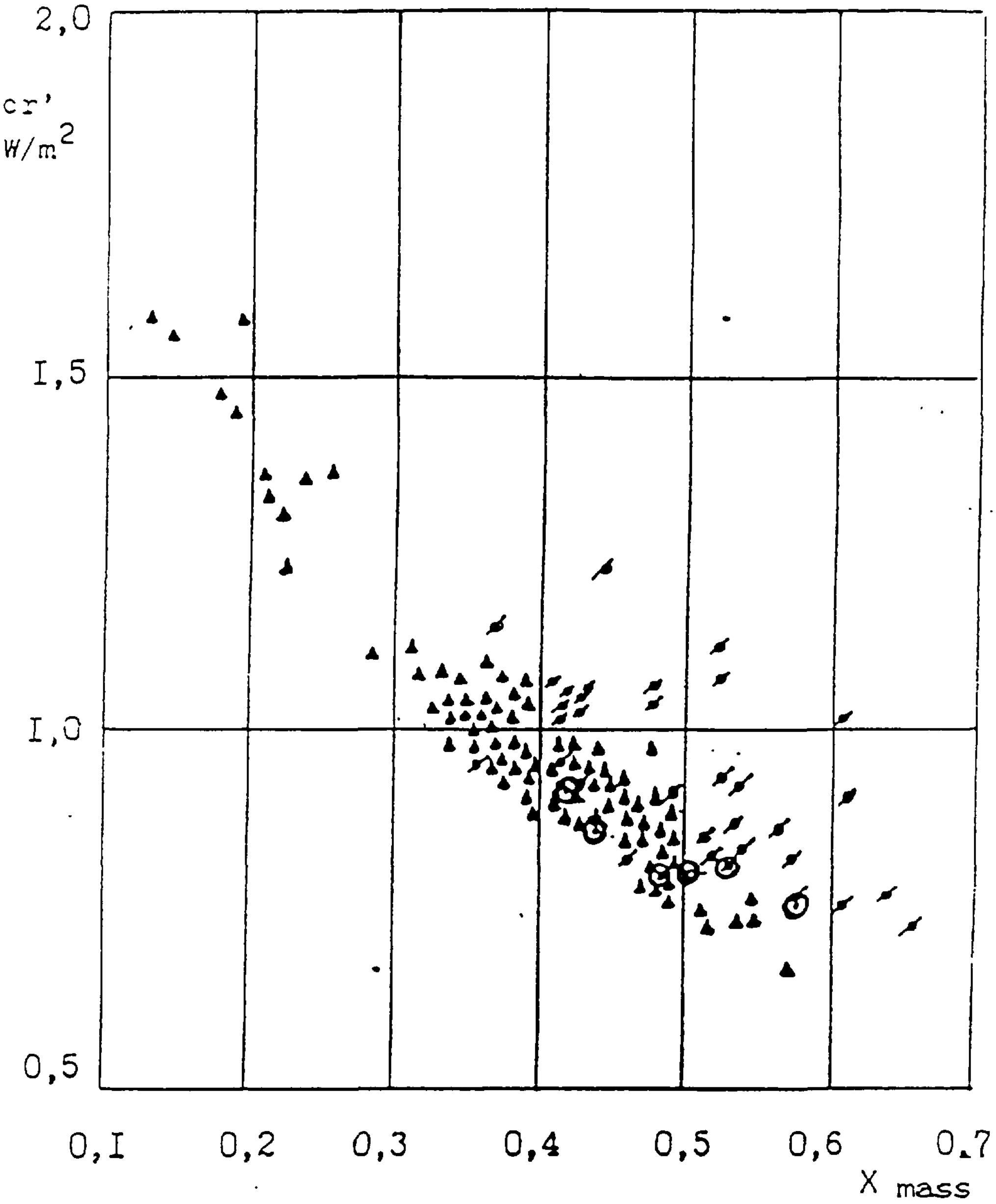


FIG. 5. Comparison of experimental rig and in-pile data on dryout in mock-up RBMK-1500 FA with standard intensifier spacers.

- *in-pile test, H_h=3.3 m,*
- *rig test, H_h=7.0 m,*
- *in-pile test, H_h=7.0 m (Leningrad NPP, upper FA).*

TABLE 3. IMPACT OF DIFFERENT TYPES OF INTENSIFYING GRIDS ON THERMAL PERFORMANCE

No.	Intensifier type	Power gain, %
1	Twisted tapes in RBMK fuel assembly	70
2	Grids with screw whittlers	40-60
3	Finned rod	35-40
4	Grids with constricted cross-section	5-40
5	Cells with outer bends	30-40
6	Grids with vertical whirling	60-75

It was established that the critical thermal load decreased by a factor of 1.5 at a change in the gap between the fuel elements from 2.3 to 1 mm, remained almost the same within a range of 2-2.3 mm and was reduced at K =1.15 as compared to K =1.06. No critical thermal load reduction was recorded in other cases. Its value exceeded that for the RBMK-1000 assemblies by a factor of 1 5-2.0 with the selected spacing and design of SG

The full-scale assembly research with heat-exchange intensifiers has shown that departure from nucleate boiling does not depend on pressure within the pressure range of 6-9 MPa.

The correlation of q_{DNB} = 2.23-2.05 X was proposed for the specific RBMK-1500 assembly design within a rate range of ρW=1000-3500 kg/m^2 s and a mass steam quality range of X = 0.3-0.9. (The mean-root square error is 4 4%) /7/ Fig. 2.

The further upgrade to the assemblies involved with an increase in the fuel burnup and implementation of sets of improvements in technology and design. These modifications were confirmed both using pre-reactor test rigs and in the in-pile conditions.

Service-life in-pile test

The preceding work on mechanical strength, thermohydraulics, autoclave and dynamic corrosion tests, justification of the material selection and tests of the material properties at hydrogen absorption allowed to proceed to the service-life in-pile test.

Initially, the service-life test was carried out on an loop rig (B-190 which specially was done for RBMK) /9/ with the core size of 3.5 m and the full-scale FA cross-section. The neutron spectrum and fast and thermal neutron flux were similar to the RBMK characteristics.

The loop test was conducted on the Beloyarsk NPP loop, the MR reactor loops at the Kurchatov Institute and the IVV-2 reactor at RDIPE's Sverdlovsk branch.

A large number of service-life tests for the RBMK-1500 FAs and subassemblies with technological and design improvements have been conducted following the commissioning of the units using RBMK-1000 reactors

Table 4 presents the basic types and results of the full-scale FSA and FA and subassembly service-life tests

The results of these tests have demonstrated the adequacy of the design approaches and have allowed a prompt introduction of progressive proposals on subassembly and FA upgrades.

TABLE 4. REACTOR TESTS. VALIDATION OF SERVICE-LIFE CHARACTERISTICS

No.	Test type	Test site	Results
1	RBMK-1000 FA	B-190	The selected gap between the subassemblies is sufficient at an average burnup up to 30 MW day/kg
2	RBMK-1500 with enrichment of 2.4% (180 assemblies) with enrichment of 3.6% (51 assemblies)	Leningrad NPP Kursk NPP	Serviceability of the assemblies with intensifiers has been confirmed
3	Measurement of gas pressure under cladding	B-190	P^{max}=1.85 MPa at a burnup of 20 MW day/kg (ql=485 W/cm P_{in}=0.5 MPa helium)
4	Testing of assembly designs: - 7m-long fuel element (39 assemblies); - Zirconium SG (30 assemblies)	Leningrad NPP	No negative signs throughout the service life
5	Different pellet configuration and density 10.2 - 10.4 and 10.4 - 10.6 g/cm^3 (142 assemblies)	Leningrad NPP	Density of 10.6 g/cm^3 was selected.
6	Testing of different welding types: magnetic-discharge welding (39 assemblies) electron-beam welding (90 assemblies) butt resistance welding (167 assemblies)	Leningrad NPP	Negative results on magnetic-discharge welding. Electron-beam and butt resistance welding is recommended
7	Testing of fuel cladding materials 139 assemblies with alloys: Zr-1% Nb Zr-1.3% Sn-0.4% Fe Zr-1.5% Sn-1% Nb-0.5% Fe	Leningrad NPP	Burnup of 18.8 MW day/kg. The Zr-1% Nb alloy was selected. The best corrosion resistance of welded joints

Special reactor test

A series of special reactor tests was conducted on the B190 loop.

1. Critical thermotechnical test (DNB). The purpose of this test was to specify the DNB for the FA with two 1700 mm-long subassemblies installed to simulate a power burst and the features of flow hydrodynamics in the gap between subassembly which, as it was earlier said, was impossible on electrically heated rigs.

The power range during the critical experiments was 1.925-3014 kW which corresponded to the assembly power of 3800-6000 kW. As the departure from nucleate boiling started at high void fractions of 48-80%, jumps in the fuel element wall temperature were recorded which were normally below 10 °C, and 70 °C in only one case. Altogether, 90 experimental points were obtained.

2. The power was 2050 kW during the supercritical experiments as the void fraction was up to 70%. The jump in the fuel element wall temperature was 70-100 °C at the coolant flow rate of 7 t/h and up to 200 °C at the rate of 5.9 t/h. No visible damages to the fuel elements were observed as the result of the inspection.

3. A thermocyclic test was conducted to research the thermomechanical pellet-cladding interaction (PCI). Fuel cladding and fuel column elongation sensors and thermocouple were installed in the RBMK-1500 assemblies to record the fuel temperature. No residual fuel element deformation was caused by power changes from 0 to 50%, but the residual deformation was recorded at power jumps from 0 to 100% with a higher rate (as compared to regulations).

The purpose of the tests conducted at the Leningrad NPP was to establish a possibility of reducing the coolant flow both for the RBMK-1000 and RBMK-1500 assemblies /10/.

The procedure for carrying out these experiments had been thoroughly developed by the Leningrad NPP personnel and RDIPE and Kurchatov Institute specialists. Primarily, the following safety aspects were considered:

- emergency protection actuation from signals other than related to the coolant flow reduction;
- detection of leaky fuel cladding;
- monitoring of the power of the fuel assembly with an increased fuel content;
- training of the personnel in charge of the coolant flow reduction.

The results of the tests are presented in Table 5 and Fig. 5 (the results in Fig. 5 refer to maximum coolant flow rate and void fraction values in the upper subassembly). Hence, the power in excess of the maximum RBMK-1500 assembly power was produced during the experiments and the RBMK-1000 assembly and thermohydraulic margins to critical heat flux were confirmed.

On the basis of the experiments conducted at the Leningrad NPP and in the B-190 loop the RBMK-1000 and RBMK-1500 fuel assemblies safe operation limit depends related to the coolant flow may be determined. It is 12.5 and 16 t/h at a void fraction of 46 % and 50 % respectively.

Acceptability criteria for fuel cladding

Fuel cladding failures may occur during accidents either on account of a thermal fuel-cladding interaction or because of a thermomechanical deformation of the cladding both at positive and negative pressure drops thereon. The first failure type is largely typical at fast and considerable power bursts (i.e. for reactivity accidents). The other failure mechanisms are related to an increase in the

TABLE 5 REACTOR TEST. SPECIAL TEST TYPES

No.	Test type	Test site	Results
1	Dryout overrun beyond the critical parameters	B-190	ql < 750 W/cm x < 80 %
2	Power surge at an increased rate, 349 cycles N=0 - 100 %, 50 cycles 60-100%, N^{max} = 2000 kW	B-190	No visible traces of damage, fuel elements are leaktight, ql^{max} = 580 W/cm. Burnup of 20000 MW day/kg
3	Coolant flow reduction: RBMK-1000 RBMK-1500	Leningrad NPP	N = 2.74 MW G_{min} = 12.4 t/h, G_{nom} = 29.5 t/h, N = 2.71 MW G_{min} = 7.65 t/h, N = 4.24 MW G_{min}=15.1 t/h, G_{nom} = 23.8 t/h

cladding temperature at different pressure ratios within the circulation circuit and under the fuel cladding. At a positive pressure drop, the heated cladding squeezes the fuel pellet column and is subsequently deformed into any gaps between the fuel pellets. At a negative pressure drop, the heated cladding swells (balloons). In any case, the being deformed cladding may locally achieve considerable deformations (i.e. local wall thinning) which are sufficient to cause a wall failure.

It was demonstrated that the following conditions are sufficient to confirm the maintenance of the cladding integrity during any accident:

- the maximum fuel enthalpy remains below 712 kJ/kg (170 cal/g);
- the maximum fuel temperature does not reach the melting point of about 2600 °C;
- the maximum cladding temperature does not exceed 700 °C (within 1 h).

These criteria are somewhat simplified but they are useful for the initial evaluation of the accident analysis results. If these conditions are met, no further analysis is required to confirm the fuel claddings integrity. If these 'initial' criteria are exceeded, it does not necessarily means that a cladding failure occurs. It means that more data is to be considered such as local pressure in the fuel channel during fuel cladding heatup, the power change rate during its pulse or other specific data of the accident scenario.

Fuel element failures which are caused by the pellet-cladding interaction are typical for reactivity accidents. No special experimental research of the kind was conducted on the fuel-cladding interaction in respect to the RBMK fuel elements. Meanwhile, the fuel for the RBMK fuel elements differs just little from the light water reactor fuel for which a sufficient experimental data base has been obtained both in Russia (pulse experiments for the WWER fuel elements conducted on the Gidra and IGR reactors /11/) and abroad (e.g. tests on the SPERT and PBF rigs in the USA and the NSSR reactor in Japan). On the basis of this data, the fuel enthalpy which corresponds to the onset of the fuel cladding failure is assumed to be equal to a value of about 1000 kJ/kg UO_2 (240 cal/g). The conservative margin assumed in /12/ suggests that the fuel enthalpy should not exceed 712 kJ/kg (170 cal/g) to avoid any fuel damages

Another criterion which determines the pellet-cladding interaction is the regulatory requirement that the fuel temperature should not exceed the melting point.

The fuel cladding squeeze occurs in accidents during which an increase in the fuel cladding temperature is observed at still an increased pressure within the circulation circuit. The fuel cladding failure conditions were experimentally determined /13/ as the function of the axial gap between the fuel pellets both for the nominal pressure of 7 MPa and with the lowest pressure inside the fuel cladding.

If a temperature increase in the fuel claddings occurs at a sudden pressure drop within the circulation circuit, the fuel cladding swelling takes place. The data in Fig. 7 demonstrates that the 'initial' temperature criterion which is equal to 700 °C is also very conservative as no cladding failures have been observed at a temperature of below 700 °C. If the "initial' criterion is exceeded, consideration is required to be made for the circulation circuit pressure change during accident development and for the in-core burnup distribution.

FA cool ability criterion

The regulatory criteria /14/ determining the maximum fuel cladding temperature which should not be exceeded /i.e. 1200 °C) and the cladding oxidation extent which should not exceed 18% of the wall thickness are the characteristics which provide for the maintenance of the FSA cool ability both in design-basis and beyond design-basis accidents.

Though these criteria are widely applied to pressurized vessel reactors, we believe that they may be too conservative for the channel-type reactors.

Operation results, post-irradiation studies (examples)

Several reports /15, 16/ were submitted by the Nuclear Reactors Research Institute to the 1993 IAEA Technical Committee concerning statistical data on the assembly operation in the RBMK-1000 and RBMK-1500 reactors. The number of leaky fuel elements in the unloaded fuel was reported to be 0.02%. The number of the failed fuel elements (which led to the premature fuel assembly unloading) was at the level of 0.013%. It should be noted that the number of failures varied among NPPs. The best result of 2-3 fuel assembly failures per one power unit per year was achieved by the Smolensk NPP personnel. As we consider, it was due to the fact that special care and thought were given to performing the fresh fuel loading operation.

As the failure analysis for different reactor types shows that a special attention should be given to avoidance of multiple assembly failures which may be due to contamination of the circulation circuit, a sudden change in the water chemistry, a mismatch between power and coolant flow in the core's local sections and an abnormal reactor condition during maintenance.

Thermometric FA were developed, produced and employed in all RBMK reactors to obtain additional data on the core condition as may be useful for NPP personnel.

Similar assemblies with still wider capabilities in respect of the obtained data scope were used during the startup of the RBMK-1500 reactor.

Emergency situations may be caused by reasons other than ones related to violations of operating rules. Such fuel accident occurred at the Leningrad NPP Unit 3 due to a loss of the coolant flow caused by the destruction of the isolation and control valve at the channel inlet. The accident led to the destruction of the channel pressure tube and a damage to fuel elements and fuel assembly. Studies in materials technology conducted by RDIPE's Sverdlovsk branch jointly with the Paul Scherer Institute have shown that the temperature of the structure material of the assembly centre carrier tube amounted to 1300 °C as the fuel cladding temperature was estimated /17/ at 1600 °C.

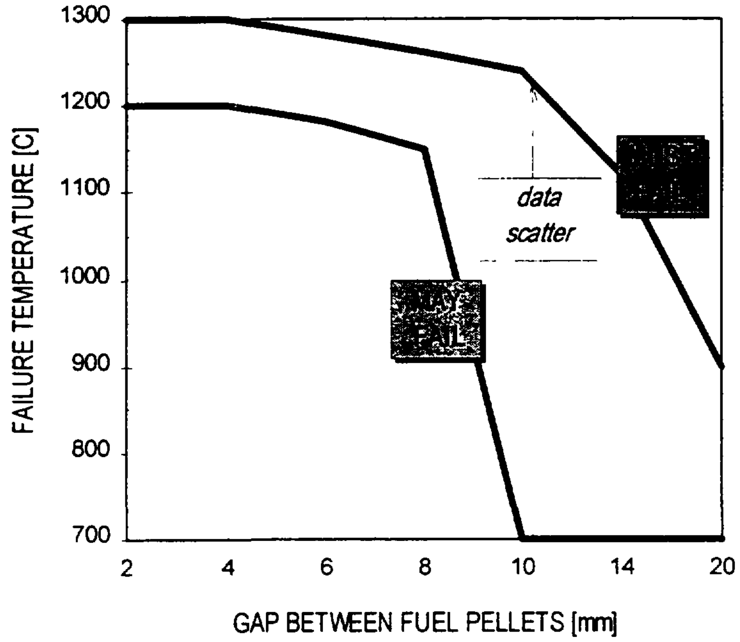


FIG. 6. Cladding failure due to collapse.

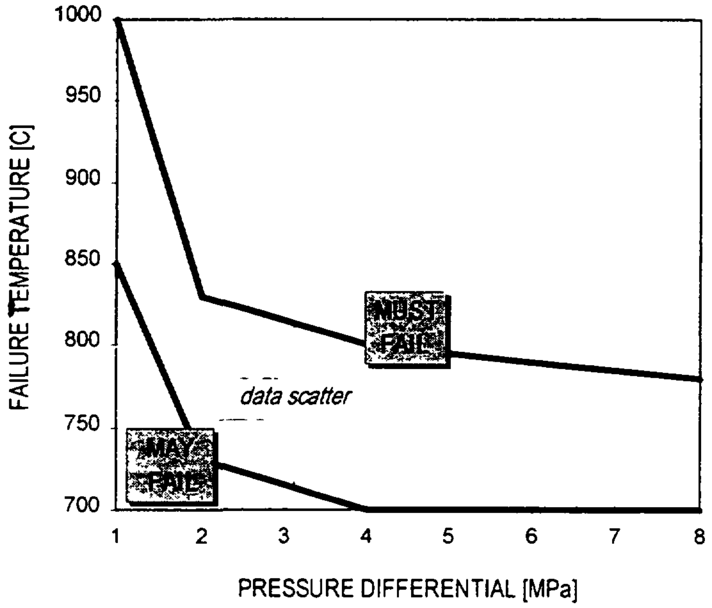


FIG. 7. Cladding failure by ballooning.

Single fuel assembly failures also occurred. They were caused by an increased hydrogen pick up at the weld joint flaw sites on the assembly's carrier end.

Failure causes were identified and the structure materials and fuel conditions were studied at RDIPE's Sverdlovsk branch.

Altogether, more than 30 assemblies were examined. The basic results of this study are as follows:

1. The cause of the fuel element plug destruction was an increased hydrogen pick-up in the heat affected zone or, possibly, the delayed hydrogen cracking.

2. It was established that increased pitting with a depth of 120-130 μm took place in the area of the lower spacer grids of stainless steel. This data refers to the fuel assembly with a burnup of 27 MW·day/kg and a service-life of 21932 hours. In some cases defect depth reached 360-340 μm.

The study of such zirconium SG areas showed that the defect depth was reduced to 100 - 120 μm.

Nodular corrosion shaped as lens with a depth of 70-90 μm maximum to 240 μm also occurs outside the spacer grids. The thickness of the oxide films is 10-30 μm in the area of the welded joints.

3. The ductility of the structure materials is at an acceptable level; δ general = 23 % at 300 °C and over 14 % at 25 °C; and δ uniform $\geq$ 3 % at fluence $\approx 10^{21}$ n/cm^2.

4. The hydrogen content does not exceed $3.8 \cdot 10^{-3}$ %; the hydride extent is 5-40 μm (in the direction of the rolling of the fuel cladding tubes).

5. The amount of the ferric oxide depositions on the fuel element surface outside the spacer grids is no more than 25 g/m^2. It is 50 g/m^2 in the spacer grid area.

6. 350 μm-deep corrosion pits were observed at the site of the contact of the assembly spacer grids with the channel tube.

7. No indications of the flaw development or changes in the material properties were detected in examination of an fuel assembly which had been in storage during 15 years /18/.

The complex post-irradiation studies of the RBMK fuel has allowed to trace the evolution of the characteristics of the Zr-1% Nb alloy.

With the permissible cladding deformation assumed to be 2%, an estimation may done for the permissible burnup increase. It will be 35 MW day/kg /19/.

We link the further work on the RBMK fuel with an achievement of higher burnup and an improved reliability of fuel element, and assembly operation in normal conditions and the study of the fuel behaviour in emergencies.

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FUEL PERFORMANCE IN ARGENTINE HEAVY WATER CHANNEL REACTOR OF PRESSURE VESSEL TYPE

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Abstract

Atucha 1 NPP is a pressurized heavy water reactor (PHWR) of 357 MW gross electrical power, natural uranium fuelled, cooled and moderated with heavy water, pressure vessel type with a moderator tank inside and 252 vertical coolant channels containing the 36 active rods type fuel assemblies (FAs) of full length (≈ 6 meter long).

The fuel rod design is basically the same as of LWR.

More than 7300 FAs (≈2,6 x 10^5 fuel rods) have been irradiated in the Atucha 1 NPP since 1974. The cumulative manufacturing fuel rod failure rate is estimated about 1,6 x 10^{-4} and the total cumulative fuel rods defect rate ranges 3 x 10^{-4}.

With the target of reduction of the front end fuel cycle cost, the gradual introduction of slightly enriched (0,85 w% ^{235}U) uranium (SEU) FAs was started on January 1995. At the present, 14 SEU Atucha 1 FAs were discharged with an average burnup of ≈10 MWd /kgU with satisfactory results and other 20 SEU FAs are in the core without abnormalities.

One project to increase (by ≈ 2,8%) the FA UO_2 weight by using an additional fuel rod and to reduce (by ≈29%) the volume of Zircaloy 4 used to manufacture the spacer grids is being performed to reduce fuel manufacturing cost.

1 INTRODUCTION

The Atucha-1 nuclear power plant (NPP) has a gross electrical power of 357 MWe and a thermal power of 1179 MWth. It is a Pressurized Heavy Water Reactor (PHWR) designed by Siemens (Germany), cooled and moderated with heavy water. It is a pressure vessel type reactor with a moderator tank inside and 252 vertical coolant channels containing the full length, fuel assemblies. The coolant channels separate the coolant from the moderator.

Some data of Atucha-1 PHWR are given in Tables 1 and 2. The reactor internals and the pressure vessel can be seen in Figure 1. The core transverse section is shown in Figure 2.

The coolant flow in different radial zones of the core is adjusted to the corresponding channel power generation by using variable flow orifices (throttles type) at the inlet of the channels. The core is subdivided in eight radial power distribution zones. The coolant flow in the fuel channel is decreased from the center to the periphery of the core according to the power channel to obtain approximately the same outlet channel temperatures. At the channel outlets there is no coolant boiling.

As natural uranium heavy water reactor, Atucha-1 NPP requires continuous on-power refuelling. The main core data are shown in Table 3. Withdrawal and loading of the fuel assemblies from and into the vertical coolant channel are performed by using one refuelling machine located on the top the reactor vessel. The control rods enter the core, between the coolant channels, in an oblique course, from the periphery of the reactor vessel head, to give space for working of the refuelling machine (Figure 1).

Power regulation is made by 6 absorber rods, three of hafnium and three of steel. There are other 23 hafnium rods for shutdown purposes.

TABLE 1.

ATUCHA-1 REACTOR TECHNICAL DATA

REACTOR TYPE:	pressurized heavy water reactor (PHWR) - pressure vessel type
THERMAL OUTPUT:	1179 MW
GROSS ELECTRICAL OUTPUT:	357 MWe
COOLANT AND MODERATOR:	D_2O
NUMBER OF FUEL CHANNELS:	252
NUMBER OF FUEL ASSEMBLIES:	252 full length
REFUELLING AND FUEL SHUFFLING:	continuous on-power

TABLE 2.

ATUCHA-1 REACTOR TECHNICAL DATA

COOLANT AND MODERATOR PRESSURE:	115 bar
INLET - OUTLET COOLANT TEMPERATURE:	265/300 °C
MODERATOR TEMPERATURE:	180 °C
NUMBER OF CONTROL RODS:	26 Hafnium + 3 Steel
COOLANT CHANNEL INTERNAL DIAMETER:	108,2 mm (Zry-4)
COOLANT CHANNEL WALL THICKNESS:	1,7 mm

TABLE 3.

ATUCHA-1 CORE TECHNICAL DATA	
FUEL TYPE:	**natural uranium dioxide**
TOTAL URANIUM LOADING:	**38,9 tU**
CORE AVERAGE CHANNEL POWER:	**4,68 MW**
MAXIMUN CHANNEL POWER:	**7,03 MW**
CORE AVERAGE LINEAR FUEL ROD POWER:	**232 W/cm**
AVERAGE FA DISCHARGE BURNUP:	**5,9 MWd/kgU**
REFUELLING FRECUENCY:	**1,3 FA/full power day**
PELLET PEAK DISCHARGE BURNUP:	**8,4 MWd/kgU**

The moderator pressure is the same that the inlet coolant pressure. Therefore, the coolant channel supports only the pressure difference between inlet and outlet. The reactor pressure is supported by the pressure vessel.

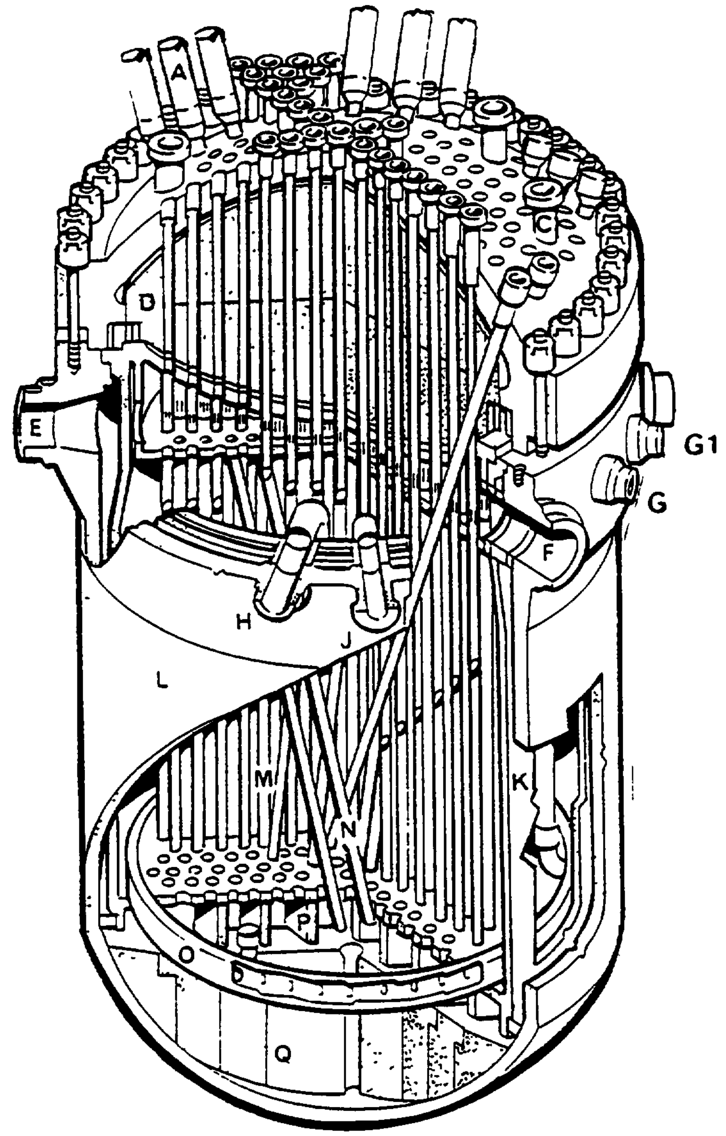
Atucha-1 NPP started the operation in 1974. The reactor power was uprated about 8% in 1977. The thermal output was increased from 1100 MW to 1179 MW and the gross electrical power to 357 MWe.

2 DESCRIPTION OF THE ATUCHA-1 FUEL ASSEMBLY

The standard FA design for Atucha-1 NPP consists of a configuration of 37 rod positions, distributed in three concentric rings and one central rod (see Figure 3). One position of the outer annulus is occupied by one supporting tube which replaces a fuel rod, as showed in Figure 4. The active length is 5300 mm and the overall FA length is 6029 mm. The main FA data is reported in Table 4.

The FA has 15 rigid (inelastic) spacer grids manufactured out of solid round plates of Zircaloy 4 by drilling and spark erosion. Three (or four, depending of spacer type) rigid sliding shoes made of Zircaloy 4 are brazed at spacer side.

The fuel rods are held in the guide holes of the spacers by wearing pads which are resistance welded to the cladding tubes at the spacer positions to prevent the fretting of the cladding. The wearing pads of nine outer fuel rods are also used to fix the axial location of the spacers. The FA is radially pressed against the wall of the coolant channel on two of the spacer fixed sliding shoes by a force from an opposite third spring-loaded sliding shoe. In such way, the support concept consists of a geometry of three points of contact between FA and the channel wall which is repeated in each spacer axial location. The springs forces are such that the relative displacement of FA in the radial direction is not allowed to avoid fretting-wear on the contact points and to reduce vibrations induced by coolant flow.The spring-



Key to reactor cutaway

A Control rods
B Channel seal
C Moderator downcomer

D Upper filler piece
E Inlet from coolant pump

F Outlet to steam generator

G Injection line upper plenum
G1 injection line downcomer

H Moderator outlet
J Fuel element detection system

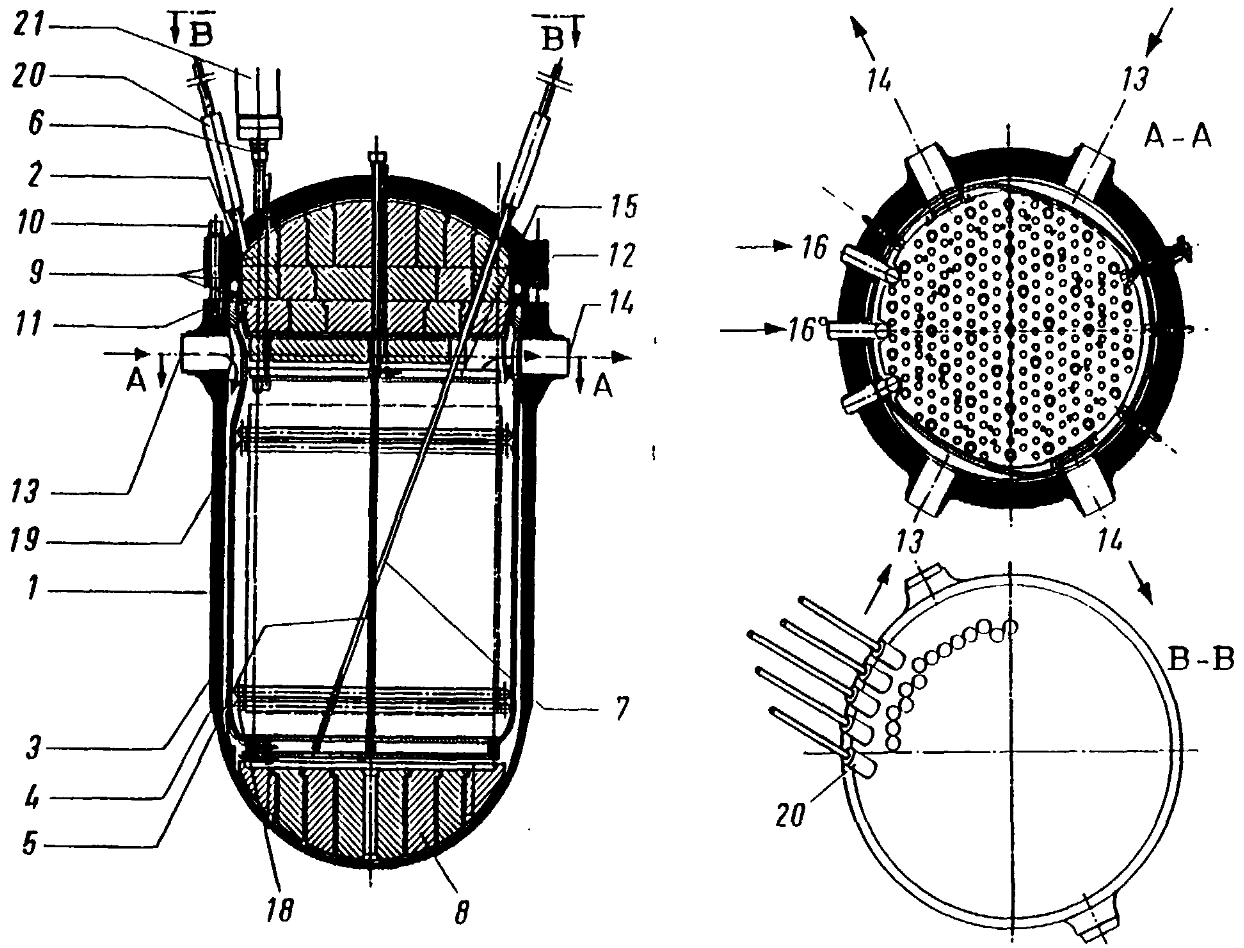
K Moderator tank
L Pressure vessel
M Fuel rods and coolant channels

N Control rod guide tubes

O Moderator distribution tube

P Support grid
Q Lower filler piece

FIG 1 Reactor pressure vessel and internals of ATUCHA 1 NPP



1 Reactor vessel
2 Vessel closure head
3 Moderator tank
4 Thermal insulation
5 Coolant channel with fuel element
6 Coolant channel closure
7 Control rod guide tube with control rod
8 Bottom filler elements
9 Top filler elements
10 Closure studs
11 Seal membrane
12 Intermediate closure ring
13 Reactor coolant inlet nozzle
14 Reactor coolant outlet nozzle
15 Moderator tank top head
16 Moderator inlet pipe
17 Moderator outlet pipe
18 Moderator inlet ring header
19 Moderator outlet ring header
20 Control rod drive mechanism
21 Refuelling machine

HEIGHT = 12,2 m

DIAMETER = 6,2 m

Reactor pressure vessel with internals

FIG. 2. Reactor transversal section of ATUCHA 1.

TABLE 4

ATUCHA-1 FUEL ASSEMBLY TECHNICAL DATA	
ASSEMBLY GEOMETRY:	circular array
QUANTITY PER FUEL ASSEMBLY:	
ROD POSITIONS:	37
FUEL RODS:	36
SUPPORTING TUBE:	1 of Zircaloy-4
RIGID SPACER GRIDS:	15 of Zircaloy-4
ADDITIONAL LOWER SPRING GRID:	Inconel 718
UPPER TIE PLATE:	Zircaloy-4

loaded sliding shoes are made of a special steel (heat treated by solution treatment and aging). They are attached to the supporting tube of Zircaloy 4 and axially distributed above each one of the fifteen spacers. The fuel rods and the supporting tube are suspended from the upper tie plate made of Zircaloy 4.

The lower end of FA suffers the highest hydraulic forces because it is near the coolant flow inlet. To prevent higher vibrations and fast wear, one additional spring-loaded sliding shoe is attached on the side of the lower spacer to increase the fuel assembly press-force against the channel at this position. In the same way, in order to press the fuel rods into each cell hole of lower Zircaloy spacer, one additional spring type grid, made of Inconel 718 is pushed into the bottom end of the FA Above the upper tie plate, the FA has an structural extension made of six Zircaloy 4 tubes It is followed by three steel rods and then by the upper coupling piece, as is shown in Figure 3.

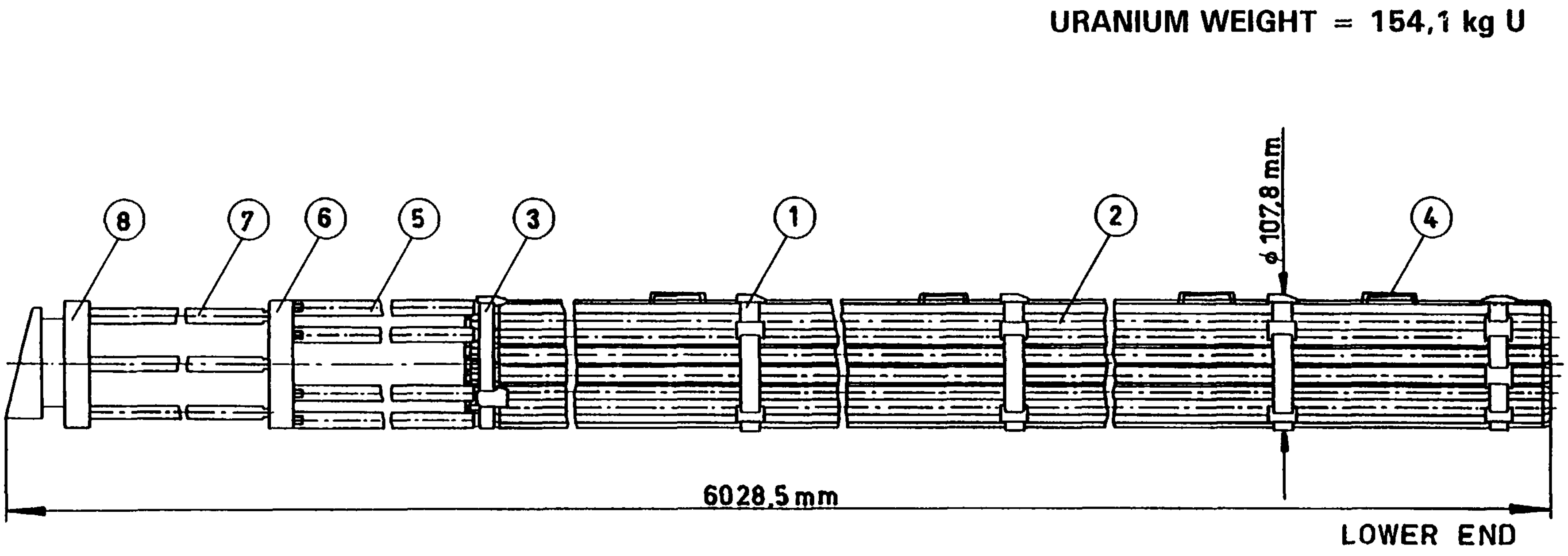
The upper end of the fuel assembly is coupled to a filler body of steel and this is also coupled to the closure end plug of the coolant channel, as is shown in Figure 5. The total fuel column is suspended from the closure plug at the upper end of the coolant channel. The full fuel column is withdrawn and reloaded as a whole in the refuelling operations.

3 DESCRIPTION OF THE ATUCHA-1 FUEL ROD

The current design of the Atucha-1 fuel rod is very similar to the design of a Light Water Reactor (LWR) fuel rod as is shown in Figure 6. The main data of fuel rod are in Table 5.

The fuel cladding is free-standing (non-collapsible). The fuel rods are pre-filled with helium gas at 17 bar through a hole in the lower end plug, which is then weld closed. Both end plugs are welded by using tungsten inert gas (TIG) process.

It is predicted by calculations that during the stationary long time irradiation up to the natural uranium burnup discharge the pellet-cladding diametrical gap never disappears and there are no mechanical interactions



1. SPACER GRIDS
2. FUEL RODS
3. TIE PLATE
4. SPRING-LOADED SLIDING SHOES
5. ZIRCALOY 4 TUBES
6. STEEL RING
7. STEEL RODS
8. COUPLING PIECE

FIG. 3. Fuel assembly of ATUCHA 1 NPP.

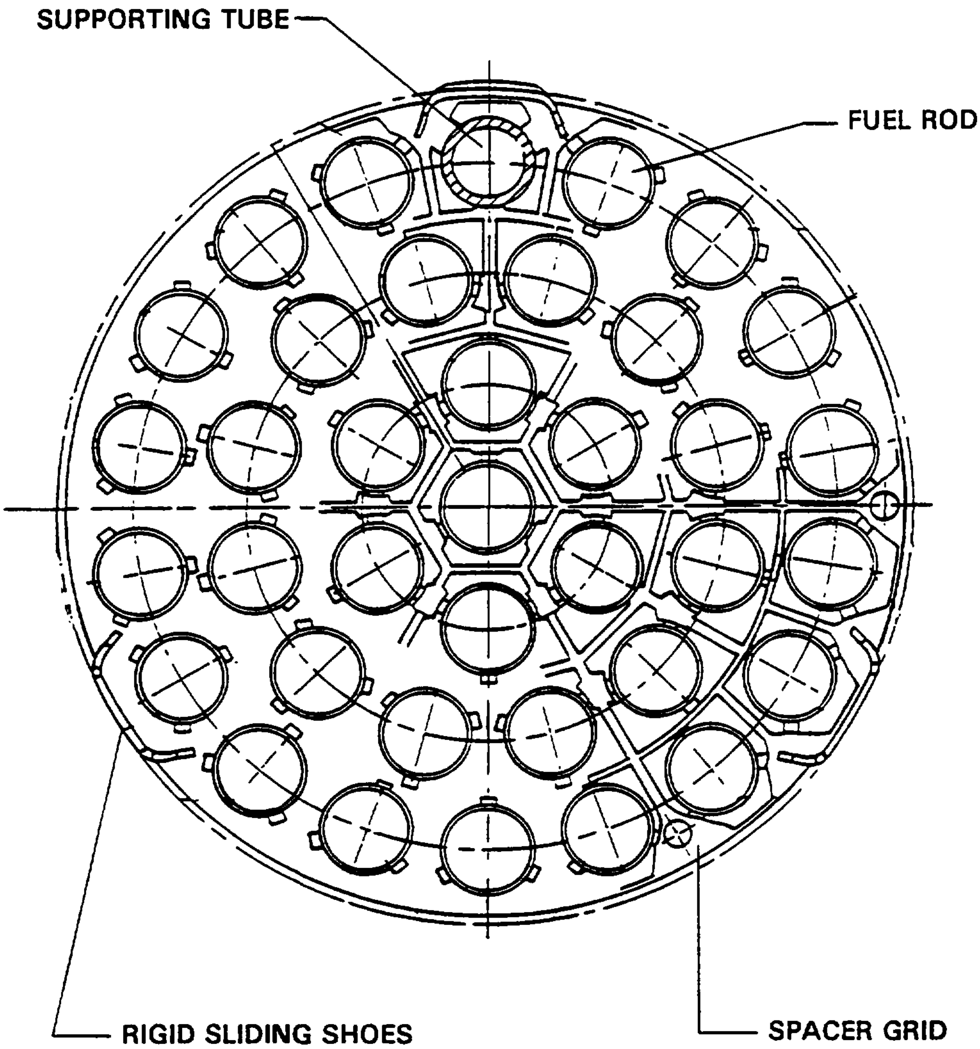


FIG. 4. Transversal section of ATUCHA 1 fuel assembly.

During short power ramps, along the burnup range corresponding to natural uranium, under the most of the design conditions, the fuel cladding is free of tensile stresses. Only under severe combination of design conditions and highest refuelling power ramps, the calculations results give a low amount of cladding total tensile strain (0,3 - 0,4 %).

4 FUEL MANAGEMENT STRATEGY

At the present, the Atucha-1 on power fuel reloading scheme for natural uranium applies a subdivision of the core on three concentric-annular zones numbered from the center (zone 1) to periphery (zone 3) as is shown in Figure 7.

The FAs exchange is done by a one-radial route and each FA is moved two times (or sometimes three times) to different locations. Fresh fuel is loaded in zone 2 (Intermediate power density) up to get an average FA burnup of about 2,7 MWd/kg U. After that, the fuel is moved to zone 1 (central, higher power density) suffering a positive power ramp. The FA stays in zone 1 up to about 5,1 MWd/kg U and then is reshuffled to zone 3 (peripheral, lower power density) up to about 5,9 MWd/kg U (peak pellet burnup of 8,4 MWd/kg U). Later, the FA is discharged of the reactor, as is showed in Table 6.

This fuel management scheme was foreseen to obtain maximum FA average discharge burnup fulfilling operational limits for local linear power density, channel critical heat flux ratios and power ramps.

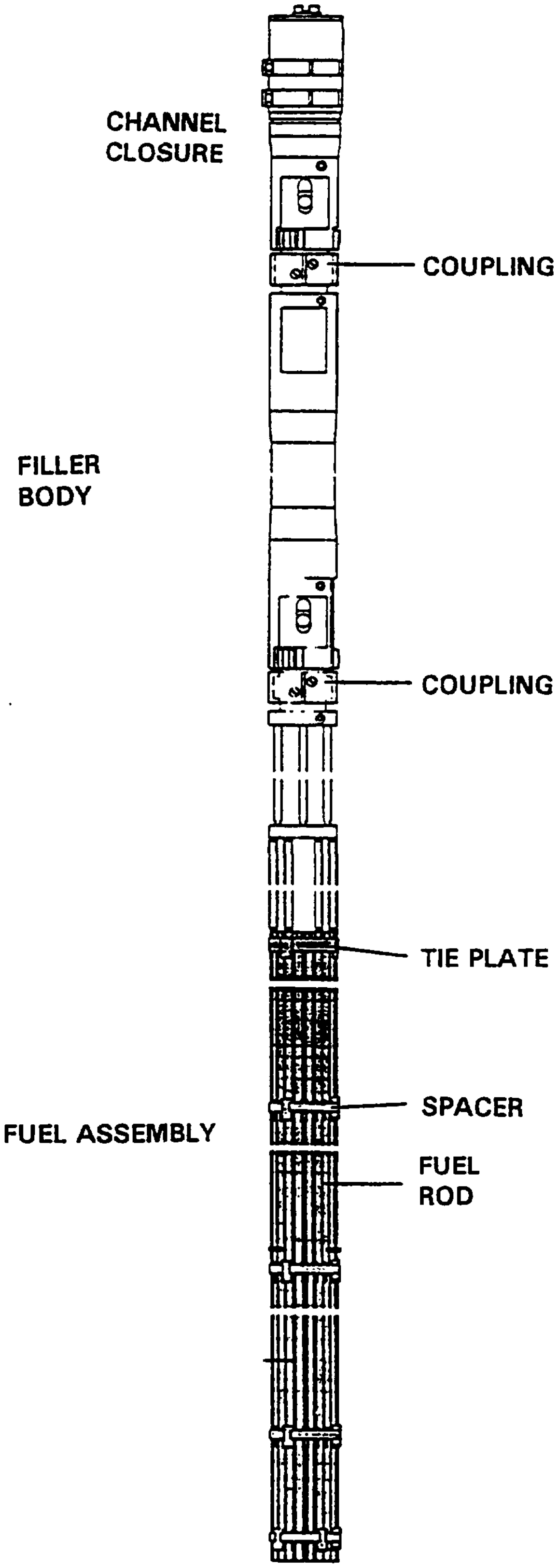


FIG. 5. Total fuel column of ATUCHA 1.

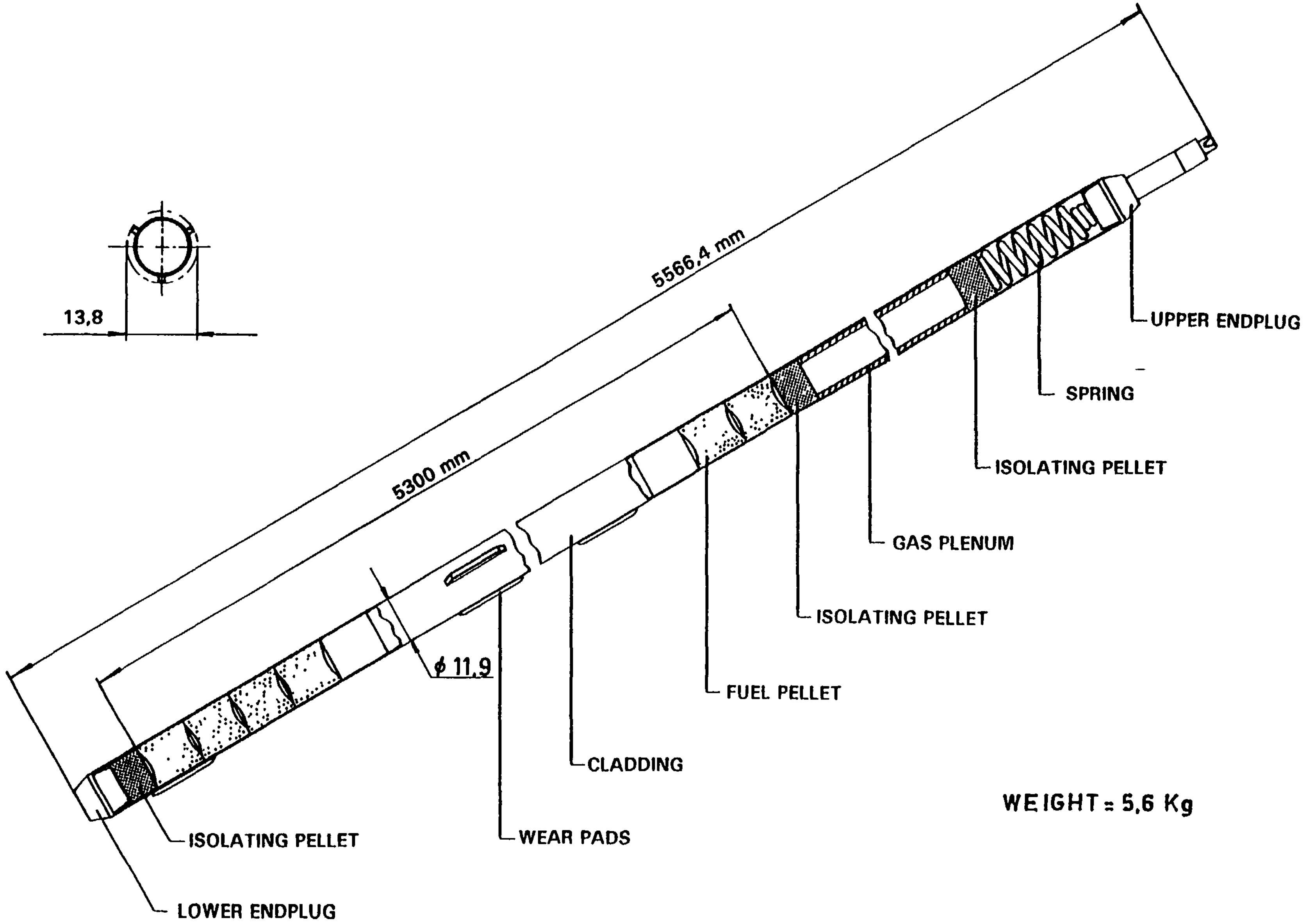


FIG 6 ATUCHA 1 fuel rod

TABLE 5.

CNEA-UPESN-IEC	TABLE 5
ATUCHA-1 FUEL ROD TECHNICAL DATA	
FUEL ROD OUTSIDE DIAMETER:	11,90 mm
CLADDING WALL THICKNESS:	0,55 mm
ACTIVE LENGHT:	5300 mm
UO_2 PELLET DENSITY:	10,55 g/cm^3
FUEL PELLET DIAMETER:	10,62 mm
CLADDING MATERIAL:	Zircaloy-4
AVERAGE FUEL ROD HEAT RATE:	232 W/cm
MAXIMUM FOR STATIONARY CONDITION:	531 W/cm
MAX. FOR NON-STATIONARY CONDITION:	600 W/cm
MAXIMUM DESIGN PEAK POWER (+15 %):	690 W/cm
INTERNAL PRE-PRESSURIZED:	17 bar (helium)

5 DIFFERENCES BETWEEN PRESENT FUELS FOR LWRs AND ATUCHA-1

The most remarkable differences between Atucha-1 fuel and modern LWR fuels (see Table 7 and 8) are as following:

Low burnup

The average FA discharge burnup is about 5,9 MWd/kg U and the peak pellet discharge burnup currently is below 9 - 10 MWd/kg U.

High linear power density

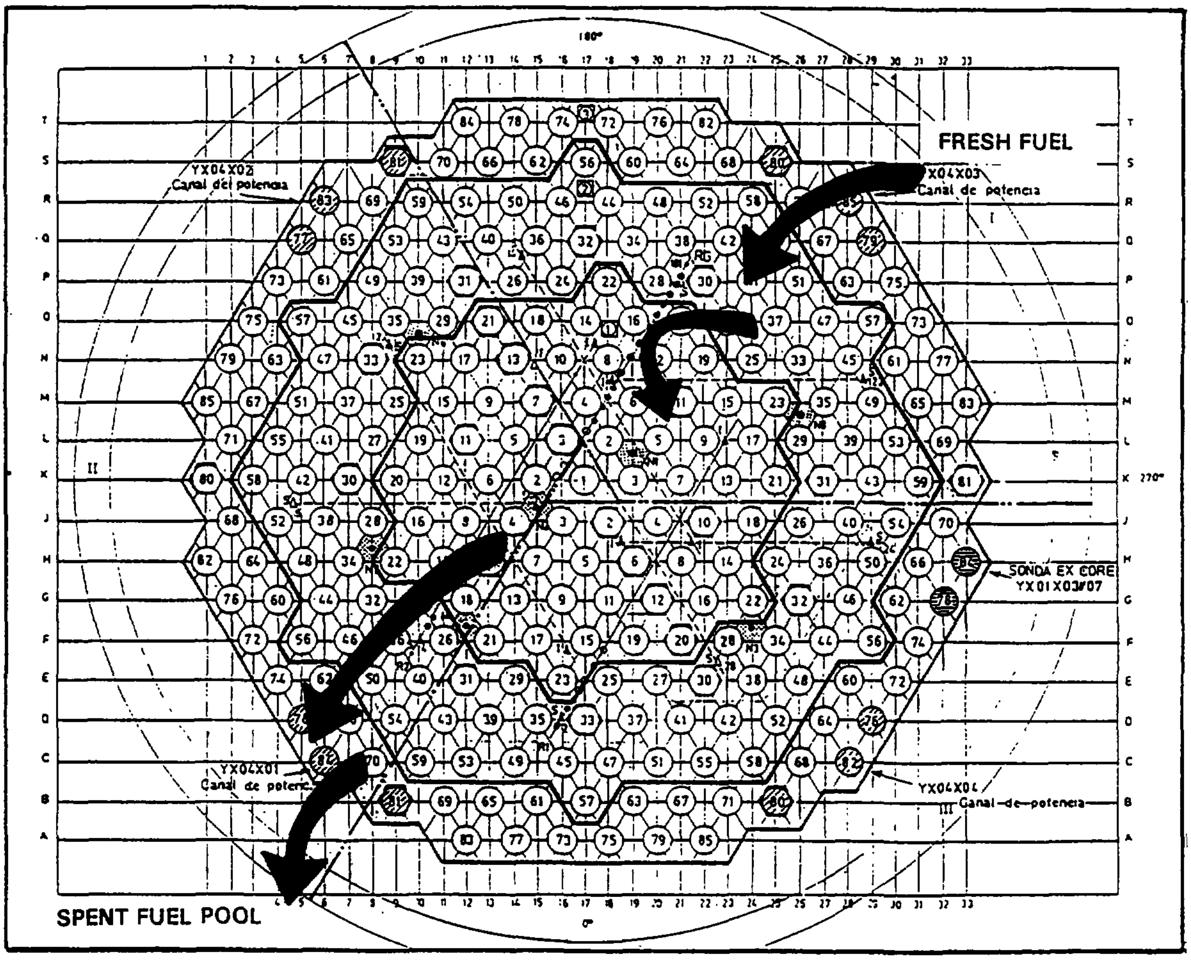
The core average linear power density is 232 W/cm and the design value for the local peak of linear power density under stationary conditions is 531 W/cm. The design value for the local peak of linear power density for non-steady state operational fluctuations is 600 W/cm (the hot channel factor is 2,59 to design verification).

Short dwelling time

The average residence time of the FA in the core is about 195 full power days (fpd).

Power ramps on fuel loading and shifting under power

Power ramp experienced by the FA when it is moved from intermediate core zone (2) to central zone (1) is of special concern because of the risk of fuel failures by pellet cladding interaction assisted by stress corrosion cracking (PCI - SCC). The most of the local power increments are in the range between 150 - 250 W/cm at an average FA burnups around 2,5 - 3,3 MWD/kg U and a peak pellet burnups of 3,6 - 4,7 MWd/kg U. The procedure of FA movement from zone 2 to zone 1 is subjected to operative restraints based on PCI - SCC fuel rod failure thresholds.



REFUELLING SCHEME:

FRESH FUEL → ZONE 2 → ZONE 1 → ZONE 3 → POOL

FIG. 7. Section of ATUCHA 1 core with refuelling zones.

TABLE 6. NATURAL URANIUM FUEL RELOAD AND SHUFFLING ON ATUCHA-1 CORE

CORE ZONE	LOADING BURNUP [MWd/kgU]	POWER RAMP ΔP	REMOVAL BURNUP [MWd/kgU]
2 (intermediate)	Fresh	$\Delta P > 0$	2,7
1 (central)	2,7	$\Delta P > 0$	5,1
3 (external)	5,1	$\Delta P < 0$	5,9
			DISCHARGE TO SPENT FUEL STORAGE POOL

Additionally, other differential features between the Atucha-1 and LWR fuels that should be pointed out are:

- Cladding wearing pads, each fuel rod has 45 wearing pads resistance welded to the cladding at spacer locations,
- Different channel coolant flows, as was indicated, the coolant velocity and mass flow of each channel depends on the different radial power distribution zones of the core,
  The worst hydraulic conditions for the FA depend on the combinations of coolant mass flow and the velocity profile at the channel inlet. The FA is required on different grades by each throttle type.

6 ATUCHA-1 NPP Overall Fuel Performance

More than 7300 fuel assemblies (< 2,63 x 10^5 fuel rods) from three different suppliers have been irradiated in the Atucha-1 reactor since 1974. The overall performance is shown in Table 9

The fuel failure rate was generally very low with some periods of enhanced failure frequency. The reasons for these enhancements were: fuel operation beyond limits, manufacturing flaws concentrations and interaction with reactor internals damaged (in 1988) and debris.

PCI - SCC failures were related with high power ramps either as result of fuel movement from intermediate to central core zone or during reactor start-up operations. Operation restrictions for fuel shuffling and reactor start-up were introduced to reduce the risk of PCI failures.

The overall rate of FA failures because of manufacturing faults is 0.56%. In our experience only in a few cases there were more than one fuel rod failed in each FA because of manufacturing flaws. Therefore the overall fuel rod manufacturing failure rate is 0.016% and almost the same, 0.014%, for the last 5 years.

Regarding the manufacturing failures, since hot cell examinations are not available, the primary causes of the failures were never exactly identified. There are some concerns on the wearing pad welding process and on the soundness of the fuel sheaths. Some recommendations to improve fuel manufacturing process and QC procedures were introduced. At the present, additional recommendations are being developed.

7 EXTENDED BURNUP BY USING SEU FUEL

The fuel discharge burnup of Atucha-1 NPP can be increased by using SEU in fuel reload. The main advantages from the use of SEU on Atucha-1 core, as illustrated in Table 10, are:

- Extension of fuel discharge burnup
- Reduction in the spent fuel volume
- Reduction in the total fuel cost
- Reduction in the frequency of (on-power) refuelling

Also, during the transition to full SEU core there is an increase in the discharge burnup of natural uranium fuel Reloading of spent natural uranium FAs to extend their original low discharge burnup is also under analysis (see Table 11).

A program for loading SEU fuel assemblies at 0,85 W % ^{235}U in Atucha-1 NPP was started in January 1995 [1]. Several changes have been introduced to the design of both, fuel rod and fuel assembly to adapt them to the SEU requirements , as is shown in Table 12. The aim of these changes was to provide more space for gas release and to assure reliably interaction between spacers and wearing pads during the whole life of the fuel. Other changes, like modifications to the specification of the fuel sheath

TABLE 7.

MOST IMPORTANT DIFFERENCES BETWEEN ATUCHA-1 NATURAL URANIUM FUEL AND MODERN LWR FUELS

- **LOW DISCHARGE BURNUP**

AVERAGE:	**5,9 MWd/kgU**
PEAK PELLET:	**< 9 - 10 MWd/kgU**

- **HIGH LINEAR FUEL HEAT RATE**

CORE AVERAGE:	**232 W/cm**
LOCAL PEAK ON NON-STATIONARY CONDITION:	**600 W/cm**

- **SHORT DWELLING TIME**

FA AVERAGE RESIDENCE TIME:	**195 fpd**

TABLE 8.

OTHER DIFFERENCES BETWEEN ATUCHA-1 AND LWR FUELS

- **POWER RAMPS AS A CONSEQUENCE OF ON - POWER REFUELLING AND FUEL SHIFTING**

LOCAL POWER INCREMENT:	**close to 150 - 250 W/cm**
AVERAGE FA BURNUP:	**2,5 to 3,3 MWd/kgU**
PELLET PEAK BURNUP:	**3,6 to 4,7 MWd/kgU**

- **DIFFERENTS CHANNEL COOLANT MASS FLOW AND VELOCITY**

  THE MASS FLOW RANGES FROM 8,5 TO 32,9 kg/s

  THE FUEL ASSEMBLIES ARE REQUIRED ON DIFFERENT GRADES ON EACH THROTTLE TYPE

- **FUEL CLADDING WEAR PADS**

  EACH FUEL ROD HAS 45 WELDED WEAR PADS

TABLE 9. ATUCHA 1 NPP FUEL PERFORMANCE

FUEL SUPPLIER	IRRADIATED FA	FUEL FAILURES							
		MANUFACTURING		PCI-SCC		OTHER (2)		TOTAL	
		FA	RATE [%]	FA	RATE [%]	FA	RATE [%]	FA	RATE [%]
RBU (SIEMENS) 1974 - 1983	3314	19	0.67	3	0.09	4	0.12	26	0.79
CNEA (PPFECN) 1978 - 1981	240	2	0.83	0	0	2	0.83	4	1.66
CONUAR 1982 - 30/11/1996	3755 (1)	20	0.53	11	0.29	14	0.37	45	1.20
TOTAL	7309	41	0.56	14	0.19	20	0.27	75	1.03

Notes: (1) Including 14 SEU Fuel Assemblies
(2) Failures for interaction with reactor internals, debris or another foreign materials

TABLE 10.

MAIN ADVANTAGES FROM THE USE OF SLIGHTLY ENRICHED URANIUM ON ATUCHA-1 CORE
• EXTENSION OF FUEL DISCHARGE BURNUP
• REDUCTION OF SPENT FUEL VOLUME
• LOWER TOTAL FRONT END FUEL CYCLE COST
• LOWER FRECUENCY OF REFUELLING

TABLE 11.

ADDITIONAL ADVANTAGES ON CORE ENRICHMENT ON ATUCHA-1 DURING THE TRANSITION TO HOMOGENEOUS SEU CORE
• INCREASING OF DISCHARGE BURNUP OF THE NATURAL URANIUM FUEL
• POTENTIAL RELOADING OF SPENT NATURAL FUEL ASSEMBLIES (STORED IN POOL) TO EXTEND THEIR BURNUPS

TABLE 12.

CHANGES TO ATUCHA-1 FUEL DESIGN FOR USING SEU (PRESENT STATUS)
• PLENUM SIZE WAS INCREASED TO REDUCE THE INTERNAL FUEL ROD GAS PRESSURE
• CLADDING DUCTILITY WAS INCREASED TO REDUCE PCI-SCC FAILURES SUSCEPTIBILITY ON POWER RAMPS
• THE MATERIAL OF SPRING-LOADED SLIDING SHOES WAS REPLACED BY INCONEL 718 TO COMPENSATE THE HIGHER RELAXATION OF COOLANT CHANNEL - FA CLAMPING FORCES

material, were directed to reduce the fuel rod susceptibility to PCI failures. A thin coating of graphite over the internal surface of the fuel sheath will be introduced in the future with the same objective.

The original material (Steel A286) for the spring-loaded sliding shoes was replaced by Inconel 718 to compensate the higher relaxation produced by the increase of the fuel life into the reactor.

PCI prevention criteria for natural uranium fuel was updated to SEU fuel.

At the present, 14 SEU fuel assemblies were discharged with an average FA burnup of about 10 MWd/kg U and other 20 SEU FA are under irradiation.

The performance during irradiation and the results of post-irradiation examination and measurements in the spent fuel pool were very satisfactory. Irradiation of the 20 SEU FAs continues at the present without abnormalities.

During the present stage of the program, the amount of SEU FAs into the core will gradually be increased up to 60 (≈24% of core) until the end of 1997. After that, the number of SEU FAs in the core will be gradually increased to a full core in about three years (2000).

Table 13 shows the comparison between the discharge burnups, dwelling times and refuelling frequency of natural uranium and SEU fuel assemblies. When the core becomes fully SEU 0,85 W% ^{235}U, it is estimated that the front end fuel cycle component of the generation cost will go down by about 20 - 25%.

8 URANIUM MASS INCREASE OF ATUCHA-1 FUEL

In order to reduce the fuel cost there is a program to increase the UO_2 weight of the fuel assemblies by using an additional fuel rod instead of the supporting tube of present design (see Figure 4 and Table 14).

TABLE 13. EXTENDED BURNUP OF ATUCHA-1 FUEL

	NATURAL URANIUM	SEU 0,85 % ^{235}U
AVERAGE DISCHARGE BURNUP [MWd/kgU]	5,9	≈ 11
PELLET PEAK DISCHARGE BURNUP [MWd/kgU]	8,4	≈ 16
FA AVERAGE RESIDENCE TIME [fpd]	195	362
REFUELLING FRECUENCY [FA/fpd]	1,3	0,7

TABLE 14.

OTHER ATUCHA-1 FUEL DESIGN CHANGES WICH ARE UNDER DEVELOPMENT
• UO_2 WEIGHT INCREASE TO ≈2,8 % BY USING 37 FUEL RODS PER ASSEMBLY (PRESENT DESIGN: 36)
• SPACER THICKNESS REDUCED TO ≈28,6 %

The spring sliding shoes made of Inconel 718 will be attached to each spacer grid to replace present spring sliding shoes mounted on the supporting tube.

The uranium content per fuel assembly will be increased by about 2,8%.

Other minor advantages by using a 37 fuel rods assembly are a reduction of frequency of refuelling and the number of spent FAs (see Table 15) In the same way, with the target of reduction the fuel cost, other design modifications are under analysis as reducing the thickness of the spacer grids and tie plate by about 28,6% The volume of Zircaloy discs feeding to manufacture the spacer grids will be reduced in the same ratio

CNEA is planning to perform low and high pressure loop tests to evaluate the hydrodynamics and endurance behavior of the 37 fuel rods assembly with spacer grids of reduced thickness. After out-of-pile testing, irradiation of a first series of SEU 37 fuel rods assemblies will be done in Atucha-1 reactor to combine the advantages of both developments.

TABLE 15

ADVANTAGES FROM THE USE OF
37 FUEL RODS ASSEMBLIES
WITH REDUCED SPACER GRID THICKNESS
FOR ATUCHA-1 NPP

- **LOWER FUEL COST**
- **LOWER FRECUENCY OF REFUELLING**
- **REDUCTION OF NUMBER OF SPENT FUEL ASSEMBLIES**
- **POTENTIAL INCREASING OF DISCHARGE BURNUP**
- **FLATTENING OF FA LINEAR POWER DENSITY**

9 CONCLUSIONS

The general fuel performance in our Atucha-1 NPP was acceptable without any significative impact on the availability of the power plant As seen, the rate of defective fuel rods because of manufacturing flaws was about 1.6×10^{-4} and the overall fuel rod failure rate was 2.8×10^{-4}.

Despite the fact that this low manufacturing rate represents only one or two defective fuels per year, performance should be enhanced through the continuous improvement of product specifications, quality control procedures and well developed manufacturing process.

The overall fuel failure rate was heavily influenced by the concentration in short periods of defective fuels from PCI-SCC, debris or manufacturing faults.

The fuel supplier and the utility are working in a close cooperation to avoid or to reduce conditions that may allow PCI-SCC or debris defects to occur.

The in-pile performance of SEU (0.85% ^{235}U) fuel assemblies at the present stage of the program is encouraging, particularly, the response to power ramps during on power refueling. The operation of the plant showed no unexpected impacts on plant operation due to the use of SEU fuel, as was predicted Although only 14 SEU FAs have been irradiated, a slight increase in the discharge burnup of natural uranium fuel was noticed by the utility.

Programs to increase the UO_2 weight of the FA by using 37 fuel rods and to reduce the spacer thickness are under going.

Irradiation in Atucha-1 reactor of the first series of SEU 37 fuel rods assemblies is planned after out- of-pile testing of the new design.

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TRANSIENT THERMAL HYDRAULIC BEHAVIOUR OF THE FUEL BUNDLES DURING ON-POWER UNLOADING OPERATION IN THE PROPOSED 500 MWe PHWR

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Abstract

One of the main objectives under design and development of fuel in water cooled nuclear reactors is to ensure fuel integrity during spent fuel handling operation. The on-power refuelling facility adopted in the Indian Pressurized Heavy Water Reactors (PHWRs) causes exposure of the irradiated fuel, during its unloading, to wide variations in its surroundings including exposure to dry gaseous environment. Detailed analyses have been carried out to assess the fuel pin temperature transients during the entire course of its passage from within the reactor to the outside surroundings to ascertain fuel integrity. The cases of normal as well as envisaged off-normal transport operations have been considered in these calculations. The forced air cooling provisions have also been worked out to mitigate the consequences of off-normal transport operation. The present paper deals briefly with the system description, method of calculations and the results obtained for the case of spent fuel handling in the proposed 500 MWe PHWR.

1.0 INTRODUCTION

The Indian Pressurized Heavy Water Reactors (PHWRs) are provided with on-power refuelling facility. During on-power fuel loading/unloading operation, the fuel bundles very often get exposed to wide variations in their environment, both within as well as outside the reactor core. Thermal-hydraulic analysis of the fuel bundles, within the reactor core, under normal operating conditions as well as during anticipated transients is carried out as a part of the reactor core design analysis. In addition, an understanding of the thermal hydraulic behaviour of the fuel bundles during refuelling operation becomes essential to ensure the integrity of the fuel under normal/off-normal operating conditions. Particularly, unloading of the irradiated fuel bundles from the heavy water filled reactor system and transporting it to the external light water environment in the storage bay is of significant importance as the bundles need to pass through dry gaseous surroundings. During this phase of transfer, due to significant decay heat generation rates in the irradiated fuel and due to poor heat transfer rates in gaseous environment, the fuel pin temperature rises sharply. For maintaining the integrity of the irradiated fuel pins the rise in the fuel pin temperature must be limited to permissible values. In addition, consequences of fuel bundles getting exposed to dry air for durations exceeding the designed transport time for normal transport operation also need to be assessed so as to arrive at effective mitigating schemes. All these aspects have been studied analytically in detail to ensure that the fuel-bundles and the refuelling / fuel transport system of the proposed 500 MWe prototype PHWR meet the specified design criteria.

2.0 DESCRIPTION OF THE FUEL TRANSFER SYSTEM

The Indian PHWRs consist of a calandria vessel with several coolant channel assemblies, consisting of horizontal pressure tube surrounded by a concentric calandria tube. The 500 MWe PHWR will have thirteen numbers of 37 rod fuel clusters (Fig. 1) housed inside the pressure tube (Fig. 2). Two fuelling machines (FM), one at each end of the reactor facilitates on-power refuelling. The spent fuel clusters are removed from the reactor core by the FM and transported to a low temperature heavy water tank (fig. 3), called Transfer Magazine (TM). From the TM, the fuel clusters are transferred through stagnant air environment to the Shuttle Transfer Station (STS), which is a light water pool. The detailed breakup of typical transport operation along with a brief description of the environmental conditions at different locations is given in Table-1. In order to analyze for the complete temperature transients during the transfer operation, it is essential to study some of the important thermal hydraulic aspects during different stages of spent fuel transfer operation.

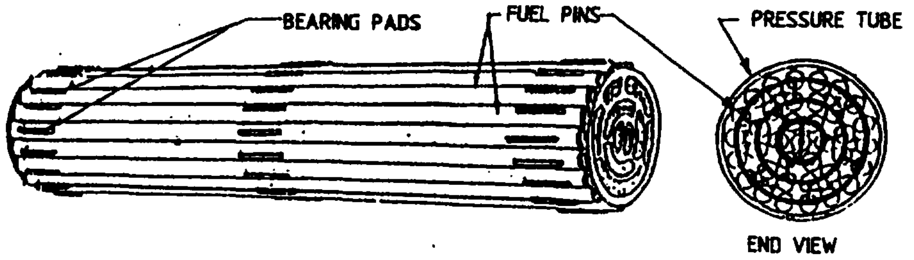

FIG. 1. 37-pin fuel-cluster for proposed 500 MWe PHVR..

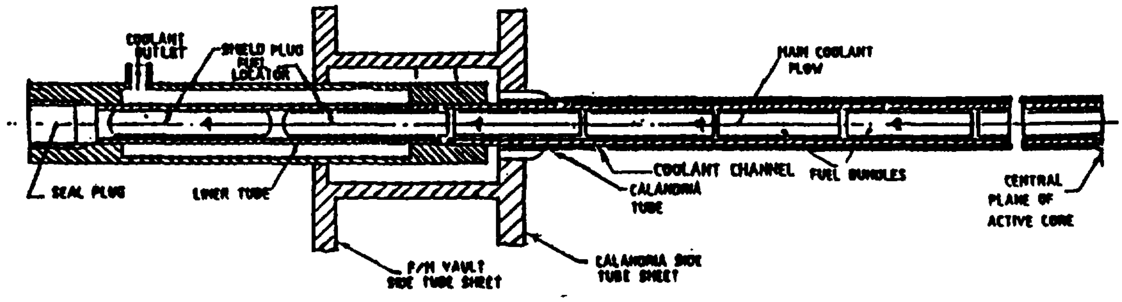

FIG. 2. Schematic of coolant channel of 500 Mwe PHNR.

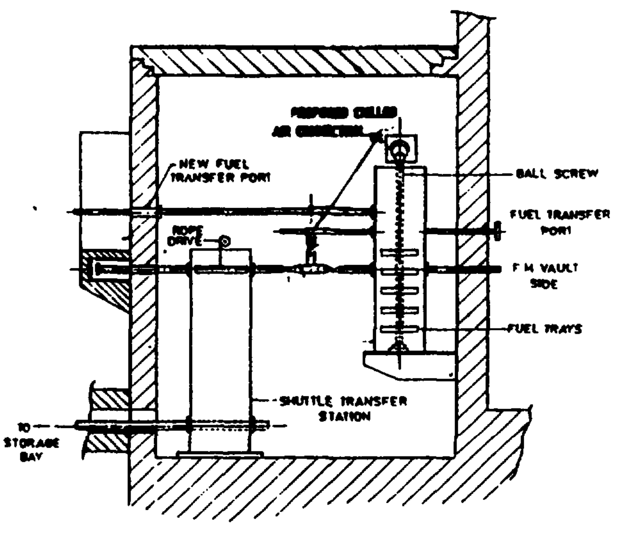

FIG. 3. Fuel transfer system layout (proposed) for 500 MWe PHVR..

TABLE 1. DETAILS OF SPENT FUEL BUNDLE TRANSPORT FROM COOLANT CHANNEL TO STS

Location of Central Bundle under Consideration	Time period (seconds)	Coolant medium and relative coolant flow rate w.r.t. normal channel flow	Temperature (° C)	Pressure (bar)	Mode of Heat Transfer
1. Coolant Channel					
a) From central location to primary coolant exit	0.0 to 320	Primary coolant 100 %	varies from 282 to 304 (*)	113	Forced covection (Turbulent flow)
b) Primary coolant exit to the exit of coolant channel	320 to 550	Fuelling machine coolant 2.5 %	inlet 93	113	Forced convection (Turbulent flow)
2. Fuelling Machine Magazine	550 to 2400	Fuelling machine coolant 2.5 %	93	varies from 1 to 3	Forced convection (Laminar flow)
3. Transfer Magazine	2400 to 3300	Stagnant heavy water	40	1	Natural convection
4. Tube connecting TM and STS	3300 to 3480 (design stipulated)	Stagnant air	50	1	Radiation

* Detailed temperature distribution obtained from steady state analysis of single channel producing maximum power.

3.0 THERMAL HYDRAULIC CONSIDERATIONS

Following are some of the thermal-hydraulic considerations which need to be considered in the transient analyses to be carried out.

a) Within the active core region of the reactor, the thermal energy produced in the fuel clusters is large due to nuclear fission, and this varies axially. Outside the active core, the spent fuel bundles produce only decay heat which is much less and reduces with time. The resulting power variations for the maximum power bundle in the hottest channel as well as the variations in the coolant flow rate and the coolant temperature as time dependent functions are arrived at considering the speed at which the bundles move. These are shown in Fig. 4.

b) From Table-1 it can be noted that the boundary conditions over the pins of the spent fuel cluster vary from forced convection inside the coolant channel and the FM to natural convection in the TM at different surrounding temperatures and finally radiation to stagnant air. These are required to be appropriately taken into account during the analysis.

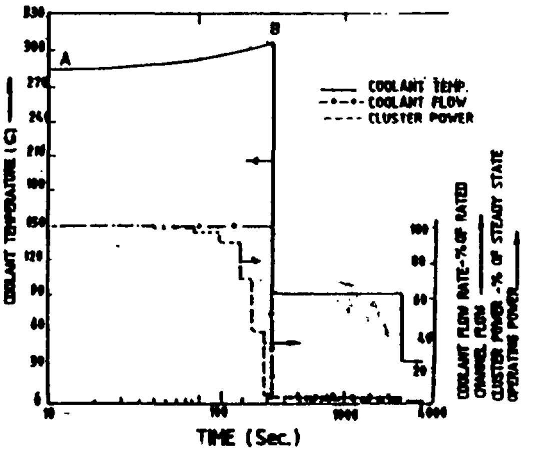

FIG. 4. Coolant flow rate coolant temperature and cluster power variation with time for central fuel cluster.

c) Since, the stagnant air environment is the worst thermal boundary condition encountered during spent fuel handling, the off-normal condition during transport operation has been postulated to be a prolonged stay of the spent fuel cluster in the stagnant air

4.0 METHOD OF ANALYSIS

The analysis involves [1] :

1) Estimation of the initial temperature distribution in the fuel cluster inside the core prior to the beginning of the transport operation.

2) Detailed transient analysis to predict the temperature transients considering the varying boundary conditions until the bundle reaches the STS.

3) Steady state analysis to predict the maximum temperature attained asymptotically by the fuel cluster following prolonged stay in stagnant air.

4.1 Initial Temperature Distribution in the Fuel Cluster

The initial temperature distribution in the central fuel cluster is obtained by carrying out steady state sub-channel analysis of the full coolant channel, considering the axial profile, channel power, coolant flow rate and coolant inlet temperature. Computer code COBRA-IIIC [2], a thermal-hydraulic subchannel analysis code, is used for this purpose. The mean coolant temperature variations along the length of the channel, obtained from the above analysis, is plotted in Fig. 4, (curve AB) as the time dependent mean coolant temperature around the central fuel cluster, considering its speed of movement.

4.2 Spent Fuel Behaviour Under Normal Transport Operation

4.2.1 Temperature Transients In The Coolant Channel, FM and TM

For further detailed transient analysis of the thermal behaviour during the movement of the fuel cluster from the core to the TM, analysis is carried out for a single pin of the cluster assuming average coolant flow per pin. The analysis is based on the solution of one dimensional transient heat conduction equation with time dependent heat generation rate and convective boundary condition. The details of the solution technique are given in [3]. The Dittus-Boelter equation is used for estimating the forced convective heat transfer coefficient in the turbulent flow regime. For laminar forced convection, a correlation for fully developed laminar flow through ducts for constant heat flux boundary condition has been used [4]. Natural convection heat transfer coefficient is obtained using the appropriate correlation for horizontal cylinders [5].

Some of the important assumptions made are :

1) The changes in the convective boundary conditions with respect to time are instantaneous.

2) The water temperature remains constant in the FM and TM. This is justified due to large size of the heat sinks provided by these water pools

3) The power generation rate in the pins located in the different rings of the fuel cluster is different. Accordingly the pins in the 37 rod cluster are classified into four groups based on the radial power factor (i.e. ratio of pin power to average pin power). It is assumed that the individual pins within each of the four groups experience identical transient boundary conditions. Transient analysis has been carried out separately for a pin in each of these groups.

Detailed temperature distribution for individual fuel pins in the different rings of the fuel cluster have been obtained separately using the above stated method. The temperature transients obtained thus, when the fuel bundle leaves the TM at the end of 55 minutes, is taken to be the initial temperature distribution in the cluster when its transfer to the STS through stagnant air begins.

4.2.2 Transfer Of Fuel From TM to The STS

During this phase, the bundle passes through an empty tube surrounded by stagnant air at 50 °C and heat dissipation from the fuel pins is mainly by radiation Analysis for this stage has been carried out using the computer code RHEINA (Radiative Heat Exchange In Nuclear Assemblies) [6]. The code is capable of estimating steady state/transient temperature distribution within a rod cluster exposed to stagnant air on the basis of 2-dimensional (r, O) model for radiative heat exchange from pin to pin, pin to the surrounding tube and from the surrounding tube to the stagnant air outside. Following assumptions have been made for this analysis:

1) The fuel bundle is concentrically located in the connecting tube (at 50 °C) between TM and STS.

2) The temperature of the surrounding air remains constant at 50 °C.

3) Decay heat generation rate is constant at 10.5 kW This is the average of the values at 55 minutes and 70 minutes.

4) The emissivity of the clad surface is 0.436 [7] and that of the connecting tube is 0.6 [8].

Transient analysis has been carried out for the exposure of spent fuel bundle to stagnant air up to about 15 minutes, even though the normal exposure time in air is about 3 minutes only, to account for the abnormal conditions. Under certain other abnormal conditions the fuel bundle may be in stagnant air for a much longer time. For such conditions, to arrive at the upper bound of the temperature reached, the spent fuel bundle is assumed to remain in air for infinite time. In such a case, the maximum pin temperature would correspond to the asymptotically attained steady state temperature. Hence, steady state analysis has been carried out for the fuel bundle (having a constant decay power of 10.5 kW) held in air, using the code RHEINA.

5.0 RESULTS AND DISCUSSIONS

The results of the transient analysis are presented in Fig. 5 which shows the variation of the pin temperature with time, over a period of 3300 seconds, for the fuel pin (fuel and cladding lumped together) in the outer ring of the cluster. It can be observed from this figure that;

a) There is a small rise in the pin temperature as the cluster moves within the coolant channel from the centre to the end of the active core, and then the temperature starts decreasing as the bundle starts moving away from the active core. The initial temperature rise is mainly due to the rising coolant temperature as the cluster moves towards the core exit. The subsequent fall in the pin temperature within the active core region is due to the sharp fall in heat flux towards the end of the active core.

b) As the cluster passes past the perforated portion of the liner tube, where it gets surrounded by the FM coolant flow at 93 °C, the clad temperature falls sharply to about 104 °C.

c) When the cluster reaches the FM magazine tube, where it is surrounded by a stagnant pool of water, the pin temperature rises sharply and attains a steady state value of bout 192 °C. Subsequently, when the cluster reaches the TM with coolant at 40 °C, the pin temperature falls sharply once again and attains a steady state value of about 60 °C.

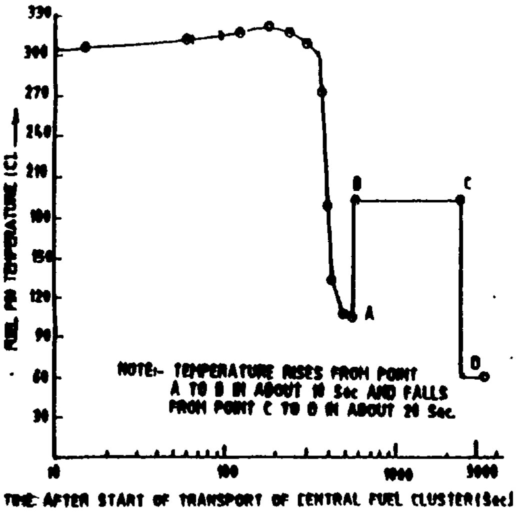

FIG. 5. Variation of fuel pin temperature for pins in the outer ring of the central cluster during transfer.

Temperature transients for the other three representative pins have also been generated in a similar way. The maximum attained temperatures in these pins are less than that for the pin in the outer ring and the temperatures of the central pin and the pins in the inner and intermediate rings at the end of 3300 seconds are 52 °C, 53 °C and 58 ° C respectively.

The transient temperature of the outer pins of the cluster, exposed to stagnant air at 50 °C from 300 seconds up to 4200 seconds, is depicted in Fig. 6. Due to the poor heat dissipation in the stagnant air, it is observed that there is a continuous rise in the pin temperature with time. The maximum attained pin temperature for the normal exposure time of about 3 minutes in air is observed to be about 305 °C, whereas for extended stay the temperature reaches to about 900 °C in 15 minutes. Results of steady state analysis for the constant bundle decay power of 10.5 kW indicates that the maximum temperature attained asymptotically will be about 1210 °C.

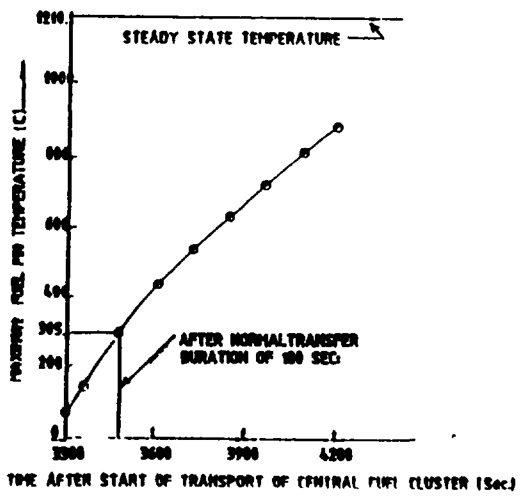

FIG. 6. Maximum fuel pin temperature transients during exposure to air.

6.0 MITIGATION DURING OFF-NORMAL TRANSPORT OPERATION

From the results described above, it is seen that extended stay of the fuel bundle in a dry air environment can lead to very high temperatures. Hence, mitigating measures need to be taken. For the proposed 500 MWe PHWR , one of the emergency measures which has been considered during such an event of off-normal transport operation is to blow chilled air through the bundle To assess the effectiveness of such a measure, detailed thermal analysis has been carried out to generate data in respect of required air flow rates, cooling times, resulting fuel pin temperatures etc. These calculations are based on the fact that as a result of the high temperatures attained by the fuel pins, the radiative heat transfer plays a significant role in addition to the forced convective heat transfer due to blowing of air. The computer code RHEINA can handle the combined boundary conditions of radiation and forced convection together. Hence, the analysis has also been carried out using the code RHEINA. Some of the simplified assumptions for these analyses are listed below.

a) Forced convection cooling by chilled air blowing is started one minute after the normal transport operation duration (i.e. 3 minutes) of spent fuel bundle in dry environment is over. Thus, the bundle remains in air for about 4 minutes before external cooling is initiated.

b) The air flows through the bundle axially, i.e. parallel to the fuel pins, and the convective heat transfer coefficient on the fuel pin surface is obtained on the basis of mean velocity of air through the bundle, using Dittus-Boelter correlation.

c) Fuel and clad are lumped together. Axial conduction in the lumped fuel pin and in the connecting tube has been neglected.

d) The temperature of air at the bundle inlet is 50 °C. (for getting conservative results)

e) The ambient temperature of air outside the connecting tube remains constant at 50 °C.

f) Emissivity of zircalloy and stainless steel are 0.436 [7] and 0.60 [8] respectively.

g) The connecting tube between TM and STS is considered to be concentric cylindrical channel around the bundle within.

The transient analysis has been carried out for different air flow rates for the maximum power bundle for which the decay heat generation rate is taken to be 10 kW

The transient temperature of the lumped fuel pin and air temperature at the exit of the bundle for different air flow rates ranging from 0 to 400 g/sec per bundle (at atmospheric pressure and at 50 °C) are shown in Fig. 7. As can be seen from this figure, the fuel pin temperature rises to a value of about 1100 °C after about 45 min. if there is no cooling air flow (curve-a). The results indicate that an airflow rate of about 50 g/sec per bundle is adequate to restrict the fuel temperature to about 400 °C. the corresponding exit temperature of air is observed to be about 200 °C. Further increase in air flow rate results in lower fuel pin temperature with simultaneous decrease in exit air temperature. In order to limit the exit temperature of air to about 70 °C, air flow rate of about 400 g/sec per bundle is required. The corresponding fuel pin temperature attained is about 110 °C.

7.0 CONCLUSION

The analysis carried out indicates that the maximum pin temperature attained during the normal transport duration of 3 minutes in stagnant air is about 305 °C which is within permissible limits. However, for extended duration of transport in stagnant air, the pin temperature keeps rising and finally it is likely to attain a maximum temperature of about 1210 °C. asymptotically. Under these circumstances, it is essential to provide adequate external cooling (i.e. chilled air) arrangement to restrict the fuel temperature within the maximum permissible limit. While air flow rate of about 50 g/sec. per bundle is found to be adequate to limit the fuel temperature to about 400 °C, an air flow rate of about 400 g/sec. per bundle is needed to restrict the exit temperature of air to about 70 °C.

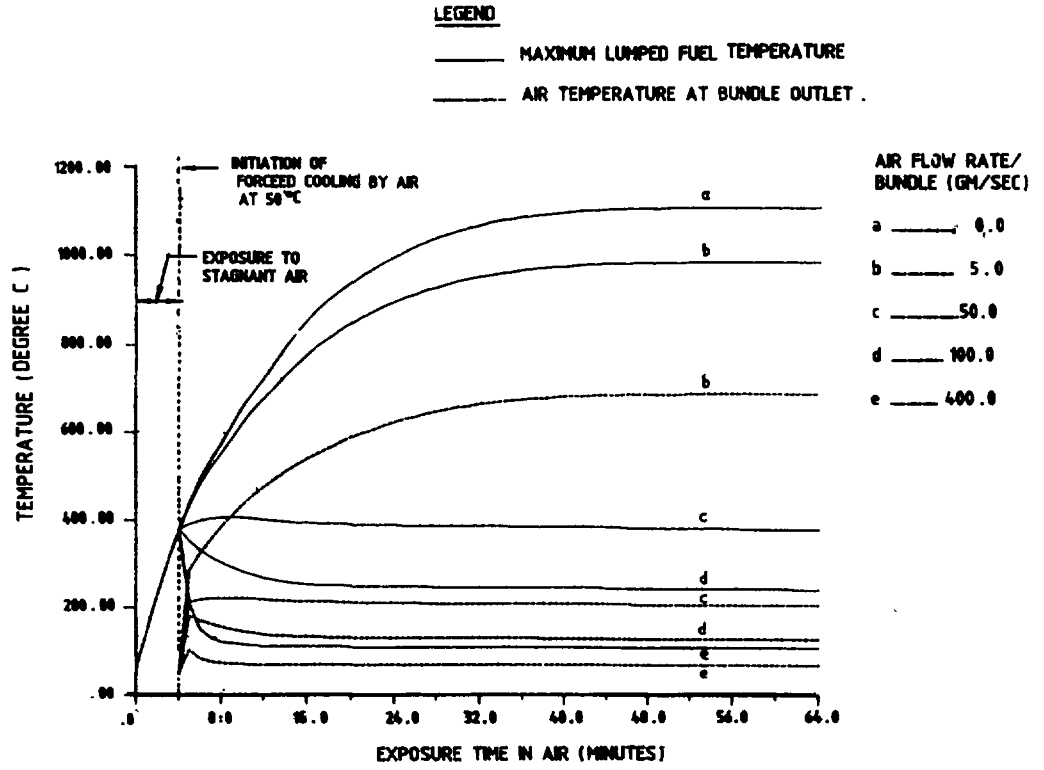

FIG. 7. Temperature transients in 500 MWe PHVR spent fuel stuck in dry environment during handling with/without forced cooling by air.

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THE EXPERIENCE OF RBMK-1000 FUEL UTILIZATION IN UKRAINE (CHERNOBYL NPP)

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Abstract

The paper presents the analysis of fuel operational experience in units 1, 2, and 3 of Chernobyl NPP with RBMK-1000 reactors for the whole time period of their operation. The data on utilization of fuel in unit 1 following previous partial burning in unit 2, which was not foreseen in the original design, are discussed. The results of the tests of technological channels material after 18 years of operation are also presented.

I. INTRODUCTION

The first operating NPP in Ukraine was Chernobyl NPP (unit 1 was commissioned in 1977, unit 2 - in 1979, unit 3 - in 1981, unit 4 - in 1983). Unit 2 was shut down after the fire in the turbine hall in 1991. On November 30, 1996 unit 1 was shut down. Unit 3 is operating at the capacity 1000 MW(e).

Because of the limited own fossil resources and reduction of its supply from Russia and also of the economical difficulties the Ukrainian NPPs produce now 40-50% of the total input of electricity compared to 16-17% during the Soviet period. That is why the questions of effectiveness of the RBMK fuel utilization as well as safety are important for Ukraine.

II. FUEL UTILIZATION IN THE UKRAINIAN RBMK

Table 1 contains the information about the burnup distribution of the unloaded fuel assemblies of the enrichment 2.0% and 2.4% during all the time of units 1 - 3 operation. Figures 1 and 2 present the same information as a chart.

Therefore this general information does not show the fuel utilization according to the design refueling scheme. After the shut down of the unit 2 it was decided to use its partially burned fuel in unit 1. 538 fuel assemblies were unloaded from unit 2 reactor and shipped to unit 1 cooling ponds:

411 of them were loaded into the reactor;

125 were damaged during the unloading and shipment and were considered as defected after the TV surface examination;

2 assemblies were leaking.

The reloading scheme of unit 3 corresponds to the design one, so it is possible to compare the information about the fuel utilization on this unit with the data for the other units with RBMK-1000 reactors, RBMK-1500 reactors and with the respective data obtained from other types of channel reactors. Table 2 and Figures 3 and 4 present the burnup distribution of the fuel assemblies unloaded from unit 3 during 01/01/92-08/01/96

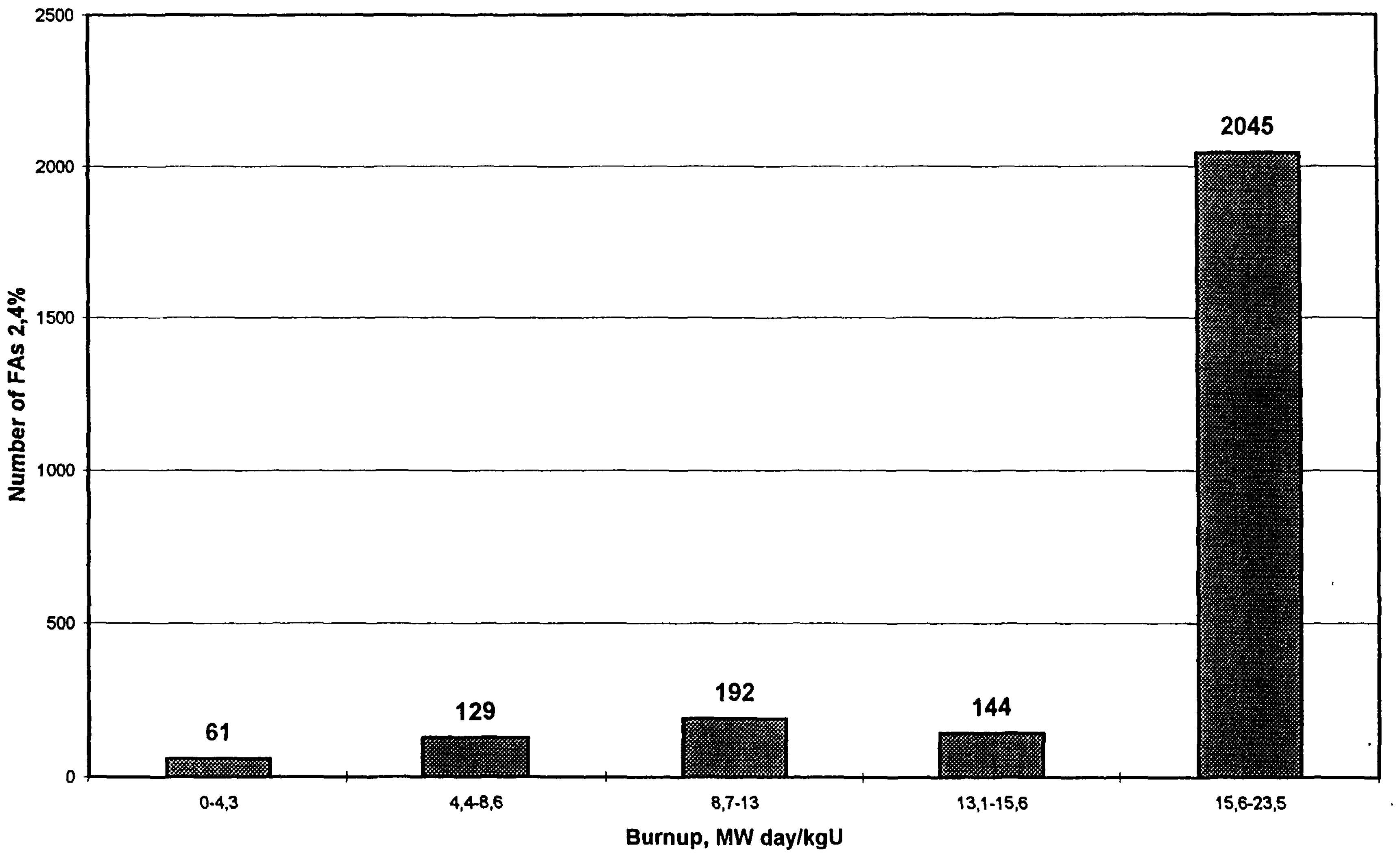


FIG. 1. Burnup distribution of discharged FAs of Chernobyl NPP during all time of operation (with enrichment 2.4 %).

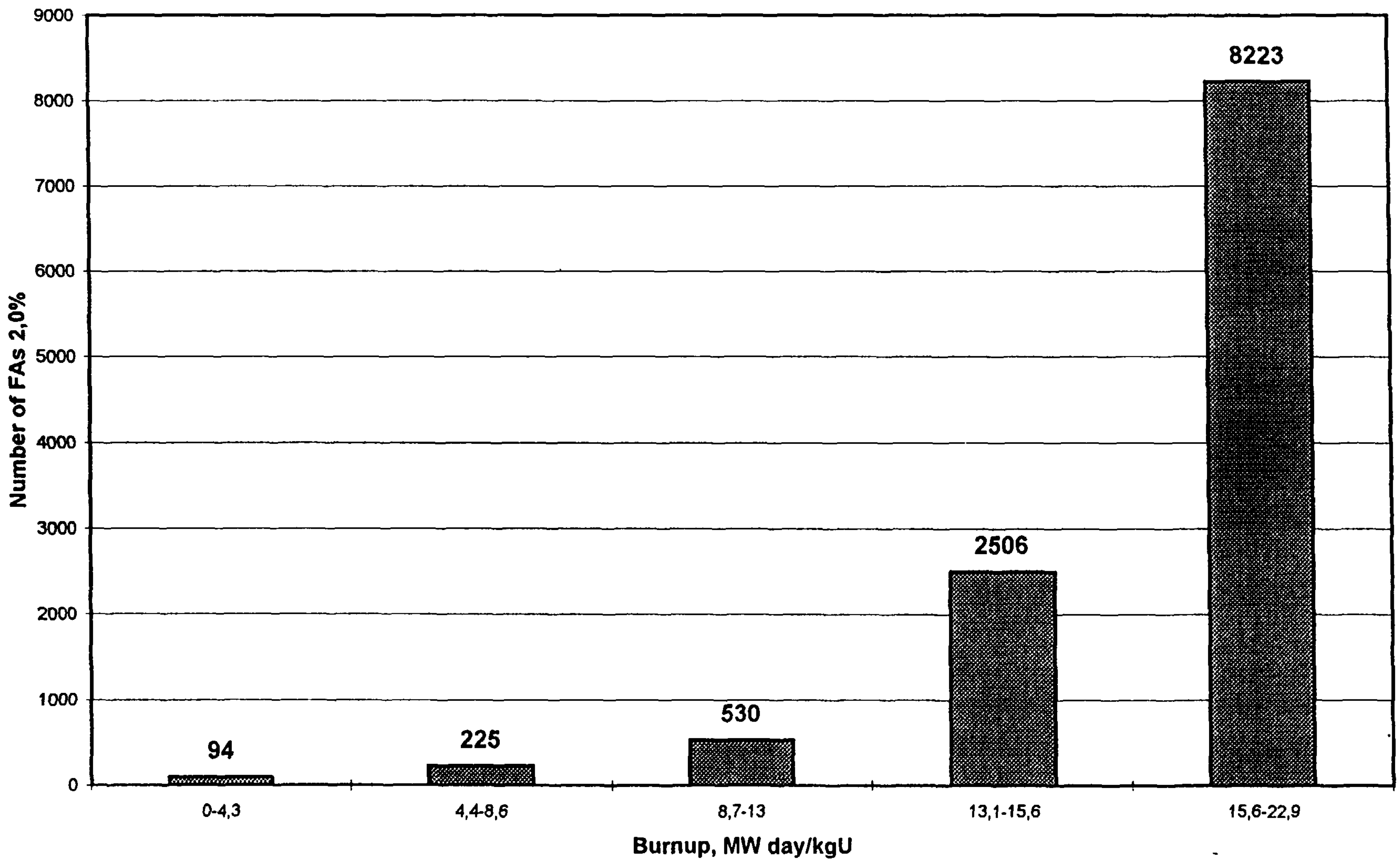


FIG. 2. Burnup distribution of discharged FAs of Chernobyl NPP during all time of operation (with enrichment 2.0 %).

TABLE 1. BURNUP DISTRIBUTION OF UNLOADED CHERNOBYL NPP FUEL ASSEMBLIES DURING ALL THE TIME OF OPERATION (MWt day/assembly/ Mwt day/kg U)

Enrichment	E max, Mwt*day/assembly / Mwt*day/kg U	Number of assemblies with E (0-500)/ E (0-4.3)	Number of assemblies with E (501-1000)/ E (4.4-8.6)	Number of assemblies with E (1001-1500)/ E (8.7-13.0)	Number of assemblies with E (1501-1800)/ E (13.1-15.6)	Number of assemblies with E (> 1800)/ E (>15.6)
2.0 %	2631/22.9	94	225	530	2506	8223
2.4 %	2701/23.5	61	129	192	144	2045

TABLE 2. BURNUP DISTRIBUTION OF UNLOADED FUEL ASSEMBLIES OF CHERNOBYL NPP UNIT 3 DURING ALL THE TIME OF OPERATION (MWt day/assembly/ Mwt day/kg U)

Enrichment	E max, Mwt*day/assembly / Mwt*day/kg U	Number of assemblies with E (0-500)/ E (0-4.3)	Number of assemblies with E (501-1000)/ E (4.4-8.6)	Number of assemblies with E (1001-1500)/ E (8.7-13.0)	Number of assemblies with E (1501-1800)/ E (13.1-15.6)	Number of assemblies with E (> 1800)/ E (>15.6)
2.0 %	2631/22.9	1	2	10	8	1089
2.4 %	2701/23.5	0	7	14	16	1160

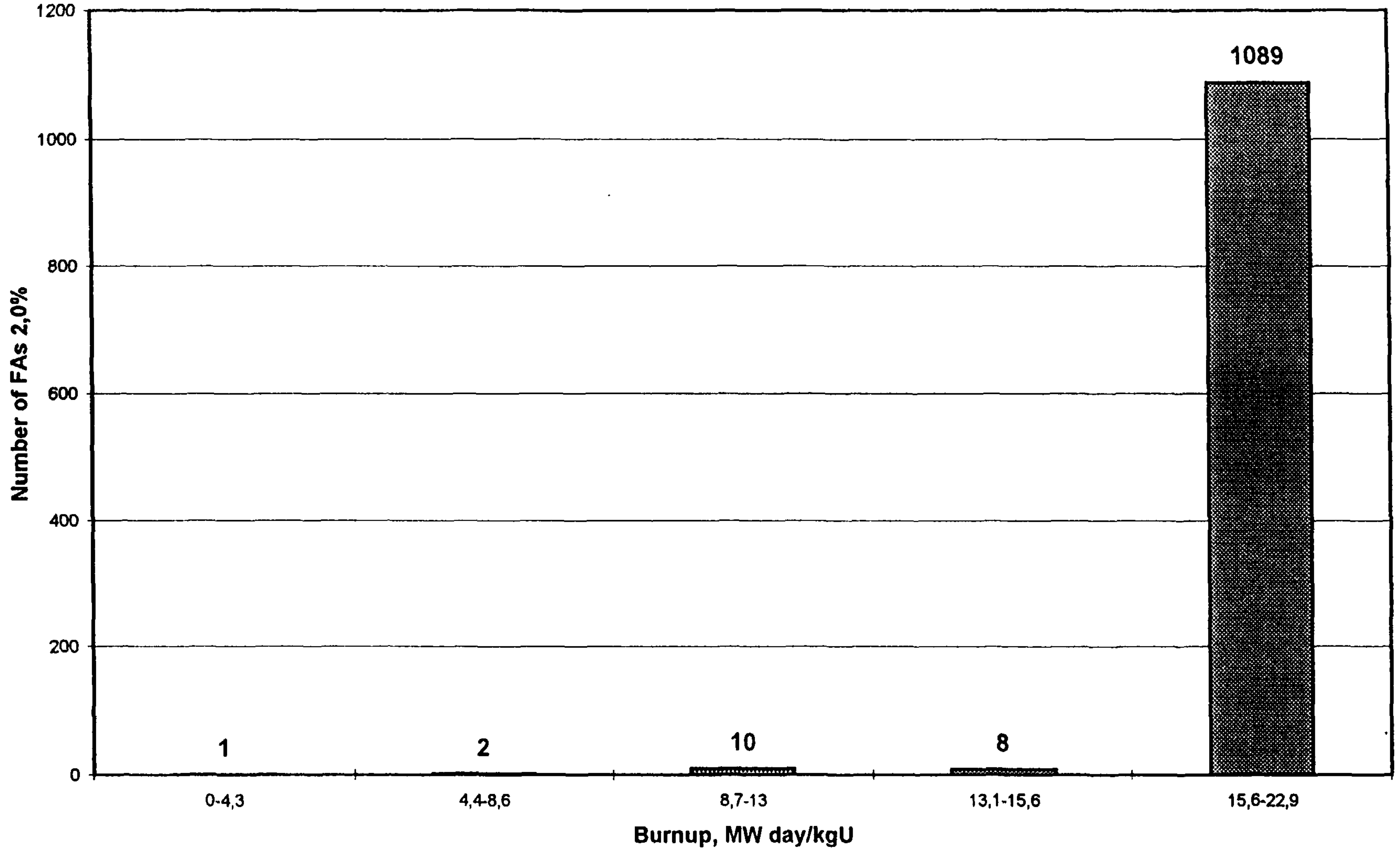


FIG. 3. Burnup distribution of discharged FAs of Chernobyl NPP, unit 3 during all time of operation (with enrichment 2.0%).

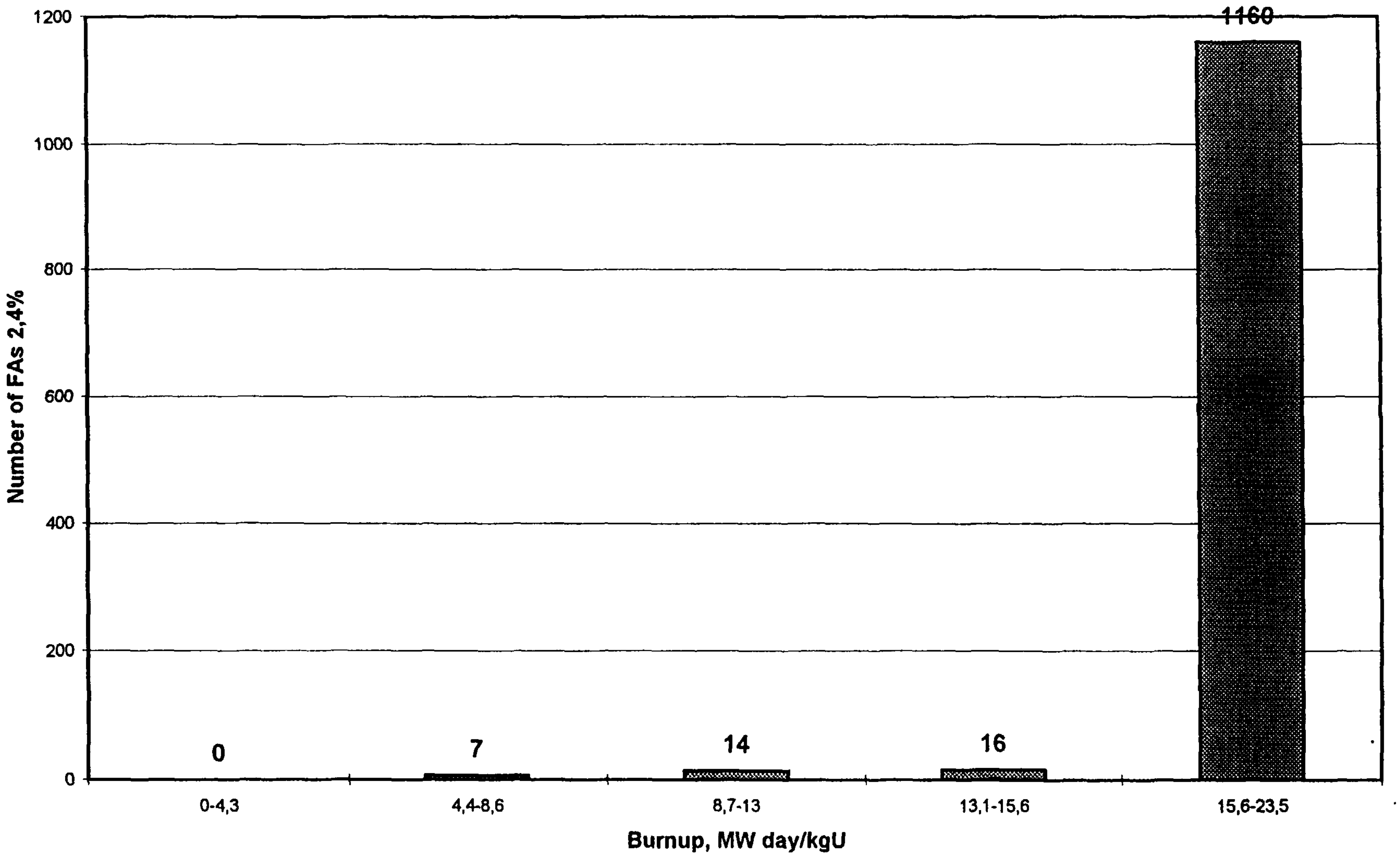
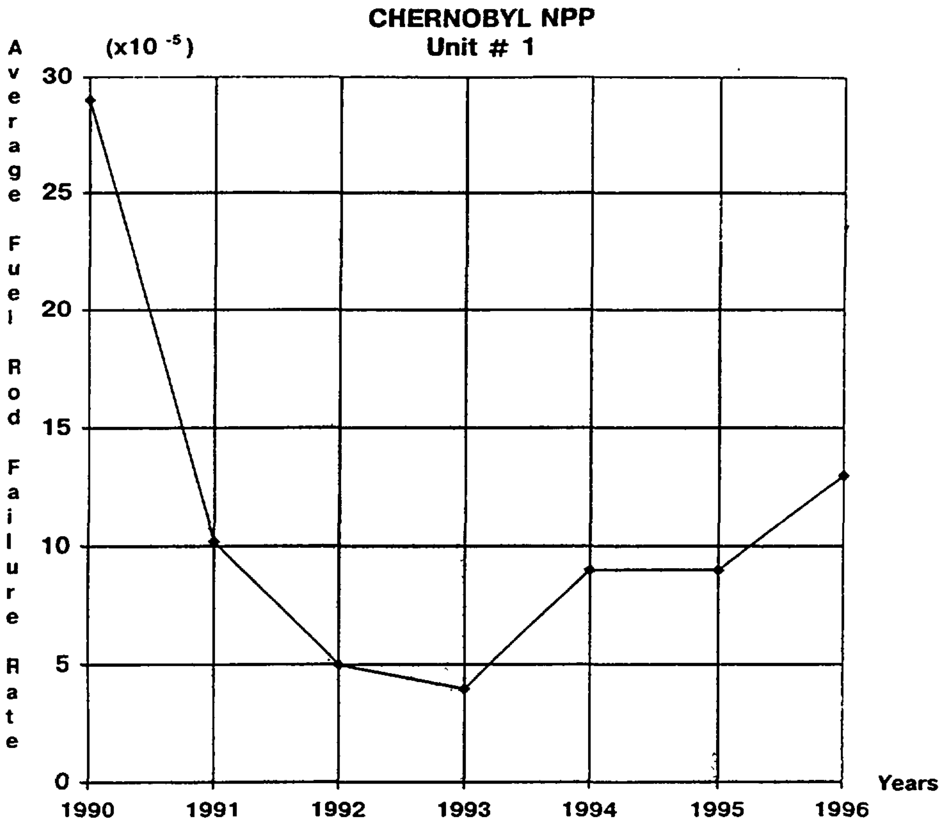


FIG. 4. Burnup distribution of discharged FAs of Chernobyl NPP, unit 3 during all time of operation (with enrichment 2.4 %).



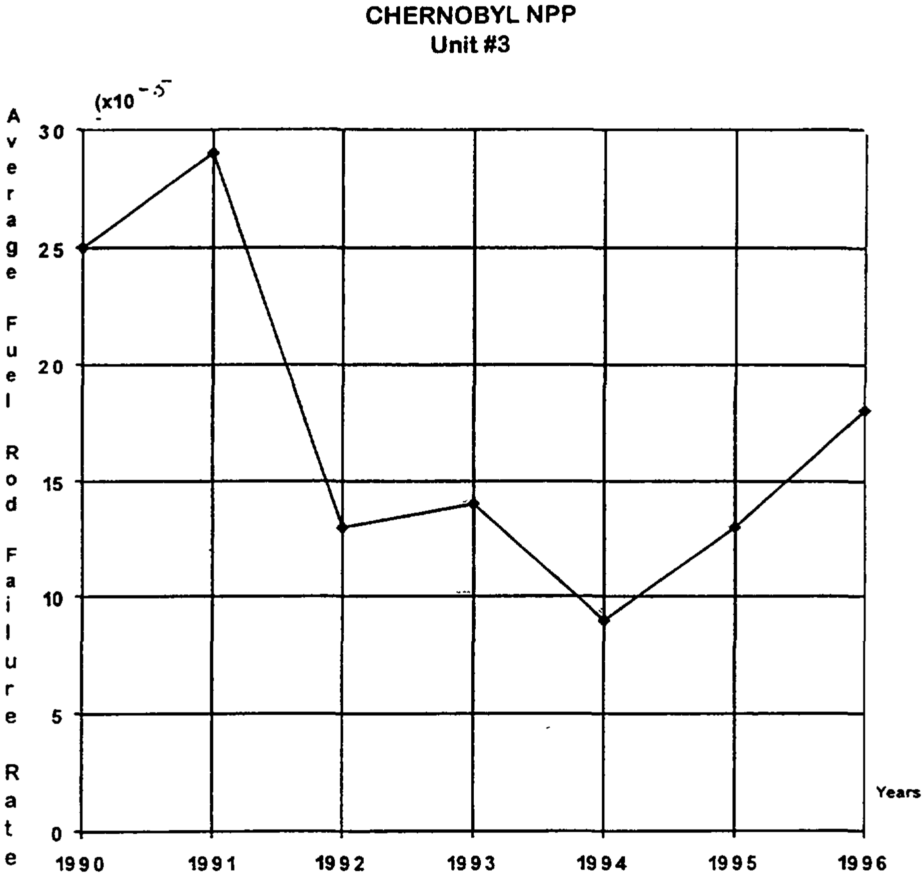


FIG. 5. Average fuel failure rate for Units 1 and 2.

III. FUEL PIN FAILURE RATE (FPFR)

For the period from 1990 till 1996 there were found 158 fuel assemblies with leaking pins, 112 assemblies from them with burnup lower than design burnup and which could be loaded for the use for the second time (early unloaded). These design burnup values were equal to E = 1500 MWt*day (13,0 Mwt*day/kg U-235) for the assemblies with enrichment 2.0% and E = 1800 MWt*day (15,6 Mwt*day/kg U-235). for the assemblies with enrichment of 2.4%.

The number of the assemblies with burnup lower than established rates for the second time use is about 70% from the total assemblies with leaking pin claddings.

The average fuel pin failure rate (FPFR) is defined as follows:

$$\overline{FPFR} = NFA / NA * 36$$

NFA - the number of leaking assemblies for the unit during one year
NA - total quantity of assemblies in reactors and unloaded assemblies for the unit during one year
36 - number of pins in assembly. It is assumed that the leaking assembly has only one leaking pin

Figure 5 presents the curves of the FPFR change for the period 1990-1996 for the units 1 and 3.

IV. SAFETY IMPROVEMENT ISSUES

The use of erbium as burnable absorber in RBMK fuel (according to the calculations done by Institute of Atomic Energy, Moscow) allows to reduce void coefficient for 0.3 - 0.6 β eff.

V. THE RESULTS OF TECHNOLOGICAL CHANNELS' ZIRCONIUM TUBES MATERIAL TESTING AFTER 18 YEARS OF OPERATION

The investigation of the mechanical properties of the zirconium alloy of the technological channel tubes of unit 1 of Chernobyl NPP after the 18 years of operation have shown:

1. The values of the rupture characteristics under the strain of the circular samples cut from the technological channels correspond to the values for the material irradiated to the rates more than $1*10^{21}$ n cm^{2} (E >= 1 MeV).

The average values of the yield strength $\sigma_{0.2}$ and the tensile strength σ_b for all the examined channels is 64 kg.mm^{-2} and 69 kg.mm^{-2} at the temperature 20° C and 49 kg.mm^{-2} and 51 kg.mm^{-2} at 280° C. The ratio of $\sigma_{0.2}$ to σ_b which characterizes the deformation strength, has grown to 0.93 at 20° C and 0,96 at 280° C for the irradiated material comparing at 0.73 and 0.79 for unirradiated samples, which indicates the earlier formation of the narrow part in irradiated samples.

During all the experiments there were no cases of rupture reduction caused by cracks.

2. After 18 years of operation the ductility of the tube material have reduced considerably. The value of uniform elongation δ_p for all the examined channels is between 0.6 ... 1.8% at 20°C (comparing at ~ 5% for unirradiated material) and 0.3 ... 0.8% at 280° C. But the dispersion of the values is not connected with different neutron fluence for each channel, because approximately the same dispersion value is observed for the samples from the same channel section.

The total elongation δ_0 of all samples tested, except two, are between 4.6 ... 6.9% for both test temperatures (compared at ~ 14% for unirradiated material).

Corrosion layer on the inner surface of channel tubes is 10-30 µm thick, in some areas it is 100-200 µm thick

REFERENCE

1. Investigation of Defected Areas of the Technological Channels from Chernobyl NPP Unit 1 After 18 Years of Operation. Mechanical Properties of Technological Channels' Zirconium Tubes. Report, Kiev Research Institute (1996).

MODERNIZATION OF THE DESIGN AND OPTIMIZATION OF THE MANUFACTURING TECHNOLOGY OF RBMK FUEL RODS AND FUEL ASSEMBLIES

XA9846751

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Abstract

The paper describes design and experience in fabrication of fuel and fuel channels for RBMK reactors (RBMK-1000 and RBMK-1500) at the JSC "Mashinostroitelny Zavod", Electrostal, Russia. The most important measures developed and undertaken since Chernobyl accident to increase operational safety of RBMK reactors are presented. Emphasis is given to modifications in fuel design and technology including U-Er fuel, rods with a new plug and fuel assemblies with Zr spacer grids.

1. RBMK FUEL DESIGN AND FABRICATION

At present 13 RBMK-1000 units and 2 RBMK-1500 units having the total rated electric power of 16000 MW are in operation in Russia, Lithuania and Ukraine; this makes 32,7% of the rated electric power of all the reactors constructed according to the Russian designs.

Fuel for RBMK-1000 and RBMK-1500 is manufactured at JSC "Mashinostroitelny zavod", Electrostal, and supplied as fuel assemblies (FAs).

RBMK-1000 fuel assemblies are in mass production starting from 1973; at present they are supplied to 13 units.

RBMK-1500 fuel assemblies are in mass production starting from 1982 and supplied to 2 units.

RBMK-1000 and RBMK-1500 fuel assemblies are designed for generating thermal power and its transfer to the coolant flow in the cores of RBMK-1000 and RBMK-1500 reactors.

During their operation in the reactors fuel assemblies are immobile. They consist of 2 bundles of pin-type fuel elements connected by spacer and end-grids, a bar with the rod-carrier, two tailpieces and a nut. The design of a working fuel assembly for the γ-chamber is similar to that of a working cassette but differs in the rod-carrier which has a central cavity for sensors.

Fuel elements for RBMK-1000 fuel assemblies are tubes made of a zirconium alloy filled with UO_2 sintered pellets and sealed with plugs by means of welding.

Components and arrangement of RBMK-1500 fuel assemblies are the same as those for RBMK-1000 fuel assemblies. The difference of RBMK-1500 FA is that 28 spacer grids-intensifiers are mounted on the upper bundle of fuel elements to increase the flow turbulence, the elements have 2 screen pellets near the plug that are made of depleted UO_2 and fuel pellets have central holes.

Fuel pellets and fuel rods for RBMK-1000 and RBMK-1500 fuel assemblies are manufactured on automatic lines, components and sub-units - on automatic and computer numerical control (CNC) machines.

Due to the high organizational level of manufacturing and control RBMK-1000 and RBMK-1500 fuel assemblies provide for safe and reliable operation of cores with RBMK-1000 and RBMK-1500 reactors.

From the moment of commissioning the first unit of RBMK type reactor (1973, Leningradskaya NPP) certain work was carried out related to the modernization and optimization of fuel rods and fuel assemblies design to increase safety and technico-economic performance of RBMK.

2. MEASURES TO IMPROVE OPERATIONAL SAFETY OF RBMK REACTORS

After the accident at Chernobyl NPP a number of technical and organizational measures was taken at NPPs with RBMK in operation that sufficiently increased their safety and reliability. Due to these measures nuclear safety of NPPs with RBMK reactors reached a new quality level which excluded the possibility of an accident with the reactor runaway.

The most important measures related to the safety increase of the units with RBMK were carried out in the period 1986-1987. The main of them are:

1. Improvement of the neutron-physical properties of the core, performed with the purpose to improve the protection of the reactor from the positive reactivity under accident and transient conditions (decrease of steam reactivity factor α_φ and scram system circuit loss of coolant effect).

2. Improving the reaction speed of the accident protection system performed with the purpose to increase the speed effectiveness of the accident protection, introduction of the second, independent, system of reactor shutdown, excluding the positive reactivity excursion at any position of control devices.

3. Replacement of fuel channels carried out with the purpose to replace the equipment gone over its service margin, provision of the design service life of a unit, provision of the possibility to prolong the service life of a unit up to 35...40 years.

4. Introduction related to the decrease of the operational reactivity margin performed to provide for the automatic reactor shutdown in case the value of the operational reactivity margin drops down to the threshold value - 30 rods of manual control.

5. Correction of the operational documentation performed to introduce into practice certain organizational measures for the prevention of the possibility for the reactor to reach unstable state and to make the wording of the "Technological operation manuals" meet the requirements of Gosatomnadzor of the Russian Federation.

6. Prevention unauthorized switching-off of the reactor accident protection devices.

7. Making stricter the operational control of the metal of the equipment and piping important from the point of view of safety.

The measures carried out related to the improvement of neutron-physical properties of the core exclude the possibility of an accident similar to that occurred in Tchernobyl. The possibility of the runaway on prompt neutrons is excluded at all design accidents (LOCA) and the shutdown of the reactor is provided for without unauthorized power growth.

That's why now the problems of the RBMK reactors safety in the broad sense are similar to the safety problems of the reactors of other types and mainly concentrate on the analysis of the possibilities of LOCA.

3. IMPROVEMENT OF THE FUEL CYCLE CHARACTERISTICS

Also, the measures aimed at the improvement of safety that are underway since 1986, have led to a certain decrease of technico-economic performance of the NPPs with RBMK and worsening of the fuel cycle characteristics: e.g. reduction of fuel burnup.

Taking into account the importance of the RBMK fuel cycle optimization, improvement of technico-economic performance of NPPs and to improve further safety and reliability of RBMK, starting from 1993, researches and developments aimed at reaching the above-mentioned targets were started.

Lately the design modernization and fuel manufacturing technology optimization for RBMK fuel rods and fuel assemblies are carried out in the followings main directions:

- using burnable absorbers -Er- in fuel;
- introducing a fixing agent of a new design made of a special alloy;
- introducing Zr spacer grids.

3.1. Manufacturing U-Er fuel for RBMK

In 1986, as a result of the actions taken to increase safety, 80 additional absorbers were introduced into the core of RBMK-1000 reactors and 54 additional absorbers into the core of

RBMK-1500 that allowed to decrease the steam reactivity α_φ down to 0,6-0,8β. Also, it simultanuously led to decreasing fuel burnup by 25-35% in relation to the design value.

Now RBMK cores are operated with this particular composition.

Further decrease of α_φ due to the addition of a larger number of additional absorbers would decrease burnup and, consequently, would decrease economical characteristics of NPPs.

To compensate for these losses experimental work aimed at improving RBMK fuel utilization had been done for several years. Lately, the researches were re-focused at using burnable absorbers which allowed to keep the fuel assembly geometry and technological channel of the core unchanged.

Having estimated the prospects of this direction of RBMK fuel cycle optimization, JSC "MSZ" made up a decision to start up U-Er fuel manufacturing with the purpose to develop the new fuel production in the volumes necessary for fuel deliveries to all NPPs with RBMK by 2000. From 1993 JSC "MSZ" together with the research institutes concerned has carried out researches and developments regarding the creation of U-Er fuel.

At JSC "MSZ" a decision is made up regarding the organization of a special production area for the new fuel fabrication. In the shortest dates JSC "MSZ" purchased modern equipment for completing the new production area (BTU furnace, "Courtoy" press). Simultaneously activities addressing the development of U-Er manufacturing technologies started at the plant.

The main difference of the new fuel manufacturing from that of UO_2 used at present is the technology of blending of source uranium dioxide and erbium oxide powders.

This operation, on the one hand, is very important because a high degree of uniformity of Er oxide distribution in the main fuel material is required, on the other hand, it's very complicated because a relatively small amount of erbium oxide is added.

Nevertheless, the technology of powder blending which provides for the required uniformity of Er distribution in UO_2 matrix (uniformity factor - not more than 10%) has been developed at JSC "MSZ" in the shortest time.

While developing the technology of U-Er pellet fabrication, the following researches were carried out:

- determination of the optimum blending mode of UO_2 powder and erbium oxide in the process of testing the different types of mixers;
- development of the modes of compacting press powder into "green" pellets;
- determination of the modes of pellet sinterning;
- development of the grinding procedures for sintered pellets;
- determination of corrosion resistance of the finished uranium-erbium pellets.

In 1995 a number of methods and instructions (erbium determination, uranium determination in U-Er fuel, determination of oxygen coefficient, impurities) was developed for quality control of uranium-erbium fuel pellets.

Thus, by the beginning of 1995 the fuel assembly design with uranium and erbium fuel, (0,47 wt% erbium oxide in UO_2) was developed at JSC "MSZ" with the participation of the Research Institutes, as well as the industrial technology of fuel assembly manufacturing, which included fuel pellets, rods and assemblies. A special production area was founded for U-Er fabrication.

The first experimental batch of RBMK-1500 fuel assemblies with uranium-and-erbium fuel in the amount of 150 pcs was manufactured at JSC "MSZ" in 1995 to prove the calculations performed by reactor experiments. This batch was loaded in the second unit of Ignalinskaya NPP in the period from June 1995 to January 1996.

In 1996 the second experimental batch of RBMK-1000 fuel assemblies with uranium-erbium fuel (200 psc) was manufactured at JSC "MSZ" and was loaded in the second unit of the Leningradskaya NPP in summer 1996. The preliminary results of reactor testing of the new fuel at both NPPs conformed to the calculations.

At present the issue of the most optimum usage of U-Er fuel in all the units with RBMK reactors is under consideration.

3.2. Manufacturing of fuel elements with a new fixing agent

It is welol known that RBMKfuel elements are operated in a more rigid mode than WWER (PWR) fuel elements because RBMK foresees the usage of a steam water mixture instead of water and on-power core reload.

Having analyzed the statistical data and reasons of RBMK fuel assembly failures (less that 2% of failures by unloaded fuel assemblies), lately JSC "MSZ" started the process of design and technology optimization for RBMK fuel elements manufacturing with the purpose to improve their reliability.

One of the fuel elements optimization directions is the introduction of a modernized fixing agent for the fuel column. The fixing agent used now is of spring type, made of zirconium alloy and has a number of drawbacks:

- widely scattered properties of the spring-part that can lead to an increased gap in the centre of fuel assemblies or to the cladding deformation;
- the material of the fixing agent (Zr-alloy) does not have enough relaxation resistance at operational temperatures.

The drawbacks mentioned initiated the passing over to modernized spring-type fixing agent of a new design of Fe-Cr-Ni alloy with an optimum spring part excluding above-mentiōned drawbacks.

The modernized fixing agent was designed on the basis of experimental researches of the mobility and the necessary compression of the fuel column.

The alloy used is compatible with the cladding material and fuel, provides for the needed elasticity and strength of the spring, it is plastic enough during manufacturing and operation and has thermal stability in the temperature range from 370 to 860°C as well as irradiation resistance during the service life of fuel elements.

3.3. Manufacturing of RBMK fuel assemblies with zirconium spacer grids

One of the possibilities to improve technico-economic performance of thermal neutrons reactors (including RBMK) is the decrease of parasitic neutron absorption in the core due to the refusal to use construction materials with large neutron absorbtion, in particular, the substitution of stainless steel spacer grids (that are currently used) by zirconium ones.

The neutron-physical calculations performed by Russian Research Center "Kurchatov Institute" showed that the substitution of stainless steel spacer grids by Zr grids allows to increase fuel burn up by about 3% without fuel enrichment increase.

Activities to develop Zr spacer grids (ZSG) for RBMK fuel assemblies started at JSC "MSZ" jointly with Research & Development Institute of Power Engineering (RDIPE, Moscow).

The following actions were undertaken:

- on the basis of R & D aimed at choosing the optimum connection type of spacer grids elements, spot welding (SW) was chosen, metallographic and corrosion studies of SW welds were carried out, welding modes providing for sufficient corrosion resistance and needed mechanical strength of welds were developed;
- Zr spacer grids strength calculations were carried out and strength, vibration and corrosion tests of RBMK Zr spacer grids of different designs were carried out. On the basis of calculations and test results, Zr spacer grid of alloy-110 of cellular type was chosen with components made of especially thin-walled tubes, and the design documentation for Zr spacer grids and fuel assemblies with Zr spacer grids was developed;
- the technology of Zr spacer grids was developed and a number of tests related to choosing the needed press tools and optimum geometry of source tubes were performed;
- several pilot batches of Zr spacer grids were manufactured and tests on fuel elements bundle assembling with Zr spacer grids were performed to estimate the damageability of fuel elements and Zr spacer grids, to determine the assembling force and the possibility to automate the assembling process. The results proved the possibility to do this. The technology of fuel assemblies manufacturing with Zr spacer grids was developed;

- a number of out-of pile tests was carried out on full-scale dummies of fuel assemblies with Zr spacer grids:
loading-unloading from the simulator of the technological reactor channel to estimate the damageability of fuel assemblies with Zr spacer grids (RBMK fuel assemblies do not have any shroud tube), life-time tests in a special rig simulating in-pile conditions (but for neutron fluence) to determine the Zr spacer grids and fuel assemblies preservation as a whole under conditions close to operational, transportation tests of fuel assemblies dummies with fuel weight simulation to estimate Zr spacer grids preservation during product delivery period. Positive results are received after all before-reactor tests;
- a pilot batch of RBMK-1000 fuel assemblies with Zr spacer grids (150 pcs) is manufactured for in-pile tests at Leningradskaya NPP.

All fuel assemblies have successfully passed the tests up to burnup values similar to those reached by mass-produced assemblies (≅20 MWd/kg U).

The results of out-of-pile and in-pile tests as well as researches conducted proved the correctness of the choice of technical solutions made during the designing and manufacturing of Zr spacer grids.

Taking into account the interest of customers in the soonest passing over to manufacturing fuel assemblies with Zr spacer grids at JSC "MSZ", with the Research & Development Institute of Power Engineering participation, a program of Zr spacer grids introduction was developed which is currently fulfilled.

3.4. Assembling fuel elements with spacer grids

RBMK fuel assemblies have 2 fuel elements bundles in which fuel rods are assembled into a skeleton. The skeleton is a tube ∅ 15,0 x 1,25 mm with spacer grids fixed on it. The RBMK-1000 skeleton has 10 spacer grids, RBMK-1500-28 spacer grids.

The need for the provision of non-damageability of the fuel element claddings and surfaces of the components of spacer grids, as well as the provision of reaching the needed matching of the pair "fuel element-spacer grid", necessary force for fuel element passing through a spacer grid, and increasing labour productivity due to the automation of fuel elements assembling required to create a special assembling technology.

JSC "MSZ" has developed a special automated technology of fuel elements assembling. In brief, it is the assembling of fuel rods preliminary covered with special coating providing the following:
preventing fuel elements claddings and spacer grids components from mechanical damages (lines, scratches, chips) during rods passing through spacer grids, sufficient decrease of the force needed for fuel element passing through spacer grids and necessary matching in the pair "fuel element-spacer grid".

Varnish on the basis of polyvinyl alcohol and special glycerin lubricant decreasing the friction during fuel elements assembling, with a specified film thickness is applied to the surface of rods.

Bundles of fuel rods are assembled on automatic rigs. Two types of rigs are used during RBMK fuel assemblies production at JSC "MSZ", that use two different methods: pushing and pulling of fuel elements.

Pushing rigs have higher productivity (by 1,5 time due to pushing 3 rods at a time instead of 2 on pulling rigs) and smaller sizes that allows to save costly producton area.

After assembling bundles of fuel rods, the special coating is washed away from the rods surface in fuel rods bundles, this is done in distilled water of a certain temperature. Futher the cleaned bundles of fuel elements are sent to fuel assemblies fabrication.

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FIRST EXPERIENCE WITH BURNABLE ABSORBER FUEL FOR RBMK-1500

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Abstract

The paper presents fuel management practices at Ignalina Nuclear Power Plant (NPP) with two RBMK-1500 units. Present thermal rating of each unit is 4200 MW. Changes in fuel management practices and reactor operation following the Chernobyl accident are described. Special emphasis is given to the reduction of void reactivity coefficient by increase of the number of control rods and transfer to the integrated burnable absorber (UO_2-0.41% Er_2O_3) fuel with increased (up to 2.4%) enrichment.

1. FUEL MANAGEMENT PRACTICES

The Ignalina NPP comprises two generating units with capacity of 1500 MW each. Each of the two reactors has a design thermal rating of 4800 MW. At present both reactors are running at the prescribed reduced power of 4200 MW. The first reactor start- up took place on 2 November 1983 and on 31 December 1983 the first turbine generator was connected to the grid. The power output achieved its design level of 1500 MWe in May 1985.

The second reactor was started up on 4 January 1987. The reactor commissioning commenced on 18 August 1987. Soon the generating power output of the second reactor came to 1400 MW and it was operated at this power until 11 August 1988. Due to the decision taken to restrict the thermal rating of the Ignalina reactors to 4000 MW, the reactors were operated at this power within three years. In August 1991 on the basis of further calculations, it was decided to raise the thermal rating limit for RBMK-1500 to 4200 MW which is still in force for both units.

As of 1 November 1996 Ignalina has generated 154,6 Billion KWh. To clearly understand the INPP contribution into the Lithuanian electricity generation, have a look at Table 1 showing the total annual generation of electricity in Lithuania, INPP electricity generation (MkWh).and its contribution (percentage wise) into the total Lithuanian electricity generation.

Table 1.

	1989	1990	1991	1992	1993	1994	1995
Lithuanian electricity generation	29130	28236	29248	18178	14081	9942	13844
INPP generation	16646	17033	17000	14638	12260	7706	11822
Ignalina contribution(%)	57,1	60,3	58,1	80,5	87,1	77,5	85,4

Following the accident at the Chernobyl Unit 4 the priority measures to improve RBMKs -safety were developed. Under this programme the measure to reduce the void reactivity coefficient was executed on the stage-by-stage terms.

As the first step the minimum acceptable ORM was increased to 30 manual control rods and ORM at stable on-power condition to 53-58 control rods. That allowed to make the void reactivity coefficient lower to +1,7β.

The purpose of the second step was to reduce the void reactivity coefficient down below +1,0 β. That was accomplished at the RBMK-1000s by stage-by-stage moving to the 2,4% enriched fuel and installing of 80-90 additional absorber (AA) rods into the core.

At the RBMK-1500s the same task was completed by inserting 50÷55 AA rods. But there was decrease in fuel burnup to 14,0÷14,5 MWday/kgU and increase in reactor refuelling frequency.

2. INTEGRATED BURNABLE ABSORBER FUEL

The next step in reducing the void reactivity coefficient down is to move RBMK-1500s to 2.4% enriched fuel with BURNABLE absorber. Although it had been suggested some time ago, the anticipated calculations did not give favorable results. It was not until the end of 1992, with the aid of WIMS code, the efficiency of burnable absorber use was proved. The different type burnable absorbers were considered to be used, namely, boron, dysprosium, hafnium and erbium. Gadolinium was not considered because some preliminary tests proved to be inefficient in neutron flux flattening comparing with boron. In addition, in case of gadolinium there was no significant decrease of the void reactivity.

From the results obtained it was specified that the most efficient neutron flux flattening between fuel channels could be achieved when either erbium or boron was used. The advantage of erbium is in decrease of the campaign average void reactivity of three times comparing with boron. The natural mixture of erbium isotopes in the form of Er_2O_3 was advised to add into fuel matrix. Among the six erbium isotopes making the natural mixture, the isotope with atomic weight of 167 has great resonance at energy 0,47 eV. This causes the decreasing of the neutron multiplacation factor under the primary circuit loss of coolant. The content of this isotope in the natural mixture is 22,9%.

The optimal weight content of erbium in the new fuel was selected on the basis of the following conditions:

- The unloaded fuel burn-up is not less 22 MWd/kg when there are no AA rods in the core;
- The void reactivity coefficient is not higher +1.0 beta;
- The maximum power level of a fresh fuel assembly with 2.4% enrichment does not exceed the corresponding power level of a fuel assembly with 2% enrichment.

The first condition defines the upper limit of the initial weight content of erbium which is equal to 0,48%. The other two conditions define the lower boundary - 0,4%. On the basis of the calculations 0,41% was selected as the value of the erbium optimal weight content.

Mechanical and thermo-physical properties of uranium-erbium fuel pellets and their compatibility with the Zr+1% Nb alloy were tested.

Finally the properties of the new enriched fuel appeared to be insignificantly different from the old fuel and therefore they came to the decision that it would be unreasonable to modify either fuel assembly or fuel element design.

The first 150 experimental fuel assemblies of the new design with 2,4% enrichment and burnable absorber were delivered to Ignalina on 9 June 1995 and on 28 June 1995 the first five fuel assemblies were loaded into Ignalina 2 to provide the flux measurement with the reference detector. The measurements results proven that the preliminary calculations for the power rate of the 2,4% enriched fuel assembly were satisfactory. The maximum deviation was not more 5% and the power

rate of a 2,4% enriched fuel assembly did not exceed the power of a fuel assembly with 2,0% enrichment. On one side it supported the accuracy of the predicted results and on the other side it was the first proof that the erbium concentration had been correctly selected.
The first 150 experimental fuel assemblies were loaded in a batch of 25. The void reactivity coefficient $\alpha_{\varphi,}$ fast power reactivity coefficient $\alpha_{w,}$ and period of the first azimuthal harmonic period changing τ_{01} were carefully measured after loading each batch.

The measurement results obtained are given in Table 2.

Table 2 Reactor Parameters obtained during the experimental FA loading

Date	17.03.95	25.07.95	08.08.95	15.08.95	08.12.95	05.01.96	24.01.96
Number of FA	0	25	50	74	100	125	150
Reactor power output, eff. days	0	90	101	106	125	147	162
α_{ϕ}, β	0,8	0,7	0,7	0,6	0,7	0,6	0,5
α_{w}, 10^{-4} β MW	-1,8	-2,2	-2.0	-2,3	-2,0	-2,1	-2,1
τ_{01}, min	22	22	26	24	22	30	26

By the end of charge of the 150 experimental FA there was decrease in fuel burnup in the core of 1%. But in accordance with the predicted results decrease in fuel burnup was estimated to be about 3%. However it should account that 24 local scram rods were replaced with rods of modified design with the seven meter replacer during the 1995 scheduled maintenance outage. As a result of the rod replacement the fuel burnup increased 1% and the void reactivity coefficient became up (~0,1 β). Summarizing the above the core data obtained progressively comply with the predicted results.

Thus after the completion of 150 experimental fuel assembly loading the void reactivity coefficient was decreased from 0,8 β to 0,5 β, the power reactivity coefficient was altered from $(-1,8)\cdot 10^{-4}$ β/MW to $(-2,1)\cdot 10^{-4}$ β/MW, period of the first azimuthal harmonic changing increased from 22 minutes to 26 minutes. There was a satisfactory compliance with the preliminary predicted results for neutronic RBMK-1500 parameters. That allowed to start the next step of the tests: loading of 500 2,4% enriched fuel assemblies with burnable absorber. At the same time a certain amount of the additional rod absorbers should be extracted.

3. MODIFIED FUEL MANAGEMENT SHEMES

To find the proper strategy for the experimental batch loading the following parameters were considered:

- the void reactivity coefficient should be within 0,4 and 0,8 β;
- the average fuel burnup should not decrease.

The procedure for loading was developed as follows:

Step 1: Loading of 45 FA, removing of 4AA rods - 4 cycles
Step 2: Loading of 60 FA, removing of 4AA rods - 3 cycles
Step 3: Loading of 96 FA, removing of 4AA rods, Loading of 44 FA- 1 cycle.

Thus, the first step includes the loading of 180 FA and removing of 16 AA rods to be completed over 4 cycles.

The second step comprises the loading of 180 FA as well as removing of 12 AA rods within 3 cycles. The third steps covers 140 FA to be loaded and 4 AA rods to be removed.

Accordingly to the experimental loading programme 32 AA rods are to be extracted from the core. The increase of the number of the experimental FA which have to be loaded before unloading the next 4 AA rods is caused by necessity to compensate erbium burnup in the previously loaded FA. The implementation of that programme commenced in June 1996 at Ignalina 2. Before the reactor was shut down for scheduled maintenance outage two cycles of the first loading step had been completed

The results of those cycles completed, the experimental reactor parameters and predicted results are given in Table 3

Table 3 Reactor parameters obtained from Experimental Programme

Date	11.06.96	25.06.96	30.06.96	16.07.96	23.07.96
Number of FA	152	195	195	236	240
Predicted number of FA	150	194	194	234	238
Reactor power output/effective days	0	11	15	28	33
Average FA burnup, MW/d	861	848	859	858	868
Number of AA, psc	53	53	49	49	45
α_{φ}, measured predicted	06 0,62	0,6 0,49	0,6 0,64	0,6 0,48	0,7 0,67
α_w 10^{-4} measured predicted	-2,0 -2,3	-2,4 -2,5	-2,2 -2,2	-2,3 -2,5	-2,3 -2,2
τ_{01}, min	26	30	26	25	29

Table 3 shows that the procedure taken for the experimental FA loading and removing of AA rods enables to maintain the void reactivity coefficient within 0,6-0,7 β which complies with the results measured

At present the Experimental Programme for 500 2,4% enriched FA with burnable absorber to be loaded at Ignalina Unit 1 has been developed and is to commence in January 1997

4. CONCLUSIONS

The predicted results indicate that the full replacement of the reactor with new uranium-erbium fuel will enable to remove all AA rods from the core The void reactivity coefficient is predicted to be decreased up to zero The magnitude of the negative fast reactivity power coefficient will increase and result in stabilizing power distribution The fuel burnup will be 30 35% up Thus, the loading of the uranium-erbium fuel into RBMK-1500 will improve both reactor safety and efficiency.

R&D ACTIVITIES ON WATER CHANNEL REACTOR FUELS

(Session II)

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Russian Federation

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BWR FUEL R&D PROGRAMS AT STUDSVIK

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Abstract

The different types of BWR fuel R&D programs at Studsvik are discussed. The R2 test reactor, its facilities for irradiation experiments and post-irradiation examinations are described.Reviews are given of some of STUDSVIK's earlier international fuel R&D programs, of STUDSVIK's defect fuel degradation experiments and of the following new upcoming international fuel R&D projects: the DEFEX II, ULTRA-RAMP, SUPER-RAMP III/10x10, STEED, Post-DO/DNB and TRANS-RAMP III Projects.

1 Fuel R&D Programs at Studsvik – Introduction

1.1 Background

Studsvik AB, the parent company in the STUDSVIK group, is partly owned by Vattenfall AB, and is performing R&D work and associated activities, primarily in the nuclear energy field. Studsvik AB is a commercial company, active in the areas of services, supply of special equipment and systems and also consulting. Studsvik Nuclear AB, which is the largest subsidiary within the STUDSVIK group, is one of the direct offsprings of AB Atomenergi, the origin of the STUDSVIK group, which was formed in 1947. The STUDSVIK group has about 580 employees and a turnover of about 425 MSEK/year.

The facilities of interest in this connection are the R2 test reactor, the Hot Cell Laboratory and various other laboratories, all located at Studsvik, 100 kilometers south of Stockholm, Sweden.

1.2 Fuel Testing

STUDSVIK NUCLEAR's R&D work in the area of fuel testing was started in the early 1960's. In a very general sense the purpose of fuel testing can be described as follows:

- Increasing of reactor availability by decreasing fuel-related operational power restrictions, defining the operational power limits.
- Acquisition of experimental data for fuel-related safety considerations.
- Decrease of fuel costs by making increases in fuel burnup possible.

The fuel testing activities can be divided into a number of well-defined steps as follows:

- Base irradiation, performed
    - in a power reactor, or
    - in STUDSVIK's R2 test reactor.
- Power ramping and/or other in-pile measurements, performed
    - in STUDSVIK's R2 test reactor.
- Non-destructive testing between different phases of an experiment, performed
    - in STUDSVIK's R2 test reactor pool, or
    - in STUDSVIK's Hot Cell Laboratory.

- Destructive post-irradiation examinations, performed

    - in STUDSVIK's Hot Cell Laboratory, or
    - in the sponsor's hot cell laboratory

Fuel examination can be performed on standard (full-size) fuel rods from power reactors, which can be investigated in the Hot Cell Laboratory. If required, some types of tests could also be performed on such fuel rods in the R2 test reactor However, due to the rather large initiation costs, such tests have not yet been performed It should be noted, however, that short fuel rodlets, suitable for ramp testing and other on-line measurements in the R2 test reactor, are routinely fabricated from irradiated full-size power reactor fuel rods by the STUDFAB refabrication process.

Fuel testing in the R2 test reactor is usually performed on fuel segments (rodlets) of 300-1000 mm length. However, tests have also been performed on full-size demonstration reactor fuel rods with up to 2.5 m length In those cases only the lower 0.6 meters were irradiated Irradiation at constant power is performed in boiling capsules (BOCA rigs) in fuel element positions, or in pressurized water in-pile loops operating under BWR (or PWR) pressure/temperature conditions, as described in Section 2.

<u>Ramp tests</u>, incorporating a very fast-responding test rod power measuring system and associated on-line measurements, such as rod elongation and noise measurements for studies of the rod thermal performance, are performed in the in-pile loops The ramp tests are a form of integral performance tests where the complex interplay between the pellets and the cladding of a power reactor fuel rod is reproduced. The primary test objectives are:

- Determination of the failure boundary and the failure threshold, see Figures 1 [1] and 2

- Establishing of the highest "conditioning" ramp rate that safely avoids failure occurrence.

- Study of the failure initiation and progression under short time over-power transient operation beyond the failure threshold.

- Proof testing of potential pellet-clad interaction (PCI) remedies.

Other, more specific test objectives have also been pursued in some ramp projects.

The <u>rod overpressure experiments</u> utilize the on-line measurements associated with the ramp tests combined with non-destructive examinations between reactor cycles and destructive examinations after the irradiation. When LWR fuel is used at higher and higher burnups the question of how the fuel might behave when the end-of-life rod internal pressure becomes greater than the system pressure attracts a considerable interest. On one hand end-of-life overpressure might lead to clad outward creep and an increased pellet-clad gap with consequent feedback in the form of increased fuel temperature, further fission gas release, further increases in overpressure etc. On the other hand increased fuel swelling might offset this mechanism In connection with such considerations the Rod OverPressure Experiments were initiated They will be discussed in Section 4.4.

The <u>defect fuel degradation experiments</u> also utilize the on-line measurements associated with the ramp tests and combine these with non-destructive and destructive examinations after the irradiation. Fretting type failures are predominant causes of the few fuel failures that have occurred in recent years in LWRs These primary failures are sometimes followed by secondary failures which frequently cause considerably larger activity releases In such cases the subsequent degradation of the defect fuel rods by internal hydriding of the cladding and by oxidation of the fuel are the common destructive mechanisms. In these tests an irradiation test scheme, adapted to the experimental conditions in the R2 test reactor was introduced. This scheme offers the possibility of executing comparative investigations of the process of degradation of commercial types of LWR fuel under simulated primary defect conditions as well as of the mechanisms involved. These experiments will be discussed more in detail in Section 5

The fuel testing projects executed at Studsvik have been organized under three different types of sponsorship:

International (multilateral) fuel projects

- Jointly sponsored internationally on a world-wide basis

- Project information remains restricted to the project participants throughout the project's duration and some predetermined time after project completion

Bilateral fuel projects

- Sponsored by one single organization, or a few co-operating organizations.

- Project information remains restricted to the sponsor, sometimes published later.

In-house R&D work

- Sponsored by STUDSVIK NUCLEAR.

Several new hot-lab techniques have also been introduced in recent years [2] The STUDFAB process for refabrication of rodlets from full-size irradiated fuel was mentioned above. Fuel ceramography can include scanning electron microscopy (SEM) and electron probe microanalysis (EPMA).

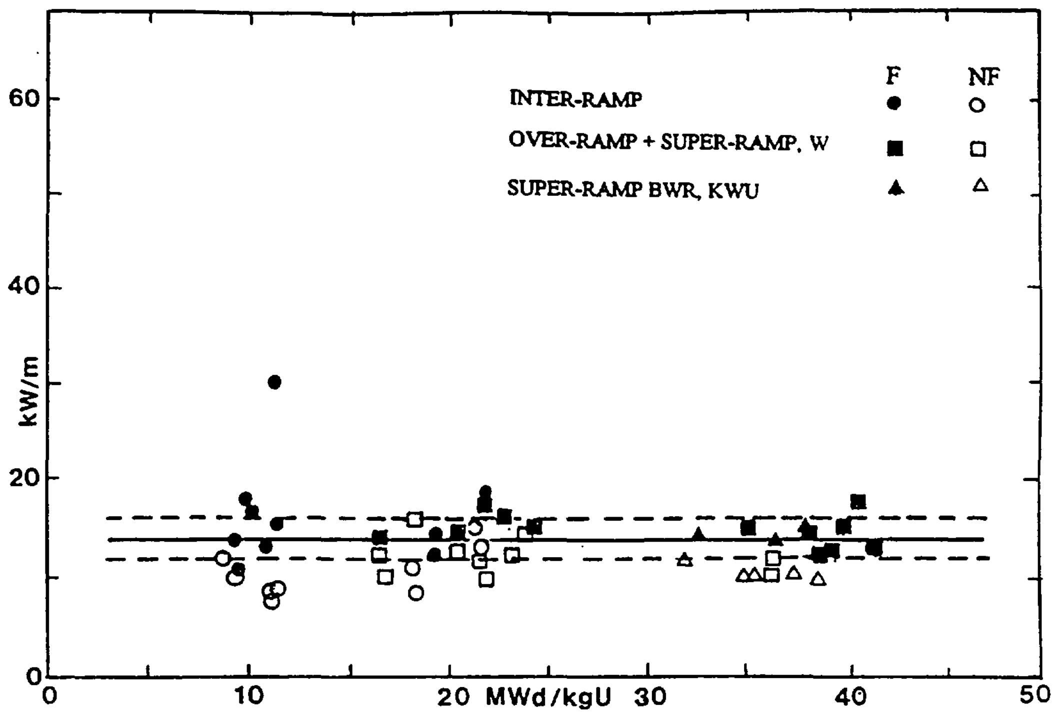
Descriptions of the fuel testing facilities and the associated techniques will be given in Section 2. The noise measurements introduced for studies of the rod thermal performance have been described elsewhere [3-5]. Several other novel testing techniques have also been introduced [6]. A very fast ramp rate, up to 3 000 W/(cm·min) can be used to obtain fast power transients and to determine the pellet-clad interaction/stress-corrosion cracking (PCI/SCC) failure boundary. Still faster, "ultra-fast" ramps, are discussed in Section 6.2. On-line elongation measurements can be performed during ramp tests, Figure 3 Test fuel rodlets can be fitted with on-line pressure transducers through a refabrication process.

Since the early 1970's, a long series of bilateral and international fuel R&D projects, primarily addressing the PCI/SCC failure phenomenon have been conducted under the management of STUDSVIK NUCLEAR [1, 6-8]. These projects are pursued under the sponsorship of different organizations; the bilateral projects mainly different fuel vendors and the international projects different groups of fuel vendors, nuclear power utilities, national R&D organizations and, in some cases, licensing authorities in Europe, Japan and the U.S. In most of the projects the clad failure occurrence was studied under power ramp conditions utilizing the special ramp test facilities of the R2 reactor As mentioned the current projects are not limited to PCI/SCC studies but also include other aspects of fuel performance, such as end-of-life rod overpressure [9-12] and defect fuel degradation [13-16]. An overview of the projects that have been completed and those that are currently in progress or planned has been published [8], recent and upcoming projects will be discussed in this paper In most cases, the test fuel was base irradiated in commercially operating light water power reactors. In some instances, however, the base irradiation took place in BOCA rigs in the R2 reactor

In general, the international fuel R&D projects can be divided into two main categories:

- Projects aimed at decreasing the fuel costs by increasing fuel utilization and reactor availability.

- Projects providing data for fuel-related safety considerations.



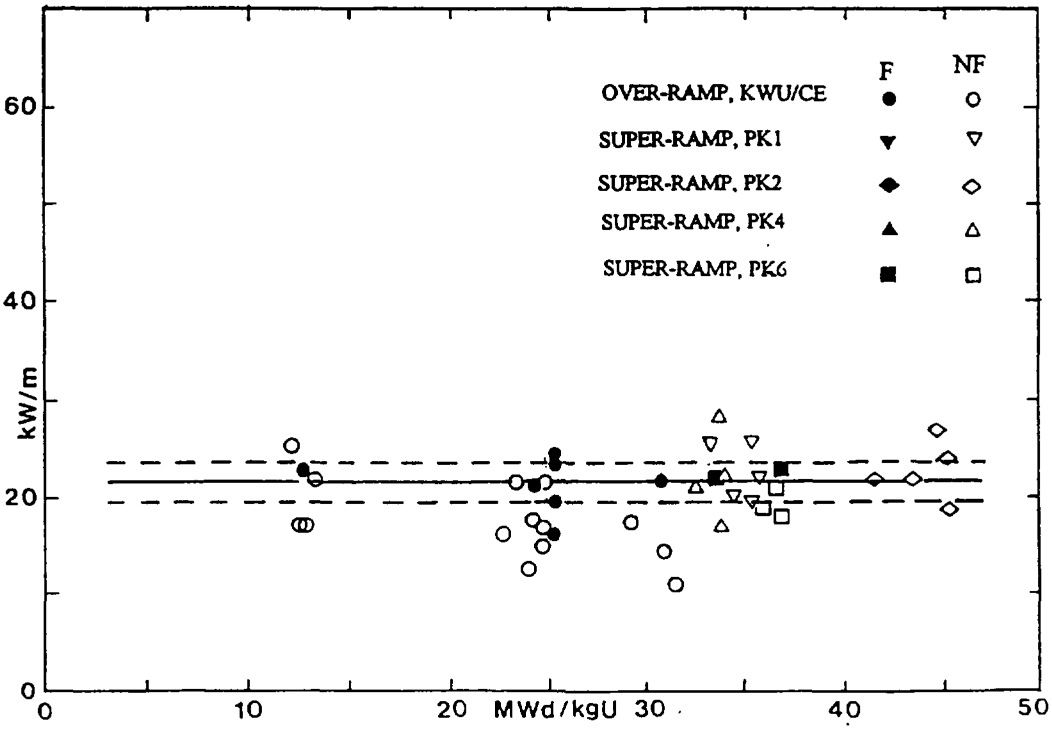


Figure 1
Summary of Some Data From the INTER-RAMP, OVER-RAMP and SUPER-RAMP Projects. The Incremental Failure Threshold as a Function of Burn-Up for Different Groups of Fuel Rods.

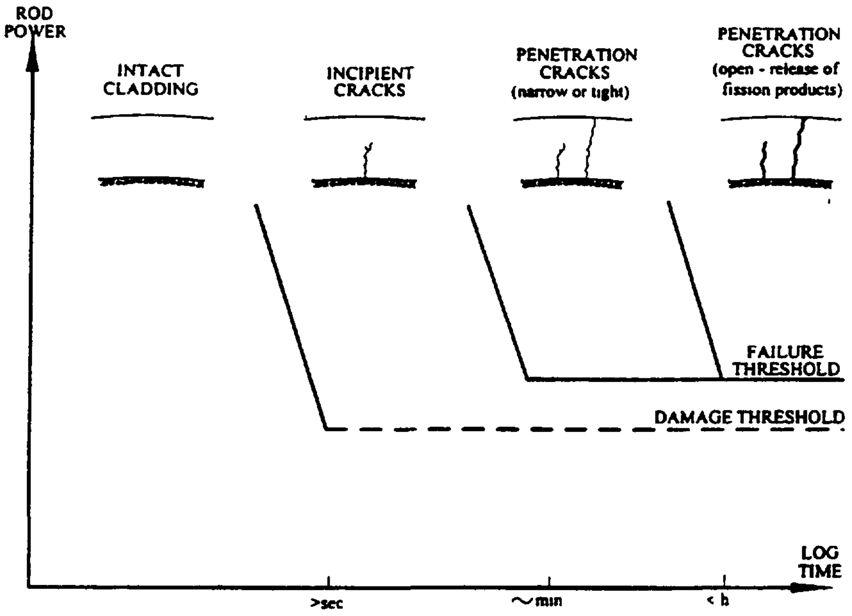


Figure 2
Schematic PCI Failure Progression Diagram.

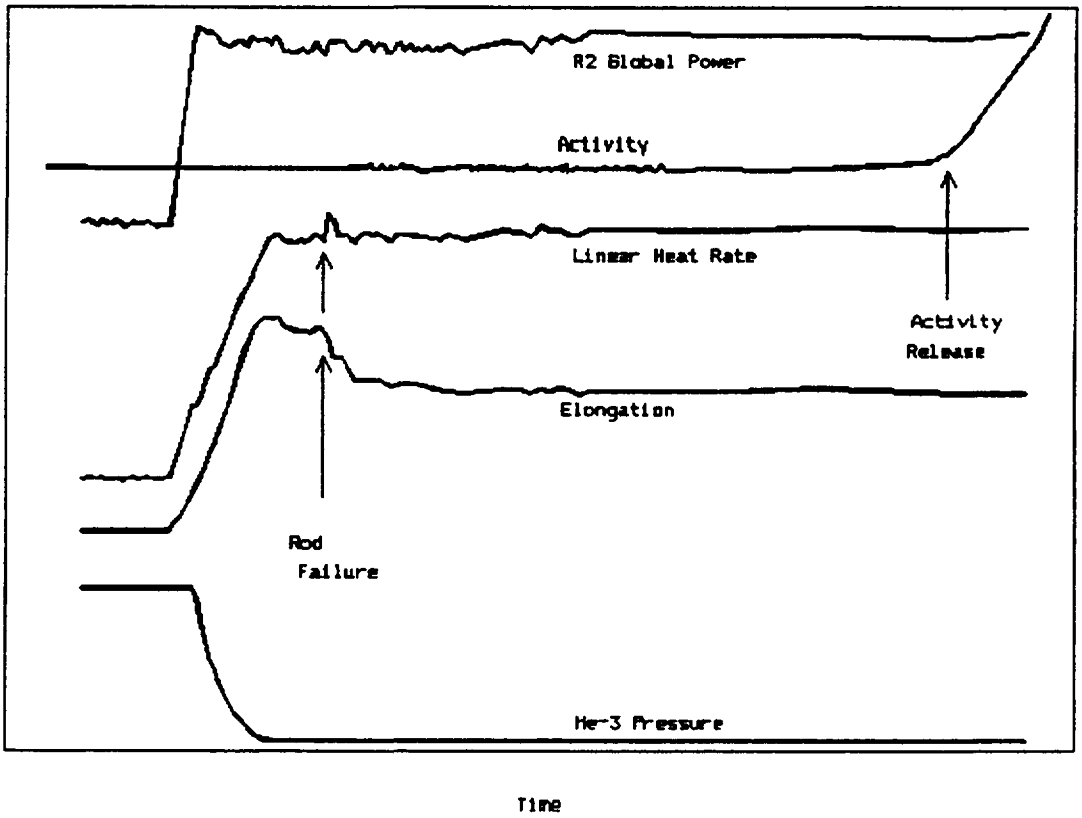


Figure 3
On-Line Measurements During a Ramp Test Showing a PCI Failure Event.

A typical example of the former category is the SUPER-RAMP project [17], where several groups of fuel from different fuel vendors and with different "PCI remedies" were tested. A summary of some of the data from this category of projects is shown in Figure 1 [1] Reviews of the projects in the latter category have also been published [18-19] Presently a new international project, combining the features of these two series, is under discussion· the ULTRA-RAMP project, to be discussed in Section 6.2

The test data are often used as "benchmarking" data in the project participants' own fuel modeling work. In recent years many ramp test data have also been analyzed with the INTERPIN code, developed by STUDSVIK [20-22]. INTERPIN is a fuel performance code which satisfies real-time simulation requirements when implemented on a minicomputer.

European, Japanese and U S. fuel manufacturers and research organizations have also for many years been utilizing the R2 test reactor and the associated hot-cell laboratories for bilaterally sponsored research[1]. ABB Atom AB has made many series of ramp tests. General Electric Co. has executed several series of ramp tests at R2, as part of the efforts to develop the zirconium barrier fuel concept. Some of the ramp techniques requested were innovative, for example the "double ramping" of the test rods. Other major customers are Exxon Nuclear Co. (later Advanced Nuclear Fuels Corporation, later Siemens Nuclear Power Corporation, now Siemens Power Corporation), B&W Fuel Company (now Framatome Cogema Fuels), Hitachi Ltd., Mitsubishi Heavy Industries, Ltd., and Toshiba Corporation. Tests have also been performed on behalf of other organizations but the results have not always been published.

STUDSVIK NUCLEAR's in-house R&D work is mainly associated with improvements of test irradiation techniques, instrumentation and post-irradiation examination, all in support of ongoing or upcoming irradiation projects Progress in these areas has made it possible to achieve important progress in fuel research. For example, the characterization of the PCI failure progression in some recent projects was only made feasible through a combination of several new techniques. These included a very fast ramp execution using a ramp rate of up to 3 000 W/(cm·min), compared to the previous maximum of 200 W/(cm·min), a prompt detection of the through-failure event using the on-line elongation detector and a subsequent special clad bore inspection technique. Another result of the in-house R&D work is the noise measurement technique mentioned above

STUDSVIK NUCLEAR has also been carrying out an in-house R&D program aimed at improving the performance of LWR fuel by the utilization of a design concept with cladding tubes which have been "rifled" on a micro-scale [23] Results of R2 irradiations of such fuel and the associated modeling work have been published [24-31].

2 The R2 Test Reactor - a Versatile Tool for Fuel R&D

The R2 reactor is a 50 MW(th) materials testing and research reactor developed in the USA and in operation since 1960, Figure 4. The R2 reactor has a high neutron flux, and special equipment for performing sophisticated in-pile experiments. An important feature of the R2 test reactor is that it is possible to run fuel experiments up to and beyond failure of the cladding This is obviously not possible in a commercial power reactor Detailed descriptions of the R2 reactor have been published elsewhere [32-33] Fuel tests are performed in the two high pressure loops in the core. These in-pile loops can be operated under BWR pressure and temperature conditions and are used for all irradiations under power changes, some of the base irradiations, some materials testing experiments and the in-pile corrosion experiments, discussed in Section 3. Most base irradiations of test fuel, i.e. irradiations at constant power, where fuel burnup is accumulated under well defined conditions, are performed in Boiling Capsules (BOCA rigs) The R2 core has an active length of 60 cm

2.2 Ramp Test Facility

Ramp testing in the R2 reactor began in 1969. In the present Ramp Test Facility, introduced in 1973, the fuel rod power during a ramp test in a loop is controlled by variation of the ^3He gas pressure in a stainless

[1] A list of available publications can be obtained upon request from the authors

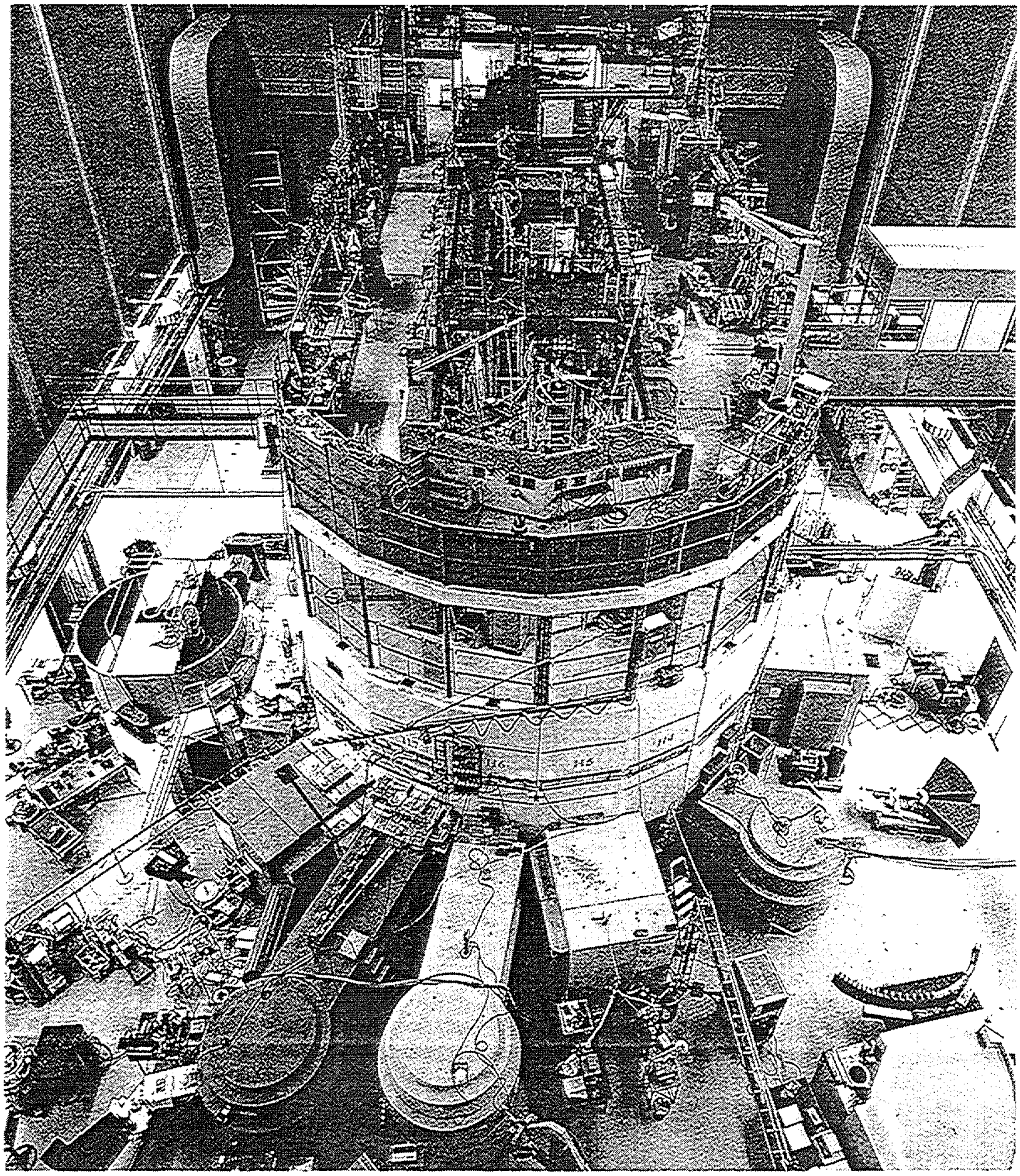


Figure 4
A General View of the R2 Test Reactor.

steel double minitube coil screen which surrounds the fuel rod test section. The principle of operation of this system is based on the fact that ^3He absorbs neutrons in proportion to its density, which can be varied as required by proper application of pressure.

The efficiency of the ^3He neutron absorber system makes it possible to increase test rod power by a factor of 1.8 to 2.2 (depending on the fissile content of the fuel). The ^3He absorber system is designed to achieve a 100 % power increase within 90 seconds, when operating with the normal pressure variation (bellows system).

In order to achieve a higher power increase than a factor of about 2, the reactor power must be increased before or simultaneously with the "^{3}He ramping". This technique with combined ramp systems is called "double step up-ramping". The technique makes it possible to increase the test fuel rod power by a factor of about 3. In the Ramp Test Facility ramp rates can be achieved in the range of 0.01 W/(cm·min) to about 3 000 W/(cm·min)

2.3 Boiling Capsules - BOCA Rigs

The Boiling Capsule (BOCA) facility is used for irradiations at constant power where fuel burnup is accumulated under well-defined conditions. It was introduced in 1973.

Up to five BOCA rigs can be operated simultaneously in the reactor. Two independent pressurization systems are available, each capable of supplying 3 to 5 BOCA rigs with water. Each BOCA rig is connected to a separate outlet circuit.

3 INCA - A New Facility for Waterside Corrosion Studies in the R2 Test Reactor

The experimental program at Studsvik has recently been extended to include investigations on clad and structural materials in an in-pile corrosion rig, INCA (In-Core Autoclave), designed to make it possible to study the in-core corrosion behavior of stainless steel and zirconium alloys

A reference electrode for long term in-pile corrosion potential measurements has been developed Such an electrode together with a water supply and analysis system will make it possible to perform long-term irradiations of materials under well-defined water chemistry conditions in a BWR environment Possibilities to move the electrode and the test specimens up and down in the core could easily be introduced in order to monitor the corrosion potential changes due to radiolysis.

Waterside corrosion becomes a more and more important phenomenon with increasing power plant efficiency The more demanding operating conditions, such as higher fuel burnup and longer in-reactor fuel cycles, make the cladding corrosion performance a limiting factor.

A number of different conditions influence the corrosion and hydriding behavior of cladding materials

- Water chemistry
- Crud deposits
- Material characteristics, e.g. annealing treatments
- Neutron dose
- Hydrogen pick-up
- Boiling conditions

The new INCA facility has been developed in collaboration with our sister company Studsvik Material AB The main feature of the new facility is the ability to control and monitor the water chemistry Therefore the facility is of the once-through type, which means that the rig is supported by a water supply system of its own and that the water passes the rig only once The desired water chemistry is created by adding impurities and additives to a purified water flow, close to the test area. This technique has successfully been used out-of-pile by Studsvik Material AB to obtain well characterized conditions The facility has a flexible design and can easily be modified to suit different types of corrosion and water chemistry experiments

The corrosion test rig, which can be seen in Figure 5, is the in-pile part of the INCA facility and is installed in one of the main in-pile loops in the R2 test reactor. It consists of two major parts, the rig tube and the electrode rod

The rig tube separates the water system of the test rig from the in-pile loop main flow. The inlet water to the rig, which is degassed high purity water, is fed from a separate water supply system. It is heated by the loop water in a preheater coil and subsequently led into the rig tube. Additives and impurities (oxygen, hydrogen peroxide, zinc etc) can be added both before the rig and inside the rig just before the in-core test

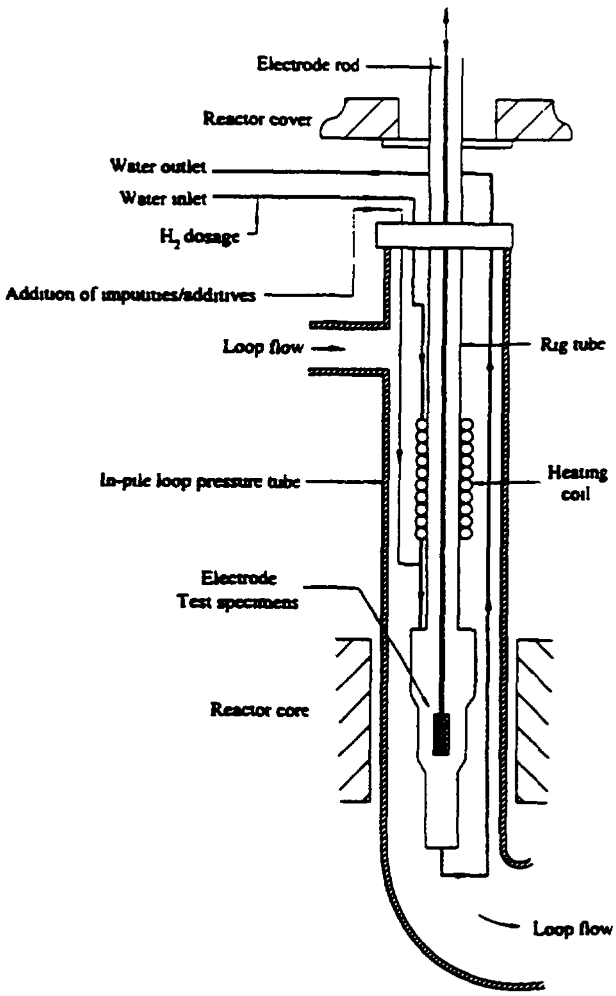


Figure 5
INCA – Conceptual Test Rig Arrangement.

area in order to establish the desired water chemistry The presently available inner diameter of the rig tube is 21 mm, but in the future it will be possible to vary the diameter or to have different diameters in different axial sections of the core.

The electrode rod is installed in the rig tube and is a carrier for the test specimens, the reference electrodes etc The tube for the injection flow is also assembled on the rod This arrangement makes it possible to change the electrode rod from one reactor cycle to another For the moment the rod is bolted on top of the rig tube, but it could easily be rebuilt to be movable up and down in the core section and below the core during operation

The INCA facility can operate under both BWR and PWR conditions Fast (> 1 MeV) and thermal neutron fluxes up to 1 9 and 2 0x10^{14} n(cm^2·s) respectively, can be achieved

The facility is suitable for different kinds of experiments, for instance materials irradiations, waterside corrosion studies and in-core materials testing, all under controlled water chemistry It has been in operation since March 1995 One of the objectives has been to develop reference electrodes for long term in-pile use. A radiolysis study where experimental measurements and computer results were compared has also been performed

4 STUDSVIK's International Fuel R&D Programs

4.1 Introduction

A long series of international fuel R&D projects, primarily addressing the PCI/SCC (Pellet-Cladding Interaction/Stress Corrosion Cracking) failure phenomenon have been conducted under the management of STUDSVIK NUCLEAR since 1975. In most of the projects the clad failure occurrence was studied under power ramp conditions utilizing the special ramp test facilities of the R2 test reactor The recent projects have not been limited to PCI/SCC studies but some of them also include other aspects of fuel performance, to be discussed below During the late 1970's and 1980's the series of international PCI ramp projects branched out in two directions. One series was initially concentrated on the PCI phenomena under normal operational conditions in different types of BWR fuel rods subjected to increased burnup. These projects were in a broad sense aimed at decreasing fuel costs by increasing fuel utilization and reactor availability. Most of the bilateral fuel projects fall into this category. The other series of international projects was concentrated on more safety-oriented issues, aimed at providing data for fuel-related safety considerations. The first-mentioned series of international projects includes the INTER-RAMP, OVER-RAMP, DEMO-RAMP I, SUPER-RAMP, SUPER-RAMP EXTENSION, SUPER-RAMP II/9x9 and SUPER-RAMP III/10x10[1] projects. The second series includes the INTER-RAMP, DEMO-RAMP II, TRANS-RAMP I and TRANS-RAMP III[1] projects. Presently a new international project, combining the features of these two series, is under discussion: the ULTRA-RAMP project[1]

In later years two other types of fuels R&D project have been introduced: end-of-life rod overpressure studies and defect fuel degradation experiments. The BWR end-of-life overpressure project was designated ROPE I (Rod OverPressure Experiment). The defect fuel degradation projects are designated DEFEX and DEFEX II[1].

Furthermore other types of projects are also under discussion, they will be described in Section 6. An overview of the 9 international BWR projects that have been completed is given in Table 1

4.2 Ramp Resistance - BWR Fuel

The first of STUDSVIK NUCLEAR's international fuel projects, INTER-RAMP, was executed in 1975-79 under the sponsorship of 14 organizations from 9 countries. The results have been described [34-35]. The main objectives of the program were to investigate systematically the failure propensity and associated phenomena of well-characterized BWR fuel rodlets when subjected to fast overpower ramps under relevant and well-controlled experimental conditions.

The DEMO-RAMP I project was executed in 1979-82. The results have been described [36]. The main objective of this program was to investigate the effects of two PCI remedies, annular pellets and niobia doping of the UO_2, on the ramp behavior, especially the fission product release and the pellet-cladding mechanical interaction (PCMI), of 8x8 type fuel rodlets.

The SUPER-RAMP project was executed in 1980-83. The results have been described [17, 37]. The main objective of this project, which was co-sponsored by 20 organizations from 11 countries, was to make a valid contribution to the general understanding of the PCI phenomenon for commercial type LWR fuel rods at (then) high burnup levels under power ramp conditions. For the BWR subprogram the more specific objectives were to

- Establish the PCI failure threshold for standard type test fuel rods on fast power ramping at burnup levels exceeding about 30 MWd/kgU.

- Identify any change in failure propensity or failure mode as compared to the failure behavior at lower burnup levels.

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[1] These projects will be discussed in Section 6

- Establish a failure-safe reduced power ramp rate for passing through the PCI failure region

In the SUPER-RAMP EXTENSION project, which was executed in 1984-86, further ramp tests were performed

In the SUPER-RAMP II/9x9 project, executed in 1987-90, the failure boundary of 9x9 type fuel rodlets, irradiated in a commercial power reactor to a burnup of 25 MWd/kgU, was determinad [38], see Figure 6

4.3 Safety-Oriented Ramp Resistance Studies

In a series of international fuel research projects (the INTER-RAMP, DEMO-RAMP II, and TRANS-RAMP I projects) it was demonstrated by means of power transient tests (intentionally interrupted power ramp tests) that when BWR test fuel rods were exposed to overpower ramps of increased severity, they exhibited a regular Pellet-Clad Interaction (PCI) failure progression (Figure 2). A higher transient peak power level resulted in an earlier fission product leakage from the fuel rods. Stress corrosion cracks (SCC) initiated promptly on fast up-ramping, i.e. within the order of seconds, and penetrated the cladding wall within about a minute Depending on the actual power "over-shoot" and the time spent beyond the failure threshold, the transient passed consecutively through a number of power-time regions defining the progressive steps of the failure process

Some of the BWR fault transients of the types that might be expected to occur once in a reactor year or once in a reactor lifetime carry a potential for causing PCI fuel clad damage or failure (i.e. through-wall crack penetration) on surpassing the PCI failure threshold. A question of prime concern is then whether a fast single transient of the type mentioned will result in fuel failure due to PCI, eventually followed by a release of radioactivity to the coolant.

In the INTER-RAMP (IR) Project, executed during 1975-79 [34, 35], BWR fuel rods were subjected to power transients of varying "over-power" levels beyond the PCI failure threshold, where cladding failure and fission product release occur after a sufficient time. The results demonstrated a systematic time dependence of the fission product release to the coolant from the failed fuel rods. An increase in the power "over-shoot" of 5 kW/m caused a decrease of the time to fission product release by a factor of about 10.

In the DEMO-RAMP II (DRII) Project, 1980-1982 [39], BWR fuel rods of intermediate burnup levels were subjected to intentionally interrupted short-time power transients at linear heat ratings a few kW/m above the PCI failure threshold. No cladding failures were detected after the transients but a large number of non-penetrating (incipient) cracks were observed. They had been formed very rapidly, within a minute These cracks could be observed by destructive post-irradiation examinations only. The crack depths ranged from 10 to 60 percent of the cladding wall thickness.

In the TRANS-RAMP I (TRI) Project, 1982-1984 [40], BWR fuel rods of intermediate burnup levels were subjected to simulated short time power reactor transients of a wide range of "over-powers" but at characteristic very fast ramp rates, in the range of 10 000 W/(cm min). The test results were similar to the DR II results and permitted a tentative interpretation of the PCI failure progression in terms of well-separated power/time boundaries defining 1) crack initiation at the inside surface of the cladding, 2) through-wall crack penetration and 3) leakage of fission products to the coolant water

The crack initiation and penetration processes can only be detected by special hot cell laboratory or test reactor techniques. The delay of the fission product release indicates that in power reactors cladding failures that occur during fast transients and terminate before any outleakage of fission products may go undetected until manifested in later operational manoeuvers

4.4 Studies of "Lift-Off" Phenomena

When LWR fuel is used at increasingly higher burnups the question of how the fuel might behave when the end-of-life rod internal pressure becomes greater than the system pressure attracts a considerable interest

Table 1
Overview of STUDSVIK NUCLEAR's International BWR Fuel R&D Projects 1975-1991

Project (duration)	Fuel Type (No of rods)	Base Irradiation (MWd/kgU)	Research Objectives	Data published
INTER-RAMP (1975-79)	BWR (20)	R2 (10-20)	Failure threshold Failure mechanism Clad heat treatment Modeling data	Yes (Ref 1,2)*)
DEMO-RAMP I (1979-82)	BWR (5)	Ringhals I (15)	PCI remedies (Annular, niobia doped pellets)	Yes (Ref 3)
DEMO-RAMP II (1980-82)	BWR (8)	Würgassen (25-29)	Failure threshold PCI damage by overpower transients	Yes (Ref 4,5)
SUPER-RAMP (1980-83)	BWR (16)	Würgassen (30-35) Monticello (30)	Failure threshold High burn-up effects PCI remedies Safe ramp rate Gd fuel	Yes (Ref 6)
SUPER-RAMP EXTENSION (1984-86)	BWR (9)	Oskarshamn 2 (27-31)	Safe ramp rate	No
TRANS-RAMP I (1982-84)	BWR (5)	Würgassen (18)	Failure boundary Crack init. and prop. Structural changes Fission gas release Modeling data	Yes (Ref 4,7)
ROPE I (1986-89)	BWR (4)	Ringhals (36)	Investigate clad creep-out as a function of rod overpressure	Yes (Ref 8)
SUPER-RAMP II/9x9 (1987-91)	BWR (4)	Dresden (30)	PCI performance	Yes (Ref 9)
DEFEX (1993-95)	BWR (6)	Initially unirradiated rodlets	Study secondary damage formation in fuel rods with simulated fretting defects	No

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On one hand end-of-life overpressure might lead to clad outward creep and an increased pellet-clad gap with consequent feedback in the form of increased fuel temperature, further fission gas release, further increases in overpressure etc On the other hand increased fuel swelling might offset this mechanism

In connection with such considerations STUDSVIK NUCLEAR initiated a limited in-house R&D activity in order to investigate the ability to detect the phenomena of interest: the ROPE Pre-project [9]. This was followed by an international Rod OverPressure Experiment (ROPE I project).

The purpose of the ROPE I project was to investigate the behavior of BWR fuel rods. ABB-ATOM 8x8 fuel rods, irradiated in the Ringhals 1 reactor to a burnup of about 35 MWd/kgU were tested [12] The rods

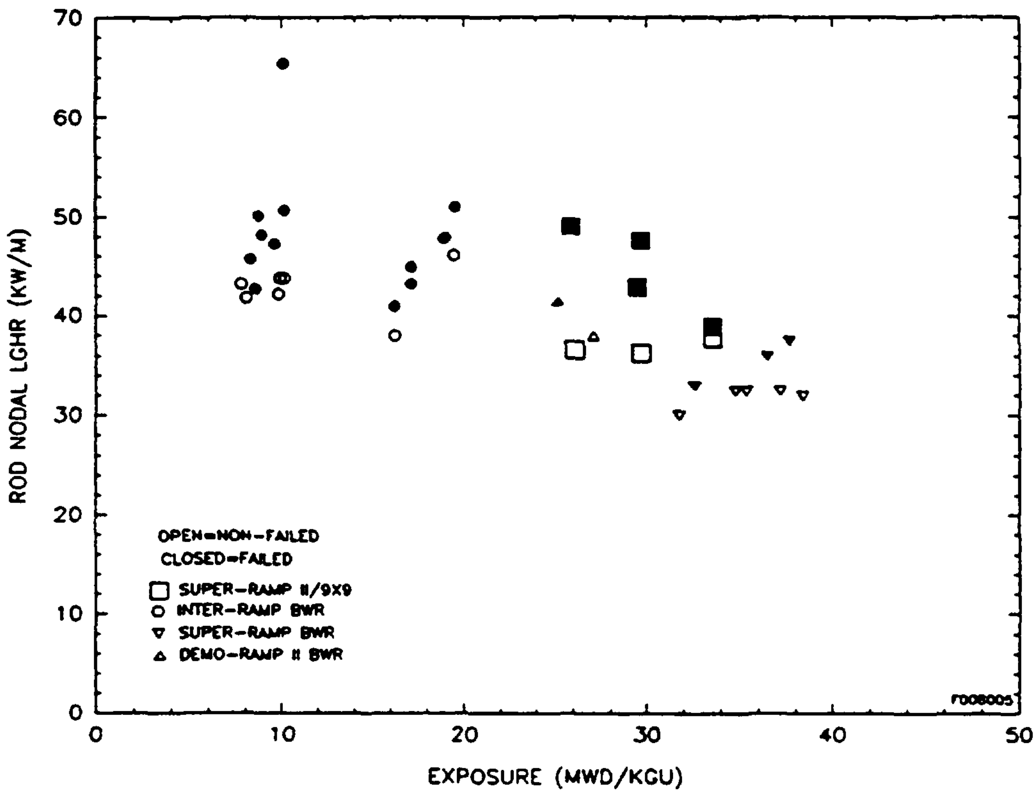

Figure 6
Comparison of 9x9 Ramp Data with Data from Other Ramp Programs.

were refabricated and pressurized to give hot internal overpressures during R2 irradiations of approximately 0.4 and 14 MPa, respectively. The clad creepout and the time dependent changes in fuel rod conductance were investigated as functions of rod overpressure. The fuel rods were irradiated one at a time during three 3-day cycles in an instrumented rig in an in-pile loop in the R2 reactor. During these cycles the fuel rod thermal response was determined on-line by noise analysis [3, 4]. Between the 3-day cycles the rods were irradiated together for a total of six 15-day cycles without the on-line measurements mentioned. In the intermissions between these cycles, profilometry measurements were performed in the R2 reactor pool. After irradiation, the rods underwent non-destructive and destructive examinations. The rod with the highest overpressure had a measured diametral cladding outward creep strain of 11 µm after 1634 hours irradiation, with no apparent primary creep This exceeded the expected pellet diameter increase attributable to fuel matrix swelling, since the average swelling rate measured in the fuel would only have resulted in a pellet diameter increase of 3.2 µm, after 1634 hours. Thus it was successfully demonstrated that a BWR fuel rod with an internal overpressure in excess of the pressure causing a cladding creepout rate as fast as the fuel solid swelling rate, can be operated at a Linear Heat Rate of up to 22 kW/m for more than 2 months without any apparent detrimental effect

5 STUDSVIK's Defect Fuel Degradation R&D Projects

5.1 Technical Background

Several instances of fuel failure in nuclear power plants have been reported [41] In a number of cases, primary defects in the fuel have led to the development of secondary defects caused by internal hydriding. An example was the failure of a liner rod in the Oskarshamn 3 BWR plant in 1988. The rod exhibited five longitudinal cracks, extending along most of its length [42]. Extensive hydriding of the cladding was observed. The primary cause seemed to be a fretting defect at the top of the fuel rod. Similar cases have been reported from other Swedish NPPs, a BWR plant in Switzerland and BWR plants in the USA, among others. Although the failure rate is small, the economic consequences can be unacceptable.

The mechanism for hydride formation has been theoretically evaluated [43, 44]. It is proposed that water penetrating through the primary crack oxidizes the fuel and the inside surface of the cladding, thus producing an excess of hydrogen. An obstruction of the axial gas communication will prevent a steady supply of steam, since the mass flow rate is proportional to the third power of the hydraulic diameter in the fuel [45]. Consequently, the criterion for hydride formation may be fulfilled in locations at some distance from the primary defect, especially if there is a constriction in between, caused, for instance, by gap closure due to a power maximum. Observations at Studsvik support the statement that the production rate of hydrogen is highest at the moment of clad defection; during the next few days, however, it levels off at a considerably lower value [46]. Massive hydride formation in the cladding will give rise to large circumferential stresses, eventually causing the cladding to split.

5.2 Technical Considerations

An experimental test scheme to be used at the R2 reactor is limited by the following non-compromisable constraints [47, 48].

- The length of the test fuel rodlets must not exceed the active core height (600 mm).

- The release of fission products and fuel material from the defected fuel rodlets has to be minimized in order not to cause intolerable contamination of the in-pile test loop.

When planning the experimental scheme it was assumed that the peaked power profile of the R2 reactor would cause a constriction of the pellet-clad gap at the middle section of the test rodlet due to mechanical interaction, if the Linear Heat Rate was sufficiently high. This situation would to a large extent prevent steam from penetrating downwards from the top of the test rodlet. In this way the conditions leading to hydriding in a power reactor fuel rod would be simulated (Figure 7).

In order to avoid contamination of the test loop, the intrusion of water was simulated by applying a water reservoir in an extended stainless steel plenum on top of the rodlet using a process developed by STUDSVIK. Consequently, no leakage of fission products to the test loop is expected to take place unless the cladding fails due to hydriding. This scheme offers the added advantage that it is possible to analyze the released fission gases and the hydrogen that has been formed, if the cladding does not rupture (Figure 8). In the experiments the amount of water in the reservoir is around 0.8 cm^3, out of which 0.2 cm^3 evaporates as the fuel rodlet is inserted into the test loop. At the beginning of an experiment the water vapor is distributed along the fuel stack. When the rodlet is lowered further into the reactor core, the fuel-clad gap, which is initially of the order of 0.15 mm, closes, provided that the Linear Heat Rate is above 30 kW/m. The rodlets are subjected to neutron radiography before and after each test irradiation in order to check the distribution of water. It has been verified that part of the water is left in the reservoir after the test.

5.3 Exploratory Defect Fuel Experiments

The two objectives of the test program were:

1. Demonstration of the feasibility to accomplish internal hydriding using the simulation scheme described above.

2. Investigation of remedies to suppress hydriding, given that the above objective is fulfilled.

The program was made up of two parts:

In Defect Fuel Experiment No. 1 (DEF-1) two fuel rodlets with two different types of cladding available from earlier STUDSVIK projects were tested, mainly for exploratory purposes. For Defect Fuel

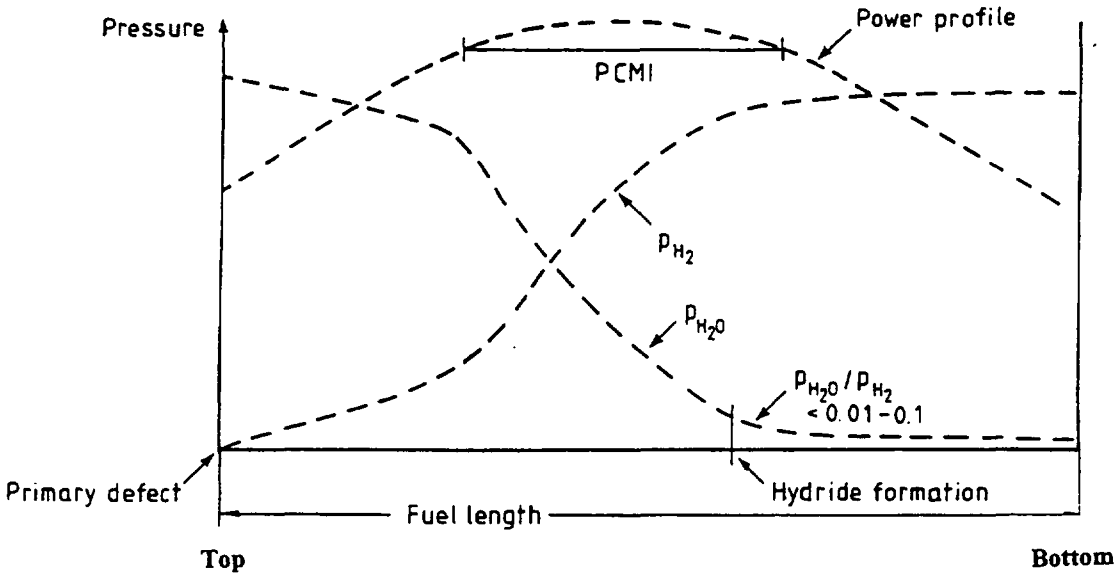

Figure 7
Conceptual Power Profile of the R2 Reactor, Distribution of Partial Pressures and Location of Hydride formation Following Water Intrusion.

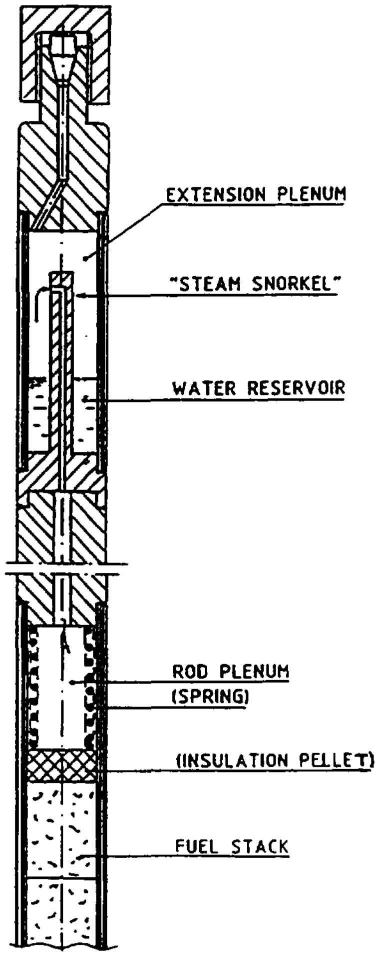

Figure 8
Array Simulating Primary Defects in Test Fuel Rodlets

Experiment No. 2 (DEF-2) five rodlets with three kinds of cladding were manufactured for the project. The rodlets were designed to be similar to 8x8 BWR fuel with Zircaloy-2 cladding In order to simplify the manufacture, only unirradiated fuel was used [47, 16].

In DEF-1 standard cladding was compared to so-called rifled cladding, which is a proposed remedy for avoiding hydriding [30]. In the second series, standard design cladding, cladding with sponge zirconium liner and rifled cladding were irradiated. Table 2 gives data for the test rodlets used in DEF-1 and DEF-2

Mainly two parameters were varied in the experiments:

1. The Linear Heat Rate was kept at 45 kW/m in all tests with the exception of DEF-2.4. For this test 50 kW/m was chosen

2. The rodlets were generally irradiated for a full reactor cycle, amounting to between 15 and 17 days. Exceptions were test DEF-2.4, which lasted 1 day, and test DEF-2.5, which went on for 5 days.

In addition, a hafnium shield was used in test DEF-2.5, reaching a level of 160 mm from the bottom of the fuel stack. In this way, the local power at the lower end of the test rodlet decreased. The closure of the gap was then expected to be less rigid than in the other tests, giving more room to the mixture of hydrogen and water vapor.

The test rodlets were investigated in the following ways:

- Elongation measurements were done on-line during irradiation.

- Profilometry was carried out on the cladding before and after irradiation.

- Eddy current testing in search of cladding defects after irradiation was done before and after irradiation for most tests

- Neutron radiography was done after irradiation in order to find out if hydriding had taken place. A check was also done before irradiation.

Destructive examination comprised determination of gas composition and metallography and ceramography investigation of the cladding and the fuel. Some results have been published [47]. The observations are summarized in Table 2.

The on-line measurements of the elongation of the test rodlets yielded seven curves which are shown in Figure 9. The elongation curves exhibit a similar shape. Initially, there is a sharp increase, when the reactor power is increased, until the Linear Heat Rate reaches its predetermined value Then follows a relatively fast relaxation phase until a minimum is arrived at. As the irradiation progresses, a monotonous increase is obtained. In tests DEF-1.1, DEF-2.1, DEF-2.3 and DEF-2.4 a distinct peak is superimposed on the elongation curve during or immediately after the relaxation period

In general, the profilometry scans look quite similar before and after irradiation The curves from tests DEF-2.3 and DEF-2 4, however, show pronounced ridging after irradiation.

The eddy current scanning of the rodlet in test DEF-1.1 shows ridging as a result of the irradiation. Moreover, there is a clear sign of a secondary defect at the bottom of the rodlet (Figure 10). Ridging is clearly shown for the rodlets used in tests DEF-2.3 and 2 4 too There is also an indication of a secondary defect at the bottom of the rodlet in test DEF-2.4.

The neutron radiograph of the rodlet from test DEF-1.1 shows that heavy hydriding has taken place at the 6th pellet level from the bottom There is also minor hydriding at the 4th and 2nd pellet levels

Table 2a
Defect Fuel Expenments 1 and 2 – Data for Test Fuel Rodlets

Test No.	Rodlet No.	Type of Cladding[1)]	Fuel Parameters
DEF-1 1	1117	Zr-2 std	L/D 1 2, dished, no chamfer, density 10.17 g/cm³, gap 0 15 mm
DEF-1 2	2050	Zr-2 nf	L/D, 1 05, no dish, slight chamfer, density 10 41 g/cm³ (ABB standard), gap 0 20 mm
DEF-2 1	2332	Zr-2 std	same standard as above, gap 0 15 mm
DEF-2 2	2330	Zr-2 lin	same standard as above, gap 0 15 mm
DEF-2 3	2328	Zr-2 nf	same standard as above, gap 0 15 mm
DEF-2 4	2333	Zr-2 std	same standard as above, gap 0 15 mm
DEF-2 5	2329	Zr-2 nf	same standard as above, gap 0 15 mm

Table 2b
Defect Fuel Expenments 1 and 2 – Test Parameters and Observation from Elongation Measurements, Eddy Current Scans and Profilometry

Test No.	LHR (kW/m) Peak	Irradiation time (days) after start	Elongation peak, hrs	Profilometry	Eddy Current[1)]	Neutron Radiography[1)]
DEF-1 1	45	16 5	5	-	R + DF	H
DEF-1 2	45	17	None	No R	(R)	No H
DEF-2 1	45	17	6	No R	(R)	No H
DEF-2 2	45	14 5	None	No R	(R)	No H
DEF-2 3	45	17 5	19	R	R	No H
DEF-2 4	50	1	18	R	R+D	H
DEF-2 5	46	4 5	None	No R	(R)	No H

1) Std = Standard
Rif = Rifled
Lin = Liner
R = Ridgning
DF = Defect
D = Small Defect
H = Massive Hydnde

In comparison, the rodlet irradiated in test DEF-1.2 did not exhibit any hydriding.

The neutron radiograph from test DEF-2.4 shows hydriding at the level of the bottom pellet.

The radiographs of the other rodlets from Defect Fuel Experiment No. 2 gave no indication of hydriding.

The metallography data published so far reveal a large hydride precipitate in the cladding at the 6th pellet level in the rodlet from test DEF-1.1. This precipitate was investigated as shown in the micrograph in Figure 11.

Phenomena influencing the growth of the cladding are in summary:

- The oxidation of UO_2 to UO_{2+x} should lead to a contraction because of the decrease of the lattice parameter.

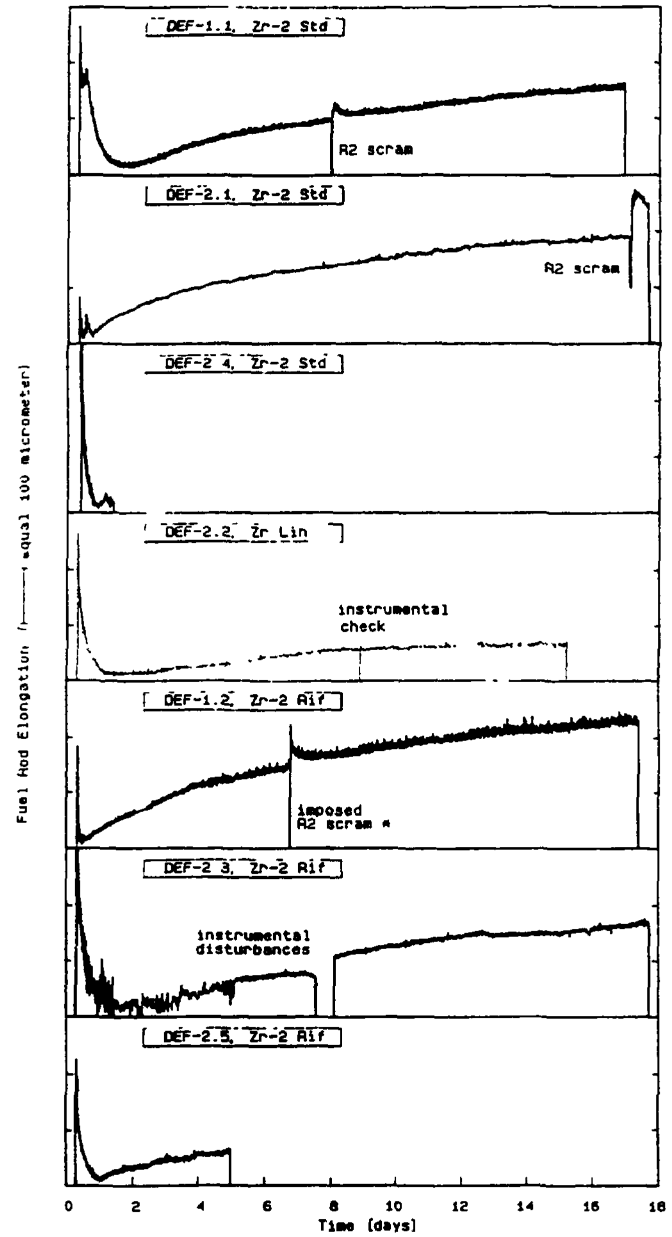

Figure 9
Compilation of On-Line Recordings for the Seven Fuel Rodlets Listed in Table 2.

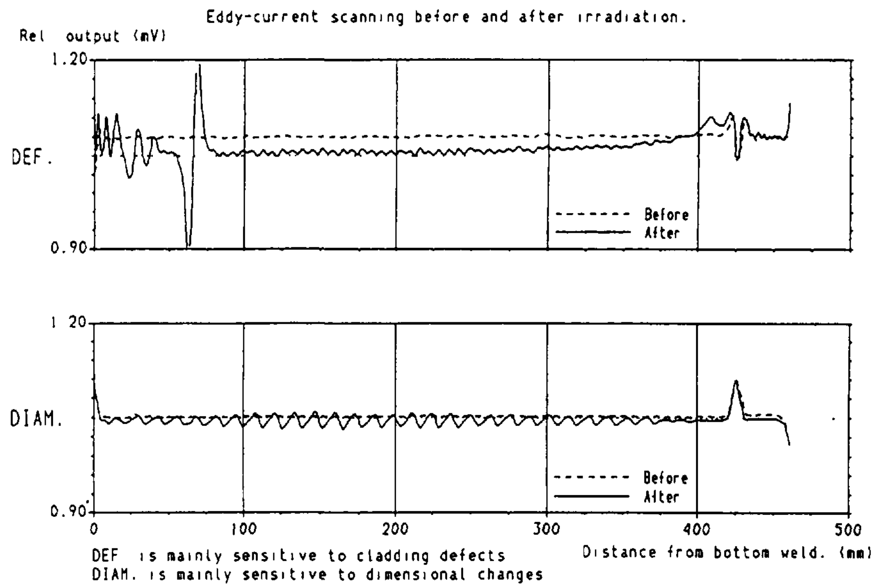

Figure 10
Eddy Current Measurement, before and after Irradiation of Test Rodlet 1117 with Standard Cladding Used in DEF-1.1.

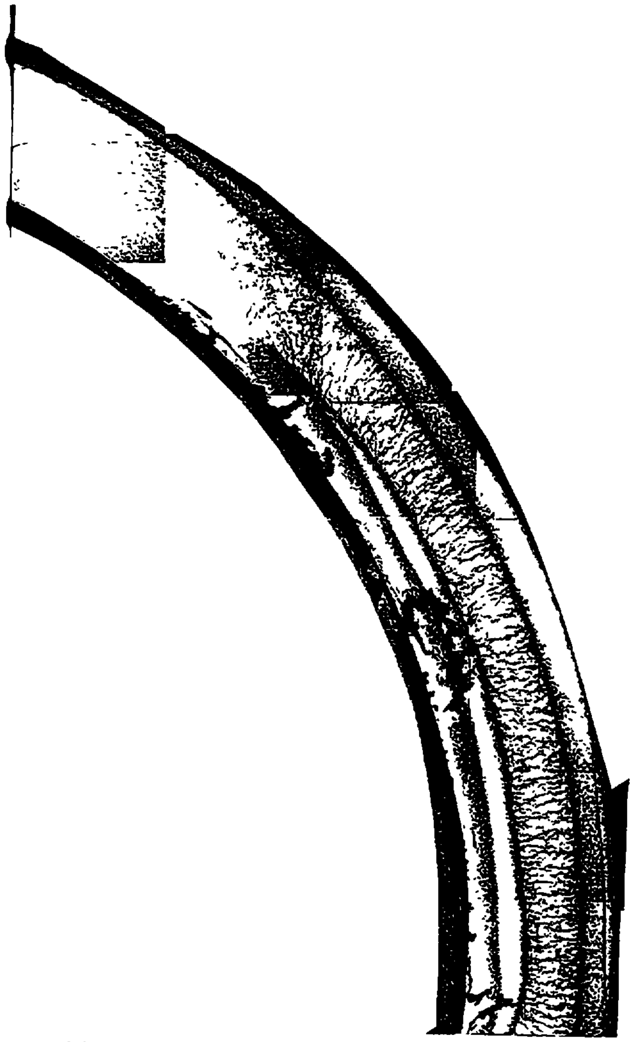

Figure 11
Micrograph of the Hydride Precipitate at the 6th Pellet Level from the Bottom of the Test Rodlet in DEF-1.1.

- The decrease of the thermal conductivity upon oxidation should lead to elongation of the rodlet.

- When helium is replaced by steam, a decrease of the thermal conductivity in the gap is expected, causing an elongation.

- When hydrogen is produced and more steam is prevented from entering, the thermal conductivity is increased, causing a contraction.

- Irradiation growth of the cladding leads to an elongation.

The elongation pattern in DEF-1 and DEF-2 differs from the behavior of a normal test rodlet without water. The rapid initial relaxation is ascribed to the production of hydrogen upon oxidation, apart from the stress relaxation.

It is suggested that the small peak appearing in some tests during or after relaxation can be explained in the following way. As the concentration of hydrogen increases, steam is reduced as it is diffusing downwards and the hydriding condition is fulfilled. When the hydrogen is consumed by the cladding upon hydriding, the rodlet starts to elongate until more hydrogen can barely be absorbed. After some time, the hydrogen content of the gap gas again rises slowly when it is replenished through the reduction of steam from above, which causes the rodlet to contract again Stress relaxation will tend to assist in the process.

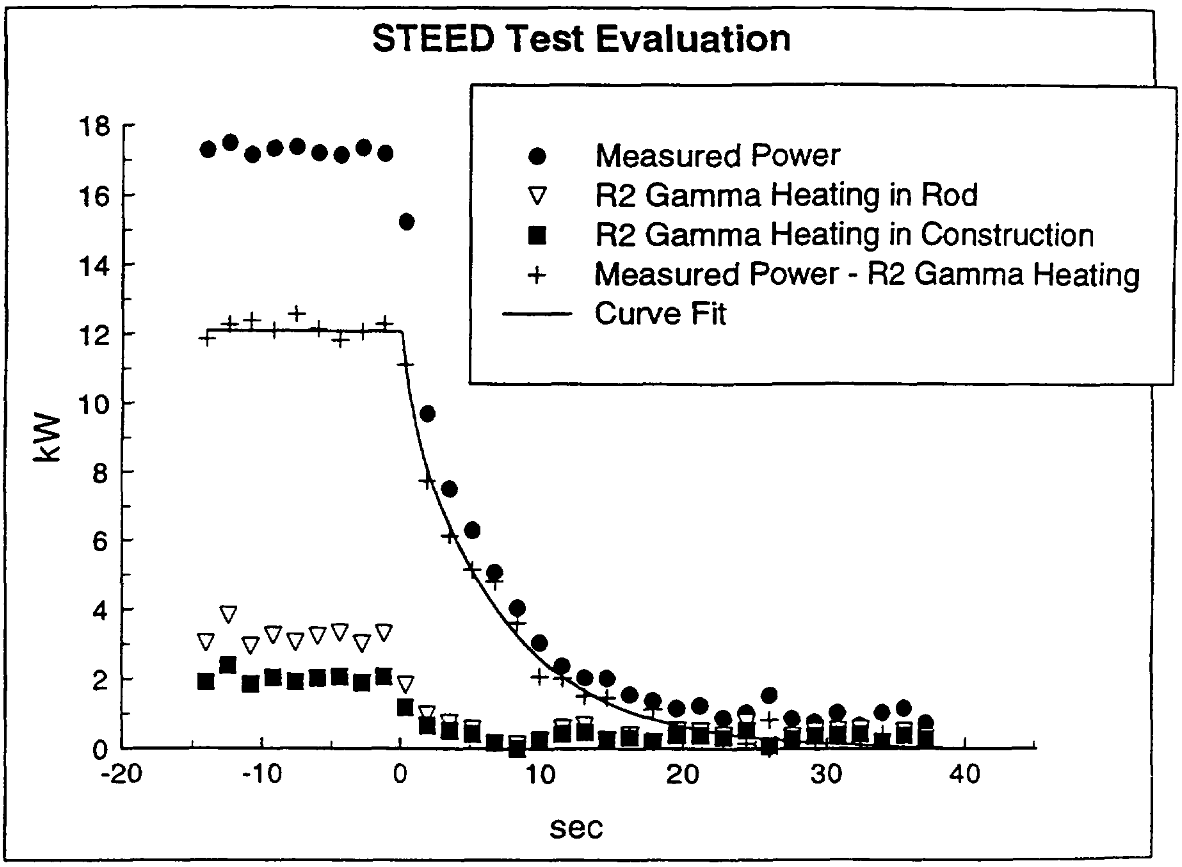

Figure 12
STEED Project – Example of Stored Energy Evaluation.
(The top curve and the two bottom curves are measured,
the second one from the top is evaluated from the others.
The stored energy is obtained from the area under the derived curve).

The long-time behavior of the elongation curve is probably due to the fact that the continuing but decreasing production of hydrogen is offset by the diffusional mixing with steam, which causes the thermal conductivity to decrease

The pronounced ridging found for the test rodlets used in DEF-1 1, DEF-2.3 and DEF-2.4 indicates that the Pellet-Clad Mechanical Interaction has been strong. In consequence, one of the pre-requisites for hydriding in the bottom part of the cladding is fulfilled.

The massive hydride in the test rodlet from DEF-1.1 can be seen as split up in an inner and an outer massive precipitate, separated by metallic Zircaloy (Figure 11). The periferal hydride, extending over a quarter of the circumference, is evidently growing at the expense of the inner hydride, which dissolves as the hydrogen diffuses radially outwards down the temperature gradient. The fact that the massive inner hydride has been covered by metallic Zircaloy along its inner surface, indicates that at some point in time it has ceased to absorb hydrogen gas and now redistributes itself radially outwards.

The following conclusions have been drawn from the seven tests in DEF-1 and DEF-2:

- Secondary failure can develop rapidly by massive hydriding, already within a few hours after the occurrence of a primary defect.

- Massive hydriding seems to cease after a time period of the order of 24 hours, probably because the hydriding condition is no longer fulfilled

- The test scheme applied is able to fulfill the objectives and can be used for further experiments.

5.4 The DEFEX Project

The conclusions that could be drawn from the Defect Fuel Experiments enabled STUDSVIK to propose and establish a larger project with international funding. The objectives of the DEFEX Project were:

* Investigation of phenomena and design parameters relevant to the degradation process in defected fuel rods, in particular:

    - Initiation of clad internal hydriding,

    - Progression of pellet oxidation, hydrogen release and clad hydrogen pick-up,

    - Clad failure by hydriding,

    - Initiation of clad fracturing.

* Providing a theoretical and experimental basis for further testing of potential remedies against secondary failure by hydriding

* Data gathering for modeling of defect failure behavior.

* Comparison to Locke type data.

The experimental scheme involves:

* Investigation of the degradation process in BWR 8x8 type fuel rodlets at near zero burnup

* Investigations under similar test conditions of.

- Standard Zircaloy-2 cladding,
- Standard zirconium liner cladding,
- Potential remedy type cladding

According to the original program, 6 BWR rodlets were to be tested in a total of 7 irradiation cycles. Non-destructive examination of the rodlets was included in the program as well as post-irradiation examination of the DEF-2 and DEFEX rodlets. At present, a continuation of the DEFEX Project is being planned, which is described in Section 6.

6 Studsvik's Upcoming International Fuel R&D Projects

Discussions are currently in progress on 6 different new international fuel R&D projects: two of them follow-ons to ongoing programs, two similar in principle to earlier projects and two completely new types of projects.

6.1 DEFEX II

The on-going DEFEX (I) project was presented in Section 5 The present plans for the DEFEX II project are tentative. The main program will include:

- Studies, analogous with the DEFEX Project, of the degradation process in irradiated fuel rods with medium (around 20 MWd/kg U) burnup with simulated primary defects.
- Testing of potential remedies against secondary failure by hydriding in order to supplement the experimental data base.
- Modeling of defect fuel behavior.
- Studies of cracking behavior of the hydrided cladding.

The primary defect would be simulated by a technique entailing delayed intrusion of water into the rodlet, thus simulating a case when a power rod is defected during operation.

A two-year BWR fuel program is envisaged, starting in 1997. The cost of the program is estimated at 30 MSEK, seeing that the test rodlets would have to be refabricated from full-length power reactor fuel. It is assumed that the scope of DEFEX II would be approximately the same as the DEFEX Project, i.e. 6 - 8 irradiation tests would be run, and destructive and non-destructive examination would include: neutron radiography, eddy current testing, dimensional measurement, visual inspection, gap squeeze measurement, gamma scanning, SEM investigation, internal gas analysis, metallography/ceramography and stoichiometry determination

6.2 ULTRA-RAMP

The question of the ramp behavior at high burnup (above 50 MWd/kgU) has been widely discussed in recent years as regards both normal and off-normal ramp rate conditions. However, only limited experimental information seems to be available so far [48].

The concern addressed appears to relate to the impact of changes in the physical properties of the fuel pellets at high burnup and their effects on the ramp behavior of the fuel rods. The fuel pellets seem to crack up in minor fragments and may no longer behave as solid bodies The fission gases will be entrapped in a magnitude of small bubbles and might cause fuel rod swelling on up-ramping. Other concerns relate to the loss of thermal conductivity and the impact of the rim zone on fuel ramp behavior.

In the SUPER-RAMP project a substantial fuel rod diameter expansion was observed (> 1 %) in a particular set of fuel rods at burnup levels of 45-50 MWd/kgU on up-ramping by 100 W/(cm·min) to close to 50 kW/m The transient tests performed in the TRANS-RAMP I and II projects indicated very short times to failure at ramp rates of approximatly 1000 W/(cm·min) up to above
40 kW/m.

The prospective ULTRA-RAMP project would constitute a combination of three groups of ramp projects· one group concentrated on the PCI phenomena under normal operating conditions in different types of fuel and the other two groups concentrated on more safety-oriented issues. Thus the ramp resistance in current fuel types would be studied both under normal operating conditions ("slow" ramps) and under off-normal operating conditions ("fast" ramps corresponding to ANSI Class II and III events and "ultra-fast" ramps corresponding to some ANSI Class IV events).

A few recent simulated RIA experiments (Reactivity Initiated Accidents) with high-burnup fuel (55 and 65 MWd/kgU) have focussed interest on Class IV events STUDSVIK is proposing a new type of "ultra-fast" ramps, faster than the fast ramps performed in earlier safety-related projects (such as TRI) but slower than the simulated RIA experiments. These new "ultra-fast" ramps could reach e.g. 120 kW/m during an 1 sec effective ramp time, corresponding to an enthalpy increase of 30 cal/g.

According to our present plans a number of high burnup fuel rodlets would be exposed to fast and slow ramp rates to preselected terminal power levels and to "ultra-fast" ramp rates to preselected enthalpy increases, all in the R2 test reactor [48]. The main objective is to identify any adverse or inadequate fuel rod behavior as for example abnormal fuel rod swelling, fast failure of the cladding, loss of fuel integrity on clad fracturing causing dispersion of fuel particles in the coolant water. Detailed non-destructive and destructive examinations (including advanced types of ceramography) would follow. These plans are quite flexible and depend on the feed-back we receive from discussions with prospective participants.

6.3 SUPER-RAMP III/10x10

In the new types of 10x10 BWR fuel the Linear Heat Rate is lower than in earlier types of fuel and the PCI resistance is presumed to be correspondingly improved. However, some utilities using zirconium liner with earlier types of fuel have raised the question whether the lower linear heat rates in 10x10 fuel really make the added resistance against PCI failure, achieved with zirconium liner fuel, unneccessary As far as is known no ramp tests have ever been performed on 10x10 fuel.

Thus the proposed SUPER-RAMP III/10x10 project would be similar to the earlier SUPER-RAMP II/9x9 project. It is planned to start and be completed in 1997.

6.4 STEED - Enthalpy Determinations

The stored energy in fuel rods during operation, the enthalpy, depends on the fuel design (dimensions, materials), operating conditions and burnup. Since the heat release from the fuel after a scram is dominated by stored energy during the first minute this quantity is an important parameter in connection with safety considerations such as LOCA evaluations. A new type of tests are now in progress, the STEED project (Stored Energy, Enthalpy Determination).

Experimental determinations of the stored energy provide a valuable alternative to fuel code calculations. Depending on the test technique used the accuracy can be better than the corresponding code calculations, especially with increasing burnup. Apart from producing valuable data for safety related analysis, experimental determinations of the stored energy provide a possibility for interesting comparisons with fuel code calculations The experimental results can also serve as excellent bench-marking opportunities for fuel modellers for unirradiated fuel.

STUDSVIK's R2 test reactor is well suited for scram experiments where the thermal response of different types of fuel can be compared. The measurements for the STEED project will be performed by analyzing the heat release from a test rod after scram, thereby using the R2 test reactor's calorimetric rod power measurement system. This system is the very same as that used in ordinary ramp experiments.

A demonstration experiment, STEED-I, is in progress on unirradiated fuel rodlets The objective is to verify the test technique and to evaluate the accuracy An international project, STEED-II, based on tests of irradiated fuel rodlets, will be discussed in 1997 An example of a "pre-project" demonstration test is shown in Figure 12

6.5 Post-DO/DNB Ramp Resistance

Early experimental data in power reactors and test reactors have shown that short periods (less than about a minute) of Post-DO/DNB operation have not caused fuel failures (defined as leaking fuel rods) The question whether such operation might cause incipient (non-penetrating) fuel cracks and a decrease in ramp resistance has never been investigated A parametric fuel modeling study performed by STUDSVIK NUCLEAR indicates that the ramp resistance might deteriorate due to such an event

STUDSVIK has planned an experiment where fuel rodlets would be exposed to short (20-30 sec) periods of Post-DO/DNB operation along a short part of their length (at the location of the highest linear heat generation rate) and then later ramp tested in the same in-pile loop

6.6 TRANS-RAMP III

Experience from the earlier safety-oriented STUDSVIK projects (discussed in Section 1), as well as from power reactor operation, shows that non-penetrating cladding cracks form readily during certain short-time power transients and cracks initiate already within 5-10 seconds In the TRANS-RAMP IV (TRIV) project, 1989-1993 [49], the influence of non-penetrating (incipient) cladding cracks on the PCI failure resistance during an anticipated subsequent transient occurring later in life was studied Seven short fuel rodlets, refabricated from full-size PWR power reactor fuel with a burnup of 20-25 MWd/kgU, were tested in the program Three of these rodlets were further irradiated in the R2 reactor after having been subjected to a first power ramp The effects of these ramps were somewhat different One of the rodlets was the only one that clearly exhibited detectable non-penetrating (incipient) cracks during the first ramp After continued base irradiation, giving an additional 4 MWd/kgU, the rodlets were subjected to a second power ramp in order to determine the residual time to PCI failure The rodlet with the incipient cracks from the first ramp showed considerably shorter time to failure than the other two rodlets

In the TRANS-RAMP III (TR III) project the influence of non-penetrating (incipient) cladding cracks in BWR fuel rodlets on the PCI failure resistance during an anticipated subsequent transient occurring later in life will be studied

ACKNOWLEDGEMENTS

The authors are much indebted to all their colleagues at STUDSVIK's R2 test reactor and Hot Cell Laboratory and especially to Mr Hilding Mogard, who initiated the work on fuel R&D projects at Studsvik The authors are also much indebted to all the sponsors of the bilateral and international projects mentioned

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SUMMARY OF THE OECD HALDEN REACTOR PROJECT PROGRAMME ON HIGH BURN-UP FUEL PERFORMANCE RELEVANT FOR BWRS

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Abstract

The basis for the Halden Reactor Project Programme is presented together with an overview of the content of the programme for the time period 1997-1999. The concept of using both separate effects studies, to determine particular fuel properties, and integral rod behaviour studies of commercial fuel is explained. Each of the items in the programme relevant for BWRs are introduced, with most being discussed in further detail.

1. Introduction

The Halden Reactor Project Programme is based on advice and guidance given by the Project participants and the overall aim of the proposed programme is to provide information and experience which will assist in enhancing safety and reliability of today's reactors

The 1997-1999 programme includes both fuel and cladding studies and makes extensive use of the inventory of in-core instrumentation and irradiation rigs available for use in the Halden Boiling Water Reactor (HBWR) These, together with installed loop systems make it possible to perform in-reactor testing under a wide variety of thermohydraulic, neutronic and water chemistry conditions.

The Project uses a two-fold approach to obtaining fuel properties data. On the one hand separate effects studies are conducted using specially designed fuel rods in tests devised for investigating specific performance parameters and on the other hand integral rod behaviour studies are carried out using commercial fuel rods supplied by the Project participants. Data from these latter tests are used to verify the performance sub-models derived from separate effects data

The separate effects tests focus mainly on determinations of thermal properties, fission gas release (FGR) and fuel dimensional stability in the high burn-up range, which together with pellect-clad interaction (PCI) are also the subjects being investigated in the integral rod behaviour studies. Included under the umbrella of the integral rod behaviour studies are also tests related to the issue of rod internal over-pressure and the clad lift-off criterion and tests simulating dry-out. It has further been proposed that the Project produce data in support of RIA evaluations and there is also a proposal to identify a test method to elucidate the processes occurring in a failed fuel rod wherein a secondary failure site forms

The investigation of cladding properties is also an important feature of the Halden Reactor Project Programme, with not only normal reactor operation conditions, especially to high burn-up, being of interest but also the behaviour of clad during post transient conditions. There is an on-going project to investigate clad creep behaviour in-pile and in the area of clad corrosion, investigations on separate effects are being proposed, for instance related to cold spots, hydrogen migration and concentration in cladding, heat rating dependence and local radiation conditions. The effects of crud deposits on corrosion will also be investigated at as well as water chemistry studies and throughout all the tests there is an emphasis to use modern materials.

2. Separate Effects Studies

Fuel performance is dependent on a variety of phenomena which need to be correctly quantified in order to obtain adequate fuel behaviour predictions. This requires that tests be suitably designed in order to extract the desired information in a direct manner. As an example, fuel thermal conductivity determinations are better performed with fuel rods having very small gaps and with fast burn-up accumulation.

Basically, the programme is designed to determine the thermal properties, FGR and swelling of standard UO_2 fuel, gadolinia fuel and MOX fuel in the burn-up range, 50 to 120 MWd/kgUO_2 Such high burn-ups are being considered with the behaviour of the rim structure in mind.

A graph illustrating the type of results that are derived from the separate effects programme is shown in Figure 1 The Halden correlation for thermal conductivity as a function of temperature is based on experimental data that have been collected from various tests already carried out in the HBWR, and this compares well with the measurements made on Halden fuel at JAERI.

3. Integral Rod Behaviour, Tests on Commercial Fuels

Whilst separate effects tests provide efficient methods for assessing specific phenomena, by implication the fuel design differs from that of standard fuel and it is important that verification be made by integral rod behaviour studies conducted with commercial fuel. The Project has developed and applied a variety of refabrication and re-instrumentation techniques in order to monitor the behaviour in the reactor of short lengths of commercial fuel supplied by the Project participants. A schematic of a re-instrumented fuel rod is shown in Figure 2, which illustrates how a re-instrumented rod can be fitted with a centre-line fuel thermocouple to monitor fuel temperature and a bellows pressure transducer to monitor fission gas release Cladding elongation can also be measured for PCI studies.

The key parameters to be studied in these tests are fission gas release and swelling/PCI behaviour as related to fuel temperature and the proposed tests will involve subjecting rods to stepwise power bumps by means of local power control executed in rigs fitted with a He-3 coil. A schematic drawing of

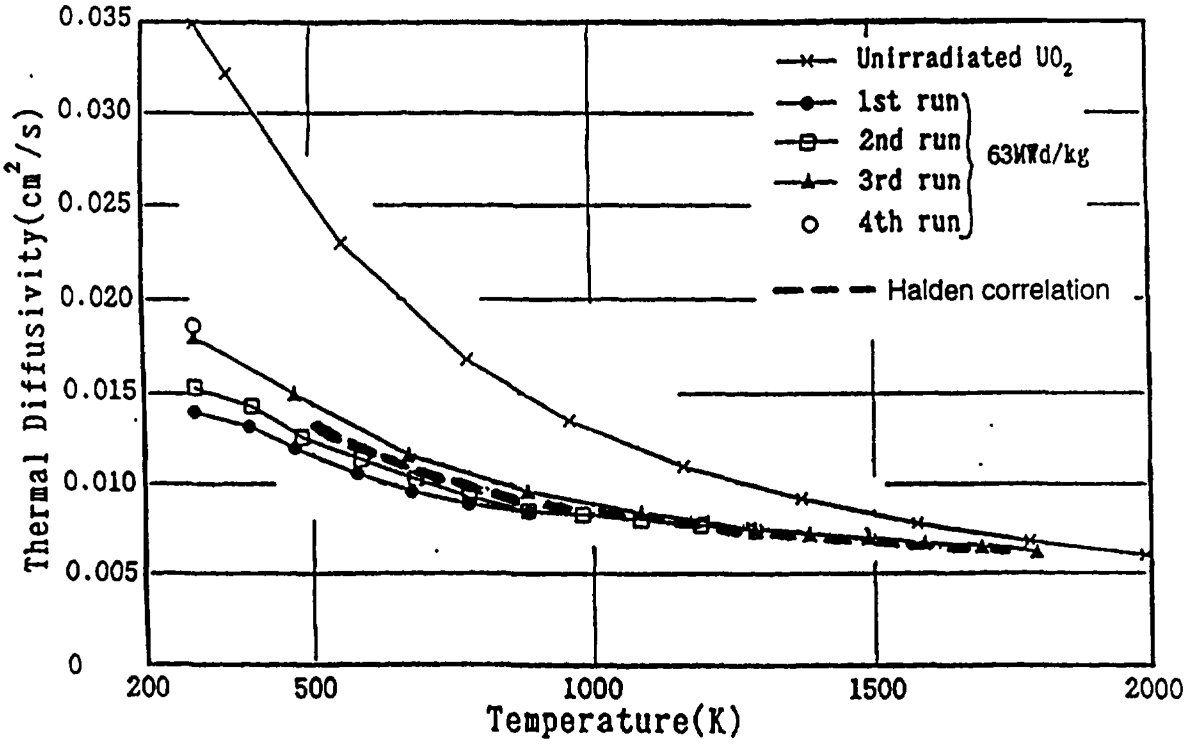

Figure 1. Thermal diffusivity measurements at JAERI on HBWR fuel at 63 MWd/kg. These data agree well with the conductivity degradation curve. Measured changes in different runs reflect annealing phenomena due to fuel heat-up.

the type of rig to be utilised is shown in Figure 3. In particular the comparative behaviour of UO_2 and MOX fuel will be determined and the burn-up range of the fuel to be investigated is 52 - 60 MWd/kg U, which is relevant to levels currently reached in commercial power plants. In one series of tests it is planned to address the effect of fuel microstructure.

In addition to these tests, integral rod behaviour studies with commercial fuel are to be performed in relation to other programme items, such as rod over-pressure and dry-out investigations and these are described below.

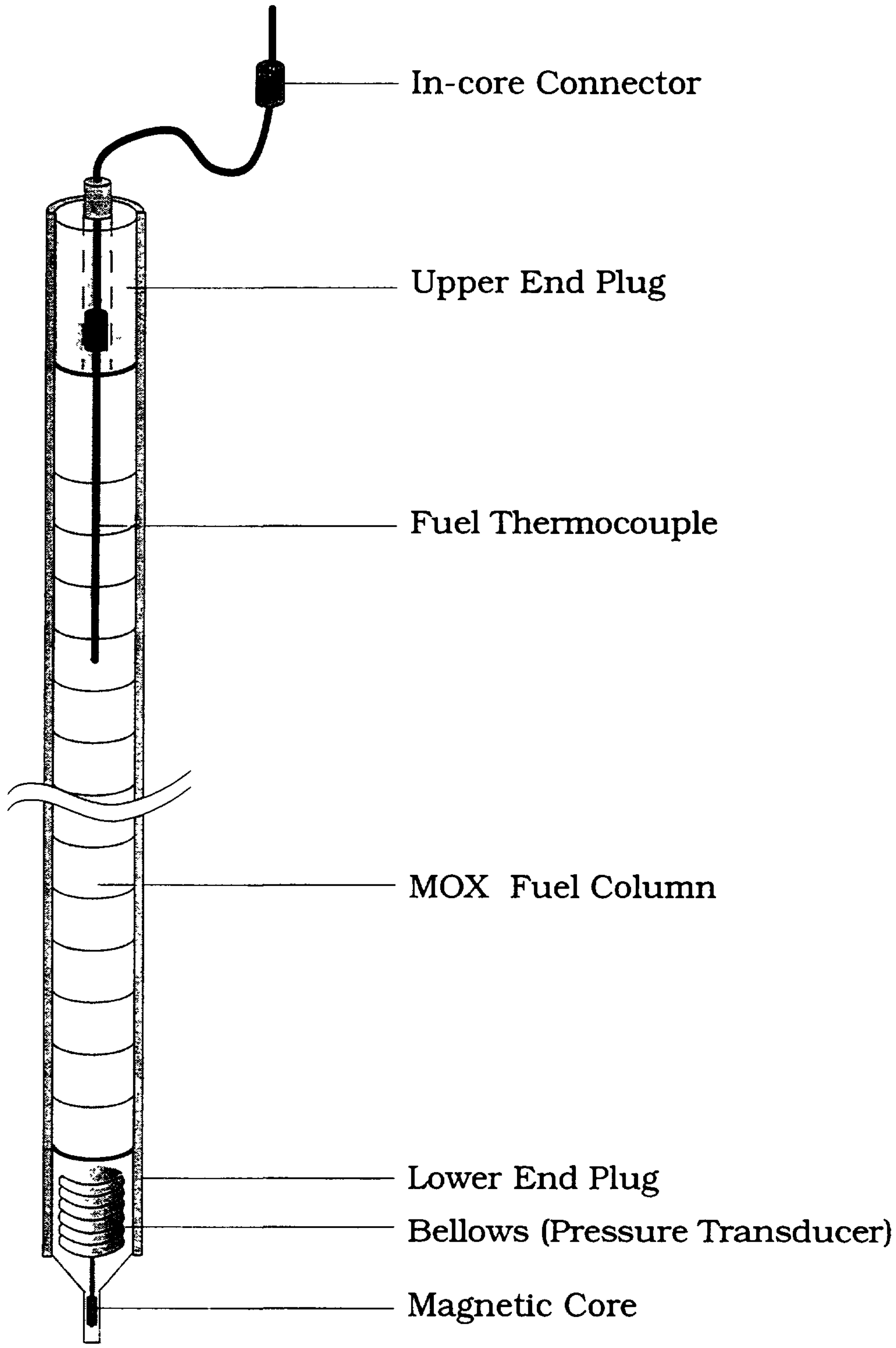

Figure 2. Schematic of a re-instrumented fuel rod for use in integral rod behaviour studies.

TEST RIG FOR POWER BUMPS OF HIGH BURN-UP
COMMERCIAL FUEL, REINSTRUMENTED SEGMENTS

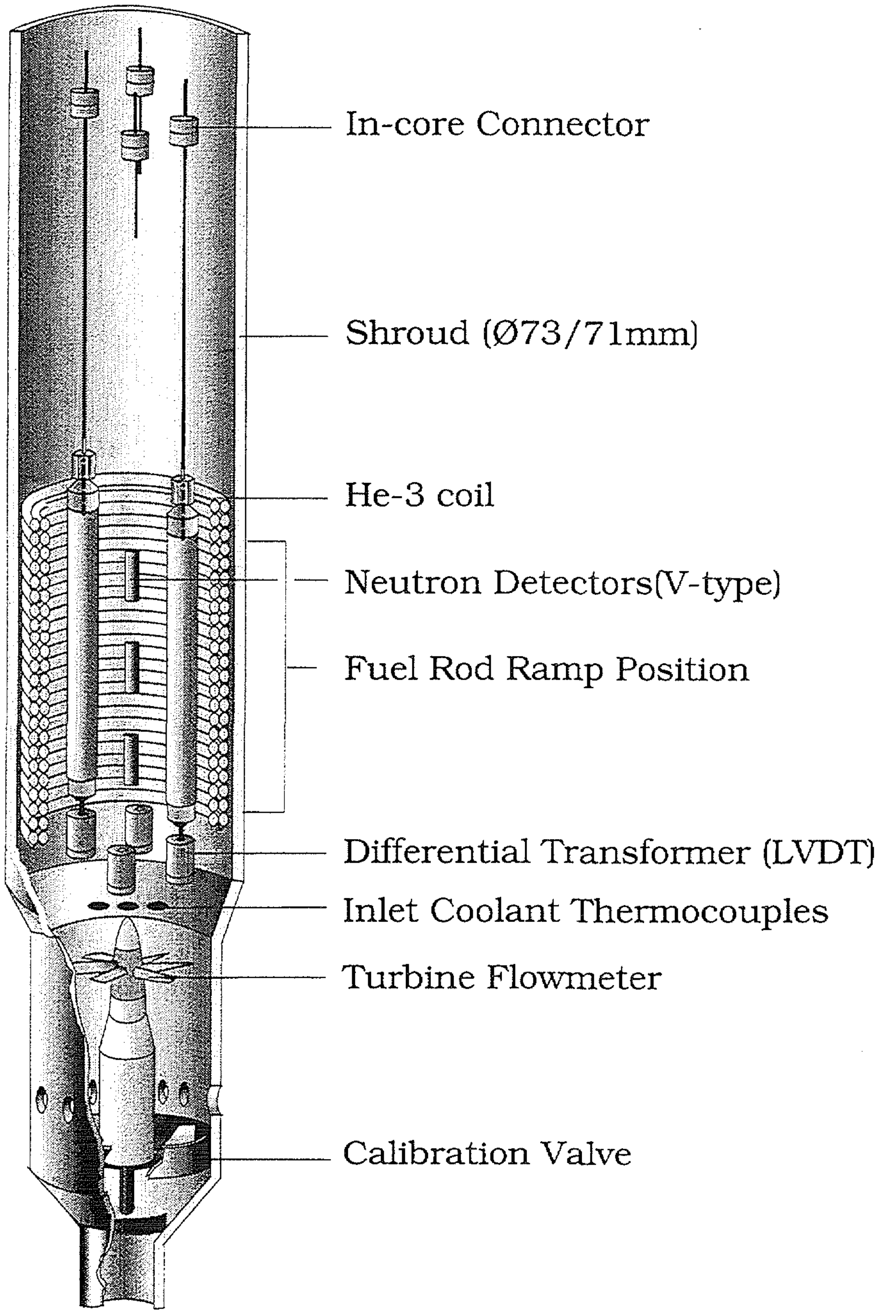

Figure 3. Schematic of the type of rig to be used in integral rod behaviour studies.

4. Rod Over-Pressure/Cladding Lift-Off

The objective of this test is to verify that rod over-pressure does not cause cladding lift-off. This is a safety related issue of great importance for high burn-up fuel: if the fuel-clad gap re-opens, fuel temperature and thus the overall performance of the fuel could be affected. The competing phenomena involved are fission gas release, which determines the extent of rod over-pressure and hence any clad creep-out and fuel swelling which tends to close the gap.

Commercial MOX fuel will be utilised in the tests, since in LWR cores MOX fuel operates at higher power than UO_2 fuel towards the end of service, The fuel rods will be re-instrumented with a fuel thermocouple and a pressure transducer

5. Dry-Out Effects

Light water reactor cores may be subjected to thermal-hydraulic transients resulting in inadequate core cooling for short periods of time The objective of the dry-out experiments is to provide information on the consequences of such an event for the fuel, by inducing short term dry-outs characteristic of those anticipated to occur from pump trips in BWRs Coolant transients aim to bring the cladding temperature into the range 550 - 750°C for typically 20 seconds The post dry-out fuel performance is to be assessed by monitoring fuel behaviour in service after the test and by carrying out extensive post irradiation examination (PIE) of the cladding to determine if there have been any material property changes

Two dry-out tests have been completed already, the temperature-time histories are shown in Figures 4 and 5, and it is planned to continue with a third series of tests

6. Activities in Support of RIA

It is not intended that the HBWR be used for RIA tests as such, considering that the energy deposition rates in RIA are orders of magnitude greater than those that could be achieved in the Halden reactor Instead studies in support of understanding the mechanisms involved in RIA are being contemplated The first priority being to understand the role of the pellet rim, where the highest inventory of fission gas is present and where the highest temperature is experienced during the transient The

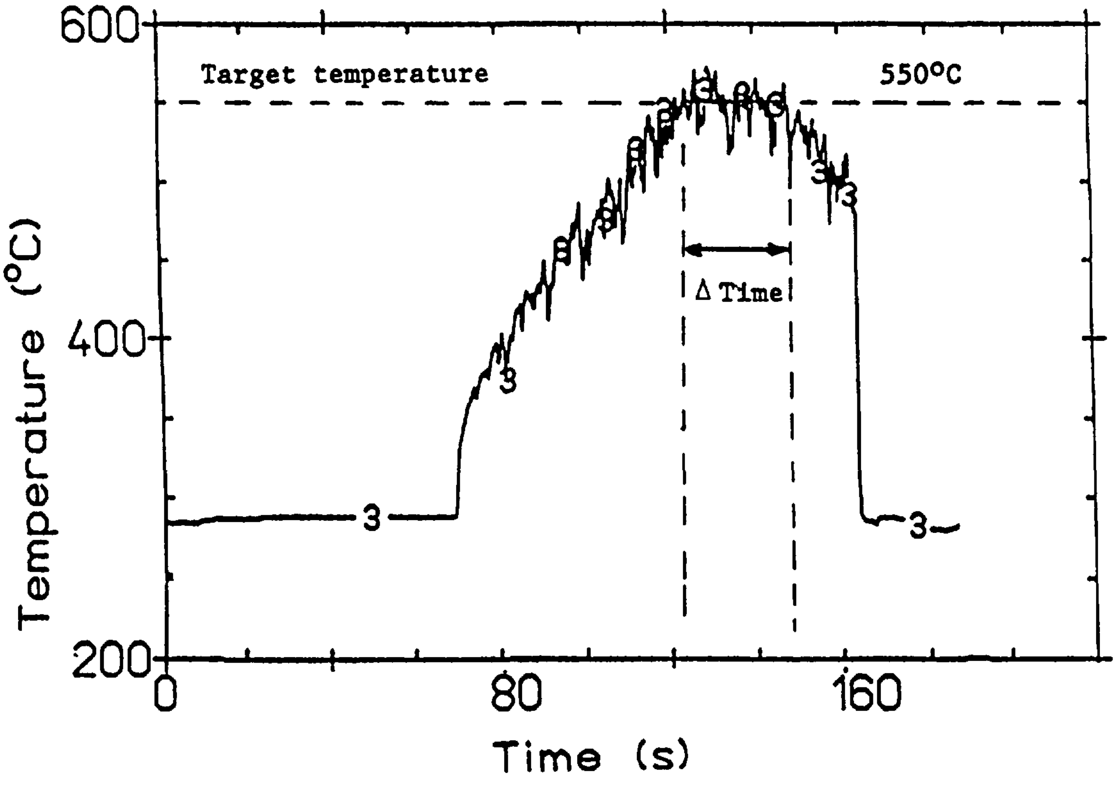

Figure 4. Temperature-time history achieved in the first dry-out test.

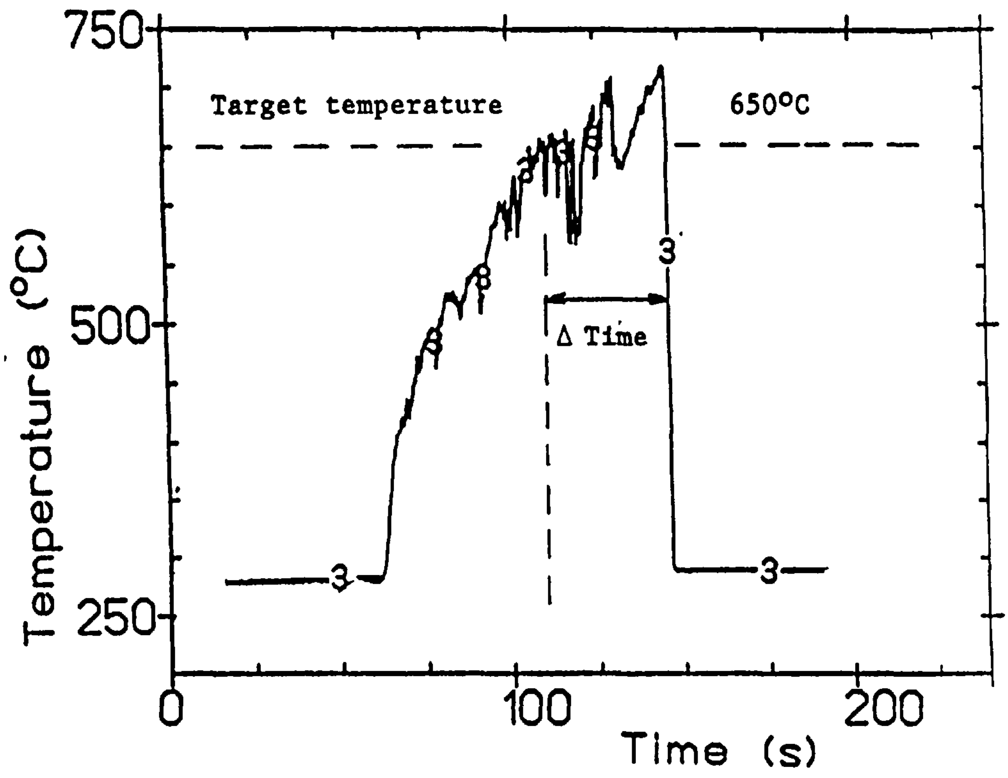

Figure 5. Temperature-time history achieved in the second dry-out test.

Project intends to evaluate the possibility of imposing rapid temperature transients on small UO_2 fuel specimens of high burn-up

7. Fuel Failure Degradation

The Project intends focusing on the mechanistic interpretation of the processes involved in failed fuel degradation and is thus proposing to carry out a test to determine the rate of fuel and cladding oxidation due to steam ingress and the rate of subsequent hydrogen uptake in the cladding This can be realised by inserting a know amount of water in an as-fabricated rod and by measuring the rod pressure during operation with a bellows pressure transducer A schematic illustration of the design of such a rod is shown in Figure 6 It is expected that the rod pressure will rise to saturation pressure and then achieve a level dependent on the balance between the rate of hydrogen production and hydrogen pick-up in the clad. A graph of the expected results are shown in Figure 6 where it can be seen that if the rate of oxidation, which liberates the H_2, is initially faster than the H_2 absorption rate, the pressure would be expected to increase above saturation. Subsequently as H_2 pick-up increases, the rod pressure would decrease.

At a later stage, the Project may look to investigating remedies related to failed fuel degradation, for which it is proposed that the gap chemistry at the secondary site be simulated in a segment of rod by way of a gas line. In this way different clad materials and designs could be tested

8. Cladding Creep Behaviour

The Project has an on-going Zircaloy creep programme which is aimed at characterising the in-pile creep behaviour of irradiated cladding. Creep-down (compressive creep), recovery effects, stress

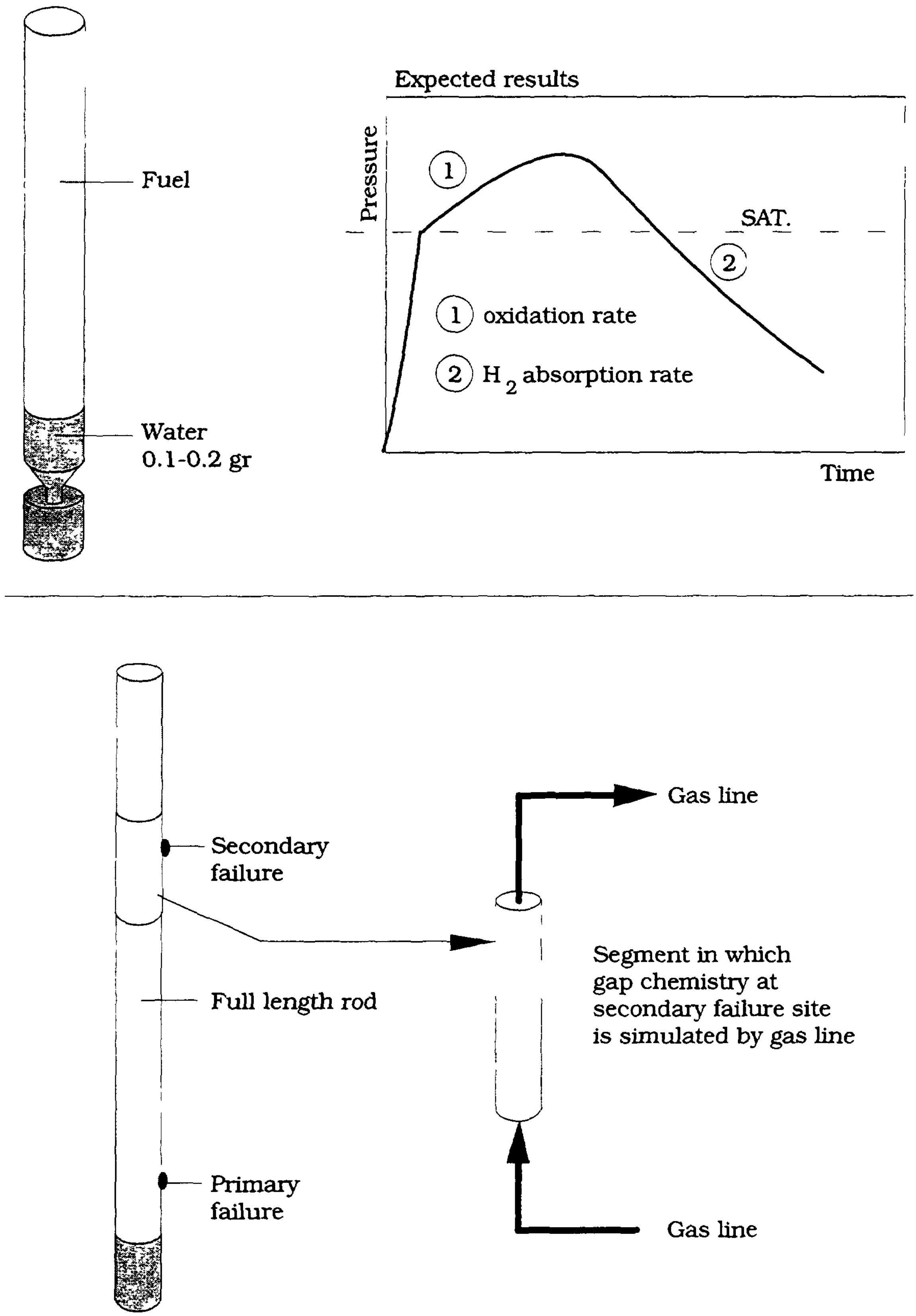

Figure 6. Schematic illustrations for the proposed tests related to failed fuel degradation.

reversals and tensile creep are all considered in the experimental programme which utilises both sealed rod segments, which experience effectively constant stress throughout the experiment and segments with gas lines attached, whereby the internal rod pressure can be varied in relation to the outer, system pressure. A study of pre-irradiated Zr-2 cladding has been carried out under BWR conditions, the results of which are shown in Figure 7, where it can be seen that several stress increments and reversals were imposed on the test rod. It is planned to use pre-irradiated modern cladding materials in future studies, which will include both constant stress and variable stress testing, with the variable stress testing to follow a similar pattern to that used in the previous Zr-2 test.

9. Corrosion Studies for BWR Fuel

Several experiments are under-way or in the planning stage for the investigation of issues related to cladding corrosion and hydriding, cladding corrosion and crud deposition and the effects of water chemistry on the corrosion process. In particular the separate effects of rod design, material and operational parameters are being explored.

Figure 8 shows the general rig design that is employed for these experiments, illustrating that both fuel rods and materials samples can be tested with the placement of booster rods around the test pieces to provide the fast flux conditions representative of LWRs.

It is also the intension within this programme to investigate the effect of variations in local conditions, and in a recent experiment (Figure 9) it was shown that by increasing the local radiation field by inserting discrete β-sources in the coolant region surrounding a test rod, an appreciable enhancement of the corrosion was observed.

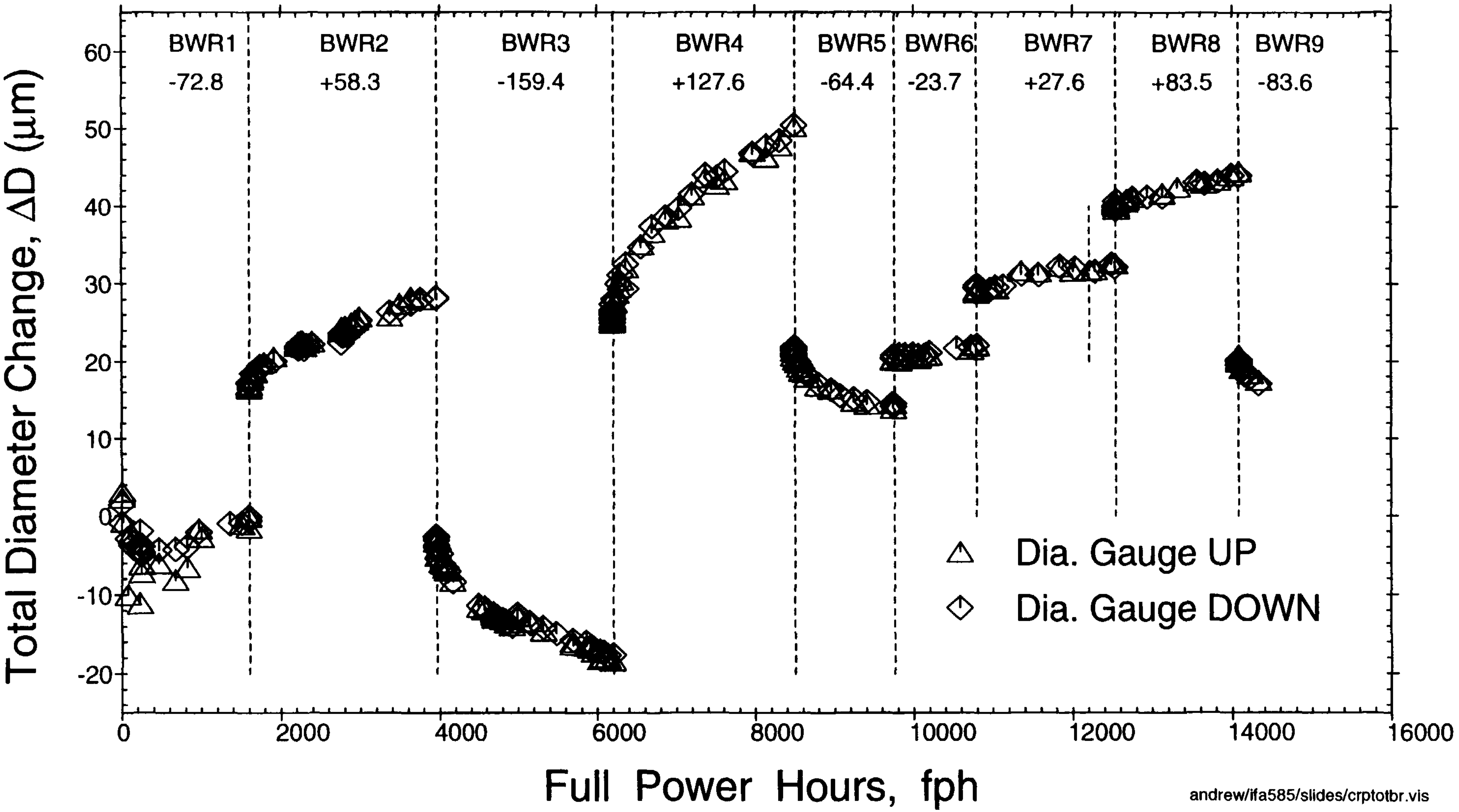

Figure 7. Results of the creep study on pre-irradiated Zr-2 subjected to varying stress conditions.

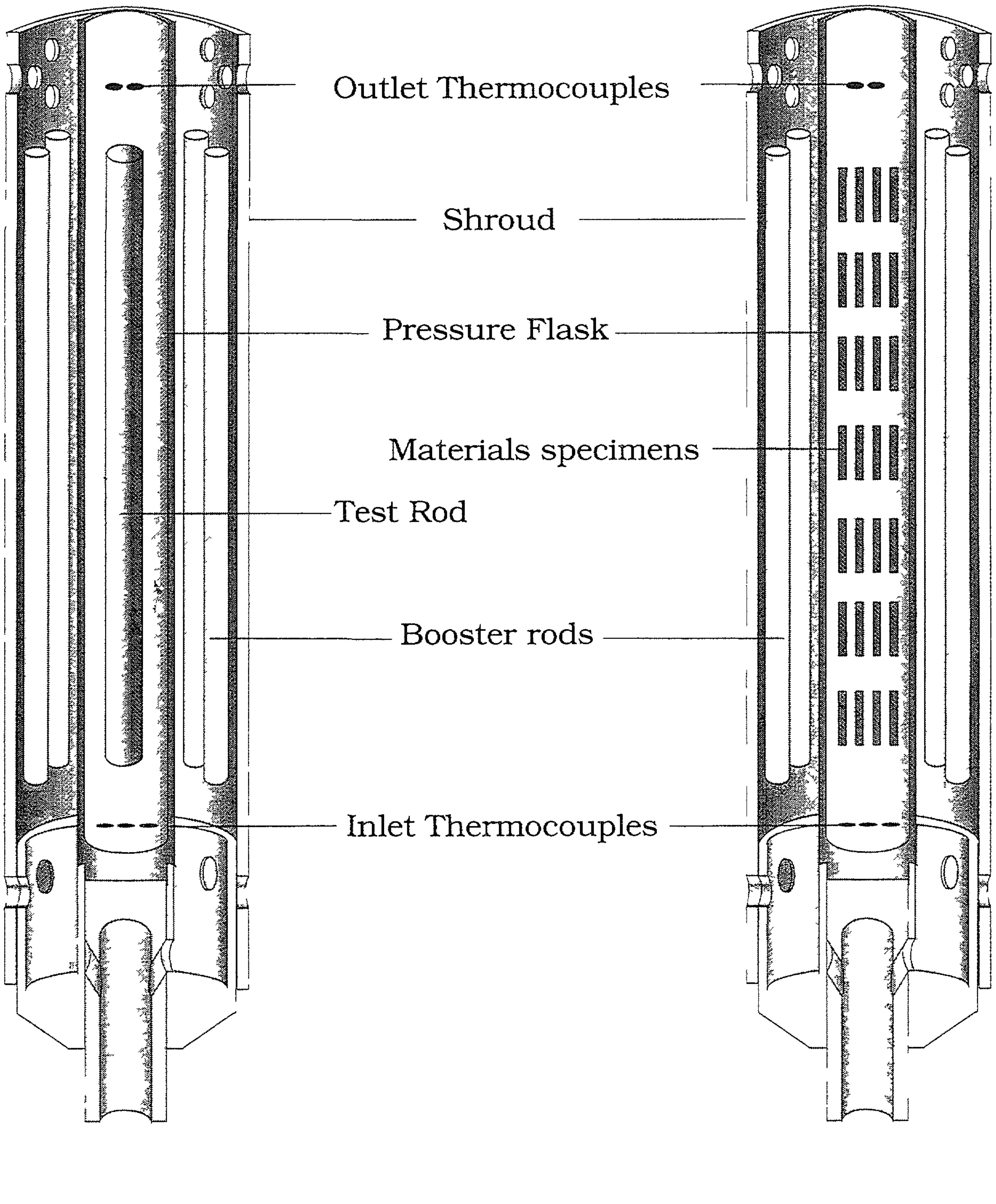

Figure 8 Schematic of test rig used for corrosion studies of both fuel and materials specimens

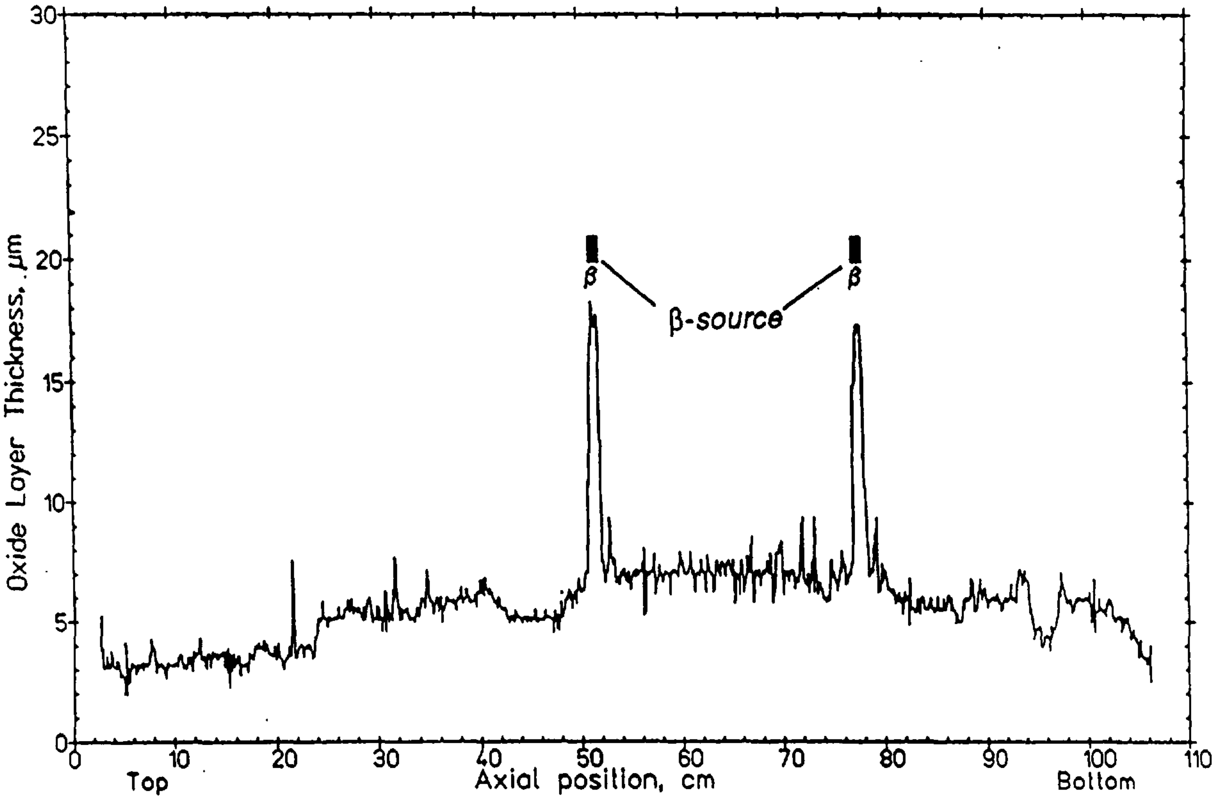

Figure 9. Interim oxide thickness measurements on a BWR rod; It can be seen that the presence of a local β-source in the coolant region surrounding the cladding produced appreciable corrosion enhancement.

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RBMK FUEL ASSEMBLIES: CURRENT STATUS AND PERSPECTIVES

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Abstract

The safety enhancement measures implemented since 1986 have led to substantial burnup reduction in the RBMK fuel assemblies and consequently to economical losses. With the purpose to compensate the losses, computer analysis and experiments were performed during the last decade. The works were aimed at the RBMK fuel charge perfection to reduce void reactivity effect and to increase fuel burnup.

The paper presents principle results of the studies which are currently under implementation or are supposed to be implemented in the nearest future.

Approach to fuel assembly design modification

As one result of the safety enhancement measures implemented after 1986, in particular, by installing a large number of additional absorbers to the core, the fuel burnup of the RBMK fuel assemblies (FAs) was reduced by 30-35 % compared to the design value. To compensate the losses, computer analysis and experimental works aimed at modifying the RBMK fuel charge were performed during the last decade with the purpose to reduce void reactivity effect and increase the fuel burnup. The activities on fuel assemblies modification were performed in two lines:

- with change in geometrical dimensions of fuel assemblies involved;
- without changing basic overall dimensions.

One RBMK FA for both the 1000 MW reactors and RBMK-1500 contains 18 fuel element of 13.6 mm diameter, all are located over two circumferences: the first radius for 6 fuel elements and the second one - for 12 fuel elements (Fig. 1).

The first line, aimed at alternating the FA design, envisaged development of two options, i.e. with fuel elements of ether diminished or enlarged diameters. The fuel elements designs 10.2 mm and 14.8 in diameter are worth noting. In the first case, the FA incorporated fuel elements of three radii, in the second case, it had two, the same as in the standard case. The reactor process channel was also envisaged to get larger in diameter where in a fuel assembly of 80 to 86 mm diameter was installed.

The analysis showed that the change for 10.2 mm fuel elements practically did not change the void reactivity effect. Thus, it makes up is ~0.9 β for the 2 % enriched fuel in the core with 80 additional absorbers, whereas for the 10.2 mm fuel elements it is ~1.0 β.

The fuel elements with diameters enlarged from 13.6 up to 14.8 mm allows additional absorbers to be discharged from the core and reduce the cost by ~15 %. In this case the fuel assembly uranium charge gets higher by ~20 %. However, bearing in mind, that actual change for channels with large

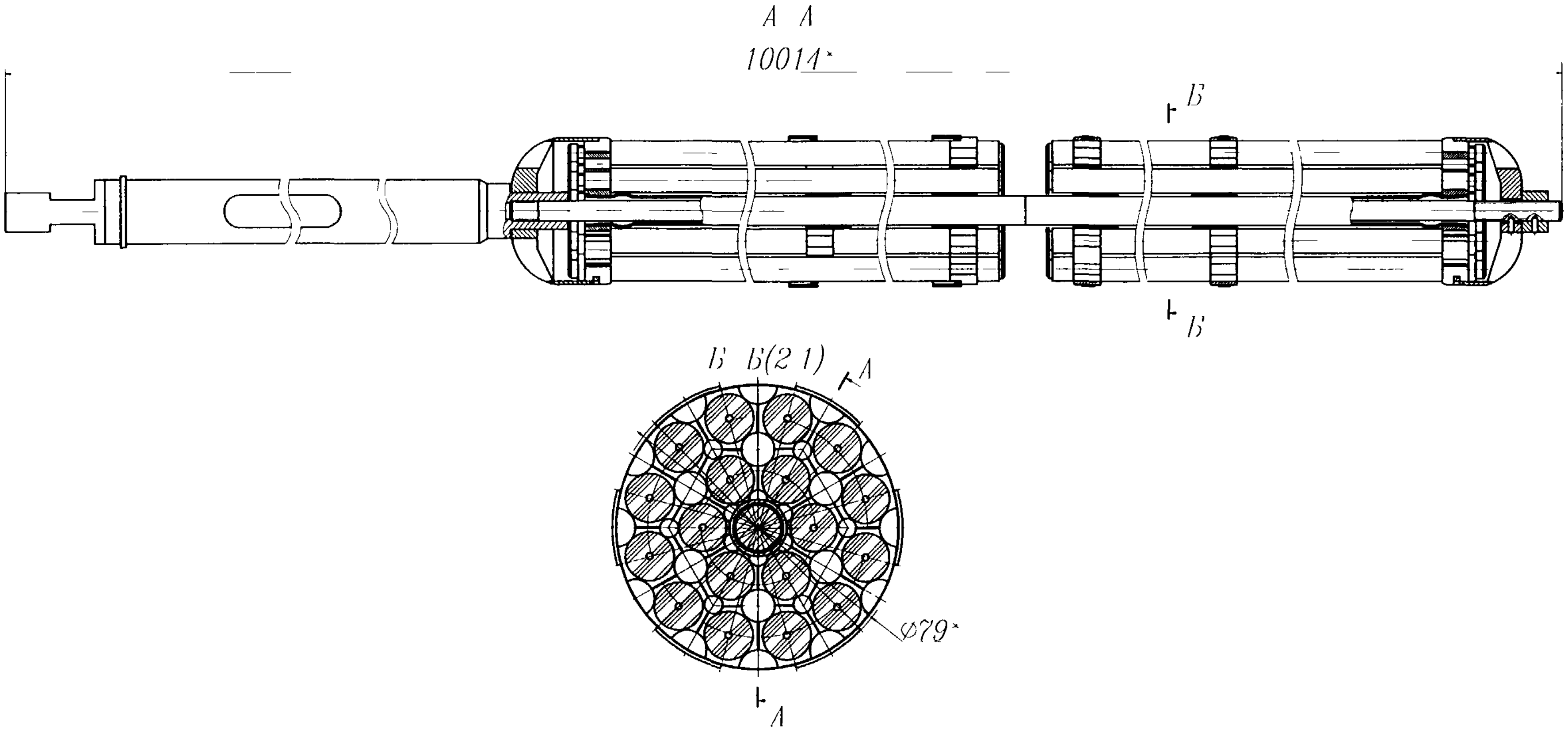

FIG. 1. Draft of the RBMK-1500 fuel assembly.

diameter may be realised only when the core is updated as well as the RBMK units service life design terms, this line did not get progression.

Therefore, the investigations were reoriented during last several years such that application of burnable absorbers should allow the geometry of the fuel assembly and fuel channel to remain unchanged

Development and implementation of burnable absorber fuel

A potentiality for application of different burnable absorbers B, Dy, Gd, Hf, Er was subjected to analysis. Also different options of their location in fuel assemblies (in structure components, partially in fuel elements, etc.) were considered. The analysis performed showed that erbium may get to be better absorber for the RBMK reactors.

The options considered for Er location in FAs allowed to conclude that the highest effect may be attained when Er is added directly to the fuel of all FA fuel elements. The calculations demonstrated that 0.4 % of Er added to the 2.4 % enrichment fuel enables to practically have the same voidage effect in the core with no additional absorbers (AA) as in the reactor with 80 AAs (Table 1).

At the same time, application of additional absorbers permits to substantially reduce (~10 %) the FA maximum power which realises at the initial stage of the FA operation (Fig. 2). This margin can be used for increasing the fuel enrichment. The calculations illustrated that Er application in RBMK allows to pass to 2.4 % enrichment without exceeding the FA maximum power limits. The higher enrichment and discharge of additional absorbers from the core lead to a fuel burnup higher by 30 %. The fuel burnup for the RBMK-1000 reactors get higher by 18 % as the enrichment is changed for 2 6%.

The presence of erbium in fuel results in a substantial (by 10-15 %) reduction in the fresh fuel reactivity as compared to standard which allows FA reload that are performed in RBMKs on power by more "mild" control. This does not require significant movement of control rods to suppress local peaks in power density.

TABLE 1. BASIC CALCULATED CHARACTERISTICS OF RBMK REACTOR CORES WITH NEW FUEL

Characteristics	**RBMK-1000**			**RBMK-1500**	
Enrichment, %	**2.4**	**2.6**	**3.0**	**2.0**	**2.4**
Erbium content by mass, %	**without Er**	**0.41**	**0.76**	**without Er**	**0.41**
Number of AA in core	**80**	**0**	**0**	**54**	**0**
Discharged fuel burnup, MW day/kgU	**20.9**	**26.6**	**28.8**	**15.1**	**21.0**
Average-over-service life voidage, β	**+0.6**	**+0.5**	**-0.39**	**+0.7**	**+0.3**
Annual economic effect due to reduction in fuel component per 1 power unit, mln. USD	**-**	**4.0**	**-**	**-**	**9.0**
Possible date of FA serial production commencement	**-**	**1997**	**1999**	**-**	**1997**

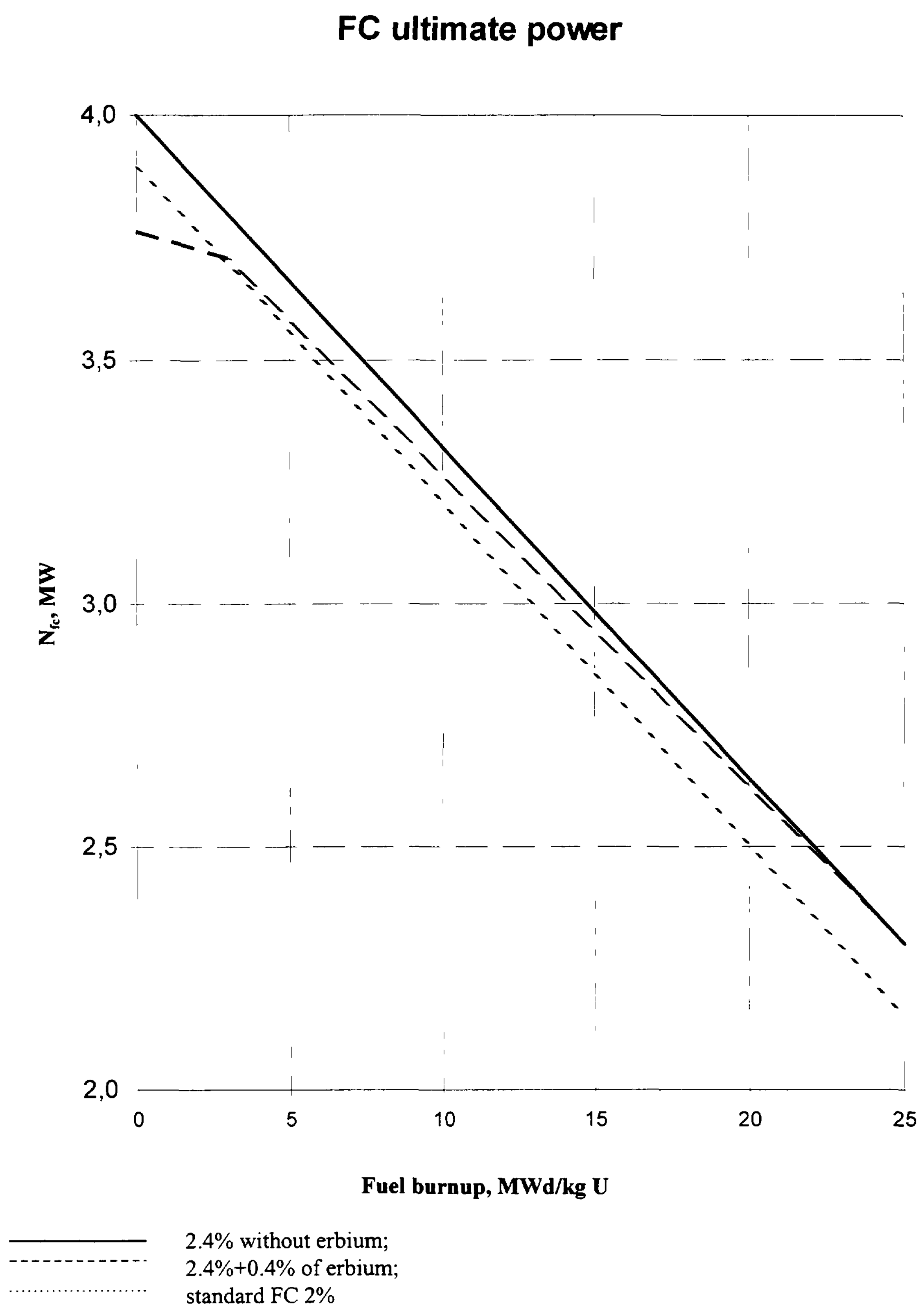

Fig. 2. Maximum FA power versus burnup.

On completing preliminary computer studies in 1993, the activities on introduction of uranium-erbium fuel were developed along the lines as follows:

- computer studies of reactor neutronics as experimental erbium FA (EFA) lots are installed into core, optimisation of RBMK units on changing for uranium-erbium fuel;
- computer studies of the core thermal and engineering reliability, analysis of emergency situations,
- fuel properties studies, fuel element (FE) design justification;
- optimisation of fuel pellet technology, techniques for their quality control, preparation of production for manufacturing experimental lots;
- computer safety analysis of cooling ponds with spent fuel discharged therein.

Basic results of the above activity lines are:

Full-scale core calculations confirmed the said neutronics characteristics of the core with uranium-erbium fuel The calculated analysis which applies the most conservative approaches showed that emergency situations in a core with uranium-erbium fuel runs in a more "mild" manner. Similarity in properties of uranium-erbium and standard fuels permitted to apply the FE design practically without any changes The pellets with a central hole are recommended for application to reduce temperature levels.

The technological flow-chart for manufacturing uranium-erbium pellets is slightly different from the standard one.

The calculations performed are quite natural to require to be confirmed by in-pile experiments. To this end, in 1995 the first lot (150 pieces) of the RBMK-type fuel assemblies with uranium-erbium (UEFA) was manufactured. It was charged into the Ignalina Nuclear Power Plant (INPP) Unit 2 from June through January, 1996 The computer prognosis predicted reduction in void reactivity effect resulting from the experimental lot charge of 0.4 β.

The EFAs charge started at INPP on June 26, 1995 Following the test program on charging every 25 EFAs, the core basic neutronic characteristics were to be measured, first of all the void reactivity effect. The results obtained are presented in Table 2

The value of the void reactivity effect reduced from 0.8 β down to 0.5 β. The maximum power of one fuel channel with EFAs did not overcome the maximum power of a FA with the standard fuel

Bearing in mind positive results of the first stage, it was decided that the tests will be expanded and the EFA (500 pieces) batch will be charged together with simultaneous discharge of additional absorbers At INPP-2 more than 100 additional EFAs are so far charged and more than 260 EFAs are presently under operation. Experimental results are also consistent with computer predictions. By the end of 1996 about 25 EFAs will attain the burnup of 10.0 MW d/kg U.

The first batch of the RBMK-1000 EFAs with fuel enrichment of 2.6 % are presently installed at Leningrad NPP. The documents for charging the EFAs lots at the Chernobyl and Kursk NPPs are under agreement.

It is scheduled that RBMK units will be transferred to uranium-erbium fuel by 2000

TABLE 2. REACTOR PERFORMANCE IN THE COURSE OF LOADING EXPERIMENTAL LOT OF ERBIUM FUEL ASSEMBLIES

Date	**17.03.95**	**08.08.95**	**15.08.95**	**08.12.95**	**29.01.96**	**25.06.95**	**30.06.96**	**16.07.96**	**23.07.96**
Number of EFAs	**0**	**50**	**74**	**100**	**150**	**195**	**195**	**236**	**240**
Thermal power, MW	**4060**	**4000**	**3420**	**3900**	**3700**	**3400**	**3370**	**3960**	**3470**
FA average power production, MW day/FA	**858**	**846**	**834**	**851**	**842**	**848**	**859**	**858**	**868**
Operative reactivity margin, MC rods	**53.9**	**54.2**	**56.9**	**53.7**	**54.6**	**55.0**	**55.9**	**54.5**	**56.4**
Number of AA	**53**	**53**	**53**	**53**	**53**	**53**	**49**	**49**	**45**
α_{φ}, β	**0.83**	**0.7**	**0.6**	**0.7**	**0.5**	**0.6**	**0.6**	**0.6**	**0.7**

Improvement of RBMK fuel reliability and burnup increase

For the purpose of attaining higher fuel burnup and FAs reliability simultaneous activities are in progress along the following lines:

- further increase in enrichment of fuel of RBMK-1000 EFAs;
- introduction of zirconium grids;
- design change of FE fastening in FA for improving fuel operating conditions, among those for operation according to scheduled variable loads;
- repair of defected FAs and their repeated use.

The works on zirconium grids for RBMK-1000 were basically fulfilled during 1985-1988, when a representative lot of EFAs underwent testing at Leningrad NPP. The subsequent post-reactor studies confirmed that the design complied with pertinent requirements. However, after 1986 the activities were stopped because they contradicted with the strategic line for improving safety by increasing the absorber amount in the core. It was shown nowadays in calculations that combination of uranium-erbium fuel with zirconium grids may produce favourable economic effect without causing any damage to safety. That is why these works are presently recommenced and it is planned that in 1998-1999 the passage to zirconium grids will be implemented.

Further increase in fuel enrichment (for RBMK up to 2.8 % with respective change in the erbium content) and associated extension of the EFA service life duration as well as increased number of power cycles during a year that takes place at a number of power units, required that the works aimed at improving fuel elements operating conditions should be performed. In particular, a FA design with fuel elements to be fastened in supporting grids in the centre of the reactor core was developed. This will allow the fuel permissible elongation during operation to be extended without enlarging the clearance between the fuel element bundles. More than that, this design will permit to eliminate a very rare but an extremely unpleasant loss of fuel element leak tightness by the "end-piece break-off"

Repair of damaged FAs

At last, one more line to increase the NPP fuel burnup relates to repair of damaged fuel assemblies and their after burning up to the design basis burnup value. Two types of FA defects can be conventionally singled out:

1. The ones related with damaged structure components and, in the first turn, grids and

2. Those related to loss of fuel elements leak tightness.

Engineering and economic study is performed by now for justifying feasibility to repair FAs by both removal of a leaking fuel element and repair of intensifying grids in FAs of RBMK-1500 Relevant technologies are developed some of which were tested in mock-ups. Certain experience was gained in FAs charging and operation with a repaired grid at INPP Repair of RBMK FAs is primarily planned for NPPs equipped with "hot cells" At the NPP having no such cells the repair works will be carried out on commissioning the spent nuclear fuel repositories of the second generation To our mind it is pssible to reuse from 80 to 85 % of damaged unburnt fuel assemblies.

STATUS AND DEVELOPMENT OF RBMK FUEL RODS AND REACTOR MATERIALS

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Abstract

The paper presents current status and development of RBMK fuel rods and reactor materials. With regard to fuel rod cladding the following issues have been discussed: corrosion, tensile properties, welding technology and testing of an alternative cladding alloy with a composition of Zr-Nb-Sn-Fe. Erbium doped fuel has been suggested for safety improvement. Also analysis of fuel reliability is presented in the paper.

1. CONDITIONS OF FUEL ROD OPERATION IN RBMK-1000 AND RBMK-1500.

The conditions of fuel rod operation in RBMK-1000 and RBMK-1500 in Table 1, 2.

2. SOME SPECIFIC FEATURES OF RBMK FUEL ROD OPERATION

RBMK fuel assembly consists of two vertically positioned fuel rod bundles that are separated in space by a discontinuity in the core centre. The RBMK reactor is operated under conditions of FA continuous reloading. These circumstances condition the two main operation features of fuel rods, namely,

- weld joints are in the core centre;
- in the region of the fuel column discontinuity a power burst (up to ~30%) is realised.

These factors define rather rigid operation conditions for both fuel rod claddings and particularly, weld joints.

One more specific feature of RBMK fuel rod operation consists in the fact that low fuel burn-up (some 2.5-3 times lower than of PWR, WWER or BWR) is combined with the adequately long operation of fuel rods (1100-1200 eff.days) without control of the coolant water chemistry for radiolytic gases at the oxygen content of water up to 20 $\mu g/dm^3$.

This circumstance gives the priority to some factors used to assess the fuel ultimate condition, specifically, nodular and general water-side corrosion, including that under spacer grids, fuel cladding overheats due to cruds and high thickness oxide films available on them, local hydriding, influence of coolant impurities on the temperature conditions of fuel rods, influence of heat flux of a fuel rod on the rate of oxide film growth on fuel claddings etc.

3. CORROSION OF FUEL CLADDINGS

As regards boiling reactor fuel rods also those of RBMK, corrosion of Zr-alloys is a dangerous phenomenon that may substantially confine the life-time of fuel rods The corrosion condition of RBMK fuel rods was studied using FA_s that have operated to different burn-ups, the design burn-up included.

It is established that along with the general corrosion as a uniform dark-coloured oxide film Zr-1% Nb claddings also develop local types of corrosion, namely, nodular corrosion, drastically increased corrosion nodules under spacer grids. The acquired data on the corrosion condition of the RBMK fuel rods are tabulated in table 3.

The nodular corrosion had a typical appearance of white coloured nodules is local thickening at the background of a dense uniform film of dark coloured Zr oxide. The nodules had a lenticular shape with a bulge in the centre. The uniform oxide film thickness varies very little along the fuel rod length, however, the sizes of the local corrosion nodules increase both under and outside the spacer grid with the proximity to the core centre (table 4). No nodular corrosion was observed in the plenum region.

It can be seen from the table the local corrosion under the spacer grid is very intensive A decrease in the fuel rod cladding cross-section in those sites may reach ~40% and more. These local thinning of claddings are stress concentrator sites and can be a cause of a crack nucleation and its further evolution to a primary penetrating defect.

TABLE 1. CONDITIONS OF RBMK-1000 AND RBMK-1500 FUEL ROD OPERATION

№	Parameter	RBMK-1000	RBMK-1500
1	Mean heat generation rate, W/cm - irradiation onset - irradiation end	 250 139	 405 215
2	Maximum heat generation rate,W/cm	350	485
3	Coolant Temperature, ^0C at FA inlet at FA outlet	 270 284	
4	Coolant pressure, MPa	6,7	6,7
5	Maximum temperature of fuel rod cladding, ^0C	367	372
6	Fuel burn-up, MW·day/t U average maximum	 26000 30000	 22500 26800
7	Neutron fluence (E≥1 Mev),n/cm^2	$1{,}3 \cdot 10^{21}$	$1{,}3 \cdot 10^{21}$

TABLE 2. CHARACTERISTICS OF PRIMARY CIRCUIT COOLANT UNDER RBMK-1000 AND RBMK-1500 NORMAL OPERATION CONDITIONS

Parameter	Value
pH at 25°C	6.5-7.8
Specific electrical conductivity at 25°C, μ Sm/cm, not more than	1.0
Mass concentration of chloride-ion, $\mu g/dm^3$, not more than	70
Hardness, μg-ef/dm^3, not more than	5
Mass concentration of SiO_3, $\mu g/dm^3$, not more than	700
Oxygen, $\mu g/dm^3$, not more than	20
Mass concentration of iron, $\mu g/dm^3$, not more than	50
Mass concentration of copper, $\mu g/dm^3$, not more than	20
Mass concentration of petroleum product, $\mu g/dm^3$, not more than	100

The dense oxide film at the outer surface is covered with a deposit layer the colour of which changes from light grey to dark-ginger depending on the thickness. Its thickness was within 7-10 µm. Oxide films and cruds are taken into account by the design analysis of the fuel rod temperature conditions.

4. TENSILE PROPERTIES OF CLADDINGS

As irradiated the fuel rod claddings have rather high tensile properties that do not essentially vary over the fuel rod active part (table 5). No substantial distinction of the metal properties was revealed outside and under spacer grids. The plenum area is characterised by higher ductility compared to the other areas of the fuel rod.

TABLE 3. CORROSION CONDITION OF FUEL ROD CLADDING AS OPERATED IN RBMK-1000 (LENINGRAD NPP)

Cladding material	Fuel burn-up, Mwt/kg	Time of operation, h	Uniform oxide film thickness, μm	Maximum size of corrosion nodules, μm		Mass fraction of hydrogen,%	
				outside SG	under SG	outside SG	under SG
Zr+1%Nb	1.3	3770	10-20	40	150	$\sim 3\text{-}6 \cdot 10^{-3}$	-
	9.9	26440	10-20	60	180	$\sim 1 \cdot 10^{-2}$	-
	19.3	29112	15-20	130	380	$\sim 1 \cdot 10^{-2}$	$\sim 1.5 \cdot 10^{-2}$

TABLE 4. THICKNESS OF CORROSION NODULES IN DIFFERENT AREAS OF FUEL ROD

Distance from the top of upper FA rod, mm	Point of measurement	Magnitude of corrosion nodules, μm
500-800	outside SG	40
	under SG	80
1550 - 3500	outside SG	140
	under SG	380

TABLE 5. MECHANICAL PROPERTIES OF FUEL ROD CLADDING AFTER SERVICE LIFE IRRADIATION

Alloy	T_{test}, ^{0}C	Mechanical properties (in fuel column region)		
		σ_d, MPa	δ_{total},%	δ_{unif}, %
Zr+1%Nb	20	630	9,2	-
	300	480	12,0	~2,5

5. RESULTS OF TESTING PILOT FUEL ASSEMBLIES WITH ZR-NB-SN-FE CLADDINGS IN RBMK-1000 AT LENINGRADSKAJA NPP

The Zr-Nb-Sn-Fe alloy was designed at VNIINM early in 70-is for application as a fuel rod cladding material both in VVER and RBMK. This alloy was also under consideration for other core components, namely, guide thimbles of control and protection system (VVER), pressure tubes (RBMK) and etc.

The experimental fuel rods clad in this alloy were successfully tested in the research reactors (MIR, MR). To generate experimental data on a large massive of fuel rods under conditions of the commercial units of NPP_s 38 pilot fuel assemblies were produced that in their designs did not differ from the standard RBMK fuel assemblies. As a cladding material of the fuel rods of those FA_s use was made of Zr+1%Nb+1.3%Sn+0.35%Fe. The pilot fuel assemblies were tested at LNPP to reach the service-life burn-up. The results of the corrosion resistance testing of the fuel rods clad in this new alloy are tabulated in table 6.

As it can be seen from table 6 the outer Zr-Nb-Sn-Fe cladding surface was not generally *subjected to nodular corrosion. The maximum local corrosion of this alloy under the spacer grids is* some 3.5 times lower compared to that of Zr+1%Nb alloy.

From the comparison between the tensile properties of the cladding materials it is established that the strength characteristics of Zr-Nb-Sn-Fe alloy are much higher and its ductility is lower than those of Zr+1%Nb (see table 7).

6. SOME ISSUES RELEVANT TO STRENGTH OF CLADDING WELDS

As it has already been mentioned the welds of the RBMK fuel rods (one weld of the top fuel rods and one of the bottom ones) are located in the core centre where due to the fuel column discontinuity a power burst is realised. These circumstances make the conditions of the weld operation very rigid.

Under these conditions the application of the progressive resistance butt welding in 1982 made the problem of welds strength more acute. End plug detachments were systematically observed even at the initial stage of irradiation at a low fuel burn-up. These events were of adequate significance. Approximately in 5% events the cause of fuel rod leakage was plug detachments. As a rule it was the bottom plug of a fuel rod in the top bundle of FA_s (table 8).

TABLE 6. CORROSIOM RESISTANCE OF Zr-Nb-Sn-Fe CLADDING OF FUEL RODS AFTER OPERATION IN RBMK-1000 (LENINGRAD NPP)

Cladding material	Fuel burn-up MW day/ kgU	Time of operation, h	Uniform oxide film thickness, μm	Maximum thickness of lokal corrosion nodules, μm		Mass fraction of hydrogen, %	
				outside SG	under SG	outside SG	under SG
Zr+1%Nb+ 1,3%Sn+ +0,35%Fe	11,4	25488	10-20	0	0	$7{,}5\cdot10^{-3}$	-
	19,3	29112	15-20	0	100	$1{,}5\text{-}2{,}0\cdot10^{-2}$	$1{,}5\text{-}2{,}0\cdot10^{-2}$

TABLE 7. MECHANICAL PROPERTIES OF Zr-Nb-Sn-Fe CLADDING OF FUEL RODS AFTER SERVICE-LIFE IRRADIATION (19.3 MWd/kg U)

ALLOY	T_{test}, ^0C	Mechanical properties (in fuel column region)	
		σ_d ,MPa	δ_{total}, %
Zr+1%Nb+ 1,3%Sn+0,35%Fe	20	720	5,3
	300	520	4,0

TABLE 8. SOME DATA OF LOSS OF TIGHTNESS BY FUEL RODS BY PLUG DETACHMENT

(upper FA, weld №1) at units №1 and №2 of LNPP (fuel rod sealing by RBW, plug detachment in confirmed by PIE in hot cells of SB R&DIPE)

№ of FA	Unit	Burn-up, MW·day/bundle	Fabrication date	Load date	Discharge date
1-20-7-26PT	1	194	10.82	03.83	06.83
1-20-7025PT	1	556	10.82	02.83	11.83
1-20-5784	1	657	08.82	11.82	09.83
1-20-6233	1	474	09.82	01.83	09.83
1-20-2319	1	308	01.82	04.82	11.82
1-24-0451	2	655	.81	01.82	11.82

Laboratory and PIE_S showed that the cause of the fuel rod failure due to a plug detachment was intensive hydrading of the weld area under conditions when the fuel column rested upon the weld burr resulting from fuel rod sealing.

The improved plug design (Fig 1) having an internal projection thus eliminating the direct contact between the hot fuel and the burr significantly improved the state of the art making fuel failures due to a plug detachment casual and being realised as a rule by the "secondary hydrading" mechanism.

From this viewpoint of significance is the analysis of the experience gained in the operation of the fuel rods of unit 1 and 2 of Ignalina NPP put into operation in 1984 The design of the RBMK-1000 fuel rods took into account the accumulated experience and the advanced design and technological solutions were realised, namely:

- use of a plug having a projection upon which two blanket pellets of a lower U^{235} content (from 0.4% to 0.7%) but not the UO_2 (2% enrichment of U^{235}) fuel column rest;
- use of fuel pellets with central holes;
- anneal of welds.

Through the blanket pellet location the region of the radial thermal mechanical pellet-cladding interaction is placed away from the plug, thus eliminating extra tensile stresses in the weld area that are induced by the side effect (Fig. 3, 4). This figure shows the distribution of the design elastic stresses simulating the radial loading of a cladding accomplished by swelling fuel (the account is also taken of the radial coolant pressure - 6 MPa) with blanket pellets available and without them. It is seen that during operation the absence of blanket pellets may result in grass tensile stresses in the weld area under the action of the side effect.

Besides, blanket pellets also lower a power burst providing an efficient protection against weld hydrading (Fig. 2).

The optimal character of this solution may be illustrated by the following data on the fuel service at INPP. At this plant during the whole period of the operation of the two RBMK-1500 units there were only four events of fuel rod failures due to plug detachments (1985, 1986, 1988 and 1995), i.e., 1 failure per ~200 thousand discharged fuel rods for the two units on the average. During the recent 8 years there was a single event of a plug detachment, i.e., 1 failure per 550 thousand discharged fuel rods. This event took place in a fuel rod of the bottom fuel bundle discharged in 1995 among 35 leaky FA$_s$. The fuel rods of those 35 FA$_s$ were overheated due to the unscheduled cooling conditions.

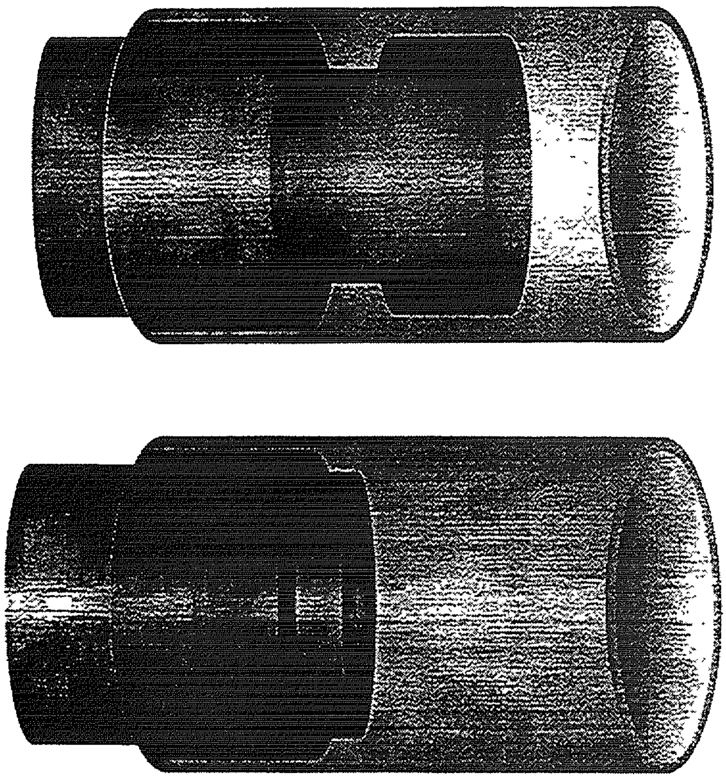
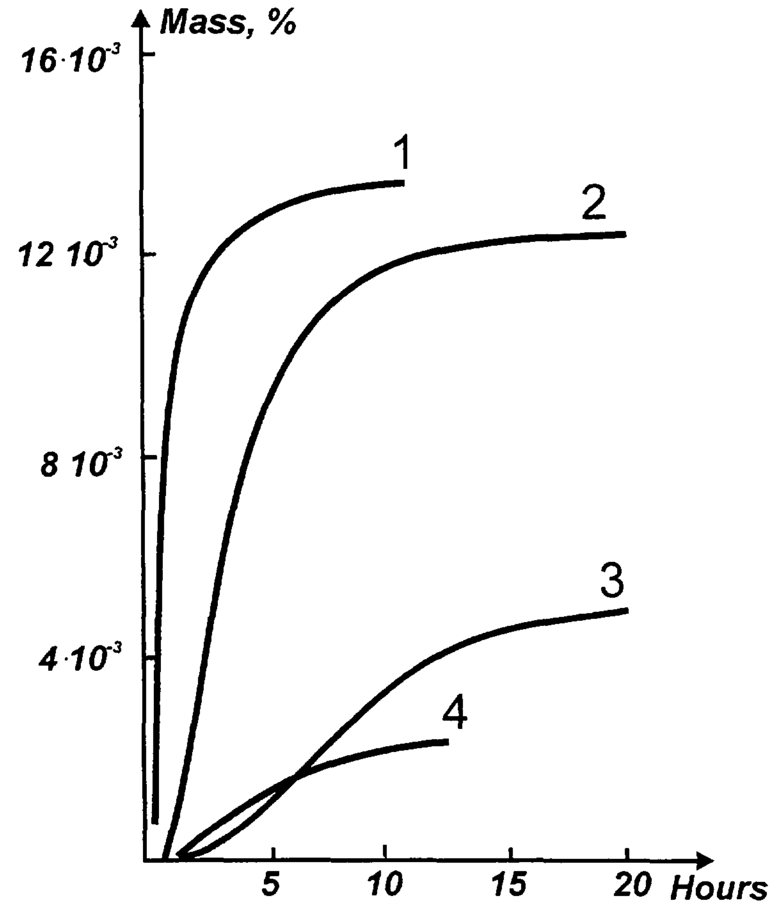


Fig. 1. Drawing of the end plug with a "mushroom" (upper) and with a boss (lower).



1-RBMK fuel welded joint with fuel on burr,
2-standard RBMK-1000 fuel welded joint with fuel on boss,
3-welded joint with "mushroom" plug having 3.5 mm diameter stem,
4-standard RBMK-1000 fuel welded joint having plug with boss and depleted (0.4% enriched) pellets.

Fig 2 Hydrogen concentration variation in resistance butt welded joints of different design in RBMK fuel rod.

It is to be noted that there is an interrelation between the events of loss of tightness by fuel rods for the reason of a plug detachment and the total quantity of leaky FA_s This circumstance may be explained by the fact that plugs are detached by the "secondary" hydrading mechanism as a result of the mixed steam-water ingress into the fuel rod interior via a small primary defect.

To-day in a pilot operation at LNPP are fuel rods having a mushroom shaped plug (Fig. 1). This mushroom plug promotes an improved resistance to weld hydriding.

7. OPERATION RELIABILITY OF FUEL RODS

Presently the operation reliability of the RBMK fuel rods differs very much between not only different NPP_s but also different units of the same NPP.

At the same technology level the operation reliability can vary between different units within 10^{-5} to 10^{-4} per year. On the one hand, this is an evidence that the operation conditions of the RBMK units have a significant influence on the fuel rod damageability, and, on the other, that those conditions are quite different in different reactor units even for the operation routine common for all NPP_s.

Therefore, the fuel damageability can be reduced via the further optimisation of the RBMK core operation parameters.

From our point of view, most significant and useful for the understanding of the problem will be the analysis of the state of the art at the two units of INPP If has already been mentioned that at this plant the fuel rods operate at heat loads a factor of 1.25-1.3 higher than in other RBMK units. Table 9 summarises the data on the number of discharged leaky FA_s of INPP submitted to us by Mr. VORONTSOV B.A., the deputy chief engineer on INPP safety. It follows from the tabulated data that during 1991-1995 the level of the failures at unit 1 was $\sim 1.4 \times 10^{5}$/year. As far as unit 2 is concerned without the account for 40 FA_S discharged in 1994-1995 for the reason of the unscheduled overheating of the fuel rods the level of the failures is $\sim 2.0\text{-}2.5 \times 10^{-5}$ /year. To-day this failure level is typical of the modern cores of NPP_S with VVER, BWR and PWR. To attain those high results, at INPP a large scope of work was implemented to optimise the core performance, namely, algorithms of stabilisation (within 15% of power rating fields) were introduced during FA_S reloading, additional absorber discharge and replacement of water columns by FA_S etc.

Also the problems were resolved relevant to a reduction of axial deviations of power density fields.

TABLE 9. NUMBER OF LEAKY FUEL ASSEMBLIES DISCHARGED FROM UNITS 1 AND 2 OF IGNALINA NPP (1988-1995)

YEAR	UNIT 1	UNIT 2
1988	10	5
1989	9	6
1990	9	8
1991	1	7
1992	3	5
1993	-	1
1994	2	5
1995	1	35

* The data are submitted by Mr. Voronzov V.A.,
Chief engineer deputy on safety, INPP

8. RBMK FUEL DEVELOPMENTS AND PRODUCTION USING ERBIUM AS BURNABLE ABSORBER

After the Chernobyl accident to improve the safety of RBMK type reactors the steam void reactivity coefficient was much reduced via loading some extra absorbers into the core which resulted in a lower burn-up of fuel as well as in an increase of the mean and maximum power of FA channels. The design studies aimed at reducing the steam void coefficient showed that an erbium burnable absorber as Er_2O_3 added to the fuel lowers down the steam void coefficient of reactivity to the level which does not require any extra absorbers to be loaded into the core. Besides, the burnable absorber available in fresh FA_S reduces significantly their power and the reload effected reactivity This simplifies the reloading procedure and control of the energy distribution within the core The lower non-uniformity of the power rating allows a higher enrichment, thus, extending the fuel burn-up /1, 2, 3, 4/.

To-day *the production of the U-Er oxide fuel has been mastered by Electrostal plant* as applied to RBMK-1000 and -1500. U-Er fuelled FA_S have been fabricated and now under test at Ignalina NPP (150 FA_S) and Leningrad NPP (200 FA_S)/. The work is under way to switch all the reactors to the fuel of this type. The erbium content of the oxide fuel is 0 41% mass. U_{235} enrichment of the fuel was increased from 2 4% to 2.6% for RBMK-1000 and from 2.0% to 2 4% for RBMK-1500

The process flow outline of the U-Er fuel pellet manufacture contemplates the use of uranium dioxide and erbium oxide powders as initial materials. The initial powders of uranium dioxide and erbium oxide in the specified ratio are blended in a vane mixer A binder is added to the as sieved powder mixture The uniformity of the erbium oxide and carbon binder is controlled. The further process does not essentially differ from the process used to manufacture the standard fuel Thus, the only distinction is the additional technological operation of blending the initial uranium dioxide and erbium oxide powders. On the one hand, this operation is very important since a high degree of uniform erbium distribution within the pellet basic material is required, on the other, it is adequately sophisticated as the quantity of erbium introduced is low.

The adopted blending process provides for the needed uniformity of the mixture. From the results of the manufacture the degree of the fuel uniformity is such that the pellet distribution in terms of their total erbium contents has a relative mean square root deviation of 2.0%. The distribution of erbium within the volume of individual fuel pellets as determined by the x-ray spectral microanalysis is also uniform.

Based on the work performed the specifications for the U-Er pellets were established, analytical techniques of control were developed and metrologically certified; they are to control the erbium content of fuel pellets, the total content of uranium, the O/Me ratio, impurities such as nitrogen, hydrogen, carbon, fluorine, chlorine .

At Electrostal plant an isolated bay was created to manufacture U-Er oxide fuel. The technological process of U-Er pellet manufacture was designed and several pilot lots of this fuel were produced of the total mass>50t for testing in INPP and LNPP

The properties of the U-Er fuel were studied both under laboratory conditions and in-pile. Laboratory scale investigations were carried out to study the thermal conductivity, thermal expansion coefficient, strength, thermal creep, U-Er fuel pellet - Zr+1%Nb cladding compatibility. The corrosion resistance of the U-Er fuel was investigated.

Measurements of the pellet thermal expansion within 25-1500^0 C showed that within the temperature range studied the linear thermal expansion of the U-Er fuel is systematically less than of UO_2 which may be related to a smaller ionic radius of erbium. However, the differences do not exceed 0,6%

The thermal conductivity of pellets was determined in the temperature range of 300-1500° C. It is found out that with the introduction of Er the thermal conductivity of the fuel is lowered in the whole temperature range studied, however, it does not exceed 10% Aside from the laboratory studies of the U-Er fuel thermal conductivity, in-pile investigations were performed Specifically, in the IVV-2M reactor the fuel temperatures were measured under irradiation using instrumented U-Er fuel rods and the standard fuel [1] The analysis of the acquired results implemented within the programme RET (TR) made it possible to introduce corrections into the temperature dependence of the U-Er fuel as derived under laboratory conditions (see Fig 5)

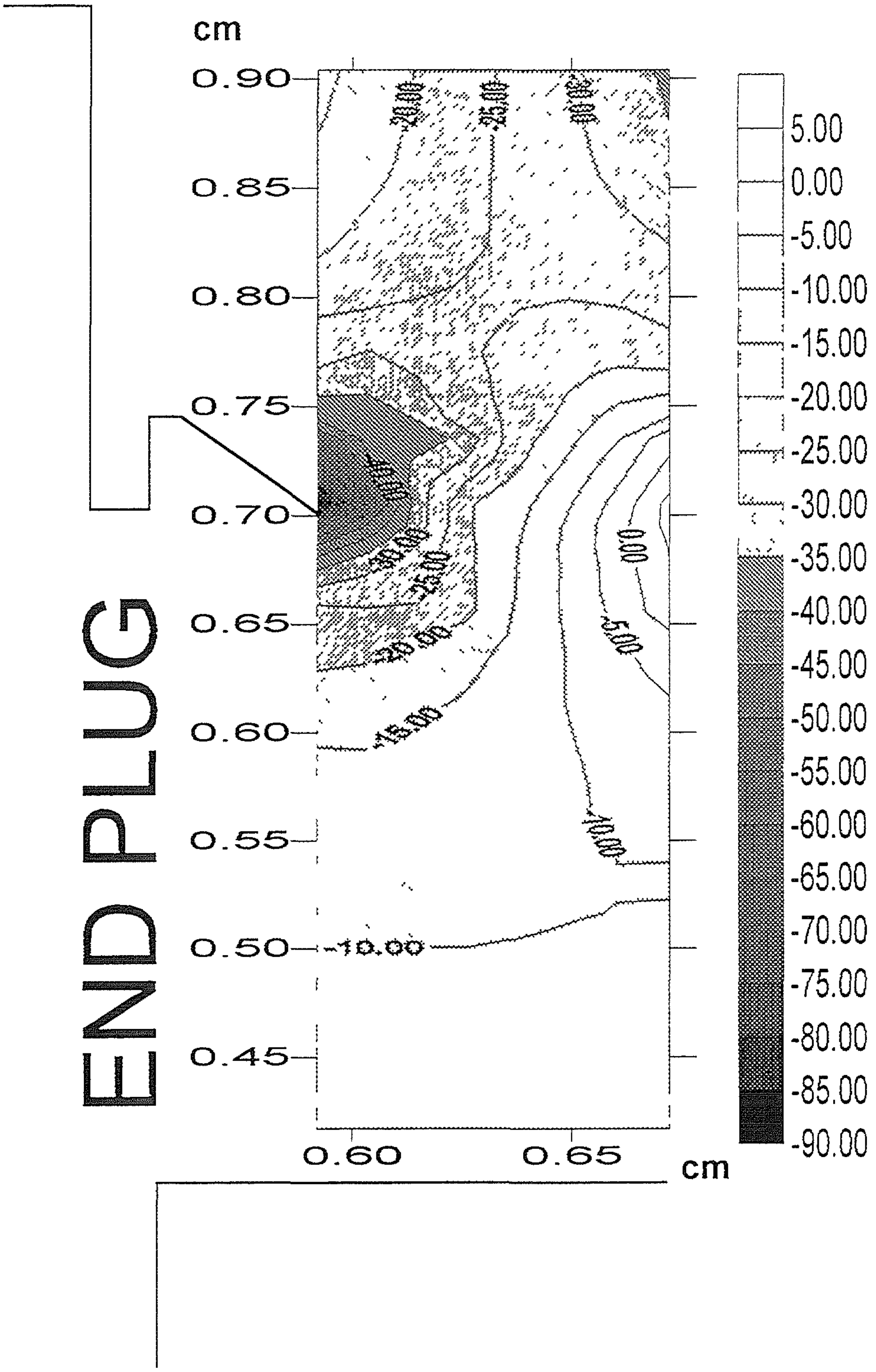


Fig 3 Longitudinal stresses in fuel rod cladding with blanket pellet in welded joint area, MPa

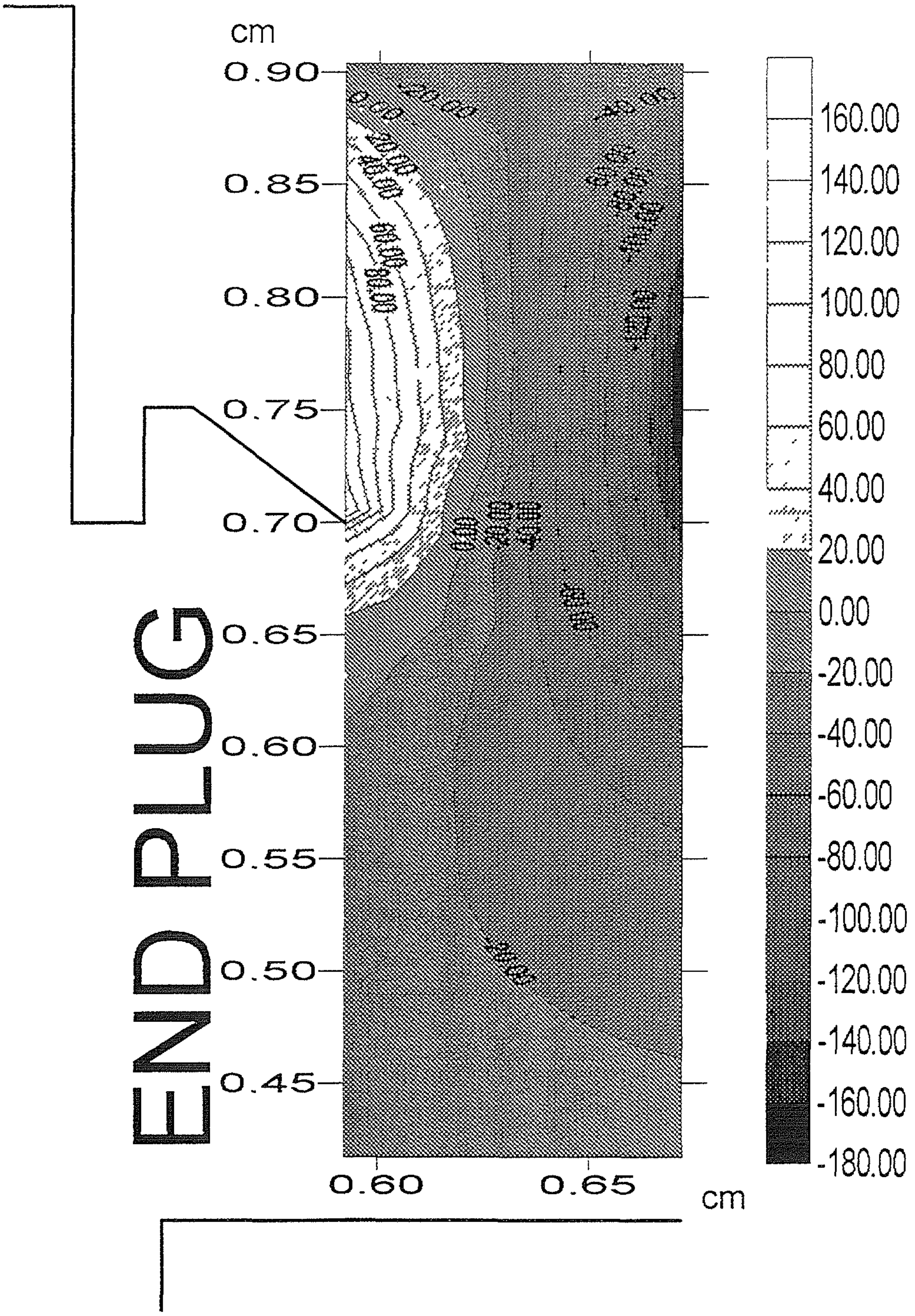


Fig 4 Longitudinal stresses in fuel rod cladding without blanket pellet in welded joint area, Mpa

The room temperature mechanical strength of U-Er fuel pellets (0 41% Er) was determined by measuring the maximum load endurable by pellets upon compression to fracture. The measurements revealed that the mean fracture strength of pellets is σ =237.7±11 MPa This value is above that of the standard RBMK pellets, i.e., σ = 211± 3 MPa.

Investigations of the U-Er fuel thermal creep were launched. The set of the data on the creep as a function of stress and temperature is qualitatively described by the relationship:

$$\varepsilon \sim \exp (Q/RT+\alpha\sigma)$$

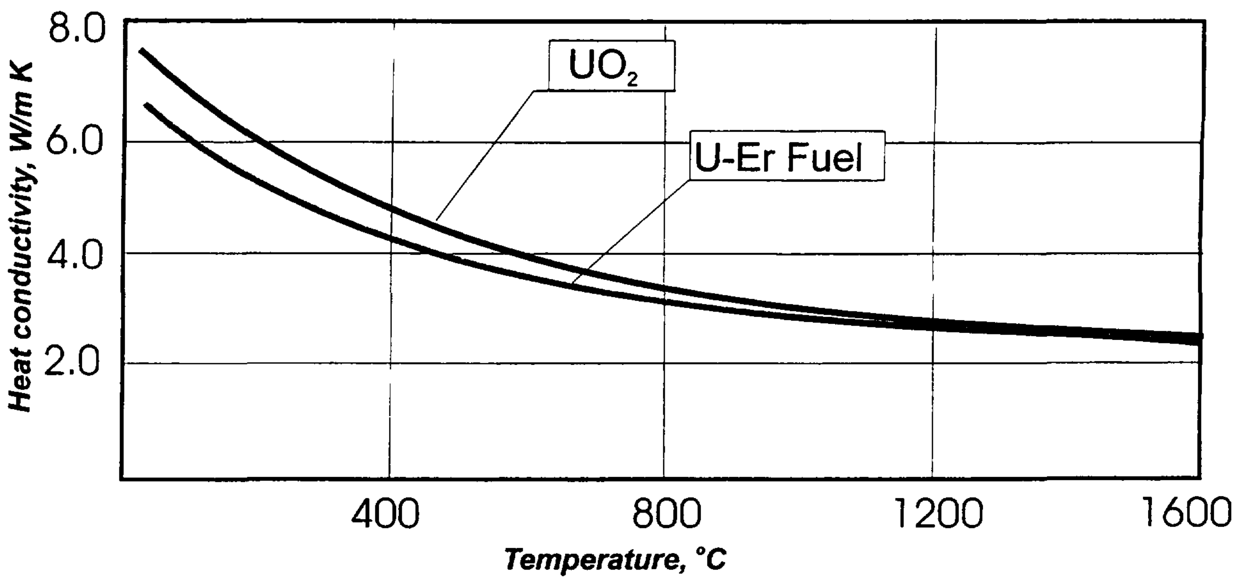


Fig. 5. Heat conductivity of U-Er fuel

where ε is creep rate, σ is stress, Q is activation energy.

The activation energy of creep is equal to 460 Kτ/mole at temperatures above 1350° C and 165 Kτ/mole at lower temperatures. The investigations are in progress.

The tests for the U-Er fuel - E110 alloy cladding compatibility at 800-1200° C and the diffusion anneal time of 4 and 9h showed that the total thickness of the interaction zone layers is smaller for U-Er fuel compared to that for the standard UO_2 fuel.

The comparative study was continued of the corrosion properties of U-Er and standard fuel (P-10-E and P-10, respectively) in water at 350° C and the pressure of 16 5 MPa for 25, 100, 500 and 1000h and in steam at 500° C and the pressure of 7.0 MPa for 25 hours (see table 10).

To study the release of fission gas products (FGP) R&DIPE and Sverdlovsk branch R&DIPE performed in-pile tests of ventilated U-Er and standard fuel rods that allowed the assessment of unstable nuclides released inside a fuel rod cladding. The analysis of the results within the programmes RET (TR) and VKI that was carried out at VNIINM showed that at the fuel temperatures of 870-1200° C and burn-up < 3.5 MW.day/kgU (the reached burn-up, the experiment is in progress) erbium introduced into the fuel does not influence the release of FGP [2]

For the analysis of the serviceability of pilot U-Er fuel rods a series of calculations was performed, involving thermophisical calculations within the RET (TR) programme, the strength calculations under normal operation conditions and calculations of the cladding stability within the programmes START-3 and BUSTER, an analysis of the fuel rod behaviour under conditions of the severe design basis accident within the RAPTA programme. When implementing the calculations the needed refinement of the programmes was accomplished for taking into account the changes in the fuel properties induced by the introduction of erbium The calculations were performed proceeding from the conditions of the highest heat density fuel operation that is the top bundle fuel rod of RBMK-1000 and - 1500 FAs. For the sake of comparison the thermophysical design of the standard fuel rod was carried out. The operation conditions and design parameters of the RBMK-1000 fuel

TABLE 10. RESULTS OF COMPARATIVE TESTS OF STANDARD (P-10) AND U-Er PELLETS (P-10-E) IN WATER AND STEAM

Type of test	Type of fuel	№ of specimen	Weigth gain after testing, mg/cm^2			
			25 h	100 h	500 h	1000 h
Water: 350 °C 16,5 Mpa	P-10	1	0,20	0,27	0,33	0,45
		2	0,14	0,24	0,30	0,42
		3	0,17	0,19	0,35	0,51
		4	0,14	0,27	0,50	0,62
		5	0,24	0,45	0,54	0,68
		mean	0,18	0,28	0,40	0,54
	P-10-Э	1	0,41	0,68	0,69	0,80
		2	0,32	0,47	0,54	0,66
		3	0,18	0,20	0,33	0,51
		4	0,33	0,63	0,66	0,77
		5	0,39	0,65	0,72	0,74
		mean	0,33	0,53	0,59	0,70
Steam: 500 °C 7,5 MPa	P-10	1	1,46			
		2	1,55			
		3	1,89			
		4	2,49			
		5	3,12			
		mean	2,10			
	P-10-Э	1	1,48			
		2	2,55			
		3	2,95			
		4	4,69			
		5	1,82			
		mean	2,70			

TABLE 11. COMPARISON BETWEEN CONDITIONS OF RBMK FUEL ROD 9 WITH STANDARD AND U-Er FUEL) OPERATION

PARAMETER	RBMK-1000 FUEL ROD		RBMK-1500 FUEL ROD	
	U-Er fuel	Standard	U-Er fuel	Standard
Mean heat generation rate, W/cm - beginning of irradiation - end of irradiation	 250 139		 360 210	 405 215
Fuel burn-up , Mw·day/tU - average - maximum	 26000 30000		 22550 26800	
Maximum heat generation rate, W/cm	350		430	485
Maximum cladding temperature, °C	367		371	-
Maximum fuel temperature, °C - beginning of irradiation *- end of irradiation*	 1366 755	 1411 760	 1660 1010	 1775 1050
FGP release (end of cycle), %	1,4	1,4	6,6	7,0
Gas pressure in fuel rod (end of cycle), MPa	1,85	1,83	3,1	3,1

rod are similar to those of erbium containing fuel rod for the exception of the standard fuel pellets without central holes. In the RBMK-1500 fuel rod the design parameters of the both fuel rods are identical, the distinctions refer to the operation conditions that at the initial moment of irradiation are more rigid (see table 11) for the standard fuel rod. Table 11 contains the design thermophysical characteristics of the standard and U-Er fuel rods of RBMK-1000 and - 1500.

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MODELLING OF THERMAL - MECHANICAL INTERACTION OF RMBK FUEL PELLETS AND CLADDINGS

XA9846757

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Abstract

The thermal-mechanical pellet and cladding interaction (TM PCI) in RBMK fuel rods leads to cladding deformation under transient operation conditions. This phenomena is practically confirmed by the full scale in-pile experiments and by the operation experience. The significant fuel rod elongation after long-term irradiation under RMBK operation conditions is observed.

What are the reasons and mechanism of this phenomenon? Based on the fuel rods behavior during operation, the calculation model of TM PCI and computer code OXPA have been developed by the authors team. It is under Russian regulatory body licencing now.

The main ideas and technique of the OXPA code are presented in paper.

Preliminary analysis

Fuel rods operation ability depends on the thermal-mechanical conditions. Experimental study of the fuel rod behavior shows that TM PCI take place under transient conditions mainly. The reasons of transients are the power increase, fuel assembly insertion in the start of operation and non-steady reactor operation.

The thermal-mechanical model is based on the following phenomenon. Under rise-to-power the pellet radial temperature gradient is increased. It is accompanied by the stress increased too. The pellet cracking is happen, the jumping effect take place and pellet's fragments put into contact with cladding. The gap between fuel and cladding is disappeared.

The further power rise leads to TM PCI both in the radial direction and in the axial direction. The interaction zone is located in the limited part along the fuel rod's axis. Fuel pin strains cladding, but fuel-cladding interaction is not hard one: the slip of fragmented fuel on the cladding inner surface take place. Fragmented fuel may be pressed practically in the contact zone

In the case of fuel pin is leaned on the fuel rod end plug, the TM PCI and fuel .thermal expansion leads to axial force appearing.

In principal, TM PCI and cladding loading may cause the stress corrosion cracking of claddings under iodine presence. This phenomena depends on the stress level and irradiation parameters The axial force which affects on the end plug can tear off the plug.

Under power cycling TM PCI leads to non-elastics strain accumulation in the cladding The total fuel rod elongations can achieve significant and dangerous values.

The team which consists of IPPE and ENTEK specialists has analyzed the experimental results and has developed the computational model of the thermal-mechanical behaviour of fuel rods with UO_2 fuel pellets under transient conditions [1,2,3,4].The calculation technique for the fuel rod operation ability was developed too.

In-pile tests under non-steady conditions

In-reactor water loop has been used for reactor tests. Experimental assembly consisted of 18 rods: 6 in the inner row and 12 in the outer row, Fig 1.

Irradiation parameters are presented in the Table l. Under 75% reactor power the fuel rod linear power is 44.4 kW/ m. The persistent monitoring of fuel rod elongation has been carried out during

TABLE 1. TEST ASSEMBLY IRRADIATION PARAMETERS

1	Cladding material	Zr+1%Nb
2	Cladding diameter and thickness	13.6x0.85 mm
3	Rod length	3644 mm
4	Fuel	UO_2
5	Pellet dimensions: diameter height	11.5 mm 15 mm
6	Maximum linear power	59000 Wt/m
7	Maximum burn up	19.600 MWt*day/kg
8	Non-uniformity coefficient: radial axial	 1.17 1.40
9	Number of rods: inner row outer row	6 12
10	Maximal neutron flux	$2*10^{13}$ n/cm^2*s
11	Coolant	water
12	Average power rate	2%/min
13	Number of power cycling (1-100%)	340
14	Staying power	50-70 h

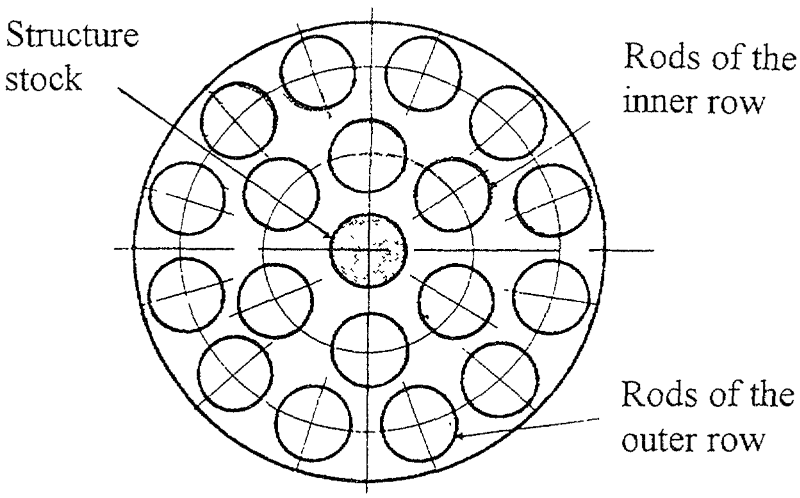


Fig.1. Rod position scheme in the experimental assembly.

irradiation. The experimental curve of fuel rod elongation during single power cycle (0-100%-0) is shown on the Fig. 2. Inelastic elongations of inner fuel rod and outer fuel rod under thermal cycling are shown on the Fig. 3.

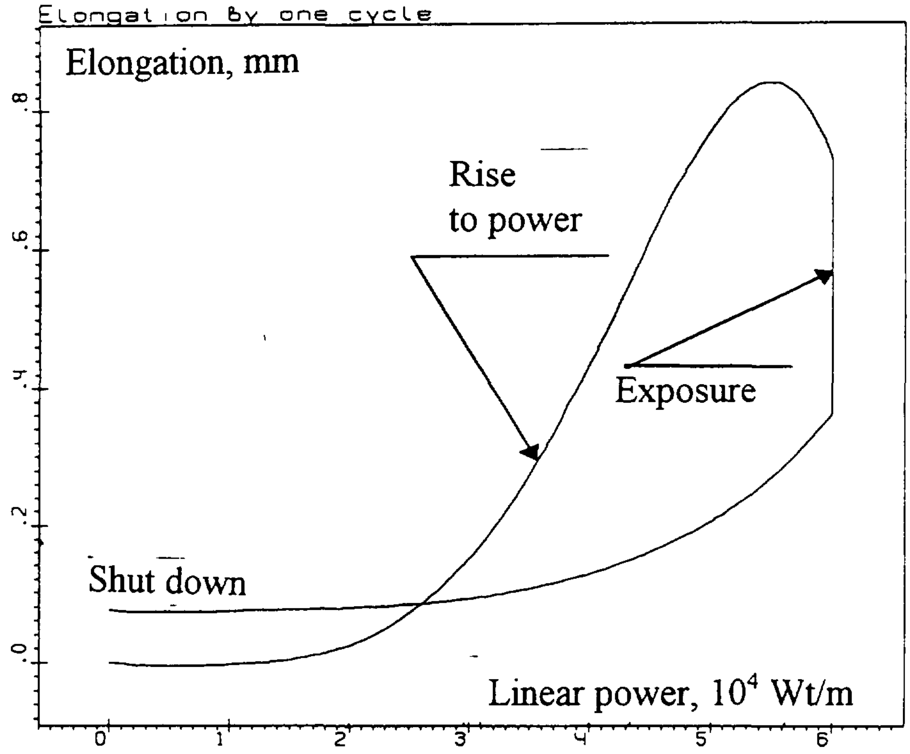


Fig.2. Experimental fuel rod growth during single cycle 0-100%.

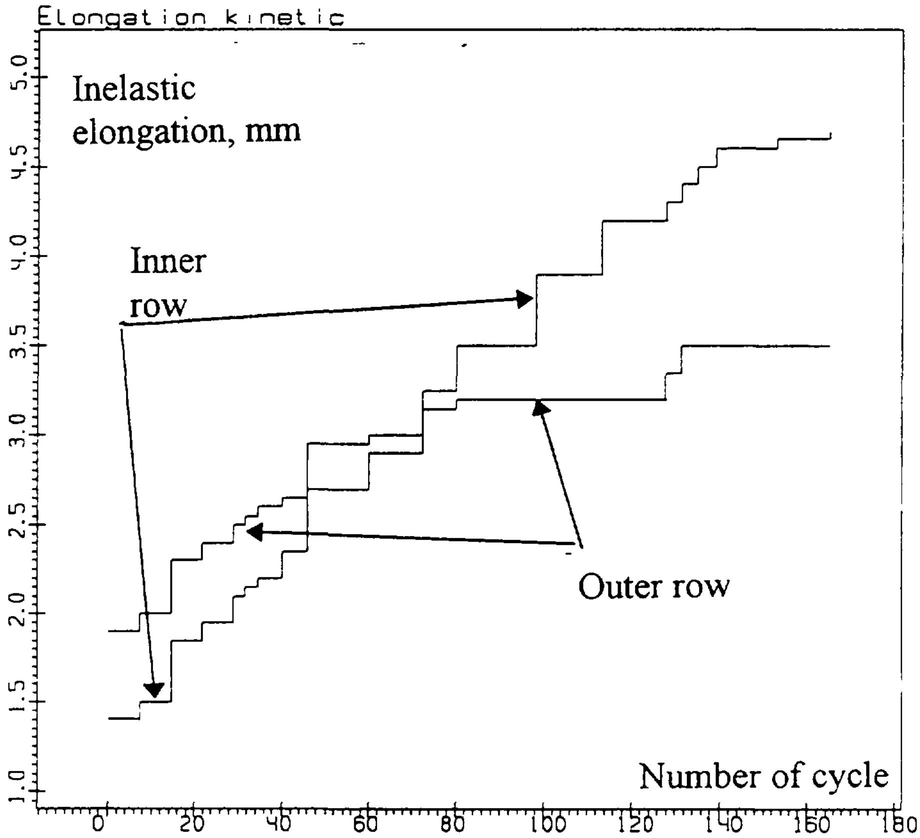


Fig 3. Inelastic fuel rod elongation versus number of cycles.

The following issues have been found after in- pile tests:

-Inelastic fuel rod deformation take place under power cycling. The reason of elongation is TM PCI.
-Inelastic deformation doesn't happen on every cycle of power rise, Fig. 3.
-Cladding-fuel slip take place in the contact area under TM PCI Thus, cladding deformation is less than fuel thermal expansion, fig.2.
-Slip rate depends on power. The power is more the slip rate is more too. The average elongation of inner fuel rods (28 kWt/m) is bigger than average elongation of outer fuel rods (44 kWt/m), Fig. 4 (a,b).
-The axial elongation value per cycle is the accidental one, Fig. 3.
-The most significant elongation rate is observed in the initial irradiation stage (Fig. 4).
-The axial gaps between fuel pellets are appeared because of pellet sticking in the different levels of the fuel rod.

Calculation model

All of above mentioned phenomenon was taken into the basis of mathematical model for fuel rod behavior under power cycling.

The accidental behaviour of fuel rod is realized by the following statistical parameters:

-The axial TM PCI level is the accidental parameter. It is changed from cycle to cycle.
-Not every power cycle is effective one. The number of non-effective cycles between effective ones is the accidental value.

Both one and another accidental functions are the experimentally selected functions. Besides, the function $F_i(x)$, which indicates the contact level, depends on previous irradiation history and cycling. It takes into account the cracked pellet sticking after previous cycles and inelastic clad deformation under previous loading. It considers spaces between stuck pellets and the end plug.

The calculation scheme of TM PCI is shown on the Fig. 5. On the Fig. 5 l_k is the axial contact level, l_i...l_k - deformed cladding space. Upper part above l_k isn't loaded by expanded fuel, because fuel pin is freely expanded into upper fuel rod value. Only external pressure affects to the cladding here.

The thermal-mechanic contact in the space l_i.. l_k is not hard one; slip take place. The slip-function Γ is an empirical one.

The slip-function Γ has been obtained from experimental data analysis. The typical elongation curve is shown on the Fig 6.

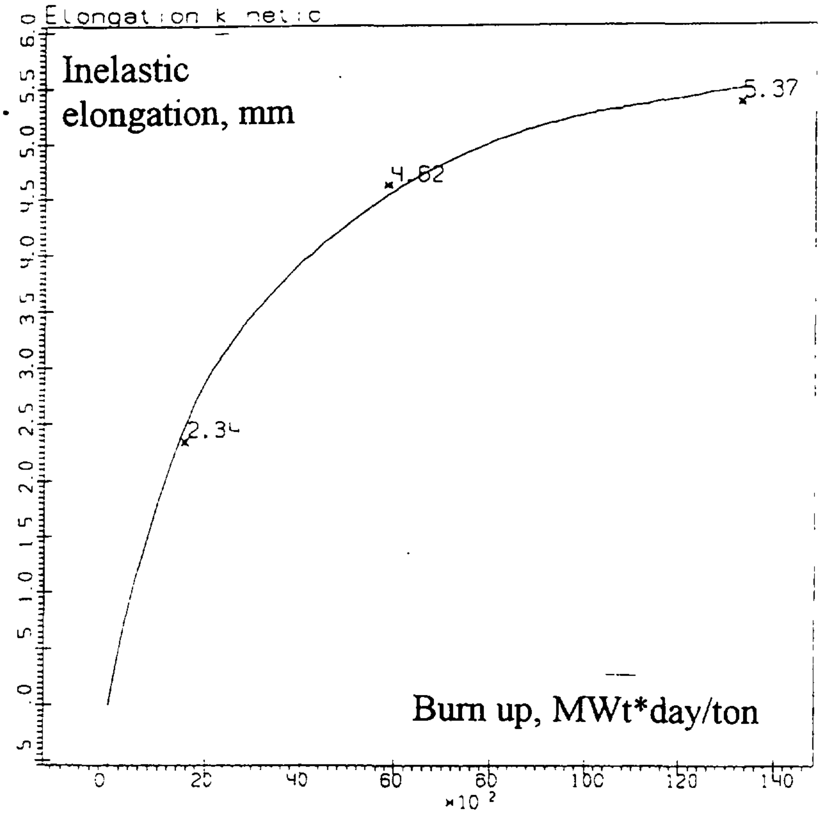
At the initial stage of the power rise the interaction force is small (Zone 1 on the Fig. 6) and slip rate is equal practically to the thermal expansions differences of fuel and cladding

Under rise to interaction force (σ_x is increased) the slip rate is reduced and is approached to the zero approximately (Zone 2 on fig.6). In this case the cladding deformation rate (Zone 3 on Fig. 6) approached the fuel expansion rate.

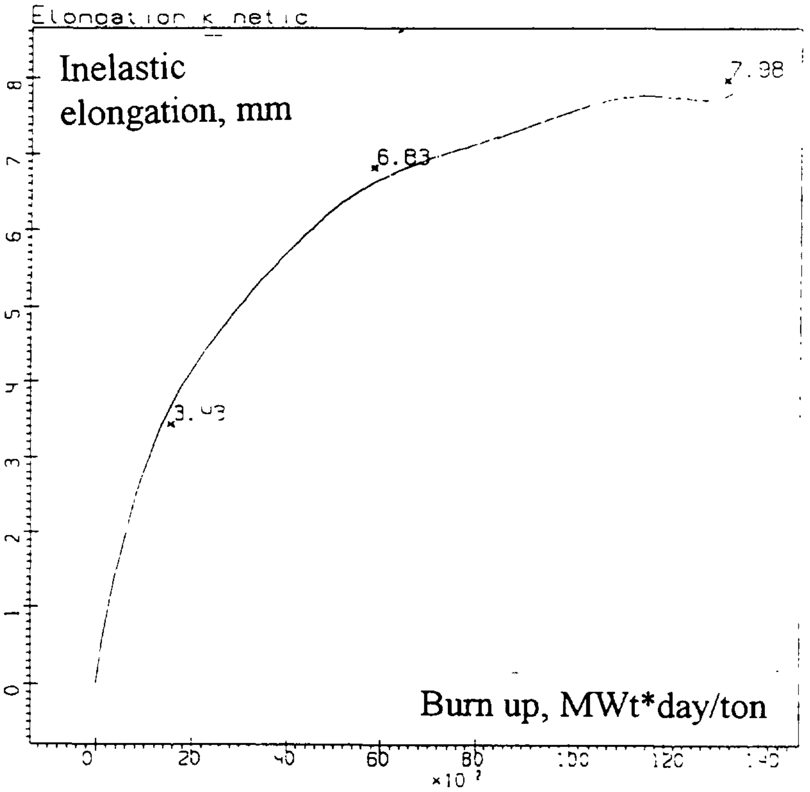
Under steady-state operation stage (Zone 4) the slip function Γ depends on the temperature and stress conditions Γ-function was found experimentally.

Calculation results

Comparative experimental and calculation data are shown on the Fig. 7. On the Fig. 8 the stress kinetics during the single power cycle is shown. The axial distribution of the hoop strain for the outer



(a)



(b)

Fig.4. Experimental elongation during irradiation for the outer rods (a) and inner rods (b); (*) - experimental data.

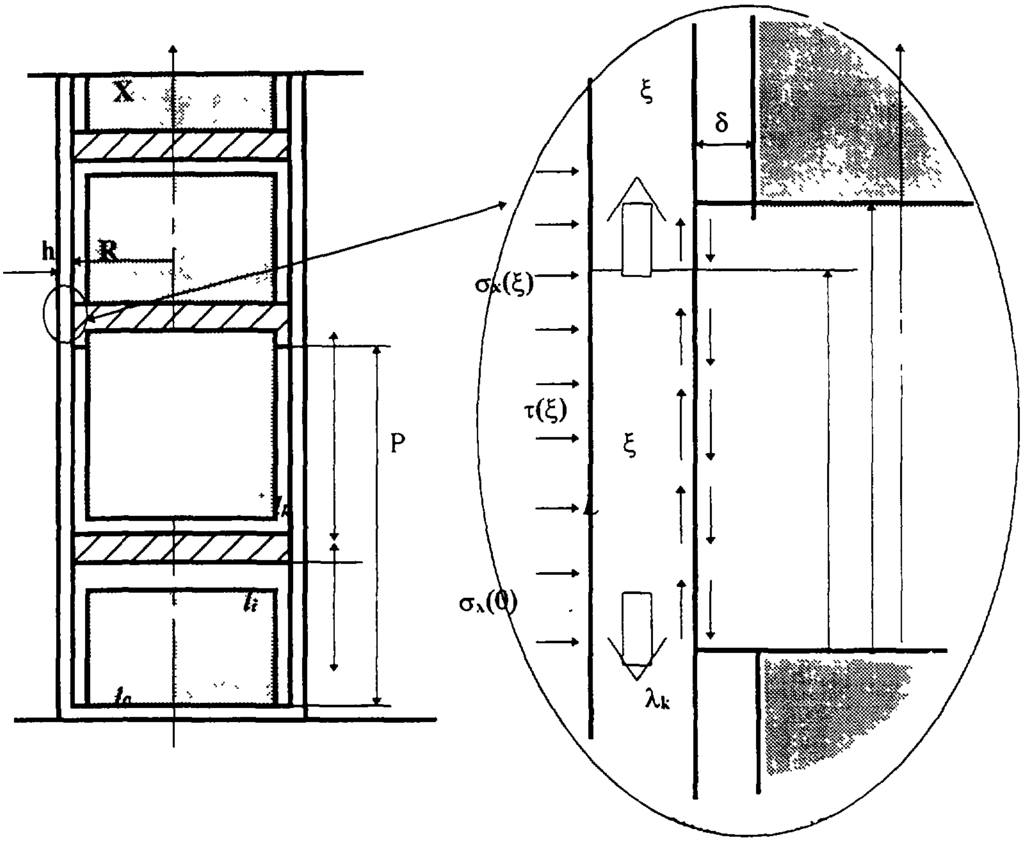


Fig.5. The calculation scheme of TM PCI.

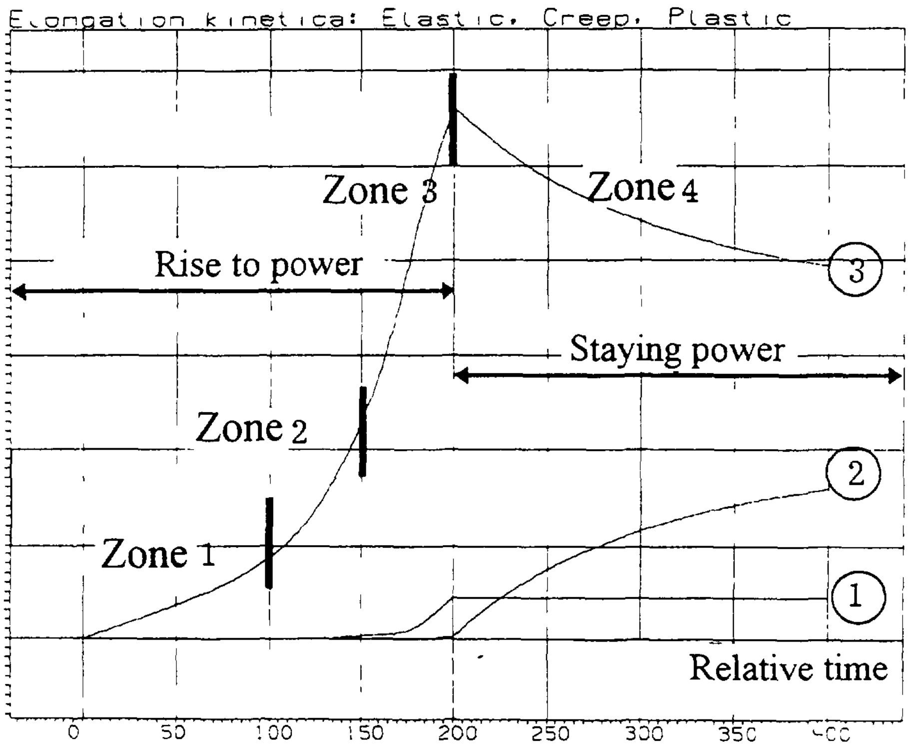
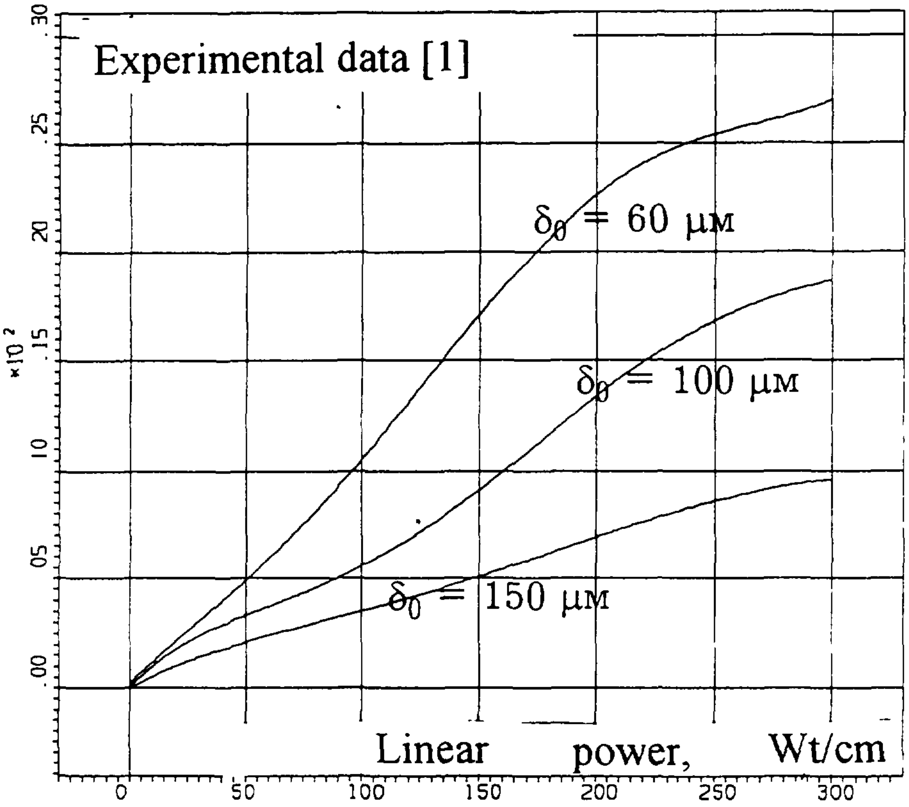
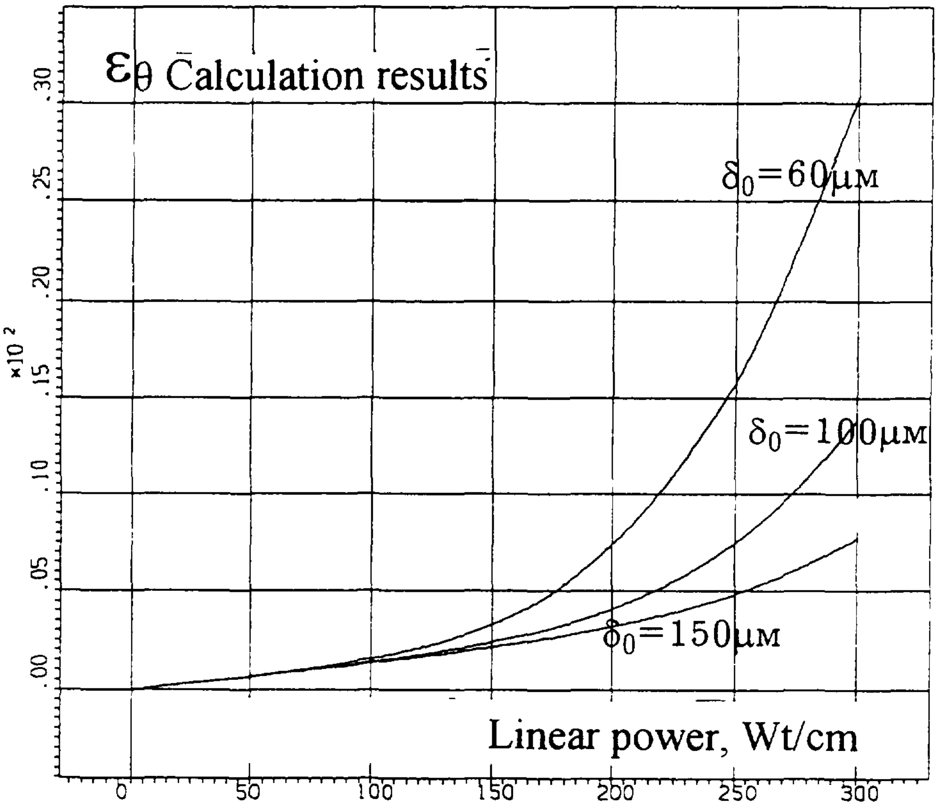


Fig.6. Typical strain curve during single power cycle,
1 - plastic elongation,
2 - creep elongation,
3 - total elongation.



(a)



(b)

Fig.7.Experimental (a) and calculation (b) cladding hoop deformation for different initial gaps in the fuel rods.

row experimental fuel rods is shown on the Fig. 9. The calculation kinetics of inelastic elongation for the outer row experimental fuel rods is shown on the Fig. 10. The calculation statistics of the inelastic fuel rod elongation for experimental fuel rods is shown on the Fig. 11

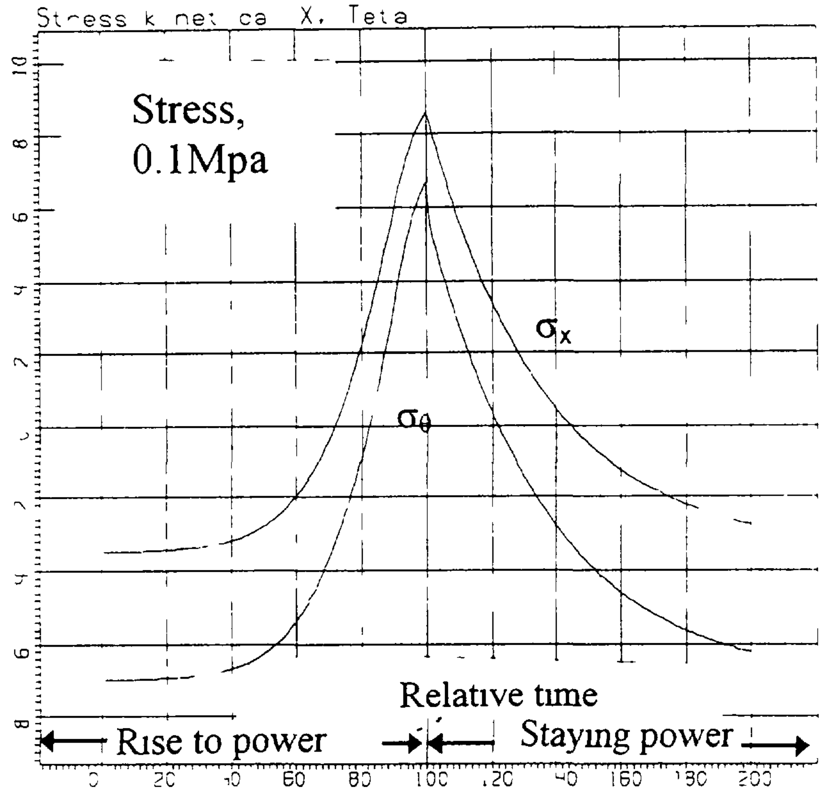


Fig 8 The stress kinetics during single cycle

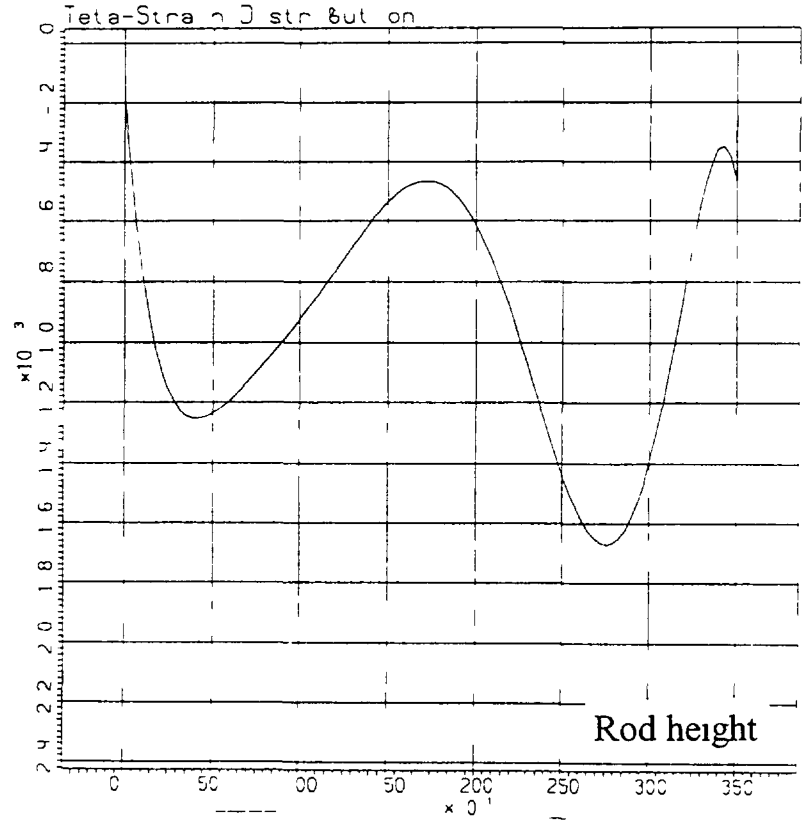


Fig.9 The axial distribution of the hoop strain for the outer row experimental fuel rods

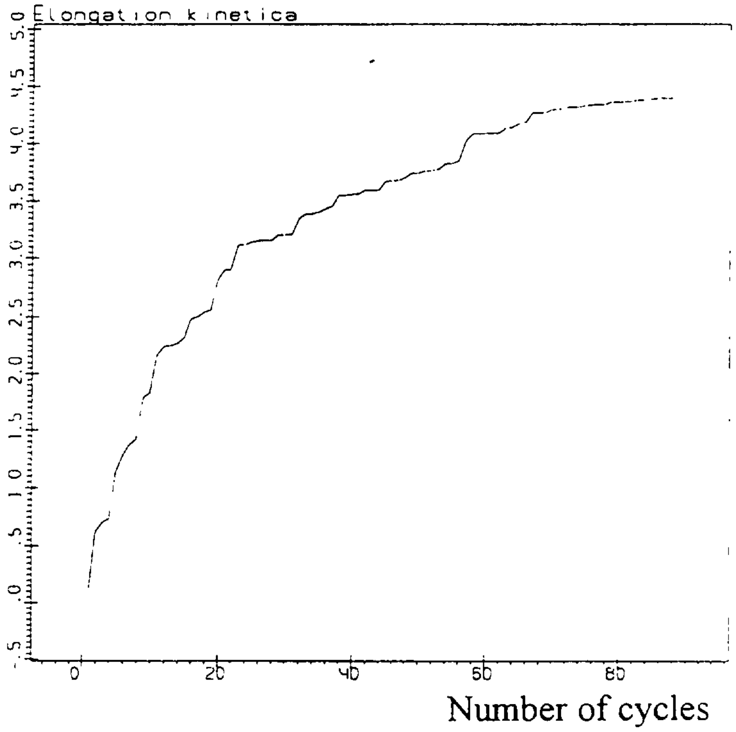
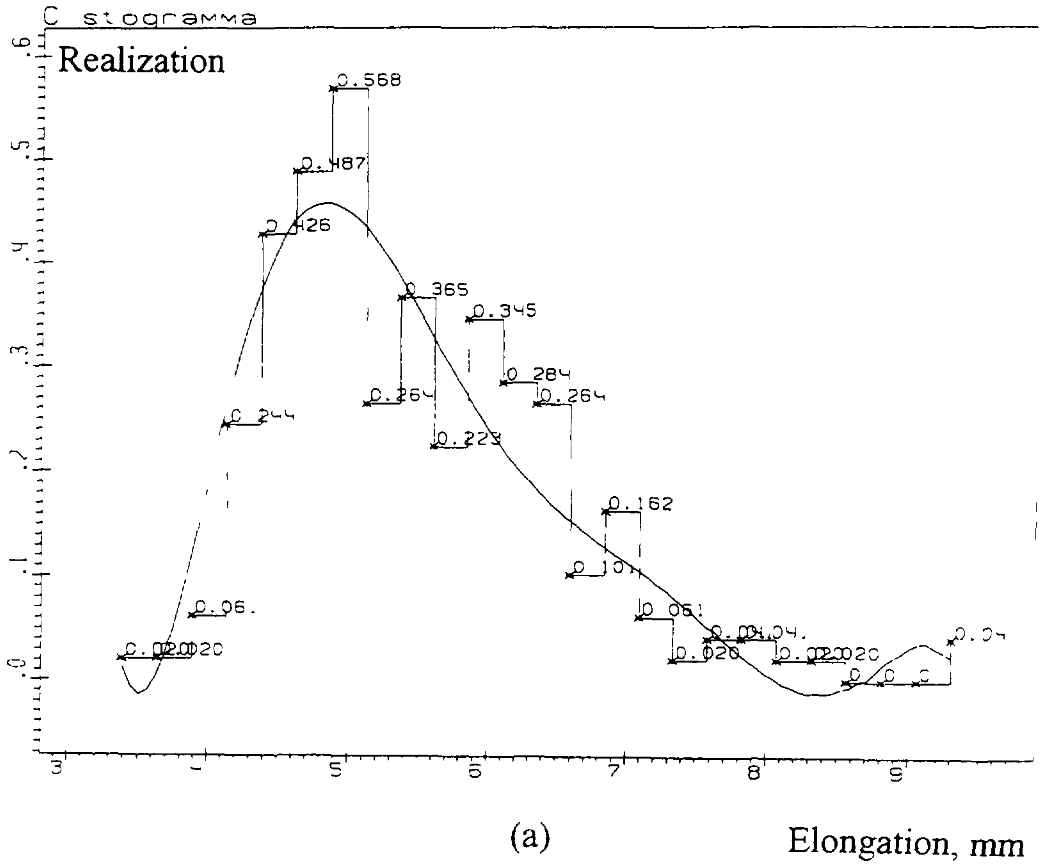
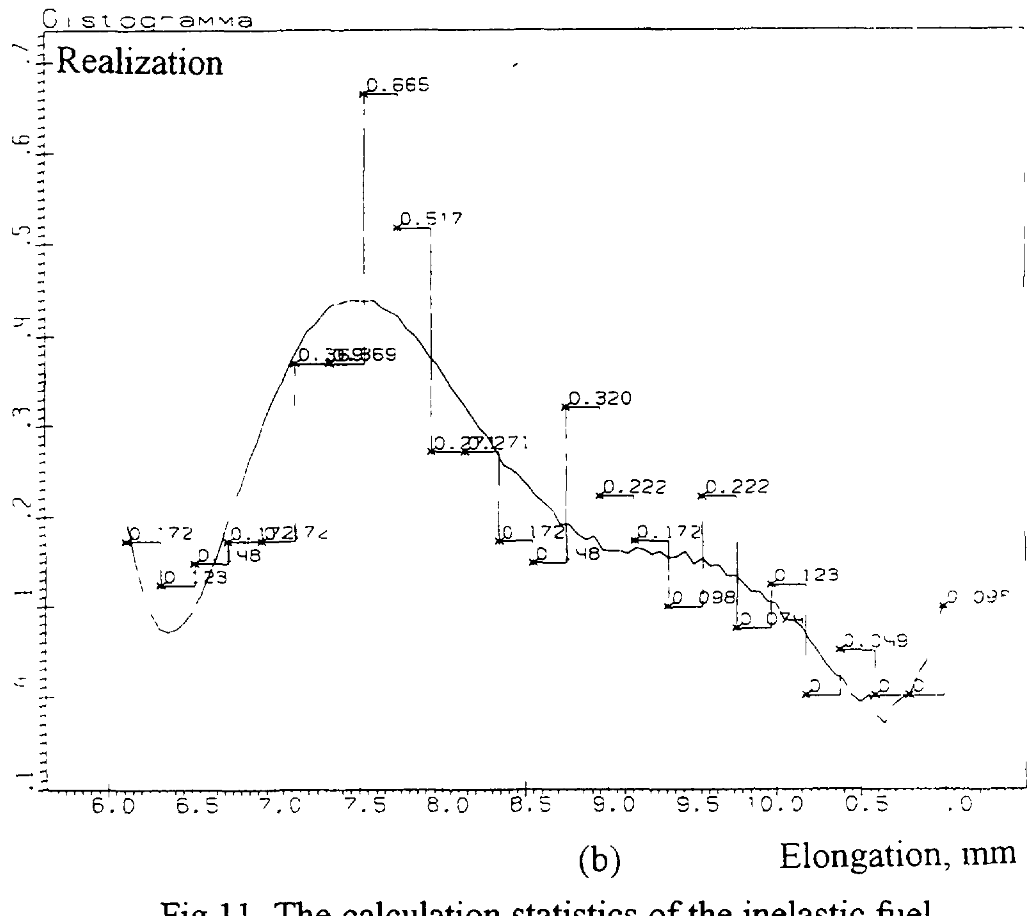


Fig.10. The calculation kinetics of inelastic elongation for the outer row experimental fuel rods.



(a)



(b)

Fig.11. The calculation statistics of the inelastic fuel rod elongation for experimental fuel rods, (a) - outer row, (b) - inner row.

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R&D ACTIVITIES ON FUEL CHANNELS

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POST-IRRADIATION EXAMINATION AND EVALUATION OF Zr-2.5%Nb PRESSURE TUBE OF FUGEN

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Abstract

Paper describes the results of post-irradiaton examination and evaluation of Zr-2.5%Nb pressure tubes of FUGEN reactor. The emphasis is given to the definition of mechanical and fracture toughness properties and hydrogen content and their relationship. Experimental data obtained are compared with design criteria.

Introduction

FUGEN is a 165 MWe prototype heavy-water-moderated, boiling-light-water-cooled, pressure-tube-type reactor having the special feature of using mainly plutonium-uranium mixed oxide (MOX) fuel FUGEN has operated satisfactorily since the start of commercial operation in March 1979

The vertical cross-section of the reactor and primary cooling system is shown in Fig. 1. Two hundred twenty four fuel assemblies are separately loaded in each pressure tube which is located through a Calandria tank filled with heavy water. Light water introduced to the core via inlet feeder pipes cools the fuel assemblies, changes into two phase flow and rises up to the two steam drums through outlet pipes. The primary cooling system consists of two independent loops: A-loop and B-loop. Each steam drum separates the two phase flow into hot water and steam. The water is circulated at 8,800 t/h of flow rate. The steam rotates a turbine generator directly and changes to condensate which returns back to the core as feed water at 910 t/h.

Specifications of pressure tubes

Zirconium alloys are widely used as material for the pressure tubes because they possess high strength and have a low absorption rate of neutrons. A Zr-2.5%Nb alloy that has been heat-treated in order to enhance its high temperature strength was chosen for FUGEN The pressure tube assembly of FUGEN consists of Zr-2.5%Nb pressure tube and upper and lower extension tubes of stainless steel, and is about ten meters in total length. The pressure tube is about five meters long, with 117.8 mm inside diameter and 4.3 mm wall thickness. The joint method of the Zr-2.5%Nb pressure tube and the extension tube of stainless steel is a mechanical joint called "rolled joint". The structure of the pressure tube assembly is shown in
Fig. 2.

The design service conditions of the Zr-2.5%Nb pressure tube of FUGEN are as follows.

design pressure	8.0 MPa (82 kg/cm^2)
design temperature	296°C
operating pressure	6.96-7.22 MPa (71.0-73.6 kg/cm^2)
operating temperature	280-286°C
neutron flux (E > 1 MeV)	approximately $3 \cdot 10^{17}$ n/m^2/sec (max.)
neutron fluence (E > 1 MeV)	$3 \cdot 10^{26}$ n/m^2 (max.)
primary coolant	pure water
heat insulating gas	pure dried CO_2 gas

The specifications of the Zr-2.5%Nb pressure tube is shown in Table 1.

Degradation mechanisms

The expected degradation mechanisms and the evaluation methods of the pressure tubes of FUGEN are shown in Table 2. Four degradation mechanisms were considered at design of FUGEN. Ductile fracture, unstable fracture and corrosion of these degradation mechanisms are evaluated through post-irradiation examinations of pressure tube surveillance specimens. Creep, one of the degradation mechanisms, is evaluated through measuring inside diameter of the pressure tube with pressure tube inspection equipment.

Table 1 Specification of Zr-2.5%Nb Pressure Tube

Chemical Requirements

Element	Composition, Weight %
Nb	2.40-2.80
O	0.09-0.13
Zr	the others

Mechanical Properties

Condition	Tensile Strength, min. MPa	Yield Strength, min. MPa	Elongation, min. %
Solution heat treated (887°C) / Water quenched Cold-drawn & Aged (500°C x 24 hr)	764	689	10

Table 2 Expected Degradation Mechanisms and Evaluation Methods for Pressure Tube

Degradation mechanisms	Design requirements	Evaluation methods
Ductile fracture	Stress intensity < Tensile strength x 1/3 and Yield stress x 2/3	Post-irradiation examinations of surveillance specimens
Unstable fracture	■ Crack tip stress intensity factor < Fracture toughness x 1/2 (for operating condition) and Fracture toughness x 2/3 (for hydrostatic tests) ■ Hydrogen content < 211ppm (30 years)	
Corrosion	Thickness loss by corrosion < 0.31 mm (30 years)	
Creep	Diametral creep strain < 2.5% (30 years)	Inside diameter measurement with pressure tube inspection equipment

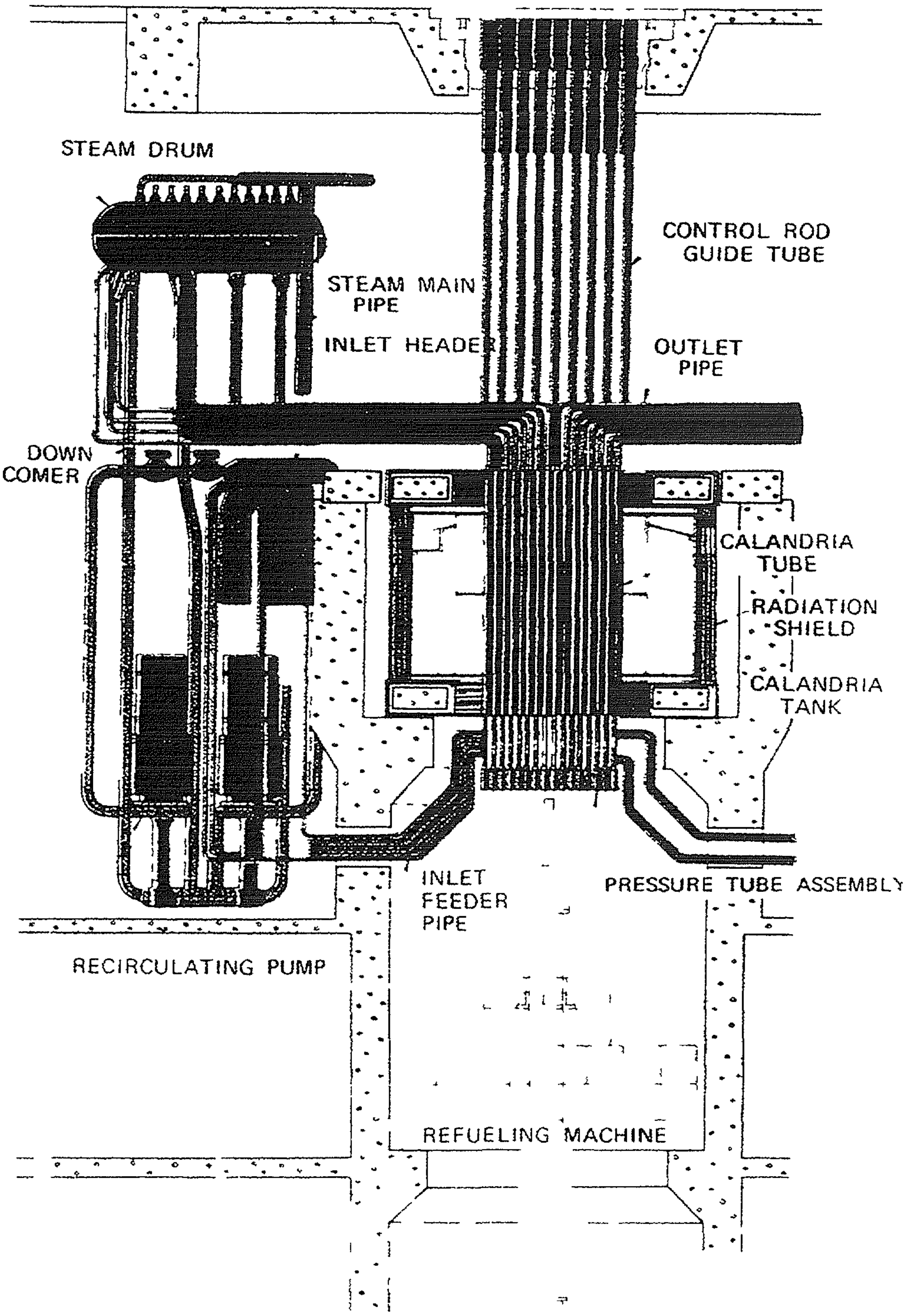

Fig.1 Vertical section of reactor and primary cooling system

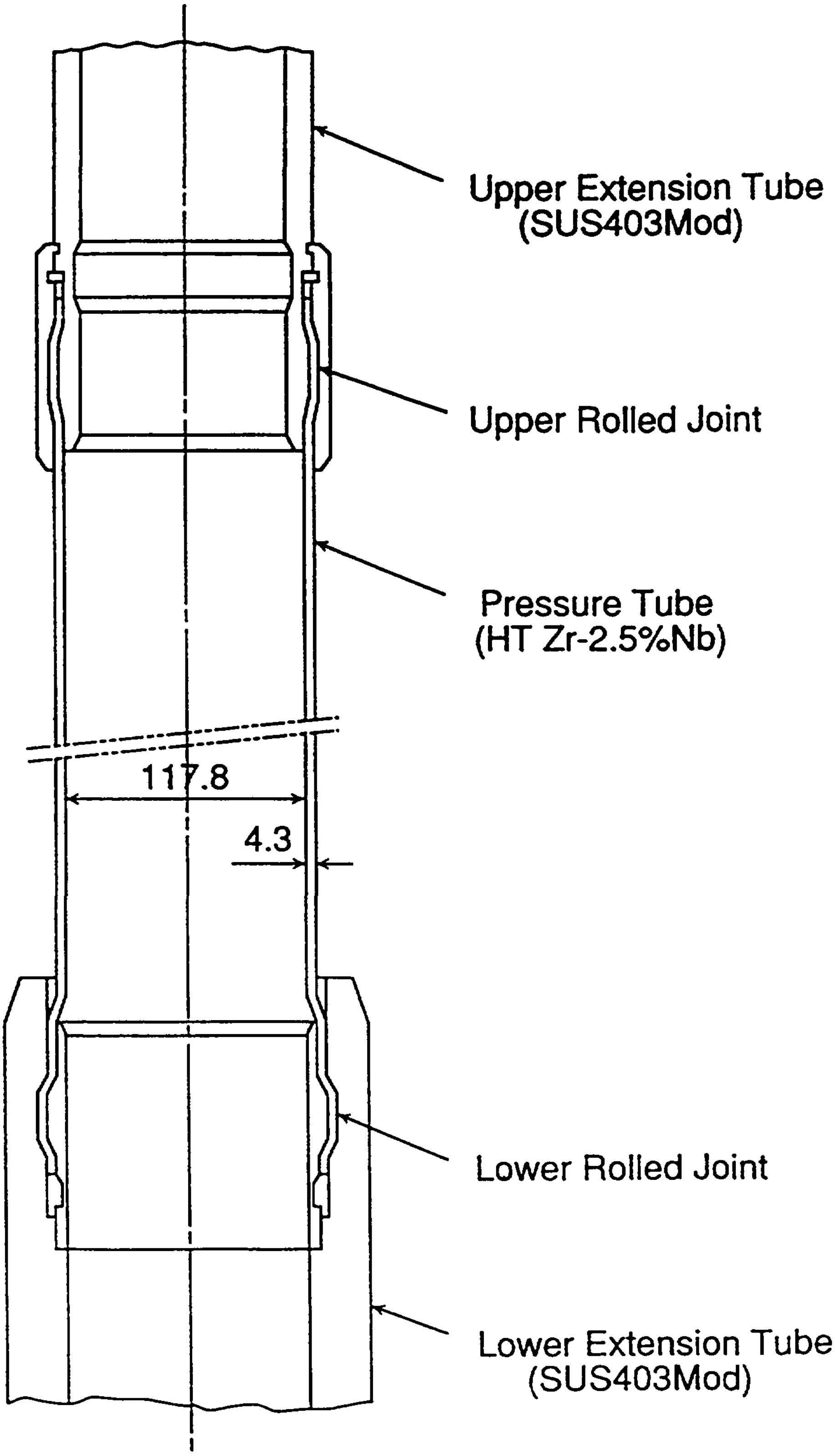

Fig.2 Pressure Tube Assembly

Results of post-irradiation examination

The fuel assembly is placed inside each pressure tube, and the pressure tubes are consequently exposed to a high rate of neutron irradiation. Therefore, it is important to evaluate the characteristic change of the material under irradiation. For this purpose, pressure tube surveillance specimens had been assembled in the inside of four special fuel assemblies since the initial stage of operation and exposed to irradiation. The surveillance specimens were transported from FUGEN to the Material Monitoring Facility in our O-arai Engineering Center, and the post-irradiation examinations were carried out three times from 1984 to 1995 for tension test, bending test, hydrogen analysis and corrosion test. Fast neutron (E > 1 MeV) fluence of the surveillance specimens for the third post-irradiation examination was $7.0 \cdot 10^{25}$ n/m^2

The tension test is carried out in order to evaluate the ductile fracture properties. As shown in Fig 3, the ultimate tensile strength (uts and the 0.2% proof stress (y show a tendency to increase. These values reach about 30% higher levels than those of unirradiated material. At fluence more than $2 \cdot 10^{25}$ n/m^2, these values become almost constant. Meanwhile, reduction area and elongation decrease during initial period of irradiation, but after that, these values also become constant.

The bending test is carried out in order to evaluate the fracture toughness properties related with the unstable fracture. Main design criteria are provided for fracture toughness Kc at operating temperature and at room temperature, as follows:

Kc (at 300°C) > maximum design stress intensity factor K_I (at 300℃) $\cdot$ 2 = 12.6 MPa m$^{1/2}$(40.6 kg/mm$^{3/2}$)

Kc (at room temp) > maximum design stress intensity factor K_I (at room temp.) $\cdot$ 1.5 = 10.9 MPa m$^{1/2}$ (35.0 kg/mm$^{3/2}$)

K_I values are determined by considering following three items The first is assumed an initial defect of 5.0 mm length and 0.4 mm depth which based detection performance of pressure tube inspection. The second is crack growth from the initial defect by fatigue in plant life time. The third is operating stress. The safety factors are based on ASME Code. During initial period of irradiation, Kc values at both temperatures show a tendency to decrease, as shown in Fig. 4 These values reach about 10% lower levels than those of unirradiated material. At fluence more than $2 \cdot 10^{25}$ n/m^2, these values become almost constant. The design criterion is satisfied in either temperature

Hydrogen concentration has a remarkable influence on the fracture toughness properties of Zr-2 5%Nb. The relationship between hydrogen concentration and fracture toughness was obtained through our another examinations It was confirmed that the fracture toughness required by the design was well ensured up to the design criteria of hydrogen concentration 221ppm, as shown in Fig. 5.

The results of the hydrogen analysis are shown in Fig. 6. The design line is based on the initial hydrogen content, and experimental data of hydrogen pick-up rate obtained by Canada. Our data through the post-irradiation examinations are smaller than the design line, and hydrogen pick-up rate seems decreasing.

The results of the corrosion test are shown in Fig. 7. The design criterion is provided for thickness loss by corrosion. The design line is based on experimental data of Canada. Our data through the post-irradiation examinations are extremely smaller than the design line

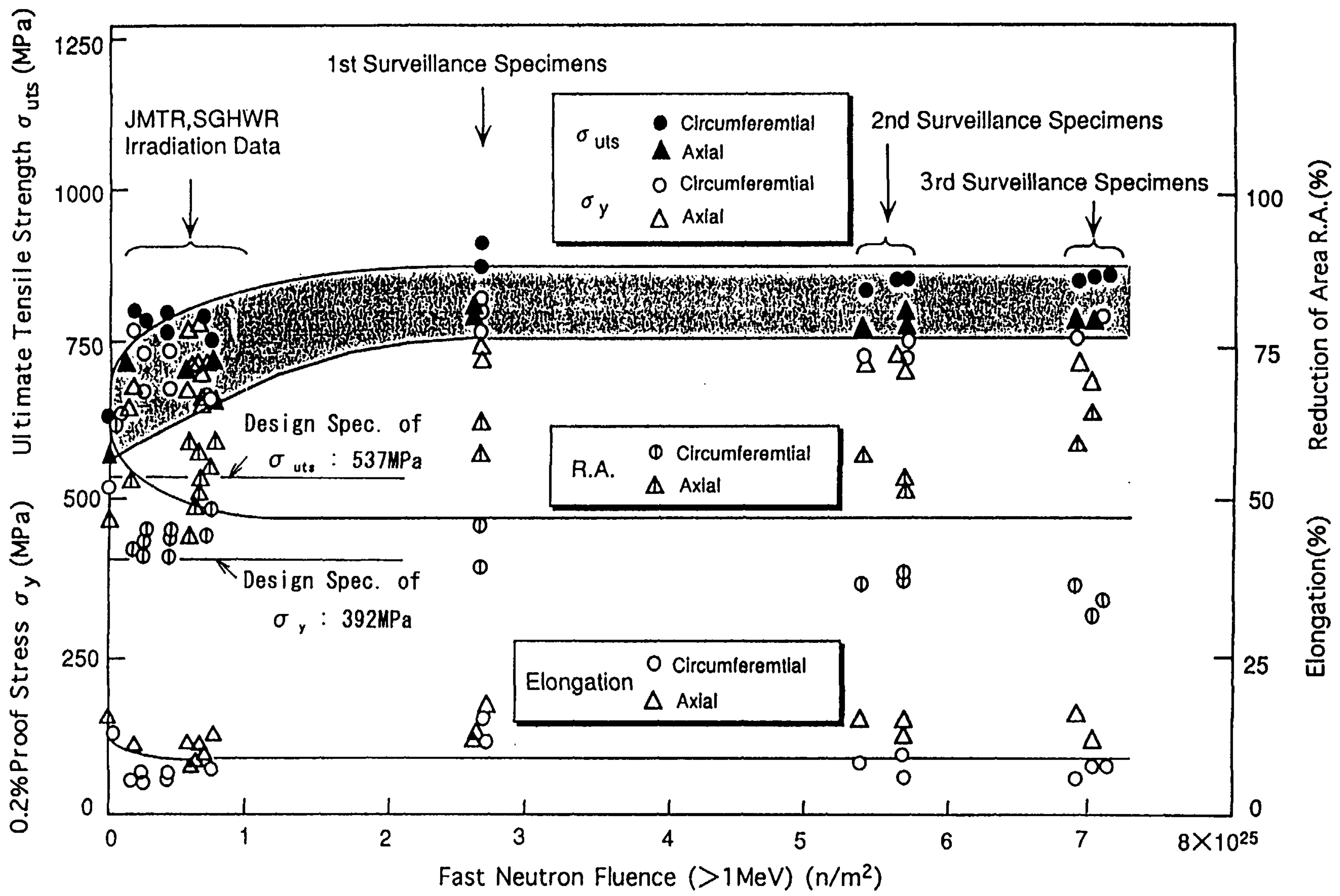

Fig.3 Tension Test Results of Pressure Tube Specimens at 300℃

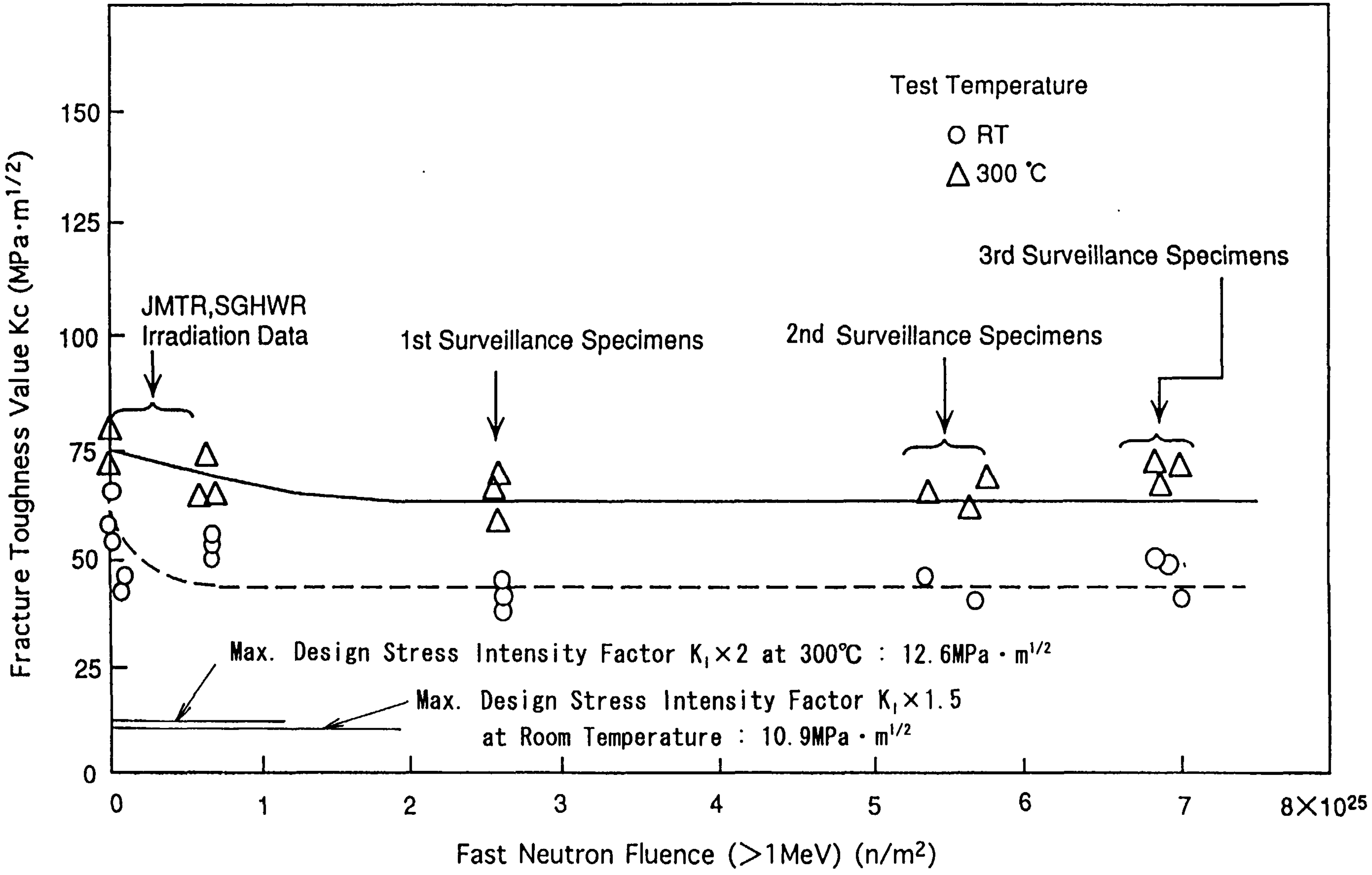

Fig.4 Bending Test Results of Pressure Tube Specimens

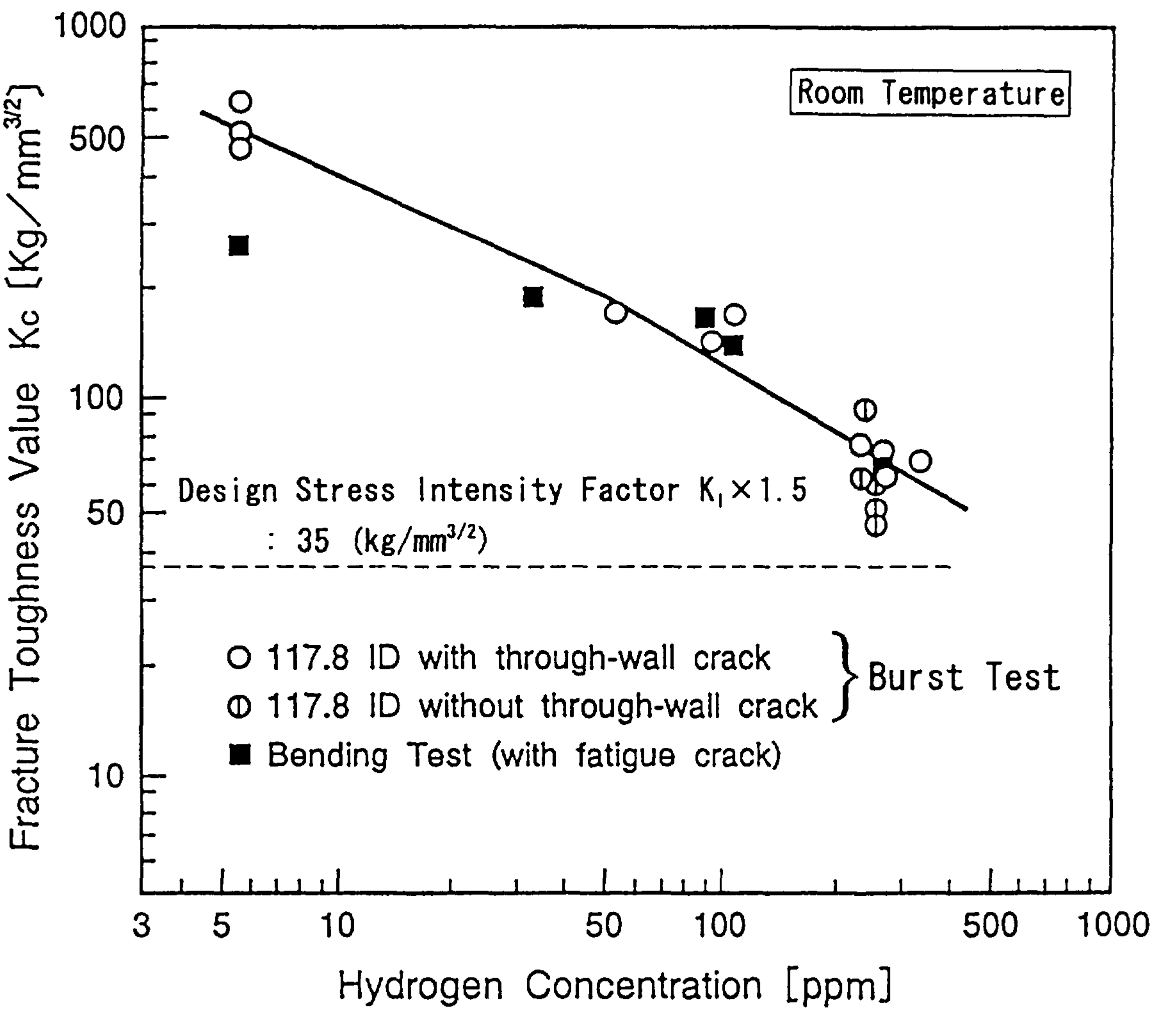

Fig.5 Fracture Toughness Values as a Function of Hydrogen Concentration.

Creep measured by pressure tube inspection equipment

Non-destructive inside diameter measurement of practical used pressure tubes is adopted in order to evaluate the creep property of Zr-2.5%Nb. The design criterion is provided for diametral creep strain. The maximum allowable creep strain is 2.5%, and it was determined lower than the initiation strain of tertiary creep written in some reports before the design stage of FUGEN. The results of the measurement are shown in Fig. 8, and the estimation line is based on the Ross-Ross's formula. Our data through the non-destructive measurement of practical used pressure tubes are corresponding very well with the estimation line.

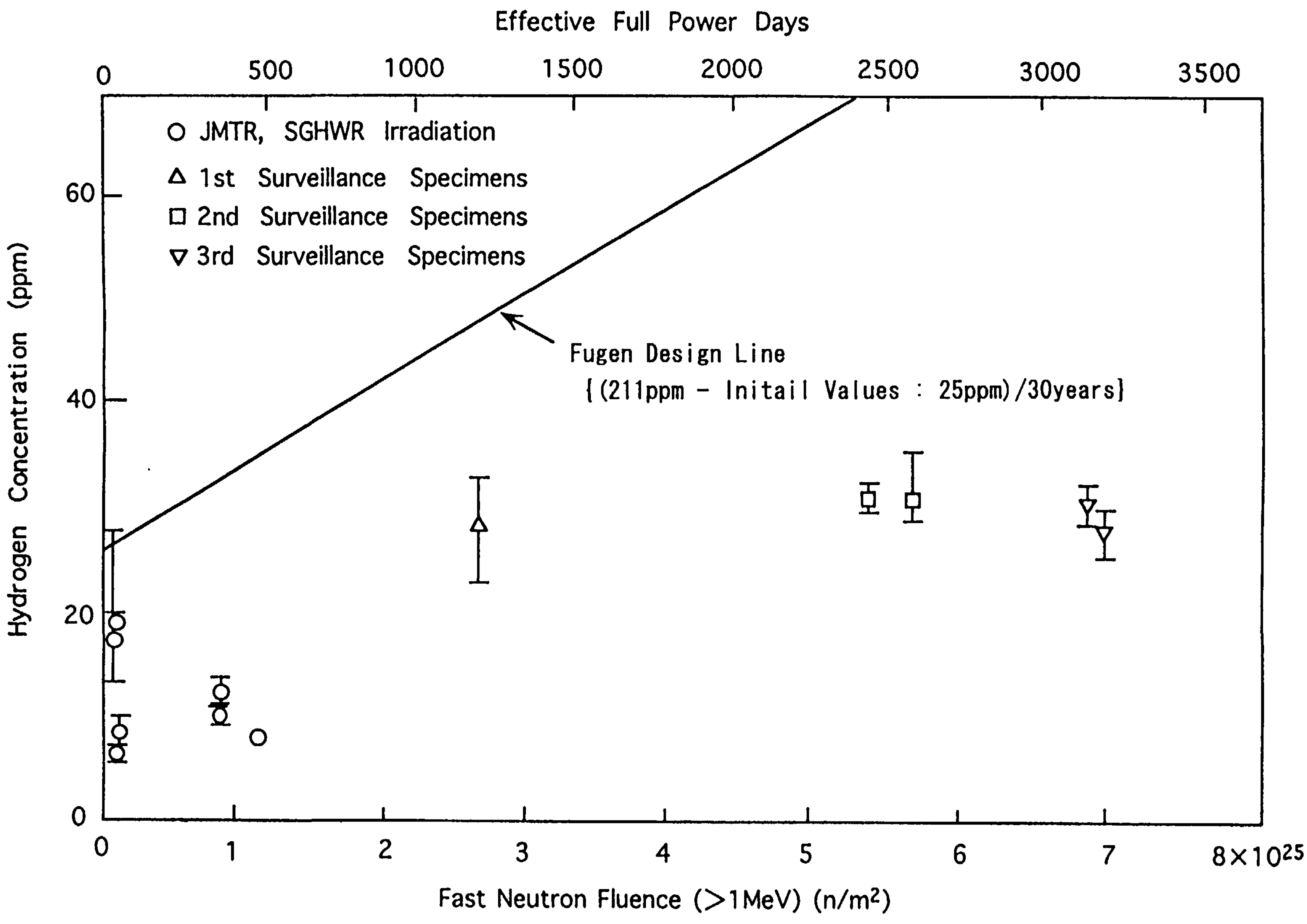

Fig.6 Hydrogen Analysis Results of Pressure Tube Specimens

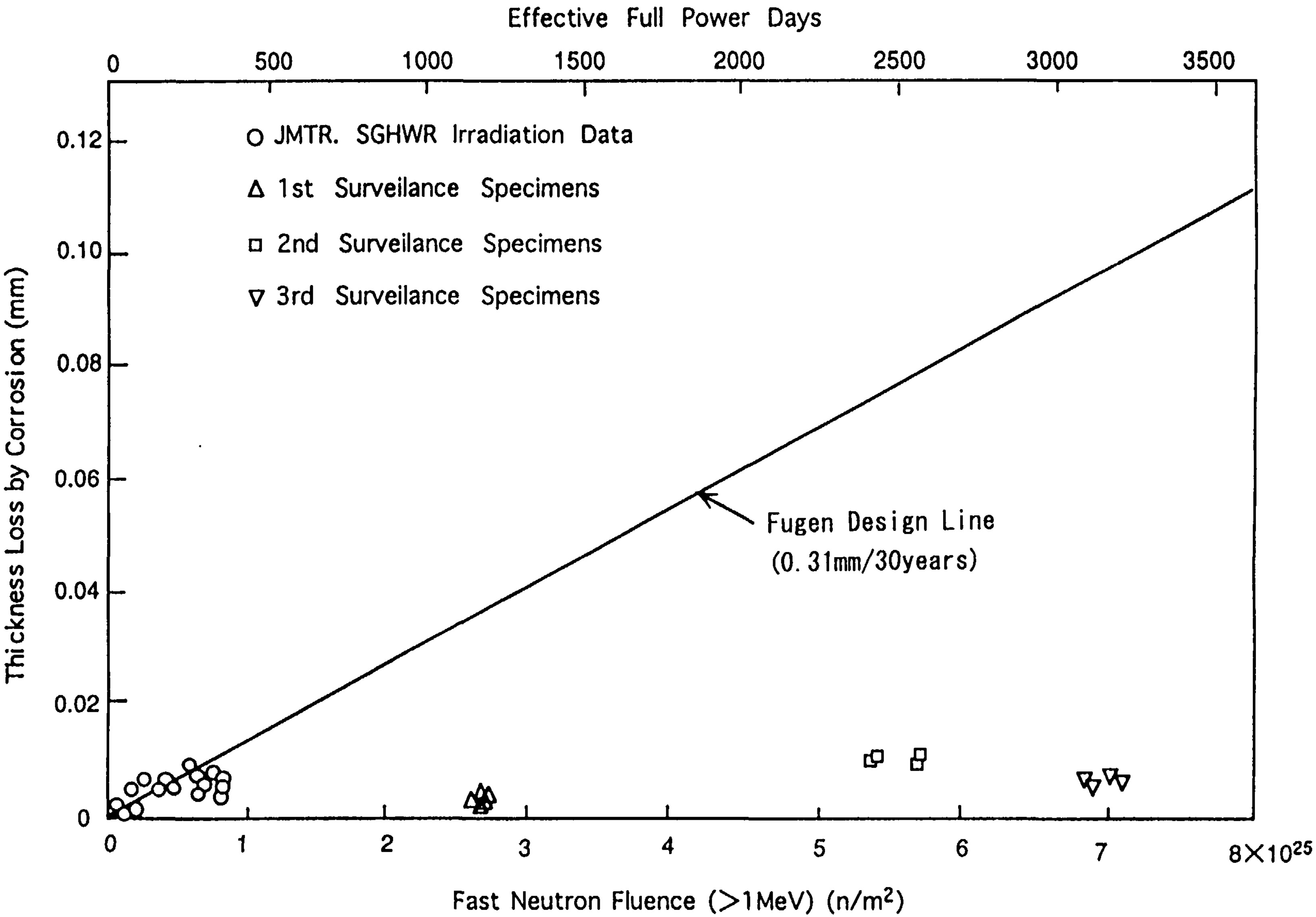

Fig.7 Corrosion Test Results of Pressure Tube Specimens

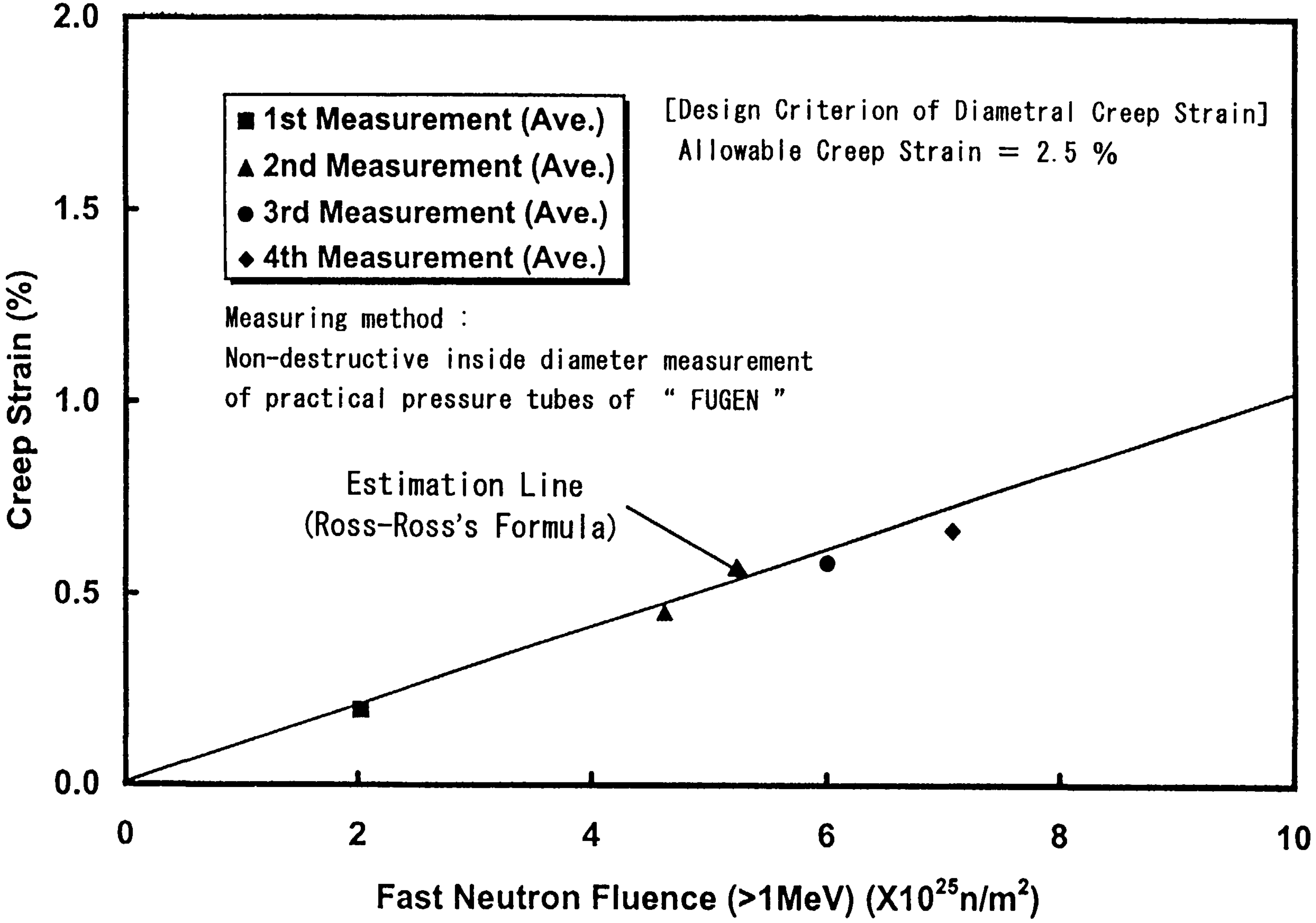

Fig.8 Results of Creep Strain Measurement

Conclusion

We have no indication that shows a possibility to exceed the limits of design criteria. We are going to continue the post-irradiation examinations and evaluation of the Zr-2.5%Nb pressure tubes of FUGEN.

RBMK FUEL CHANNEL AND WAYS TO INCREASE ITS LIFETIME

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Abstract

Design and main technological features to fabricate fuel channel components (steel-zirconium and zirconium-zirconium welded joints) are described and information on operating experience is given. Fuel channel fracture resistance characteristics (critical crack sizes at failure and start of hydrogen-induced delayed cracking) are presented. Measures on enhancing quality and improving quality control over fuel channel zirconium tubes taken to increase fuel channel lifetime are outlined.

RBMK FUEL CHANNEL DESIGN AND TECHNOLOGY

RBMK fuel channel is an important structural component related to reactor core, which operates under severe operating conditions Its central part manufactured of zirconium alloy with 2.5 % Nb, is exposed to high temperature (design temperature is 304 °C), pressure (around 8 MPa), as well as neutron flux (maximum neutron flux is $2 \cdot 10^{13}$ n/(cm^2·s), E ≥ 1 MeV)

"Steel-zirconium" transition joint (Fig. 1) is a special structural component of the fuel channel to join the zirconium alloy central tube with steel parts.

"Steel-zirconium" transition joint [1] is made as individual part by diffusion welding the steel and zirconium parts in vacuum.

Internal part is made of Zr-2.5 % Nb alloy, and external embracing part is made of austenitic stainless steel.

Surface relief of the parts to be welded looks like thread to increase contact surface and form mechanical coupling between the steel and zirconium.

Besides, stepped change in diameters of the parts assures gradual stress distribution along joint length

Thus, combined approach including mechanical and welded joint is implemented.

CURRENT FC TECHNOLOGY

Concerning such joints, all thermal and operating stresses are, in essence, taken up by mechanical coupling, and the welding operates as mainly sealing part to provide metallurgical bound between the steel and zirconium.

Solid phase steel zirconium welding process that assured high-quality joints satisfying entire set of requirements being imposed on, including 100 % objective quality control of <u>each</u> welded joint under conditions of extensive production, was used as the basis

Transition joint-to-zirconium pressure tube connection is carried out by electron-beam welding in vacuum. This process is related to transformation of jointing zirconium parts from solid phase to liquid one. After cooling the liquid phase provides considerable changes in metal structure of welding zone.

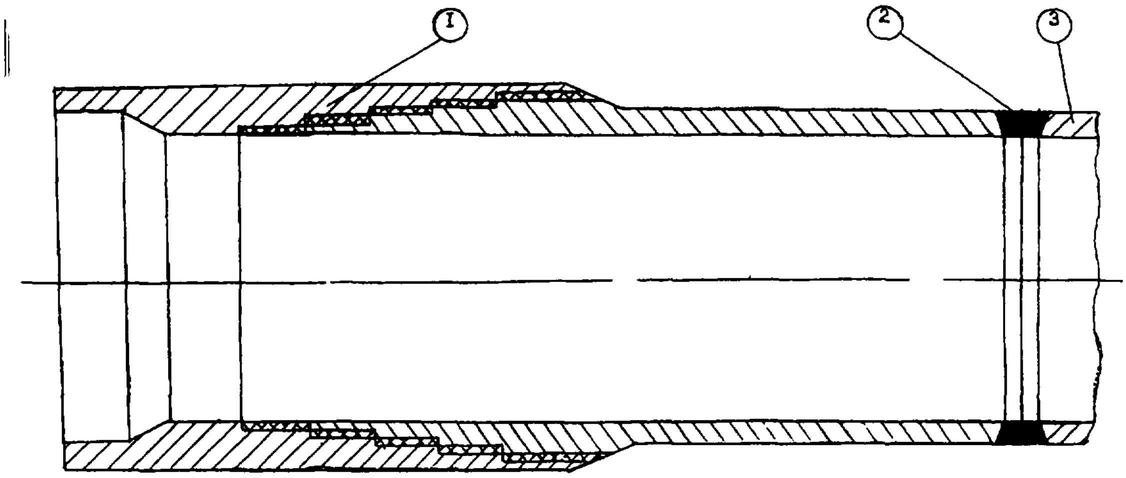

Fig.1. Scheme of connection of main fuel channel central parts

1 - Zr-2,5% Nb alloy - stainless steel 06X18H10T diffusion-bonded joint
2 - electron beam weldment
3 - Zr-2,5% Nb alloy pressure tube

Engineering approaches that allow to solve the problem have been developed, justified and implemented in the course of commercial production to ensure the required level of corrosive and mechanical properties of the welded joints.

Main of the above requirements {2, 3, 4] are

- use of filler material of composition differing tube metal to be welded;

- apply surface plastic working followed by thermal treatment of the welding zone.

As indicates the experience gained in operation of 23500 fuel channels with 47000 transitions joints and electron-beam welds used, no one fuel channel failure along Zr-Zr weld has been found for more than 200 reactor-years of operation, but the events of transition joints leakage that took places were related to the use of stainless steel with lower resistance to intergranular corrosion.

More severe requirements were imposed on the production process, in particular, titanium-to- carbon ratio in the steel applied was increased, the specimen preparation procedure for intergranular corrosion testing was changed on revealing the reasons of transition joints leakage.

The provisions taken allowed one to exclude, in essence, the use of poor steel and possibility of in-service leakage of the transitions joints.

IMPROVEMENT OF FC PERFORMANCE AND LIFETIME

The study recently performed, allowed to highlight the factors to regulate fuel channel service lifetime, including brittle fracture resistance of the metal as one of the most important factor.

Nowadays "leak before break" concept, which states the complete failure will not occur after part-wall flaw growth in channel and turning in through-wall flaw in the process of operation, if through-wall flaw length does not exceeded critical value [5].

Surface crack critical sizes for fuel channel tubes are presented in Fig. 2.

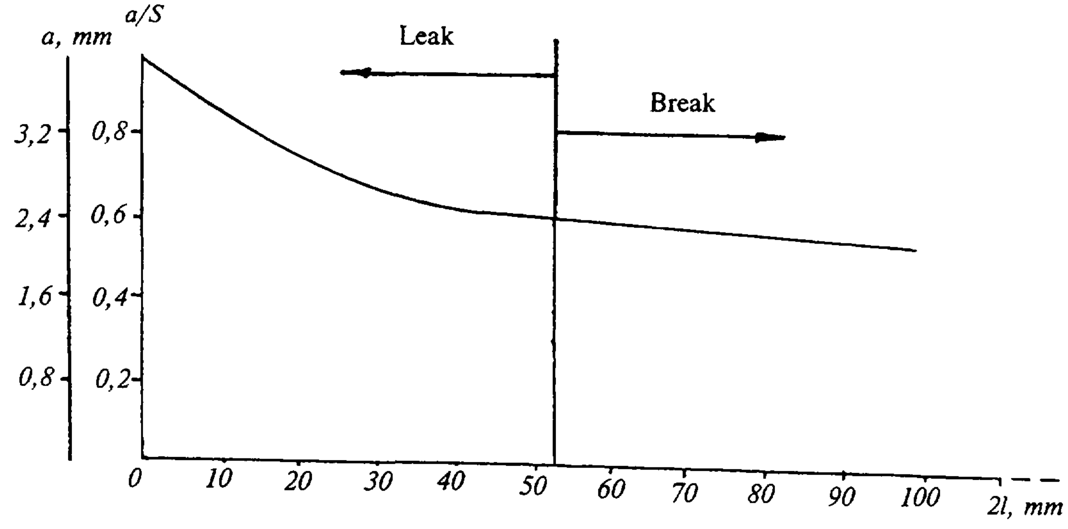

Fig.2. Surface flaw critical sizes in zirconium pressure tubes

2l - flaw lenght
a - flaw depth
s - tube wall thickness

As indicated, surface flaw critical depths are rather large even for large extent for 0.5 to 0.6 wall thickness Complete failure may occur with through-wall crack length exceeding 55 - 56 mm only

As evident from the analysis performed [6], leakage through such cracks may be easily detected, and the reactor shut down timely. Thus, "leak before break" criterion is satisfied

Crack propagation in the tubes may take place under mechanism of hydrogen-induced delayed cracking (HIDC). This problem has been studied thoroughly in a number of Russian papers [7], [8], [9].

Hydrogen pickup of tubes withdrawn from RBMK reactor after different operation period [10] is shown in Fig. 3.

Maximum hydrogen concentration is 55 - 60 ppm after 18.5 years of operation, that is close to limiting hydrogen solubility in zirconium at operating temperature. Excess in this value may result in metal fracture according to the mechanism of HIDC with flaws of specified sizes, and excess in threshold level of stress intensity coefficient.

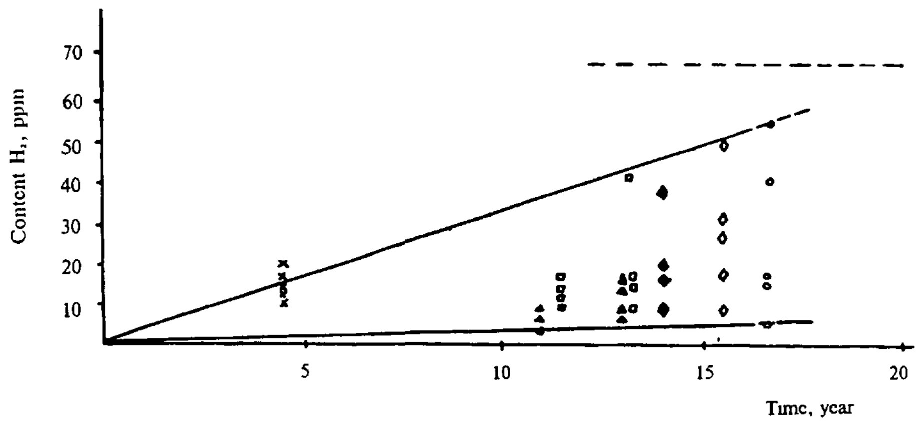

Fig.3 Hydrogen pick up in zirconium pressure tubes after some operation period

Geometry check

Surface quality examination

Mechanical property tests at 20 and 350 °C

Ultrasonic examination

Eddy current examination

Chemical analysis for determination of content H_2 and N

Metallographical examination of structure and inclusions at tube ends

Radiographical examination of structure homogeneity along tube length

Radiographical examination of residual stress along tube length

Fig.4. Types of fabrication examination for zirconium pressure tubes of fuel channel RBMK

As clear from estimates, extended surface flaw may be developed according to the mechanism HIDC with threshold value of K_{th} = 6MPa $\sqrt{M}$ under stress state caused by operating conditions and residual process stresses (90 and 100 MPa respectively), if flaw depth exceeds 224 μm.

Taking into account the above mentioned factors, the decisions were made to revise requirements to fuel channel tubes made by industry, in terms of decreasing allowable initial hydrogen concentration in tube metal, decreasing the size of allowable flaw detected by ultrasonic examination, limiting the level of residual process stresses on tube outer surface.

For the purpose of this, zirconium hydrogen pickup in the process of tube manufacturing was studied beginning from charge material to finished product. Changes in specifications have been made and hydrogen concentration limited within 5 ppm as a result of the study

Methodology was developed to increase sensitivity of flaw ultrasonic examination, resulting in limiting of the value of the flaw being detected within 100 μm.

Hoop residual process stresses were examined on pressure tube outer surfaces. Technique, hardware and software has been developed up to day to provide on-line non-destructive X-ray examination of hoop residual stresses ranging from minus 200 MPa to plus 250 MPa at error within (10 - 30) MPa. The technique and hardware allows one to detect (at up to 5 mm localisation) and diagnose discontinuities or stress concentrators above allowable level followed by revealing plastic deformation zones.

With respect to the fact, that all zirconium properties are susceptible to its structure, it is important to ensure and examination whether the structure is homogeneous or not along entire length of each tube in a batch.

A technique and hardware providing on-line non-destructive examination of metal structural state along entire 8 meter inner surface of zirconium tubes have been developed on the base of modern state-of-the-art of X-ray difractometry.

As resulted from the use of the technique, hardware and software, on-line examination of grain recovery degree, phase state and metal chemical compositions is possible under conditions of pressure tube extensive manufacturing. Currently, the examination unit is introduced into the system of product inspection to estimate the degree of structure homogeneity along entire tube length

Types of examination used currently in manufacturing zirconium pressure tubes for fuel channels are presented in Fig. 4.

Above measures tended to make the requirements to pressure tubes and introduction of additional inspections more stringent, allow one to increase fracture resistance of fuel channel metal and increase tube service lifetime, and will increase fuel channel serviceability up to 20 - 25 years.

In conjunction with the concept of interim replacing fuel channels taken in Russia, this may increase RBMK reactor service life up to 35 - 40 years.

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ASPECTS OF SAFETY ASSESSMENT OF RBMK-1500 FUEL CHANNELS

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Abstract

Factors influencing the residual life time of RBMK-1500 reactor are analyzed in the paper. The cases of cracking formation in transitional joints (zirconium alloy-stainless steel) and in pressure tubes are indicated Acceptable and critical dimensions of cracks are calculated by the methods of fracture mechanics, as well as crack growth dynamics under the conditions of corrosion environment.

Introduction

Functional features of the RBMK-1500 reactors of the Ignalina NPP depend on interaction between the graphite moderator and fuel channels (FC) of the reactor. The graphite stack and fuel channels of a RBMK type reactor are subjected to irregular temperature fields and neutron irradiation. This causes development of additional stresses and strains induced by changes of dimensions and conditions of materials used as the constructions experience temperature, radiation and mechanical loading.

One of the key components of the RBMK reactor is the assemblage of fuel channels. Their strength and durability substantially determine serviceability of the reactor as a whole. The main element of the fuel channel is the tube made from Zr-2.5 % Nb alloy. The wall thickness of this tube is 4 mm and outer diameter is 88 mm. In this tube a fuel assembly is placed cooled by water and water-steam mixture. The zirconium tube of the fuel channel is set into the central hole of the graphite block using graphite ring system. Each ring is tightly fit on a pressure tube or in a graphite block. Alloy Zr-2.5% Nb has good physical, chemical and mechanical properties as well as nuclear features. Specific distinctions of this alloy are: creep, hydrogen saturation, influence of irradiation on: creep, strength duration and resistance to brittle fracture. Essential dependence mechanical characteristics upon production procedures and techniques and subsequent thermal treatment is another salient feature. These peculiarities determine beforehand the necessity of detailed studies of strength of zirconium tubes of fuel channels.

As the creep process persists, the gap between graphite plates of the moderator diminishes This can affect heat transfer between graphite and surrounding water. Compression of tubes of fuel channels by graphite plates can bring about two problems: increase of load on welded joints and splitting of graphite plates. To avoid this condition replacement of fuel channel tubes is applied in practice. It is assumed, that the expansion of graphite takes place within duration of service life of fuel channel tubes and diminishing of the gap between graphite and fuel channel decelerates or ceases. Shrinkage of graphite blocks and of the whole graphite stack affects all dimensional changes Dimensions of the graphite blocks decrease more at their ends than at the centre. Temperature and flow gradients can lead to bending and crooking of whole column of plates. Operation mode of the reactor exerts great influence upon dynamics of gap closing At present time RBMK-1500 reactors operate below their maximum project power.

Factors influencing to aging process

Importance aims to be achieved are preservation of fuel channels in intact condition and provision of the yield sufficient for reliable fixation by available means. There are several mechanisms of aging of materials that determine noticeable influence on ability of fuel channels to execute their functions. Main aging mechanisms of graphite stack and fuel channels are /1/:

- thermal and radiation creep,
- radiation growth,
- radiation hardening and embrittlement,
- static and cyclic metal damage,
- hydrogen ingress,

corrosion.

The thermal and radiation creep and radiation growth are brought about by dimensional changes. Problems of creep and strength of zirconium tubes of fuel channels assume great importance as power of the RBMK reactors increases. It tends to increase intensity of high-velocity neutron flow, the magnitude of which substantially affects creep speed of zirconium alloys. Irradiation results in increase of creep speed, which depends also upon active stresses. Rise of temperature results in noticeable increase of the creep speed. An interesting feature is that the creep speed of zirconium alloys under radiation is greater than one outside the reactor. This means that evaluations of creep properties according to experimental results outside reactor will be conservative /2/ Radiation growth of zirconium alloys might occur in connection with considerable anisotropy of alloy's /3/. As it is known, in tubes, made of isotropic material and loaded by inner pressure, axial creep speed is equal to zero and therefore any extension of their length is not possible.

Under static and cyclic metal load defect sizes can increase. Presence of defects in the fuel channels while in operation influenced by cyclic changes may cause disintegration of the tube, as defect's grove and reach their critical values. Under periodic loading or from exposure to corrosion the process of disintegration consists of three stages. Initiation of a crack, stages of stable and unstable growth are described by various mathematical models. Operational experience shows, that zone of diffusion welding joining zirconium and steel pipes is dangerous one, where occurrence of growing defects may be expected /4,5/.

Corrosion process decrease wall thickness of fuel channel. This is important factor influencing longevity of fuel channels. On ground of analysis of corrosive action it was established, that transitional joints of fuel channels - diffusion-welded joints are least corrosive-resistant. As a result of condition inspection of RBMK-1000 fuel channels circumferential defects in transitional joints were located /4-6/. For all transitional joints examined it was established, that steel shows tendency to intergranular stress corrosion cracking (IGSCC). For the zirconium tube are characteristic also axial cracks, located from outside and growing due corrosion and DHC /4/

Radiation hardening and embritlement change the mechanical properties: ultimate strength and yield limits, fracture toughness and etc. Hydrogen ingress is also related with embrittlement and DHC. An important problem of the strength of fuel channels is to ensure their resistance to brittle fracture down /7/. It is common knowledge, that under influence of irradiation and hydrogen ingress the critical temperature of brittleness of zirconium alloys, determined by testing specimens for impact bending, shifts itself to the region of positive temperatures. In these circumstances it appears to be necessary to carry out, evaluation of the resistance to brittle fracture of zirconium alloy. This evaluation is carried out within the framework of fracture mechanics /8/. As conditions for plane deformation in thin-walled fuel channel tubes are not fulfilled, critical crack depth is used as a criterion of failure of tubes The hydrogen ingress lowers strength of tubes noticeable in low temperatures region.

Failure cases of fuel channels

During the operational process of RBMK type reactors several different modes of channels' failures are observed: disintegration of the zirconium tube in active zone, damaged transitional joints and cracks in the weld joining the casing of upper part of fuel channel. The rupture of zirconium tube in active zone region usually happens as a result of development of longitudinal crack by a mechanism DHC. An initiating technological defect is the necessary condition for development of a crack. Development of this crack is facilitated by increased contamination of the sections with release of second phase material, increased concentration of hydrogen and high level of residual stresses. Damage of lower transitional joints usually begins on the inner surface of the transitional joints on the boundary between steel and zirconium's alloy as presented by Fig 1 As the crack reaches steel part of transitional joints (steel 08X18H10T) it can grow by the mechanism of IGSCC

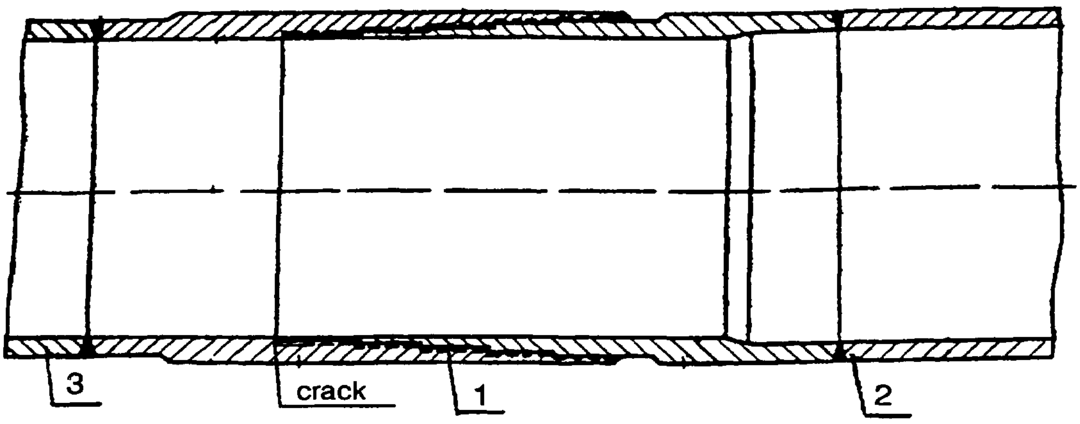


Fig. 1. Location of IGSC cracks in the transitional joint: 1 - diffusional weld joint, 2 - zirconium tube, 3 - steel tube (08X18N10T)

Defects in weld zone, where upper casing of fuel channel is joined, may be divided into three groups: defects in the weld, in metal below the weld and in the junction of slots, /9/ as shown in the Fig. 2. In the weld zone A defects have a pattern of circumferencial oriented cracks. They are caused by underweld's /10/. In the main body of metal (zone B) few separate circumferencial oriented cracks and numerous axially oriented cracks developing from inner surface, were observed. At transition to slot's point (zone C) circumferencial-oriented cracks, developing from outer surface, were observed

Fracture mechanics analysis

Presence of defects and action of the mechanisms of material aging of fuel channels becomes one of the main factors determining the service life of the reactor. The results of the corrosion investigation of the fuel channels after application in reactor evidence growing influence of corrosive condition and hydrogen ingress upon serviceability of fuel channels. Evaluation of ultimate sizes of fuel channel defects and analysis of sensitivity to parameters above mentioned is important for upgrading of safety of operation of the RBMK reactors. In this paper the analysis of admissible and critical sizes of defects, located in fuel channels and variously oriented, is performed. As starting data for calculations material characteristics shown in the table 1, and loading conditions, presented in the table 2, were used Two defects in the shape of cracks were examined In the first case the crack was situated in axial direction in the zirconium tube. In the second case the circumferencial-oriented crack was situated in transitional joint between the last step of zirconium's alloy and steel 08X18H10T and continued to grow in the steel part of the fuel channel (Fig.1)

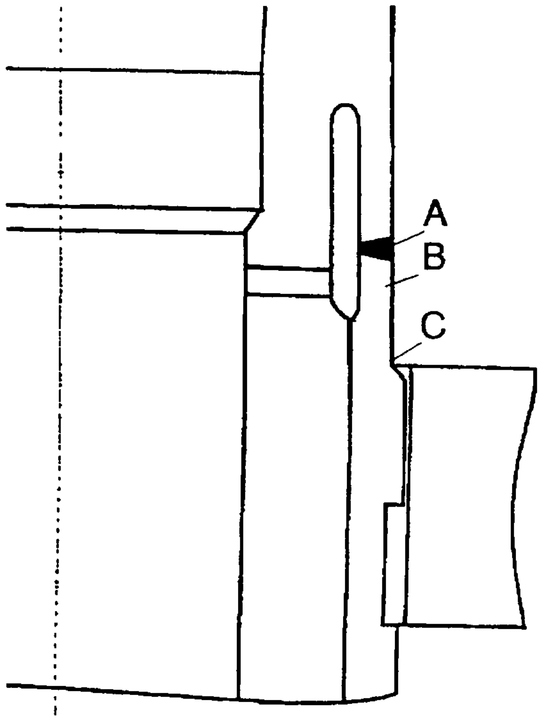


Fig. 2. Location of defects in weld region of the joint of the upper section of the fuel tube. A-in weld, B-in the main body of metal, C-at transition to slots point

Table 1

Calculated material characteristics /5,11,12/

Materials properties	Zr+2.5%Nb		08X18H10T	
	T=20 °C	T=300 °C	T=130 °C	T=300 °C
$\sigma_{u,t}$ MPa	630	510	461	412
$\sigma_{y,t}$ MPa	605	500	206	177
δ_c mm	0.007	0.007	-	-
K_{Ic} MPa$\sqrt{m}$	-	-	350	350

Table 2

Operational parameters /1/

Loading conditions	Temperature °C	Pressure, bar
Normal operation	288	82
Hydrostatic test	130	105

For evaluation of ultimate sizes of the crack in the first case the level 2 of the method PD6493 /13/ was used. By thus method ultimate condition of the crack may be represented by the curve FAD, shown in Fig, 3.

$$\sqrt{\delta_r} = S_r \left[\frac{8}{\pi^2} \ln \sec\left(\frac{\pi}{2} S_r \right) \right]^{-0.5} \qquad (1)$$

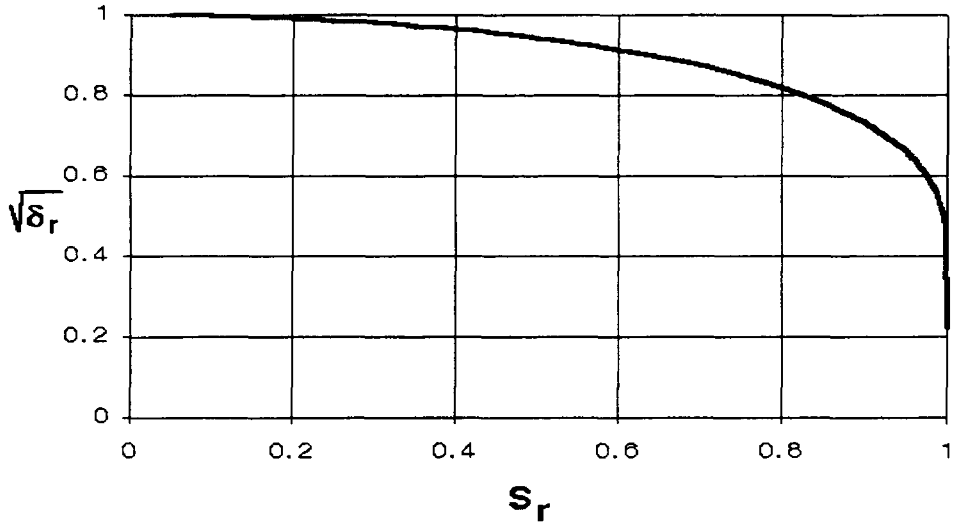


Fig. 3. Failure assessment diagram

In equation (1): $S_r = \dfrac{\sigma_n}{\sigma_{flow}}$ $\qquad \sigma_{flow} = \dfrac{\sigma_{ys} + \sigma_u}{2}$

$$\sqrt{\delta_r} = \sqrt{\frac{\delta_I}{\delta_{cr}}} + \rho\,, \qquad \delta_I' = \frac{K_I^2}{\sigma_{YS} E}$$

$$\begin{cases} \rho = \rho_1; & \dfrac{\sigma_n}{\sigma_{flow}} < 0.8 \\ \rho = 4\rho_1\left(1.05 - \dfrac{\sigma_n}{\sigma_{flow}}\right); & 0.8 < \dfrac{\sigma_n}{\sigma_{flow}} < 1.05 \\ \rho = 0; & \dfrac{\sigma_n}{\sigma_{flow}} > 1.05 \end{cases}$$

$$\begin{cases} \rho_1 = 0, & \chi \le 0 \\ \rho_1 = 0.1\chi^{0.714} - 0.007\chi^2 + 0.00003\chi^5, & 0 \le \chi \le 5.2 \\ \rho_1 = 0.25, & \chi > 5.2 \end{cases}$$

$$\chi = \frac{K_{Is}\sigma_n}{K_{Ip}\sigma_{ys}}\,, \qquad K_I = K_{Is} + K_{Ip}$$

there σ_n - is the effective primary net section stress, σ_{ys} - yield stress, K_{Ip}, K_{Is} - primare and secondary stress intensity factors.

he results obtained as critical sizes of cracks under conditions of hydraulic testing are presented in Fig. 4. As presented results indicate, the ultimate dimensions of the crack are noticeably influenced by residual stresses and the decrease of the wall thickness due corrosion.

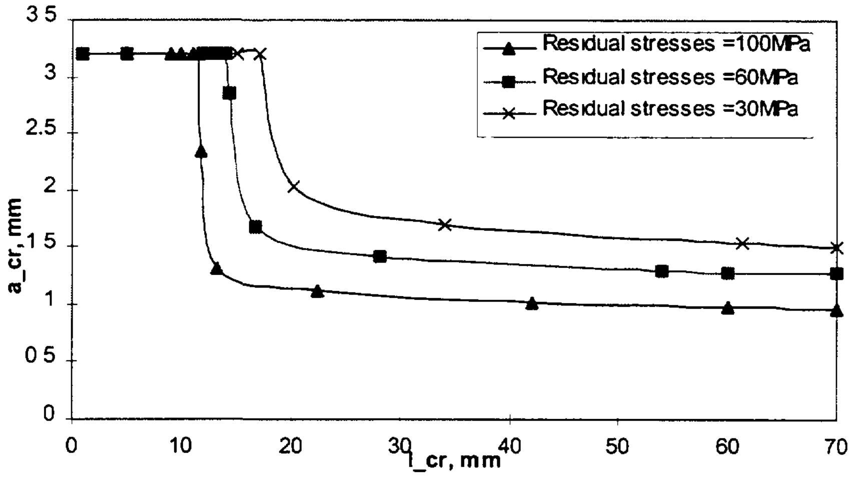


Fig. 4. Relation of critical dimensions of the axial outer crack, occuring in the central part of the fuel tube.

In the second case discussed the crack situated inside the transitional joint between zirconium and steel and having the circumferencial direction. Transitional joint of zirconium alloy and corrosion resistant steel is made using diffusion welding in vacuum techniques. Operating conditions afford reliable metallic bond of metals. The internal part of the transitional joint is made from zirconium alloy. The external part of the reducer joint is made from steel 08X18H10T. During the process of diffusion welding on the contact surface of parts to be joined a thin layer of diffusion products is formed. Quality of the diffusion welding is monitored by means of NDT and metallography. Joining of channel tubes and zirconium reducer part is performed by electron-ray welding. After welding the joints underwent to hardening and thermal treatment. Steel parts of reducers were joined with upper and lower parts of fuel channels by argon are welding. The external side is metallized with aluminum.

The analysis of the process was performed for evaluation of dynamics of the growth of the crack under influence of corrosion. Method R6 /13/ and the program SACC /14/ were used to determine allowable and critical sizes of defects. Ultimate sizes of cracks, corresponding to material mechanical characteristics according to the table 1 and loading conditions during hydraulic tests are presented in Fig. 5. Safety factors in this case were following :

- by primary loading -1;
- by secondary loading -1;
- by yield limit -1 5;
- by fracture toughness -3.16

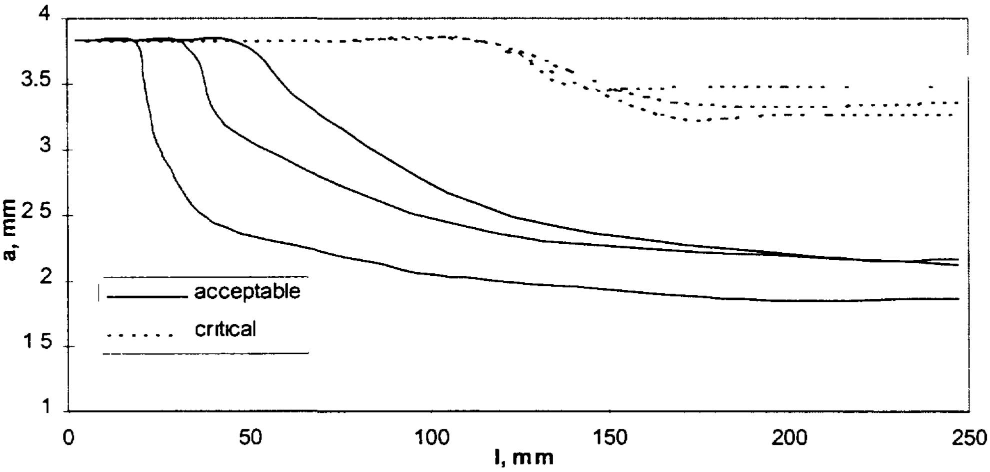


Fig. 5. Allowable and critical sizes of circumferencial surface crack depending on residual stresses (30 MPa, 60 MPa and 100 MPa correspondingly)

The analysis of a crack growing due IGSCC in stainless steel 08X18H10T was carried out using the crack growth equation:

$$\frac{da}{dt}=\left(\frac{K_I}{K_0}\right)^{\alpha}10^{-11}\quad m/s$$

there: a - dimension of the defect, t-time, α, K_0 - constants, depending on material and environment (for stainless steel characteristic α=2.161, K_0=6.02 Mpa$\sqrt{m}$). In Fig 6 dynamics of corrosive growth of a crack, depending on the magnitude of residual stresses is shown.

It's established, that the heaviest influence for growth of a crack is had by residual stresses and the residual stresses in weld seams after thermal processing up to size decrease $(0.15\div0.20)\sigma_y$. While in service channels under action of temperature and neutron flux occurs relaxation of residual stresses. For evaluation of the time interval, in which the surface crack reaches ultimate dimensions, analysis of the effect of residual stresses and initial dimensions of the crack was carried out, results of which are shown in Fig. 7. Dependence of service time when the mechanism IGSCC is active upon the parameter, proportional to surface area of the crack and residual stresses, is presented. Points represent results of calculations, and curves are envelops for all the cracks with equal residual stresses.

The evaluation of duration of the development of through crack to guillotine rupture presents certain interest from the stand point of operational safety of reactor.

Some time interval and some flow magnitude surpassing the threshold values of registering equipment are needed for detection of leakage of coolant through developed crack. The crack growth duration as a function of initial length and residual stresses is shown in Fig. 8. From this figure we notice that residual stresses as well as initial depth of the crack strongly influence the growth speed of the crack through the mechanism of inter-crystallitic corrosion. It should be mentioned that at low sizes of initial surface crack it grows in such a way, that in the moment of appearance on surface its length to width ratio approaches 2. It is clearly visible in Fig. 5. Therefore it is not dificult to determine the length of through crack.

For more detailed evaluation of the safe exploitation were provided a analysis of the dependence of fluid mass flow rate a through circumferential crack. For the analysis were used the computer code SQUIRT (Seepage Quantification of Upsets In Reactor Tube)/15/. The SQUIRT program was created by Battelle as an account of work sponsored by IPIRG and the USNRC. SQUIRT is a computer program that predict the leakage rate and area of crack opening for cracked

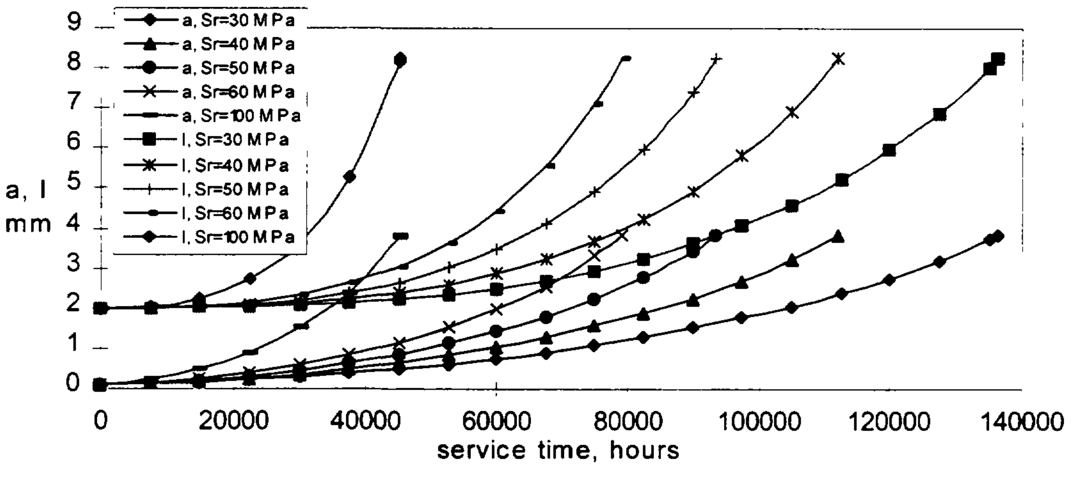


Fig. 6. Dependence of the increment of dimensions of the surface crack (a - depth, l - length, S_r - magnitude of residual stresses)

pipes in nuclear power plants. In all cases the fluid in the piping system is assumed to be water at either subcooled or saturated conditions. Development of leak-rate estimation methodology was initiated in response to IGSC cracking and fatigue in boiling water reactor piping. This program use the Henry-Fauske two-phase flow model with a friction factor. In the first step was established a minimal crack length for which flow could be established. It is assumed, that safety systems detecting leakage through wall cracks, should detect fluid volume flow rate 3.8 l/min. (or fluid mass flow rate 0.064 kg/sec). This fluid mass flow rate is achieved when crack length is 5.6 mm in lower transitional joint and crack gap is 0.22 mm (this gap is minimal for the program to calculate flow the through crack). Other used and calculated values are presented in table 3.

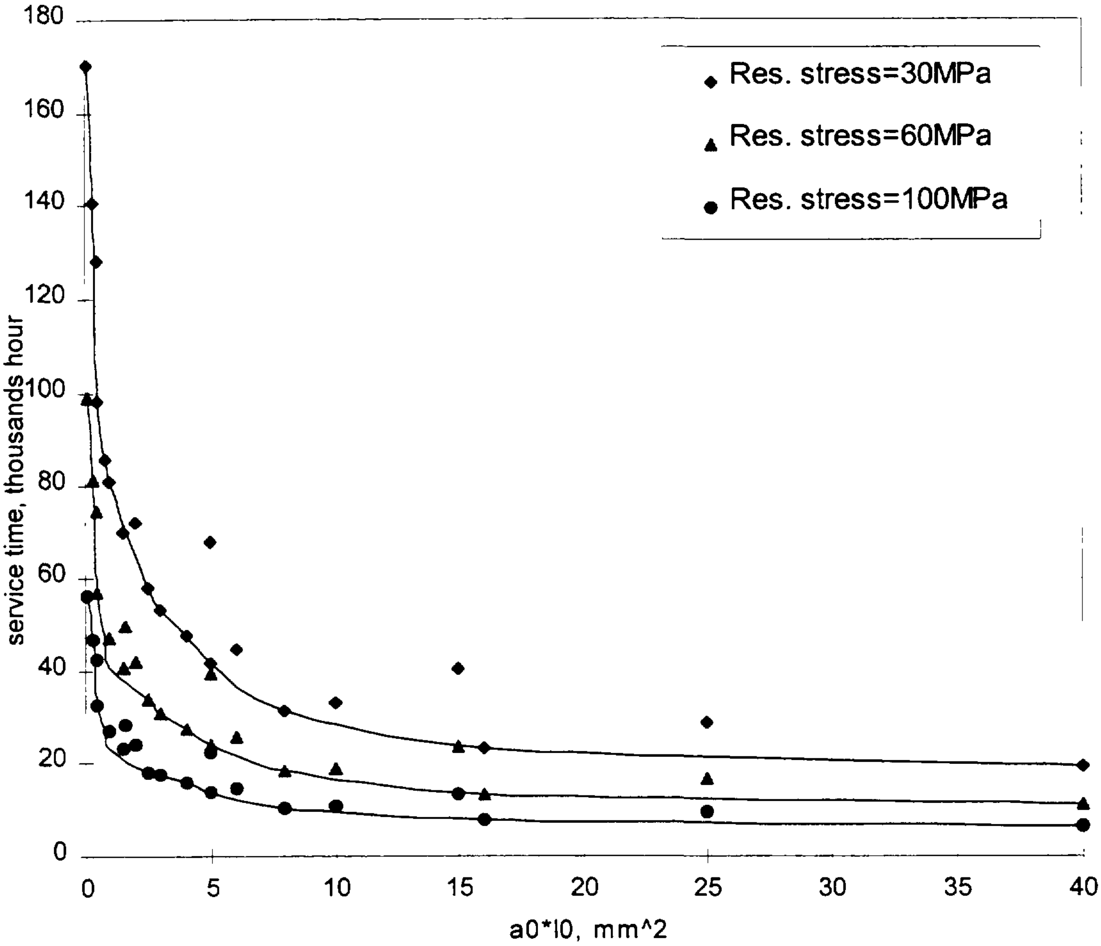


Fig. 7. Service time as a function of the crack surface area and residual stresses at acting mechanism IGSCC.

Table 3

Leak rate estimation using SQUIRT computer code

Location of the transitional joint	Fluid pressure, MPa	Fluid temperature, ^0C	Exterior crack length, mm	Exterior crack gap, mm	Exterior area of crack opening, mm^2	Fluid mass flow rate, kg/sec	Fluid volume flow rate, l/min
lower	7.33	264	5.6	0.22	0.97	0.0633	3.8
			25.14		4.34	0.281	16.9
			73.4		12.7	0.821	49.2
upper	6.73	284	6	0.22	1.04	0.0633	3.8
			25.14		4.34	0.263	15.8
			73.4		12.7	0.768	46.1

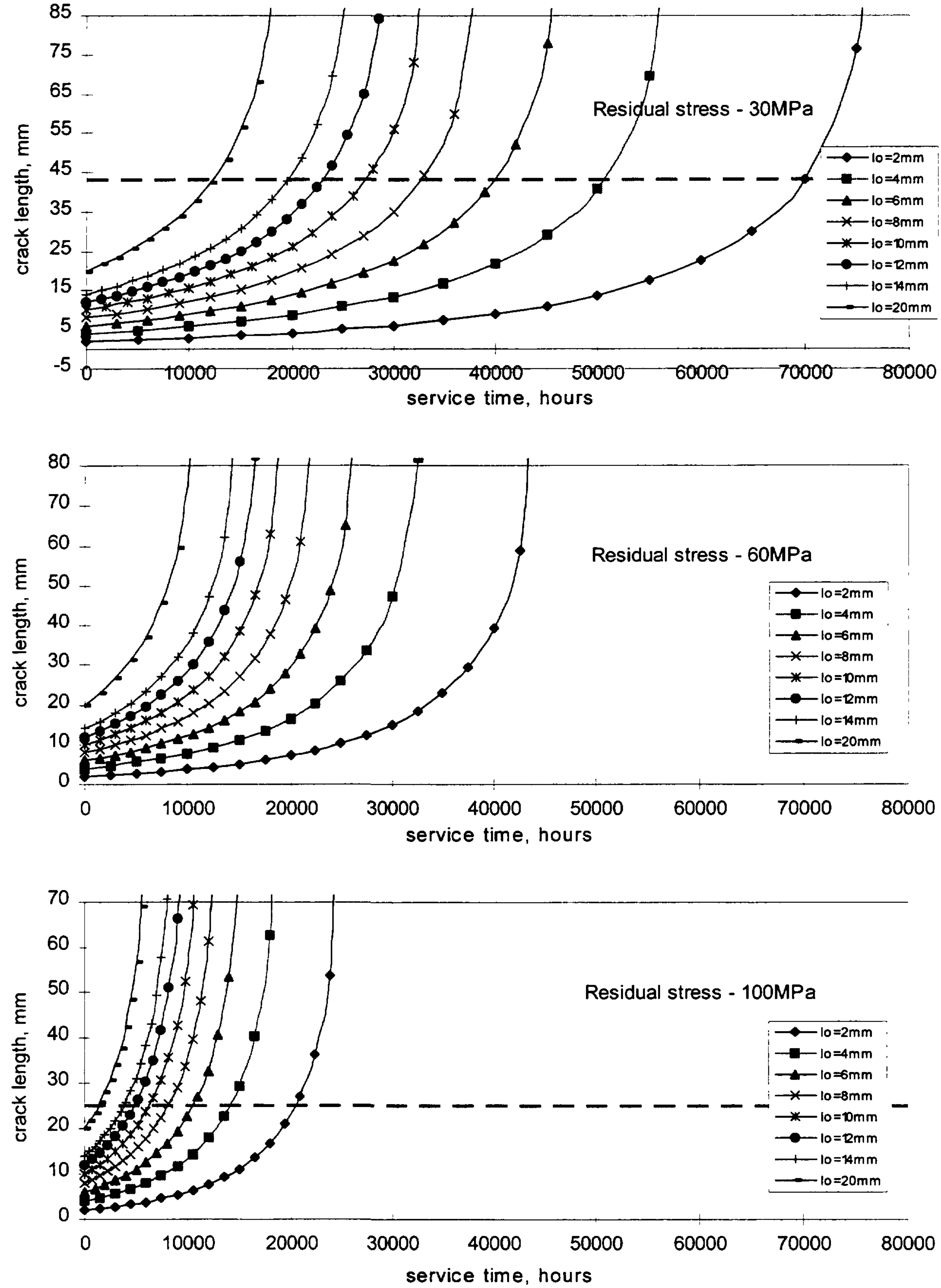
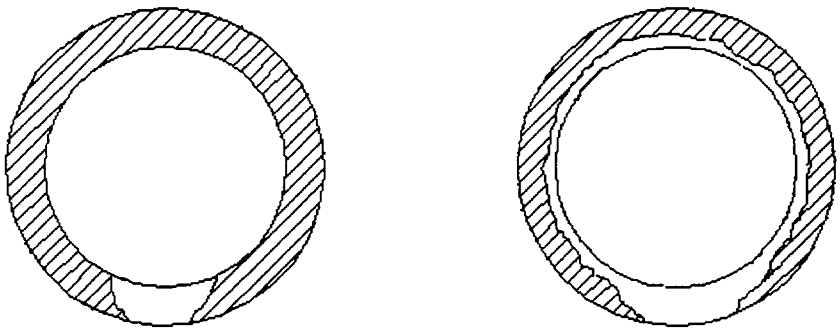


Fig. 8. Growth of a through wall crack due to IGSCC

The analysis of the acceptable and critical sizes the through cracks was carried out of the leakage rate for the most dangerous case, that is when residual stresses are 100 MPa and crack is in lower and upper of the transitional joint the fuel channel (Table 3). It should be seen from the obtained results (table 3 and Fig.8), that after establishing flow through crack, crack will grow up to acceptable length of crack (25.14 mm) during 1000 hours. Safety factor is equal to 10 for safety systems detecting leakage and should detect fluid volume flow rate will be 38 l/min. Because fluid flow rate is linear dependence from crack length, to this flow will be detect for 56 mm length through crack in the lower transition joint. In this case after establishing flow through crack, crack will grow up to critical length of crack (73.4 mm) during about 500 hours This time is acceptable for detect through crack to the critical crack length.

Presented relationships of development surface and through wall cracks are characteristic to IGSC. It is known, that stainless steels have become have propensity to IGSCC at some combination of a chemical structure of a metal, electrochemical characteristics of environment, loading conditions and another operational parameter. The reasons of occurrence IGSCC in transitional joints of FC is completely not investigated /4/. The submitted results of valuations of development of through wall cracks concern to cracks, close under the form to shown on Fig 9a The initial length of those cracks is not more as allowable length of a surface crack. This size makes 20-50 mm depending on size of residual stresses. If the lengths of a surface crack more, then a transfer on a through wall crack can have the form, shown on Fig.9b. Such opportunities of development of a corrosion crack considerably complicate the evaluation of duration of a crack growth up to the limiting sizes.



a) b)

Fig. 9.The hypothetical IGSC circumferential crack shapes in the transitional joint

Conclusions

In the present work some factors, influencing on residual resource of the RBMK-1500 of Ignalina NPP are considered. It is shown, that important parameters, limiting by a life time of reactors are material properties of FC as well as a rate of gas gap closing.

The cases of FC damage in a zone of a transitional joint and in zirconium pipe are indicated. By use the methods of fracture mechanics are calculated the limiting cracks' sizes.

Analysis of an influence of the initial crack's sizes and residual stresses on character of growth a circumferential crack on the IGSCC is executed. Results of analysis, enabling to predict a time of crack's growth up to the critical sizes are submitted. As presented results indicate the ultimate dimensions of the crack are noticeably influenced by residual stresses.

A valuation of leak-rate through circumferencial crack in a zone of a transitional joint is conducted. The controllable duration leakage before achievement of limiting lengths of a crack is appreciated.

It's marked, that the opportunity of development of cracks IGSC on all circumference of connection zirconium-steel can considerably complicate a technique of a prediction of operation time of FC defect.

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ASPECTS REGARDING THE LIFETIME OF A FUEL CHANNEL IN A CANDU NUCLEAR POWER PLANT

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Abstract

The paper presents the analysis of factors influencing upon the time life of a fuel channel of CANDU reactors built in Romania. Fuel channels are made of Zr-2.5%Nb alloy. Means and methodology to detect cracking of fuel channels are described, as well as improvements to increase life time of Chernavoda NPP fuel channels and national programme in this area.

1. INTRODUCTION

From 5 CANDU nuclear units which Romania started to construct, the first one has been in commercial operation since December 1996. For a good operation and maintenance of these units it is necessary to be known and prevented the defects which could occur. But to prevent the occurrence of such defects you have to know the causes.

One of the most important system of a CANDU power plant is fuel channel system. During the life time of a power plant an expected defect is the damage of the fuel channel.

2. SHORT DESCRIPTION OF FUEL CHANNEL ASSEMBLY

A fuel channel assembly consists of a pressure tube, two end fittings and associated hardware. The pressure tube is connected at each end by expanded joints to end fittings and is located inside the calandria tubes of the calandria assembly, so that annular gap between of these may be filled with a dry and inert gaseous atmosphere. In the annulus there are 4 spacers (called garter springs) which support and centralize the pressure tubes in the calandria tubes. The end fittings are supported on two bearings at each end of the fuel channel and are sealed by a channel closure which can be removed, stored and re-installed by the fuelling machine. In each end fitting there is a shield plug which provides the required shielding.

3. ELEMENTS INFLUENCING UPON THE LIFE TIME OF A FUEL CHANNEL

The most important component of a fuel channel is the pressure tube. In the same time with the building of CANDU higher power reactors, the pressure tubes have been improved in order to accommodate the more severe operating conditions. The properties of the pressure tubes change during service due to the high neutron flux, high pressure and temperature. In order to be compatible with new operating conditions Zircaloy 2 was changed with Zr-2.5%Nb.

There are a lot of main factors which have the potential to affect the service life of a pressure tube:

A - corrosion and change in a hydrogen concentration
B - dimension changes
C - changes in mechanical properties
D - integrity change

A. Corrosion and change in a hydrogen concentration

Zirconium oxide and deuterium are products of the chemical reaction between the pressure tube and heavy water The loss of metal from corrosion is structurally insignificant but some deuterium is absorbed by the tubes. If deuterium is present in the annulus between the pressure tube and calandria tube, deuterium can also enter the tubes from the outside surface of the tube if the oxide on the pressure tube loses its effectiveness as adequate barrier. When the hydrogen concentration is above the terminal solid solubility, delayed hydride present in the tube material could be cracked.

The cracking of the delayed hydride depends on the stress intensity for crack initiation, the rate of crack propagation and the temperatures at which cracking can occur. Increasing of deuterium concentration the possibility of delayed hydride cracking increases.

B. Dimension changes

Creep is defined as deformation due to stress and irradiation. Fast neutron irradiation induces the pressure tube zirconium alloys to creep in both the diametral and also axial directions. It also causes the pressure tube sag. Creep measurements show that axial creep rate is linearly proportional to fast neutron flux and stress. Creep sag is defined as the deflection of the tubes in the vertical plan due to a loss of stiffness caused by flux, stress and temperature.

Creep sag is limited by the number of channel annulus spacers and locations to prevent contact between pressure tubes and calandria tubes. Another limit for the creep sag is the passage of the fuel through pressure tubes.

C Changes in mechanical properties

Fast neutron irradiation increases the strength of cold-worked Zr-2.5%Nb and reduces its ductility.

Hydrogen in solution has very little effect on mechanical properties but when it precipitates as hydrides it can reduce the fracture toughness and the ductile-brittle transition temperature. The amount of the reduction depends on the number of hydrides and their orientation according to the stress direction.

D. Integrity change

The unforeseen mechanical effects (lap flaws, scratches) change the integrity influencing the life time of a tube Cold-worked Zr-2.5%Nb is susceptible to a failure mechanism caused by high stress and propagates during a thermal cycle by a solution of hydrides.

If a region of the pressure tube wall (which is hot) touches the calandria tube (which is cold) it gets cool. If the temperature is below the terminal solid solubility, hydride precipitation will occur and the thermal gradients will allow further hydrogen to migrate to the relatively cool zone and build up a mass of hydride (bleaster).

Sag of the pressure tube could result in contact with the calandria tube. The volumetric expansion blister and also the stress can make the blister fracture. Both the probability of blister fracture and also the depth of the crack in the blister increase with blister depth When a critical depth determined by the stress is reached the crack outside the blister is propagated. Propagation of a such crack in blister is dependent by the stress intensity developed by the acting stress (residual+service stress) and the crack size. The higher hydrogen concentration the larger blister size will be.

In order to prevent the blister formation is important to know.

- hydrogen concentration,
- position of the spacers to prevent the possibility of the contact between the pressure tube and calandria tube,
- sag size.

Warning of possible blister formation is best achieved by measuring the deuterium ingress of tubes that have high initial hydrogen concentrations

4. CRACKING CONDITIONS AND WARNING MEASURES

A pressure tube can be cracked under the following conditions

- presence of high residual and service stress,
- flaw on a pressure tube,
- high hydrogen concentration,
- presence of blister,
- a great sag which determines the contact between pressure tube and calandria tube.

The following measures has been taken in design in order to prevent the crack of the pressure tube

- replacement from Zircaloy-2 to Zr-2.5%Nb (the increase of deuterium from the oxidation reaction is less in Zr-2.5%Nb and hydrating is at an acceptable limits),
- limitation the hydrogen quantity at the tube's manufacture,
- zero-clearance rolled joints between the pressure tube and end fitting in order to decrease the residual stress,
- installation of 4 spacers instead of 2 between the pressure tube and calandria tube to decrease the possibility of contact between them,
- surveillance of the operating conditions of the fuel channel.

If the crack of a Zr-2.5%Nb pressure tube occurs, it can not reached at the critical length (when the crack is unstable) because as soon as the crack penetrates the pressure tube wall the leakage is detected by the gas annulus system.

Gas annulus system is sealed and has a gas which is circulated and monitored for moisture content. The system is sensitive to the ingress of small amounts of moisture and leaks can be detected by dew point measurements.

Techniques and equipment have been developed to measure the operating conditions of the channel

- eddy current inspection for defects, spacer location, oxide thickness,
- ultrasonic inspection for defects in the rolled joint region,
- channel elongation which is measured by specific instrumentation attached to the fuelling machines at each end of the channel,
- measurement of sag.

An equipment so called CIGAR (Channel Inspection and Gauging Apparatus for Reactors) has been designed and manufactured in Canada and uses ultrasonic to measure diameter, volumetric inspection in the body of the tube and the rolled joint, eddy current to determine the position of the spacers, inclinometer to measure the sag

5 IMPROVEMENT REFLECTIONS OF THE LIFE TIME INCREASE OF A FUEL CHANNEL AT CHERNAVODA NPP

The Romanian CANDU 600 NPP includes all up-to-date improvements. They are as follows:

- Pressure tube of Zr-2.5%Nb instead of Zircaloy-2
- Manufacture of pressure tube material with a hydrogen contain limited to 11 ppm by weight
- Zero-clearance rolled joints between the pressure tube and end fitting
- 4 spacers instead of 2 between pressure tube and calandria tube
- Tight spacers on pressure tube and gap between spacers and inside of calandria tube
- Checking of spacer location after installation by eddy current techniques
- Closed gas annulus and circulated gas
- Use of CO_2 only.

6. ROMANIAN PREOCCUPATION CONCERNING THE LIFE TIME OF FUEL CHANNEL

In Romania there are preoccupations to be well understood and learned the phenomena which determine the life time of a fuel channel.

Our Institute CENTER OF TECHNOLOGY AND ENGINEERING FOR NUCLEAR PROJECTS together with the Research Institute Pitesti are preoccupied by the following aspects

- Development of replacement technology of an activated fuel channel
- Design and manufacture of tools devices and equipment for fuel channel replacement
- Application of stipulations of the recent edition of the standard regarding the periodic inspection
- Development of methods equipment and procedures for in-service inspection of pressure tubes for knowledge of the following phenomena:
    - Evolution of manufacturing flaw and service induced flaws (e.g. lap flaws, heavy scratches, cracks)
    - Creep ductility/stress rupture (dimensional stability)
    - Blister formation
    - Material properties degradation
    - Delayed hydride cracking
    - Manufacturing/commissioning/service induced flaws
    - End fitting/pressure tube disbands.

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- COG reports concerning failure of pressure tube G16 from Pickering 2

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