IAEA-EBP-RBMK-05



# RBMK FUEL CHANNEL INTEGRITY

A PUBLICATION OF THE EXTRABUDGETARY PROGRAMME ON THE SAFETY OF WWER AND RBMK NUCLEAR POWER PLANTS

January 1999



INTERNATIONAL ATOMIC ENERGY AGENCY

The originating Section of this publication in the IAEA was:

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

RBMK FUEL CHANNEL INTEGRITY
IAEA, VIENNA, 1999
IAEA-EBP-RBMK-05
ISSN 1025-2754



Printed by the IAEA in Austria
January 1999

## FOREWORD

The IAEA initiated in 1990 a programme to assist the countries of central and eastern Europe and the former Soviet Union in evaluating the safety of their first generation WWER-440/230 nuclear power plants. The main objectives of the Programme were: to identify major design and operational safety issues; to establish international consensus on priorities for safety improvements; and to provide assistance in the review of the completeness and adequacy of safety improvement programmes.

The scope of the Programme was extended in 1992 to include RBMK, WWER-440/213 and WWER-1000 plants in operation and under construction. The Programme is complemented by national and regional technical co-operation projects.

The Programme is pursued by means of plant specific safety review missions to assess the adequacy of design and operational practices; Assessment of Safety Significant Events Team (ASSET) reviews of operational performance; reviews of plant design, including seismic safety studies; and topical meetings on generic safety issues. Other components are: follow-up safety missions to nuclear plants to check the status of implementation of IAEA recommendations; assessments of safety improvements implemented or proposed; peer reviews of safety studies, and training workshops. The IAEA is also maintaining a database on the technical safety issues identified for each plant and the status of implementation of safety improvements. An additional important element is the provision of assistance by the IAEA to strengthen regulatory authorities.

The Programme implementation depends on voluntary extrabudgetary contributions from IAEA Member States and on financial support from the IAEA Regular Budget and the Technical Co-operation Fund.

For the extrabudgetary part, a Steering Committee provides co-ordination and guidance to the IAEA on technical matters and serves as forum for exchange of information with the European Commission and with other international and financial organizations. The general scope and results of the Programme are reviewed at relevant Technical Co-operation and Advisory Group meetings.

The Programme, which takes into account the results of other relevant national, bilateral and multilateral activities, provides a forum to establish international consensus on the technical basis for upgrading the safety of WWER and RBMK nuclear power plants.

The IAEA further provides technical advice in the co-ordination structure established by the Group of 24 OECD countries through the European Commission to provide technical assistance on nuclear safety matters to the countries of central and eastern Europe and the former Soviet Union.

Results, recommendations and conclusions resulting from the IAEA Programme are intended only to assist national decision makers who have the sole responsibilities for the regulation and safe operation of their nuclear power plants. Moreover, they do not replace a comprehensive safety assessment which needs to be performed in the frame of the national licensing process.

## *EDITORIAL NOTE*

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## SUMMARY

The fuel channel integrity in the RBMK NPPs is an issue of high safety concern. To date, three single fuel channel ruptures have occurred. Fuel channel rupture results in release of radioactivity to the reactor cavity and may lead to a release of radioactivity to the environment if the confinement safety system does not function properly. A multiple fuel channel rupture exceeding the venting capacity of the reactor cavity overpressure protection system poses a major impact on the plant safety. Further, due to incorrect prediction at the design stage the gas gap between the fuel channel pressure tube and the graphite blocks closes after approximately 17 years of plant operation. There is no safety justification available for the continued plant operation in this condition and the reactors are being retubed to avoid operation in this out of design condition, which may have negative impact on the fuel channel integrity. The loss of the mechanical integrity of fuel channel pressure tubes is a major safety concern for RBMK reactors since it may lead to overpressurization of the reactor cavity and consequently develop into a severe accident.

In this report, information on the main design features of the RBMK reactor related to the fuel channel integrity is given. Further, detailed information on the fuel channel pressure tube and the graphite blocks with respect to their design, manufacture, in-service inspection, operating experience, ageing behaviour including degradation mechanisms is discussed in detail. The behaviour of the system fuel channel–graphite core including the corrective actions developed and implemented is discussed. Both normal operating conditions and accident conditions are addressed, considering also the gas gap closure process and its impact. The report also covers the fuel channel ducts.

It is concluded in the report that for RBMK-1000 reactors and the adopted retubing strategy, limited local gas gap closure occurs at the time of pressure tube replacement. The safety justification for short term operation in this condition has not been documented for normal operation or operational transients. There is, however, no straightforward indication that the impact of a small percentage of channels operating in a closed gas gaps condition for a "short" period of time will increase the risk of an unstable fuel channel pressure tube failure under normal operating conditions. The approach developed by the reactor designers appears currently to be the most useful available criteria for retubing. For normal operation and with the retubing approach adopted for RBMK-1000 reactors, no fuel channel material degradation mechanisms have been identified that could influence the fuel channel integrity. The related safety analyses are still to be completed including the development of applicable acceptance criteria.

Based on the information available during the preparation of the report, a set of detailed technical recommendations has been developed, addressing in particular aspects related to in-service inspection and safety assessment.

# 1. INTRODUCTION

The reviews performed within the framework of the IAEA's Technical Co-operation and Extrabudgetary programmes identified that fuel channel integrity in the RBMK NPPs is an issue of high safety concern. To date, three single fuel channel ruptures occurred due to water flow blockage or power flow imbalance at the operating RBMK plants. Fuel channel rupture results in release of radioactivity to the reactor cavity and may lead to a release of radioactivity to the environment if the confinement safety system does not function properly. A multiple fuel channel rupture exceeding the venting capacity of the reactor cavity overpressure protection system poses a major impact on the plant safety. Further, due to incorrect prediction at the design stage the gas gap between the fuel channel pressure tube and the graphite blocks closes after approximately 17 years of plant operation. There is no safety justification available for the continued plant operation in this condition and the reactors are being retubed to avoid operation in this out of design condition.

In order to assist Member States operating RBMK plants and at a request of the Government of Lithuania, a Workshop on RBMK Fuel Channel Integrity was organized by the IAEA in co-operation with the Lithuanian Nuclear Safety Inspectorate (VATESI) and the Lithuanian Energy Institute, in Kaunas, Lithuania, 13–17 May 1996. The objective of the workshop was to serve as a forum for exchange of experience on aspects and measures related to fuel channel integrity in channel type reactors and to provide basis and guidance for regulatory reviews in the countries operating RBMK reactors. The presentations and discussion during the workshop covered aspects related to fuel channel integrity such as Zr-Nb pressure tube itself, bottom and top stainless steel piping sections of the fuel channel, welds and graphite stack. The workshop was attended by 47 experts from Canada, France, Germany, Japan, Lithuania, Russia, Spain, Sweden, Ukraine, the UK and the USA. A draft report, summarizing the main aspects discussed during the workshop, was prepared.

With respect to the safety significance of the issue and in order to finalize the draft report, a consultants meeting on the subject was organized by the IAEA in Vienna, 16–20 June 1997. Following the presentations complementing those made during the workshop in 1996, the meeting focused on the finalization of this report. Sixteen experts from Canada, France, Germany, Lithuania, Russia, Sweden, the UK and the USA participated in the meeting.

Based on the information presented and the discussions held during the workshop and the consultants meeting, this report has been prepared. Copies of the presentations made both during the workshop and the consultants meeting are available upon request.

# 2. BACKGROUND

The RBMK reactor core consists of a graphite stack approximately 8 m high and 14 m in diameter weighing about 1700 tons. The graphite structure houses 2488 channels. The configuration of the core varies from reactor to reactor, however, a typical RBMK consists of 2044 channels for fuel rods, control rods and instruments surrounded by 444 channels at the periphery of the core which are filled with graphite to act as reflector (Fig. 1). Radial core support is provided by 156 water cooled support tubes situated at the outer periphery. To prevent graphite oxidation and to aid the conduction of heat from the graphite to the coolant, the core is protected by a helium–nitrogen gas blanket.

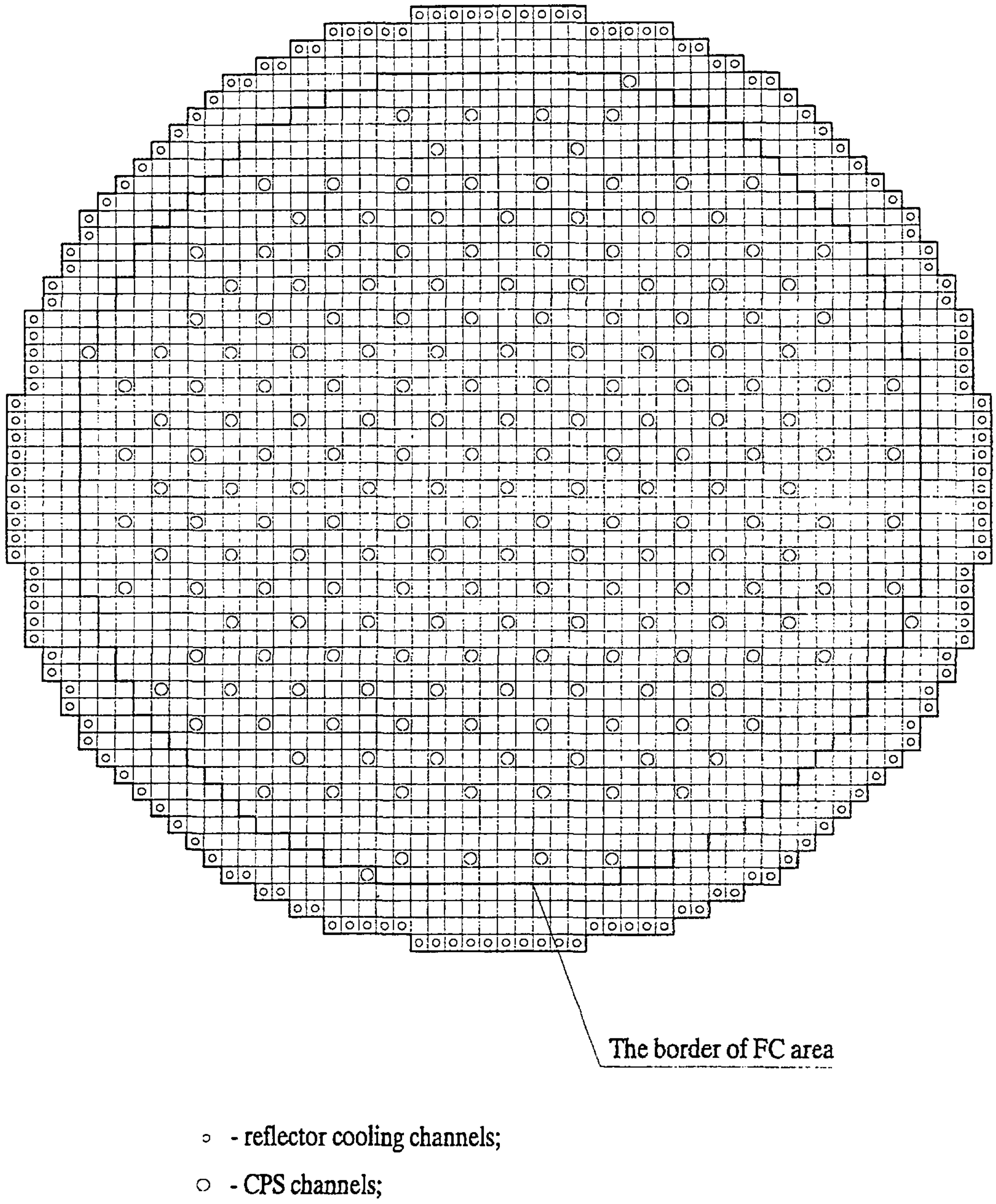


*FIG. 1. Graphite stack plan.*

It should be noted that this report deals only with the integrity of fuel channels. Even though there were problems observed with other channels, such as the control rod channels, during operation of RBMK plants, these are not addressed here.

Each individual graphite column, along with its associated internals, is referred to as a cell. A cell consist of a stack of graphite bricks joined together by a cup and cone arrangement through which a zirconium-2.5% niobium fuel channel pressure tube passes. The fuel channel consists of Zr-2.5% Nb pressure tube and stainless steel end pieces, joined by diffusion bonded transition joints. The stainless steel end pieces are attached to upper and lower ducts, which penetrate the upper and lower steel structures "OR" and "E". Thermal contact between the graphite bricks and Zr-2.5% Nb fuel channel pressure tube is provided by a system of slotted graphite rings, which alternatively make contact with the bore of the graphite brick or the fuel channel pressure tube (Fig. 2). In this arrangement, there is a dimensional allowance to accommodate diameter changes referred to as the gas gap. The slots in the graphite rings are aligned to provide a passage for the flow of the nitrogen helium gas which forms part of the fuel channel leakage monitoring system.

The steel–zirconium transition joints situated at the top of the core and the bottom of the core are made by a diffusion bonding process. The transition joint is then welded by a zirconium alloy to zirconium alloy weld and stainless steel to stainless steel weld either side of this joint, (Figs 2 and 3) to form the fuel channel.

The original design of the RBMK reactors did not foresee the closure of the gas gap between the fuel channel pressure tube and the graphite.

At the beginning of the 1980s, it was however found during regular in-service inspections (examination of a fuel cell by fuel channel tube and graphite hole diameter measurement after tube removal) at Leningrad Units 1 and 2 that the design predictions for fuel channel pressure tube and graphite dimensional change were not correct and that gas gap closure would occur.

The RBMK designer concluded that the plant operation with a large number of gas gaps closed is not allowable. Various corrective measures were considered and analysed including the replacement of channels that had closed gas gaps. This required direct measurement of the gas gap for which a device was developed, which worked well under laboratory conditions but unsatisfactorily in a reactor. Based on economy, safety and feasibility considerations, an approach of retubing the whole core was adopted. Considerations favoring full retubing included the radiation dose to the staff, need to completely cool down the core, and the possibility to combine the retubing with other reactor upgrading.

For the Leningrad Unit 1 retubing, a maximum of 10 tons withdrawal force was recorded, which became the criterion for removal force. This force equals approximately to 50% of the strength of the upper diffusion bonded joint. However, results obtained during retubing of Leningrad Unit 2, where an increased number of tubes was withdrawn without graphite rings (approx. 16%) and forces up to 15 tons were needed to pull several channels, indicated that the state of gas gap closure was undesirable. Therefore, an additional criterion was proposed which required that not more than 10% of the fuel channel tubes should be withdrawn without rings.

The criteria were then correlated with the total power production of the reactor. At the required level complete retubing is recommended.

It should be noted that this approach leads to short term operation with a small number of channels with closed gas gaps. This situation is considered acceptable by the designer.

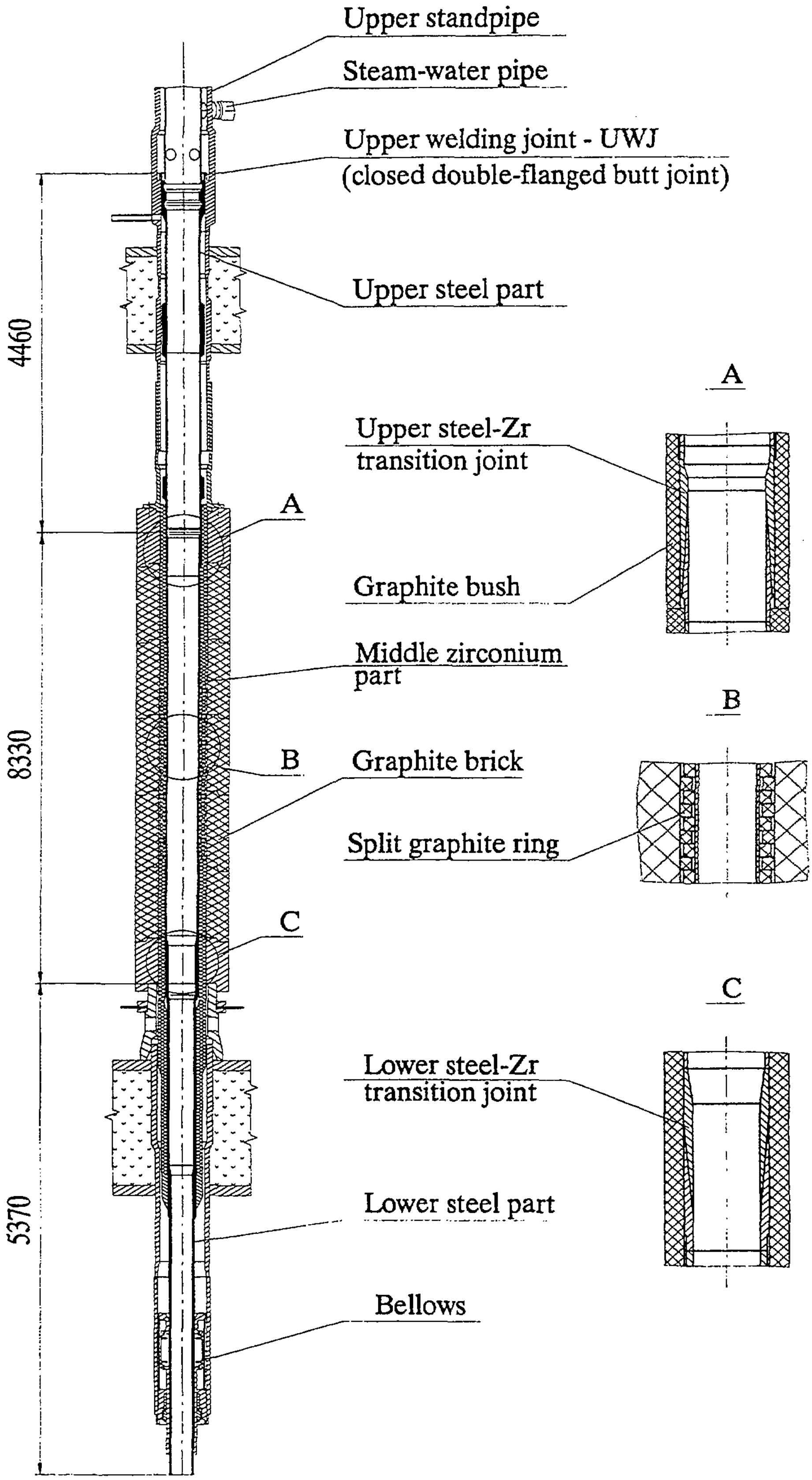


*FIG. 2. Fuel channel.*

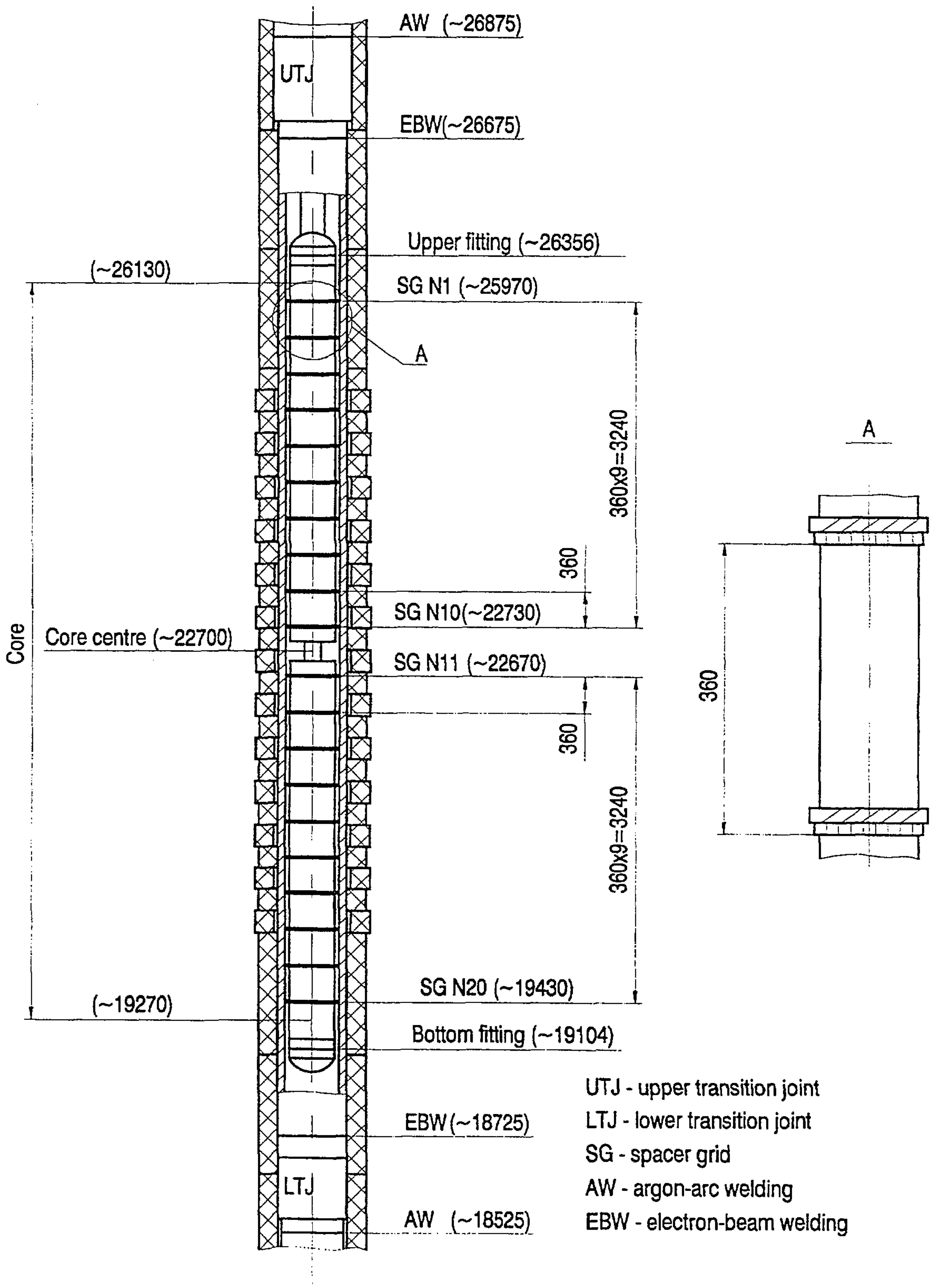


*FIG.3. Fuel channel and fuel assembly components.*

## 3. SAFETY CONCERN

The loss of the mechanical integrity of fuel channel pressure tubes is a major safety concern for RBMK reactors since it may lead to overpressurization of the reactor cavity and consequently develop into a severe accident.

The original design basis of the reactor cavity overpressure protection system was the postulated break of a single fuel channel pressure tube. The number of simultaneous fuel channel pressure tube ruptures that the system can cope with has been increased since the Chernobyl accident to three, in some plants to nine. In case of fuel channel pressure tube ruptures in excess of these numbers, which might be caused by multiple pressure tube rupture accidents (beyond the design basis), the upper core structures could be lifted causing control rod withdrawal and destruction of all the fuel channels.

Loss of mechanical integrity of channel tubes may be caused by excessive internal pressure at nominal tube temperatures, by excessive heat up under internal pressure loads, and due to thermomechanical interaction with fuel or graphite.

Mechanical integrity of pressure tubes might also be affected by the mechanical interaction of the channel tubes and the surrounding graphite structures. In order to avoid this interaction, the reactor was originally designed for operation with open gas gaps.

The accumulated experience have shown that operation of the reactor after about 15 years will lead to some channels having closed gas gaps. Continued operation will lead to an increased proportion of the channels in this out of design condition.

Operation with closed gas gaps would increase the probability of graphite failures under normal and accident conditions.

Long term operation of the reactor with global gas gap closure may:

- lead to uncontrollable interaction between graphite and pressure tubes;
- increase graphite stresses, cause graphite cracking and increased graphite temperatures;
- change the overall core geometry, induce bowing in both fuel channels and control and protection system (CPS) channels;
- jeopardize the insertability of control rods as a result of excessive CPS channel bowing;
- increase stresses at the upper pressure tube welds and transition welds;
- affect the probability of propagation of tube failure after single tube rupture;
- reduce sensitivity and reliability of fuel channel integrity monitoring system.

The prediction of gas gap closure of individual channels, which is based on calculations and measurements, is subject to significant uncertainties due to manufacturing tolerances, variability and uncertainties in material properties, operational data, statistics, scope and accuracy of measurements.

A safety related definition of the permissible extent of gas gap closure along an individual fuel channel, the allowable number of fuel channels experiencing gas gap closure and the allowable period of time for out of design operation is still lacking.

Further, calculation tools and material properties required to properly predict channel and core behaviour under closed gas gap conditions are currently not available. Therefore, the retubing approach adopted for RBMK-1000 reactors, which excludes long term operation with a large number of fuel channels with closed gas gaps, is based on energy production of the total core and does not consider local effects.

# 4. FUEL CHANNEL PRESSURE TUBE

## 4.1. DESIGN

The following are the main functions of the fuel channel considered at the design stage:

(1) to provide integrity of the coolant circuit under normal and abnormal operation within the design lifetime;

(2) to maintain the required conditions that ensure serviceability of fuel assemblies (FA) in terms of heat removal, charge ability and vibration loads;

(3) to remove heat from the graphite stack and ensure the graphite normal temperature does not exceed 750°C;

(4) to enable fuel channel replacement using standard engineering facilities and approved repair procedures.

RBMK fuel channels integrity under normal steady state and transient operating conditions is supported by the following 4 levels of protection:

(1) quality of design, material and manufacturing technology as well as pre-service inspection after manufacture and installation, and in-service inspection;

(2) elimination of the initiation of defect growth as a consequence of operational stresses and strains;

(3) low rate of subcritical growth of defects (cracks);

(4) assurance of leak before break[1] behaviour. This behaviour, justified by the characteristics of the delayed hydride cracking mechanism in Zr-2.5 Nb pressure tube alloy ensures that any crack progression will cause a leak upon through wall penetration before rupture will occur (see also Section 4.7).

## 4.2. MANUFACTURING

The processes used in the manufacture of fuel channel pressure tubes determine the quality of the product and the existence or development of defects.

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[1]The term "leak before break" used in this report describes a behaviour but shall not be confused with the leak before break concept as applied to large diameter high energy piping of PWRs and some other reactors in line with the respective requirements, e.g. USNRC SRP, Russian procedure M-LBB-01-93, IAEA-TECDOC-774. This applies in particular to the assumption, conditions, safety factors, margins and the possible use in the plant safety assessment.

Any processes that leave high residual stresses in the zirconium fuel channel pressure tube are not acceptable. A particular example from operating experience is tube straightening which must be well controlled to avoid defect initiation and growth. The use of polyurethane covered rollers in the straightening skew roll mill has led to the reduction of the residual stresses to less than 100 MPa. A 100% ultrasonic examination in tube manufacturing including intermediate operations (at the ingot and round forged billet stages) provides for a high quality of tubes.

### 4.3. IN-SERVICE INSPECTION

The parameters that are necessary to establish the condition of fuel channels and determine their integrity (availability of defects, properties and their degradation) are estimated by in-pile in-service inspection (ISI) and post irradiation studies in hot cells.

In-pile ISI is carried out in accordance with regulatory requirements. Separate regulatory requirements have been developed for RBMK-1000 and RBMK-1500 reactors.

In-pile ISI includes:

- ultrasonic examination of the pressure tube and its welds including stainless steel to Zr-2.5Nb transition joints;
- visual inspection by TSU-24M television monitor or RVP-489 periscope;
- wall thickness measurement (currently carried out at Ignalina NPP only).

At Ignalina NPP, equipment for ultrasonic examinations are used. However, it is reasonable to underline the necessity of development of new equipment for destructive and nondestructive testing of fuel channels.

The scope of in-pile examination and hot cell investigation in accordance with regulatory requirements is shown in Table I.

TABLE I. REQUIREMENTS FOR IN-PILE EXAMINATION AND HOT CELL INVESTIGATIONS OF RBMK-1000 FUEL CHANNELS

| No. | Investigation method* | Number of channels subject to ISI | | | |
|---|---|---|---|---|---|
| | | Stages** | | | |
| | | 1 | 2 | 3 | 4 |
| 1. | Ultrasonic examination | 50 | 10 | 30 | 30 |
| 2. | Visual inspection | 50 | 10 | 30 | 30 |
| 3. | Hot cell investigations*** | – | 1 | 1 | 2 |

*Fuel channel sizes examination methods are discussed in Section 6.2.3.
**No.1 (incoming inspection): during installation and full scale replacement;
No. 2: after 8000–10000 hours of fuel channel operation;
No. 3: every 30 000 hours from the beginning up to 15th years of operation
No. 4: every 8000–10000 hours after 15 years of operation up to fuel channel full-scale replacement or decommissioning of the reactor;
***The number of fuel channels to be cut out for examination may be increased depending on the results of the previous inspection.

In the case of detection of surface defects with sizes in excess of the permissible values, the scope of the ultrasonic examination is increased.

Besides the above nondestructive test methods, hydrostatic tests are used to evaluate the integrity of fuel channels. All of the fuel channels in a set are subject to hydrostatic tests at the fuel channel manufacturing stage.

During operation hydrostatic pressure tests are used after any primary circuit pressure boundary repair and also during any primary circuit ISI.

## 4.4. OPERATING EXPERIENCE

At the present time, more than 20 000 fuel channels are operating in RBMK reactors.

Seventy-three fuel channels were replaced because of leaks or suspected leaks during reactor operation. They were mostly results of deficiencies in the manufacturing process. These events took place in the time interval up to 10 years from the beginning of channel operation. After 10 years and up to 17.5 years of operation (maximum achieved time for fuel channel at Kursk Unit 1) there were no leakages from RBMK fuel channels. Special corrective measures were always implemented in the fuel channel pressure tube fabrication process after determination of the causes of defects.

Hot cell evaluation and measurement is required to determine the following information:

- the effect of service conditions on mechanical properties such as tensile strength, ductility, and fracture toughness;
- assessment of extent and trend of material loss by corrosion and evaluation of potential for localized material loss;
- measurement of hydrogen concentration (effect on fracture toughness and delayed hydride cracking) and mapping of hydrogen pickup along pressure tube;
- evaluation of microstructure changes and distribution of hydrogen in the tube wall, evaluation for any unknown features;
- confirmation of dimensional change.

More than 45 fuel channels have been investigated in hot cells. The general results of these investigations are as follows:

- from the point of view of fuel channel integrity there are no factors which would determine that the lifetime of pressure tubes has been reached at the present time;
- among the potential factors which could be life limiting for operation of fuel channel pressure tubes in the future, are the following:
- hydrogen ingress into the fuel channel pressure tubes which potentially could lead to hydrogen related failures;
- local fretting corrosion in the regions of fuel channel pressure tubes opposite spacer grids of the fuel assemblies, which could lead to unacceptable thinning of the pressure tube wall, local increases in hydrogen ingress and reorientation of precipitated hydrides.

## 4.5. MATERIAL PROPERTY CHANGES DURING OPERATION

The dimensions of the fuel channel pressure tube change because of irradiation creep and growth. This process can be considered to be linear with cell energy generation. The direction and magnitude of the changes depend on material parameters such as texture, microstructure and dislocation density. The diameter increases and the length may increase or decrease by a small amount. The pressure tubes have shown no significant bending. The significant effect is diameter expansion, which is discussed in following sections.

The properties of the fuel channel pressure tube change because of the reactor operational environment. Typically, the mechanical properties change as shown in Table II and the toughness of the tube, measured from crack opening displacement tests, change as shown in Fig. 4. In addition, corrosion of the inside surface produces an oxide film and part of the hydrogen generated by that reaction enters the tube. The hydrogen concentration, if it exceeds the terminal solid solubility at operating temperatures, can precipitate as a solid hydride phase and render the material susceptible to the delayed hydride cracking phenomenon. For this reason, the hydrogen concentration in RBMK is required to not exceed 70 ppm.

TABLE II. MECHANICAL PROPERTIES OF PRESSURE TUBE, LENINGRAD UNIT 1

| Time of operation (years) | Fluence (n/cm$^2$) for E>1 MeV | Mechanical properties | | | |
|---|---|---|---|---|---|
| | | Tensile strength (MPa) | Yield strength (MPa) | Elongation (%) | |
| | | | | Total | Uniform |
| 0 | 0 | 490–500 | 400–410 | 19–20 | 5.5–6 |
| 4.5 | $1.4 \times 10^{21}$ | 675–685 | 590–600 | 9.9–10 | 3.6–3.7 |
| 10 | $4.4 \times 10^{21}$ | 655–660 | 580–605 | 11–12 | 3.8–3.9 |
| 13.5 | $5.3 \times 10^{21}$ | 560–650 | 560–650 | 5.6–7.6 | 0.4–1 |
| | $5.5 \times 10^{21}$ | 585–610 | 560–590 | 4.8–5.8 | 0.4–1 |
| 15.5 | $5.6 \times 10^{21}$ | 585–610 | 560–590 | 6.7–7.9 | 0.6–0.8 |
| | $6.4 \times 10^{21}$ | 540–565 | 470–550 | 6.4–8.6 | 0.5–1.9 |
| | $6.7 \times 10^{21}$ | 550–610 | 535–610 | 6.9–8.0 | 0.5–1 |

The above properties, combined with the presence of a suitably sized defect can make the tube susceptible to cracking. However the parameters of the delayed hydride cracking process determine that cracks will not occur when defects and stress intensity are below a threshold size and value. If cracking does occur, the cracks can be detected at penetration and the reactor shutdown.

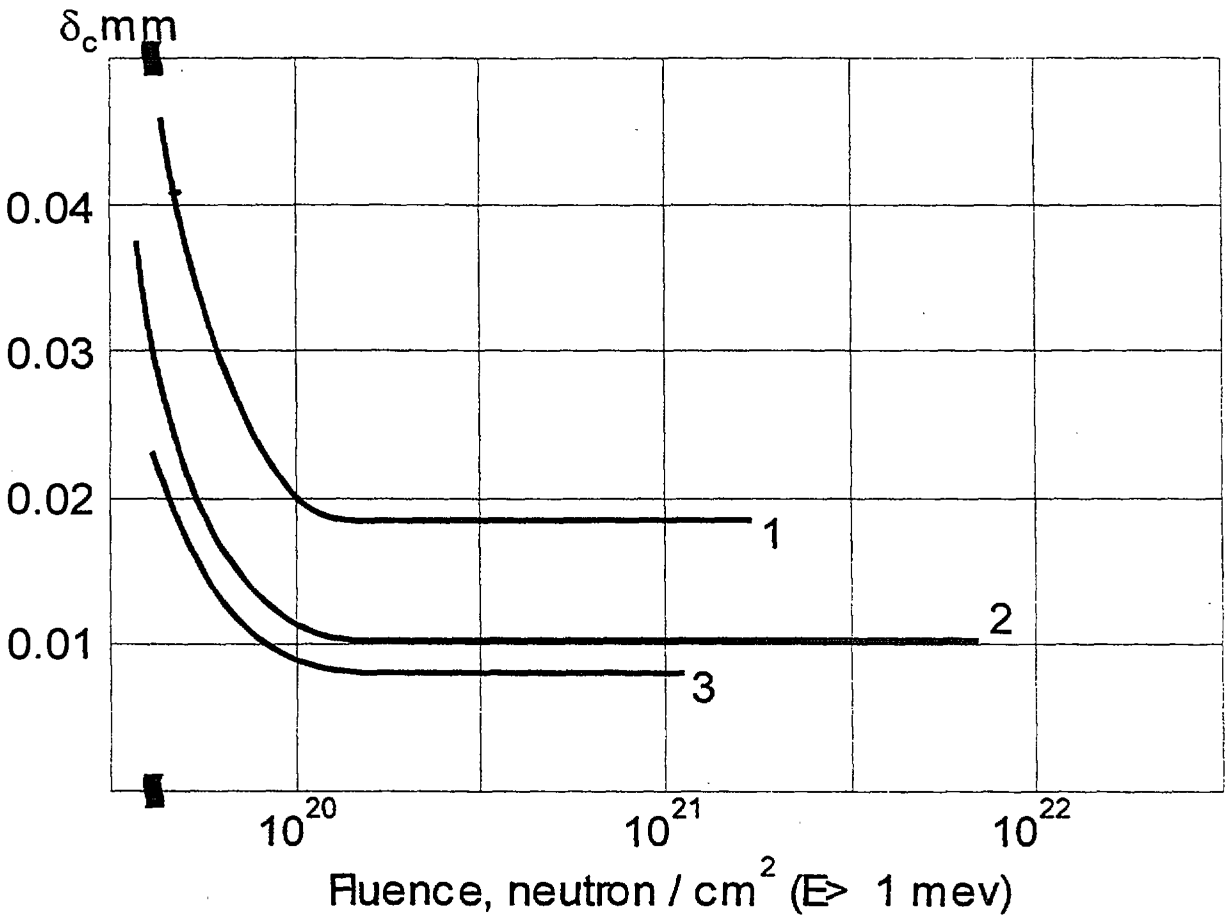


*FIG.4. The effect of neutron irradiation on the fracture toughness of RBMK pressure tubes in terms of $\delta_C$, T=300$^oC$, 1-TMO 2, 2-annealed, 3-TMO 1).*

## 4.6. LEAK BEFORE BREAK BEHAVIOUR

Leak before break occurs when a crack penetrates the wall and the resulting leak is detected with sufficient margin of time to prevent its growth to a size where the tube could rupture. The reactor is then cooled down, and the tube removed. Providing such failures are not frequent, safe operation is assured. If such occurrences become frequent, because of generic initiation of defects or if a crack reaches a critical length before a leak is detected because of material property changes, leak before break behaviour is not assured and tube rupture will occur. In that case, shutdown of the reactor must take place. The leak before break behaviour is ensured by the following aspects:

- the length of a surface crack as it progresses into a through-wall crack, or of a through-wall crack, is less than the critical crack length;

- the leakage through a subcritical crack is reliably detected by the leakage monitoring system;

– the time from the moment of leakage detection to reactor shutdown is less than the time for the crack progression to a critical size.

Consequently, assurance of leak before break behaviour is provided by the existence of satisfactory data on hydrogen concentration, defect quality and mechanical properties (including toughness) of the fuel channel tubes. Adequate values of these parameters are thus the main safety parameters that define the lifetime of fuel channels, and are discussed in the following section.

Accepting a strategy of RBMK-1000 fuel channel replacement after 15–20 years of operation, the most important factor determining safe operation of fuel channel is the capability of meeting leak before break behaviour. On this basis, the following degradation mechanisms will be of prime priority: hydrogen ingress, which potentially determines delayed hydride cracking; radiation hardening and embrittlement; and, finally, corrosion as the factor which determines general and local wall thinning. Currently hydrogen concentrations are measured on removed pressure tubes. The application of a scraping technique for removal of micro samples for analysis should be investigated because potentially a greater number of channels can be sampled.

## 4.7. DEGRADATION MECHANISMS

Degradation mechanisms and RBMK fuel channel functions are related as shown in Table III. All the degradation mechanisms identified are related to fuel channel integrity. Four degradation mechanisms are related to leak before break behaviour.

TABLE III. POTENTIAL RELATIONSHIP BETWEEN THE AGEING PROCESS AND FUEL CHANNEL FUNCTIONS AND PROPERTIES

| Ageing mechanisms | Factors affecting serviceability | Fuel channel functions | | | | | |
|---|---|---|---|---|---|---|---|
| | | Integrity | FA cooling | Leak before break behaviour | Graphite cooling | Maintenance of straightness | Replace-ability |
| Radiation hardening and embrittlement | Change of mechanical properties | + | – | + | – | – | – |
| Static and cyclic metal damage | Increase in surface defect sizes | + | – | + | – | – | – |
| Hydrogen ingress | Hydrogen embrittlement and DHC | + | – | + | – | – | – |
| Corrosion | Change of wall thickness | + | – | – | – | – | – |
| Radiation creep and growth | Dimensional changes | + | + | + | + | + | + |

Thermal and radiation creep are important degradation mechanisms, and are discussed in the following section of this report.

Corrosion does not influence leak before break behaviour of Zr-2.5Nb pressure tubes directly but the associated hydrogen ingress can indirectly influence leak before break behaviour through, increased risk of delayed hydride cracking.

Estimates of serviceability of channels are conducted according to the criterion of rupture resistance which includes an assessment of leak before break behaviour and leak prevention.

These estimates are based on the results of the following work:

– investigation of fuel channel specimens in hot cells to define mechanical properties, hydrogen content, size and distribution of hydrides, as well as the corrosion state;

– evaluation of critical parameters on the basis of investigation results mentioned above.

The degradation mechanisms in pressure tubes could be prioritized based on the effect on fuel channel functions (Table IV).

Currently, the longest achieved service life of an RBMK-1000 fuel channel tube is 18-19 years and there are no property degradation mechanisms observed in that time that can reduce this service life. As was demonstrated recently, the RBMK-1500 reactors fuel channel may operate for a longer time. Thus priorities of degradation mechanisms may need to be determined for NPPs with RBMK-1000 and RBMK-1500 reactors separately.

*RBMK-1500 fuel channels with TMO-1 tubes*

If fuel channel tube replacement is retained as an option the most important life determining factor is the gas gap within the fuel channel graphite stack system and consequently, radiation creep and growth of fuel channel tubes are the highest priority mechanisms. Of important priority are all the degradation mechanisms which contribute to leak before break behaviour assurance: radiation hardening (which determine changes in mechanical properties including reduction in fracture toughness), hydrogen ingress and the associated potential for delayed hydride cracking. General corrosion is an important deterioration mechanism acting on TMO-1 pressure tubes. At the upper and lower parts of tube (low flux regions) the corrosion rate is high. After 15 years of operation, the corrosion problem must be investigated. Wall thickness measurements performed during ISI over the length of the fuel channel tube have been initiated at Ignalina NPP.

*RBMK-1500 fuel channels with TMO-2 tubes*

Radiation creep and growth of TMO-2 fuel channel tubes at Ignalina Unit 2 results in a smaller change of diameter compared to TMO-1 tubes and thus closure of the gap within the fuel channel graphite stack system for this reactor may not occur. It is reasonable to require that the number of in-pile measurements to provide sufficient confirmation of this prediction should be increased (see Section 6.2.3). Therefore, prioritization of degradation mechanisms for this reactor's fuel channels is the same as for the fuel channels of RBMK-1000 reactor (see above) with life extension.

TABLE IV. PRIORITIZATION OF DEGRADATION MECHANISMS FOR RBMK FUEL CHANNELS

| Significance | RBMK-1000 reactor (second set of fuel channels after 20 years of operation) | RBMK-1500 reactor | |
|---|---|---|---|
| | | TMO-1 tube | TMO-2 tube |
| Most important | Leak before break assurance:<br>– $H_2$ ingress (DHC)<br>– irradiation hardening and embrittlement (fracture toughness, tensile properties)<br>– corrosion (wall thickness) | Irradiation creep and growth (gap between fuel channel and graphite stack)<br><br>Evaluation of corrosion | Leak before break assurance:<br><br>– $H_2$ ingress (DHC)<br>– Irradiation hardening and embrittlement (fracture toughness, tensile properties<br>– Evaluation of corrosion |
| Important | Irradiation creep and growth (geometry/size/changes) | Leak before break assurance:<br><br>– $H_2$ ingress (DHC)<br>– Irradiation hardening and embrittlement (fracture toughness, tensile properties) | Irradiation creep and growth (geometry size changes) |

TABLE V. PRIORITIES FOR FUEL CHANNEL TUBE PARAMETER MEASUREMENT

| Pressure tube type/ Significance | RBMK-1000 | RBMK-1500 reactor | |
|---|---|---|---|
| | | TMO-1 tube | TMO-2 tube |
| Most important | (1) ISI:<br>– US inspection<br>– wall thickness<br>– $H_2$ ingress (by scraping)<br>(2) PSE in hot cells:<br>– tensile properties<br>– fracture toughness<br>– $H_2$ content<br>– DHC | ISI: measurement of tubes and graphite stack geometry | (1) ISI:<br>– US inspection<br>– wall thickness<br>– $H_2$ ingress (by scraping)<br>(2) PSE in hot cells:<br>– tensile properties<br>– fracture toughness<br>– $H_2$ content<br>– DHC |
| Important | ISI: measurement of geometry changes (diameter, length, bowing) | (1) ISI:<br>– US inspection<br>– wall thickness<br>– $H_2$ ingress (by scraping)<br>(2) PSE in hot cells:<br>– tensile properties<br>– fracture toughness<br>– $H_2$ content<br>– DHC | ISI: measurement of geometry changes (diameter, length, bowing) |

Taking into account potentially longer operation of Ignalina Unit 2 fuel channel tubes, leak before break behaviour assurance becomes the most important factor determining fuel channel serviceability. Table V is summary of the priorities for fuel channel parameters measurement, derived from the priority of degradation mechanisms described in Table IV.

# 5. GRAPHITE CORE

## 5.1. DESIGN

The graphite core acts as the main moderator material as well as a structural component. Approximately 5% of the heat generation in a graphite moderated reactor is generated in the graphite due to gamma and neutron heating. This heat must be removed to the coolant. In an RBMK reactor, this is done via the graphite rings as described in Section 2.

The graphite core cannot be easily replaced and its life is therefore one of the life limiting features of an RBMK reactor.

In the original design it was envisaged that the gas gap between the graphite rings and pressure tube would not close throughout the plant life. However, experience has shown that the gas gap will close at between 15–22 years depending on reactor design and operation (including TMO-1 tube). In retubed reactors or at Ignalina Unit 2, it is possible that the gap may not close.

## 5.2. MANUFACTURING

The graphite bricks are manufactured from GR-280 graphite. This is an extruded needle coke graphite. This gives the graphite bricks anisotropic properties, i.e. the graphite material properties are different in the direction of extrusion (axial direction) from the properties perpendicular to extrusion (radial direction). The initial graphite GR-280 properties are given in Table VI. The size of a typical brick is 600 mm high by 250 mm × 250 mm square, with a central hole 114 mm diameter (see Fig. 5). The rings are manufactured from a better quality graphite GRP2-125 with a higher density.

TABLE VI. CODE REQUIREMENTS ON INITIAL PROPERTIES OF GRAPHITE GR-280

| Physical property | Graphite GR-280 grade averaged value from "Code for strength analysis ..." |
|---|---|
| Density, $g/cm^3$ | 1.71 |
| Compressive strength, MPa | 34/24* |
| Tensile strength, MPa | 7.6/6 |
| Compressive strain at failure, % | 1.6/2 |
| Tensile strain at failure, % | 0.2/0.2 |
| Young's modulus, GPa | 6.5/5 |
| Coefficient of thermal expansion, $10^{-6}.K^{-1}$ (at 400°C) | 4/5.4 |
| Thermal conductivity, W/m.K (at 650°C) | 48/37 |
| Electrical resistivity, $10^{-6}$ Ohm.m | 10/13 |

*Axial/radial direction.

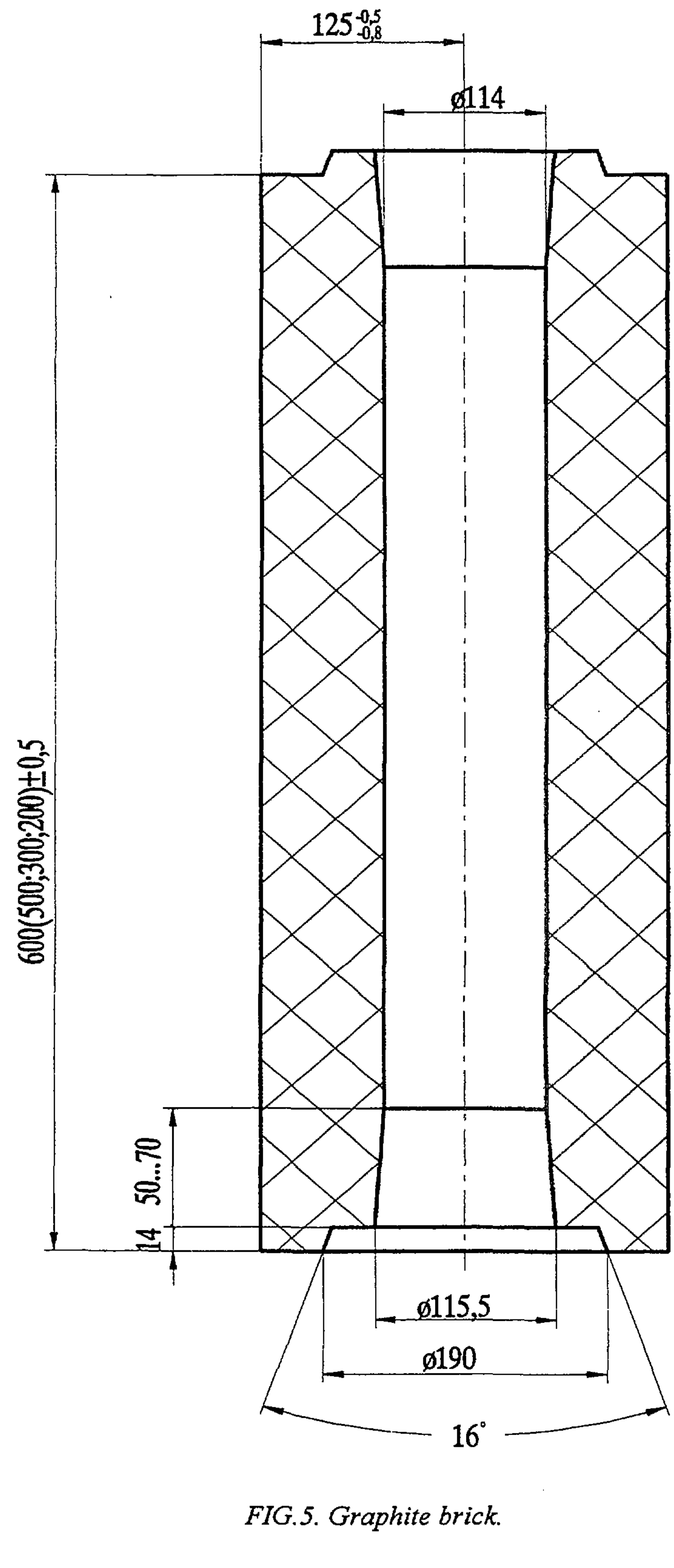


*FIG.5. Graphite brick.*

## 5.3. IRRADIATION BEHAVIOUR OF GRAPHITE

The dimensional and material properties change as a function of irradiation dose and irradiation temperature. These changes are summarized below. After 30 years of operation for the reactor RBMK-1000 the maximum fluence at the inside of a peak rated brick is expected to be approximately $1.9 \times 10^{22}$ n/cm$^2$, E >0.18 MeV. Typical temperature range for a RBMK-1000 core is 500–650°C (750°C maximum under normal operating conditions). The proportion of helium in the nitrogen helium cover gas is a factor determining the graphite operating temperature.

### 5.3.1. Dimensional changes

After an initial small expansion the graphite shrinks as a function of irradiation dose and irradiation temperature before turning around and expanding again. Eventually the original volume is reached and at this point the graphite starts to degrade (Figs 6 and 7). This is known as the critical fluence as described later.

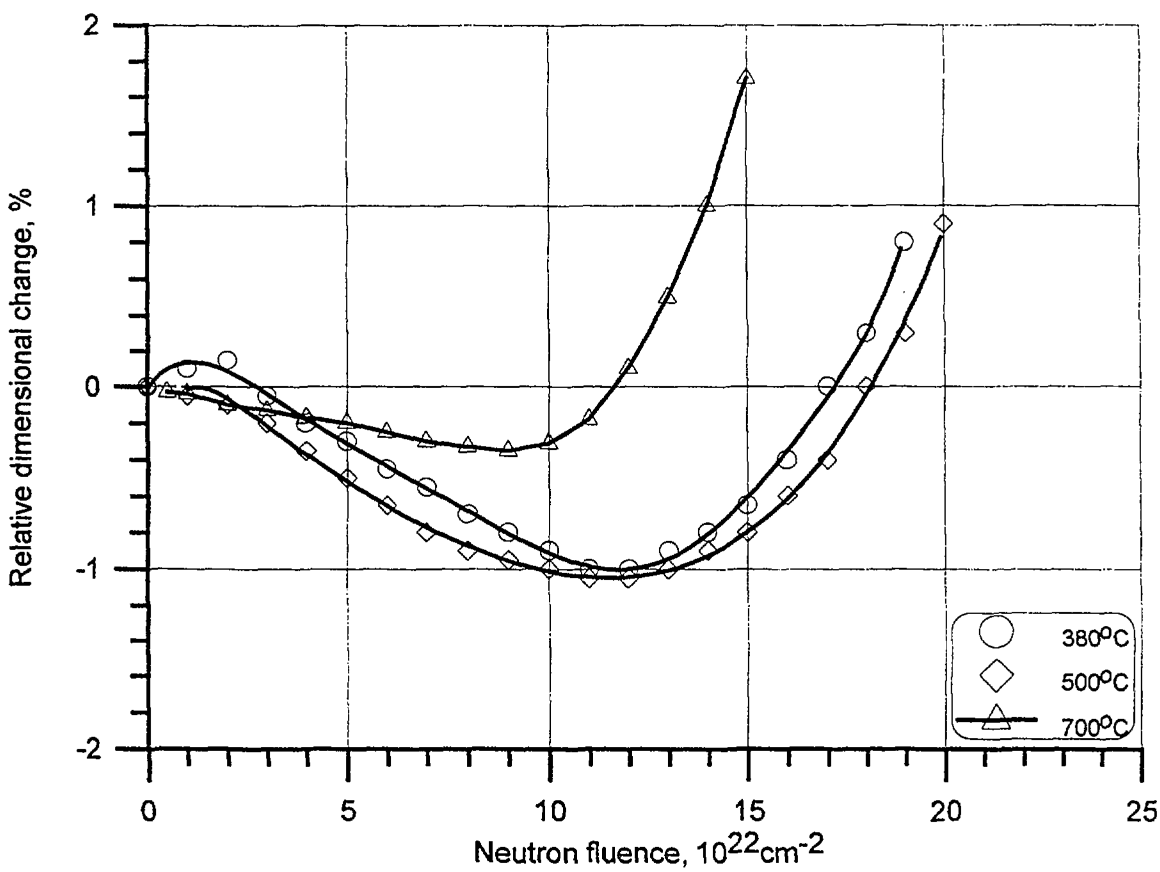


*FIG.6. Relative dimensional change of reactor graphite samples (GR-280) in radial direction.*

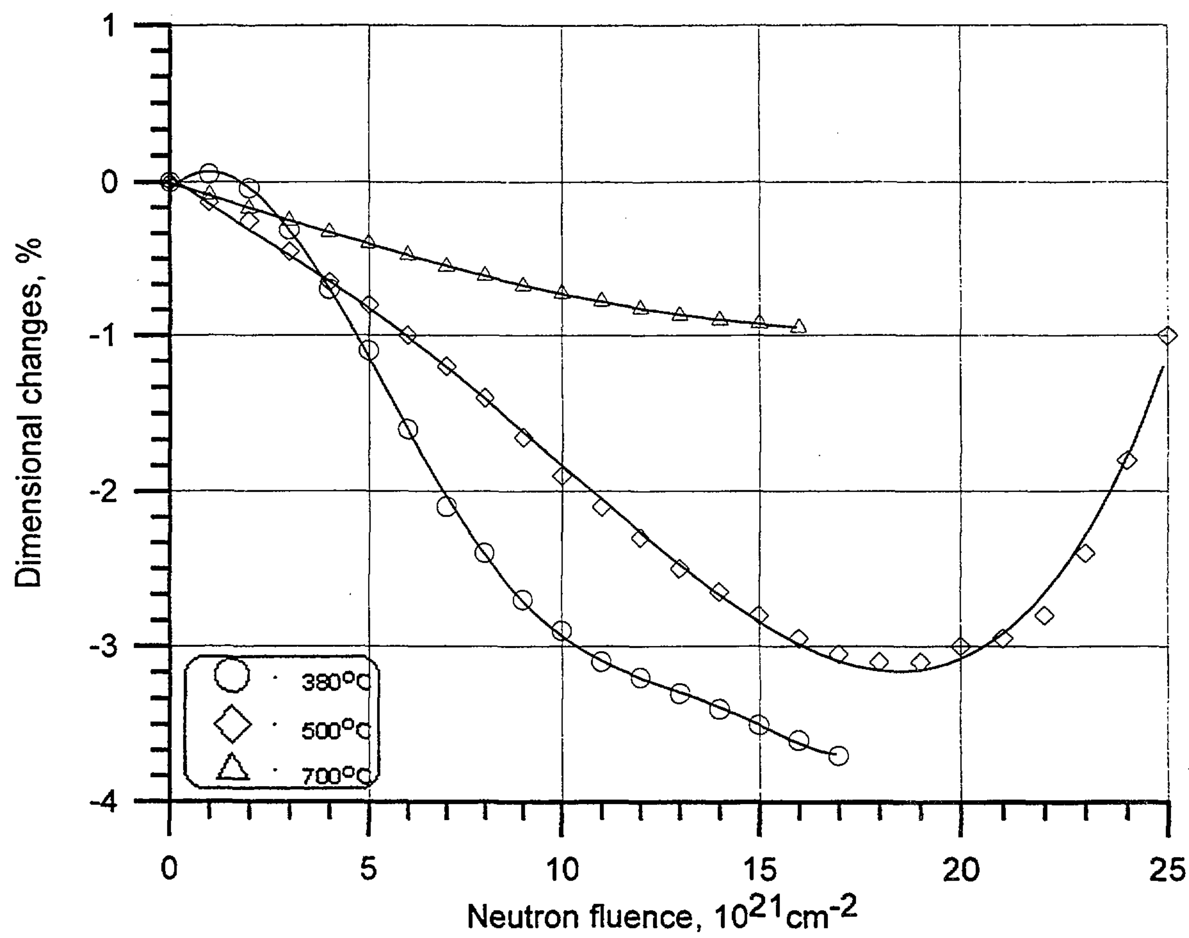


*FIG. 7. Relative dimensional change of reactor graphite samples (GR-280) in axial direction.*

In the case of the graphite bricks the general trend in dimensional changes reflects that predicted from small graphite samples. However the dimensional changes of the brick are more complex due to the variation of irradiation dose and temperature within the brick. The graphite also creeps under irradiation which is described later. The whole brick shrinks and the diameter of the central hole decreases. In addition the brick takes up a limited barrel shape profile. Under flux and temperature gradients, such as exist near the reflector elements or near absorbers the whole brick will bow. But these deformations will only be significant towards the end of reactor life.

### 5.3.2. Mechanical properties

The modulus and therefore the strength first increases with dose and then decreases initially slowly and then rapidly at very high doses (approx. $2.2 \times 10^{22}$ $n/cm^2$, E >0.18 MeV) (see Fig. 8).

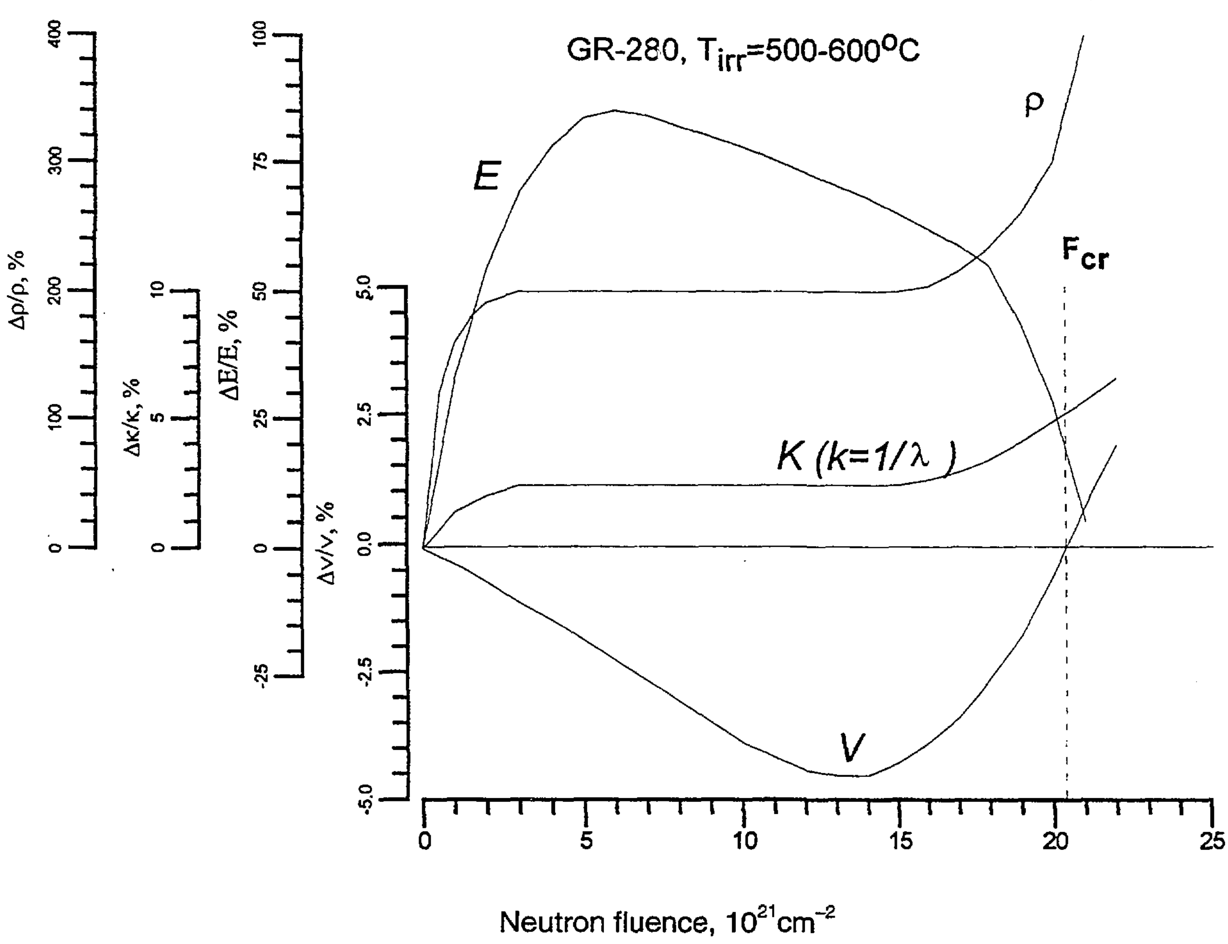


*FIG.8. Determination of neutron fluence critical values for graphite physical properties changes (V-volume, E-elastic modulus, k-thermal resistivity, ρ-electrical resistance).*

### 5.3.3. Thermal properties

The coefficient of thermal expansion slowly increases up to 40% at high doses under RBMK irradiation conditions.

The thermal resistance (the reciprocal of thermal conductivity) increases rapidly with dose before saturation. However, at very high irradiation doses as the graphite ages the thermal resistance increases again (see Fig. 8).

### 5.3.4. Irradiation creep

In the presence of irradiation graphite creeps at a much faster rate than it would otherwise. Irradiation creep reduces internal shrinkage stresses and thermal stresses generated within the graphite bricks due to thermal and dose gradients. In the absence of creep these internal stresses would lead to early brick failure.

On reactor shutdown, the thermal stresses which are relieved by irradiation creep during operation reappear as the brick cools down to a constant temperature. In the long term (at approximately 30 year operation for RBMK-1000) these thermal stresses may lead to brick cracking.

These complex changes in material properties coupled with irradiation creep make the assessment of stresses and deformations in graphite components complex. The graphite bricks have a large cross section which is best analysed using finite element techniques. The assessments must be time integrated from non-linear relationships to take into account the non-linear material changes with time and the visco-elastic behaviour.

These techniques have been used in Russia and the UK to assess the behaviour of graphite in a number of graphite moderated reactors including the RBMKs.

Analysis of expected deformation of the graphite using finite element methods do not correlate with actual results of measurements. To overcome this problem, possibly some adjustments to both the Russian and UK models were required. These finite element techniques are also used to predict the time at which graphite bricks will fail. Therefore it is important that these techniques are verified. This may be done by applying them to other older graphite moderated reactors and comparing calculated values with measured values.

## 5.4. CRITERIA FOR GRAPHITE STACK OPERATION

### 5.4.1. Criterion for graphite as a structural material

The limit on the degradation of the physical and mechanical properties is defined by the critical fluence. This defines the time when the graphite, after shrinkage, returns to its original volume (see Fig. 8). From this stage of irradiation all the properties of graphite begin to deteriorate greatly (decrease in modulus and strength, increase in thermal resistivity, etc.).

Also the critical fluence (the point where the graphite returns to its original volume) will be reached earlier at higher graphite temperatures than at lower temperatures.

Russian evaluation of the time when the value of critical fluence could be achieved showed that with average temperature of the graphite stack operation 500–600°C and at the achieved rate of energy production of RBMK-1000 reactor, it will be near 40 years.

The cracking of the graphite bricks with subsequent increase of core temperature and bowing of graphite columns could lead to a decrease in the 40 years estimate for critical fluence.

### 5.4.2. Criteria for graphite stack as a whole

Dimensional changes and changes to the thermal expansion coefficient lead to internal stresses in the individual bricks. Eventually this will lead to brick failure (cracking). This will greatly accelerate the bowing deformation of the whole core. The criteria for stack operation in this case is the ultimate bowing of an individual column, most probably at a peripheral channel or cells. This could lead to difficulties in operation of the control and protection system.

# 6. NORMAL OPERATING CONDITIONS

## 6.1. PRESSURE TUBE BEHAVIOUR

### 6.1.1. Design condition

During initial operation, the fuel channel tubes will increase in diameter and possibly elongate from irradiation creep and growth. The water chemistry results in a relatively rapid rate of corrosion to produce an oxide layer and a slow rate of hydrogen pick. The initial corrosion in core is uniform. After a period of time, the form of corrosion in RBMK-1000 enters a transition phase and becomes quasi-nodular. Eventually the corrosion assumes a near uniform thick condition.

In the out of core regions of the fuel channel tube the corrosion is uniform. The corrosion rate is different for each reactor type. In the RBMK-1000 fuel channel tubes, the out of core corrosion for the alpha annealed condition of the fuel channel tube is less than the in core corrosion.

The mechanical properties (tensile and yield strength) will increase and the elongation and toughness will decrease to constant levels.

The stresses on the tube are those expected from design provided the tube is manufactured free from excessive residual stress. Crack propagation in the tube can only occur if a defect size is large or stress intensity exceeds a threshold value. At low hydrogen concentrations, crack propagation from a defect by delayed hydrogen cracking can only occur at low temperature (<200°C) if stress intensity is higher than threshold value. After 17.3 years of operation, the mechanical properties including toughness are at a saturation level. The hydrogen concentration in the tubes is in the range 15–47 ppm, and the remaining metal thickness varies from 3.57 to 4.24 mm.

### 6.1.2. Out-of-design condition (gas gap closure) for short term operation

With the pressure tube diameter expanded to cause contact between the fuel channel tube, graphite rings and the graphite block, the following conditions will change from the designed condition:

- the hoop stress on the tube in the contact region will decrease and some loading will be transferred to the graphite blocks;
- the axial stresses in the upper part of the fuel channel tube will depend upon the length of contact which will increase with time;
- the fuel channel tube after some time may locally creep into the slotted region of the surrounding rings;
- the temperature gradient through the wall of the pressure tube will change because of increased heat transfer from the graphite;
- the cell (graphite block rings, fuel channel tube) will become more rigid;

- at this stage of operation the corrosion layer will have reached a thickness of up to 300 μm and the hydrogen concentration may be approaching ~50-60 ppm;
- the form of contact will not be uniform even along one graphite block because of flux profile variation and variations in contraction of the graphite brick.

With the above conditions existing in fuel channel tubes, it is likely that the probability of a rupture of a channel is smaller than for the conditions when a gas gap is present because the stresses are reduced, the defect needs to grow to a longer length and there is a higher probability of the leak being detected even with the closed gap condition. If a defect would develop to an unstable length and rupture, the consequences are not clear but the evidence from a rupture in Leningrad Unit 3, which occurred in a near closed gap condition, is that the consequences may be similar to the open gap condition.

Long term operation with gas gap closed is currently not recommended by the designer. Should operation continue longer than the time defined in Section 2, the conditions specified above would generally become worse and a situation of graphite brick cracking and deformation would make the predictability of consequences of such operation unreliable.

## 6.2. GAS GAP CLOSURE

In this section, the prediction methods and issues related to the gas gap closure and the implication of this behaviour on the safe operation of an RBMK reactor are discussed. There are two assessment techniques used related to gas gap closure.

The first is the methodology to predict gas gap closure. This is based on measurements of fuel channel diameter and graphite bore diameter taken from the reactors. These measurements are used along with statistical analysis to predict the time at which the gas gap will close. These predictions have to be adapted for each reactor to allow for the particular operating mode in particular with respect to the graphite temperature.

The second assessment technique involves the assessment of the core life and consequence of gas gap closure. These techniques involve the use of material test reactor data and finite element techniques. This second technique also takes into account the graphite brick shrinkage and thermal stresses and enables the time to brick failure to be predicted. This is important as it is the only method available to determine these brick stresses.

### 6.2.1. Process of gas gap closure

The initial nominal size of the gap is 3.0 mm on the diameter. Due to the graphite shrinkage and the zirconium pressure tube expansion the gap between the graphite and the fuel channel pressure tube closes. The rate of closure depends on channel pressure tube type (manufacturing process), reactor power and graphite temperature. The larger the size of the gas gap, the higher the graphite temperature becomes due to the worsening of the heat transfer from the graphite to the fuel channel. The higher the graphite temperature, the sooner the graphite will reverse its dimensional change and the higher the rate of subsequent expansion.

After it was known that the gas gaps would close, it was decided in Russia to perform complete retubing of the whole core. This strategy permitted a reduction in time and dose expenditure compared to partial retubing, and to combine the retubing with upgrading of other part of reactor.

The RBMK designers now recommend that the tubes should be replaced before a large fraction of the gas gaps have closed in the core. This has been carried out in Leningrad Units 1 and 2. Retubing has also begun in Kursk Unit 1 and Leningrad Unit 3. It should be noted that after retubing, the gas gap will contract before starting to open, however, gap closure is not expected to occur.

### 6.2.2. Prediction of gas gap closure

To enable the time for gas gap closure to be predicted, measurements are made at the operating plants of fuel channel pressure tube internal diameter and graphite brick internal bore diameter. Because the graphite bore diameter measurement requires the fuel channel pressure tube to be removed, the frequency of these measurements is much less than the fuel channel pressure tube measurements. There are some uncertainties in the original component dimensions, the material properties, graphite temperature and irradiation dose so it is possible that there will be some fuel channel tubes locally jammed into the graphite bricks at the time of retubing.

For Leningrad Units 1 and 2, the time to retube were predicted on the basis of a linear extrapolation of the measured graphite data as a function of cell energy production.

Linear extrapolation of the graphite measurements for the RBMK-1000 will result in plant operation with limited gas gap closure. In the case of the RBMK-1500 units at Ignalina NPP, preliminary calculations suggests a non linear relationship between shrinkage and energy generation, and in the process of estimating gas gap closure time, this non linearity should be taken into consideration.

After completion of Leningrad Unit 2 retubing, the obtained experience was used as a basis to develop recommendations for retubing of subsequent units.

One recommendation made by the reactor designers is to retube at such a core average exposure where the maximal force for withdrawing of the fuel tube is not more than 10 tons. This was derived from the load bearing capacity of 20 tons of the upper welded joint, with a factor of safety of two.

Another recommendation the designers made is that the number of fuel channels withdrawn without graphite rings should not exceed more than 10 percent of the core. This criterion is based on the considerations given below.

There are three major factors that could affect on separation of graphite rings from fuel channel tube during fuel channel withdrawal:

- quality of the connection of the lower supporting bushing with fuel channel tube;
- mutual location of the minimal graphite bore diameter and maximal tube diameter;
- existence of contact between fuel channel and graphite brick.

The first two factors are random and due to large number of fuel cells the number of cells with similar characteristics should not differ from unit to unit.

Therefore the number of channels which were withdrawn without graphite rings could reflect the number of fuel channels with closed gas gap.

On the basis of experience accumulated during retubing Leningrad Unit 1 and 2, and using in addition similar data obtained in the process of retubing of Kursk Unit 1 and Leningrad Unit 3, the value of reactor energy production at which it is necessary to begin retubing was defined, see also Section 6.4.

This methodology should be used in combination with analysis of diameter measurements results because this data give opportunity to check if the rate of graphite shrinkage is the same as in previous cases.

Using the two recommendations given above it is possible to estimate a limiting value of the reactor energy production and subsequent time of fuel channel replacement. However, it is still necessary to take account of plant specific operating mode, particularly graphite temperatures, for the different units.

Both of these recommendations require core retubing towards the lower range of predicted gap closure as defined by average channel power production.

### 6.2.3. Methods for measuring gas gap and fuel channel deformation

At each of the RBMK plant, there is equipment for measuring fuel channel tube and graphite stack deformations. Measurements that are made are:

- ultrasonic measurement of the fuel channel pressure tube wall thickness (only at Ignalina NPP);
- outside diameter of fuel channel pressure tube (only at Ignalina NPP);
- internal diameter of the fuel channel pressure tube;
- pressure tube length;
- fuel channel bow;
- graphite brick internal hole diameter (only after removal of the fuel channel);

The present requirements for geometry measurements of RBMK-1000 reactors during fuel channel in-service inspection are shown in Table VII.

TABLE VII. GAS GAP MONITORING REQUIREMENTS

| Time | Number of fuel channels measured | Number of graphite bricks bores measured |
|---|---|---|
| Start of operation | 50 | 10 |
| 8000–10 000 hours of operation | 10 | 1 |
| Each 30 000 hours from start up to 15 years of operation | 30 ( 90 in 15 years) | 1 (3 in 15 years) |
| After 15 years each 8000–10 000 hours of operation | 30 | 10 |

## 6.3. CONSEQUENCES OF GAS GAP CLOSURE

If the RBMK reactors are operated for a long time with fully closed gas gap in the core the consequences may be:

- premature cracking of the graphite bricks;
- elongation of the fuel channel tube;
- fatigue loading of upper welds due to different thermal expansion of the fuel channel tube and of graphite during start up and shut down or fluctuations in power;
- swelling of the tube into the slots in the graphite rings raising locally the tube stress;
- changes in the graphite temperature associated with broken bricks;
- bowing of fuel channels;
- deformation of the core as a whole.

Some of these consequences, such as bowing of the channel, could have safety implications. For this reason it would appear that the RBMK plants should be either retubed or shut down shortly before significant gas gaps closure has occurred.

However, it would be expedient to assess the consequences of operating an RBMK reactor with closed gas gaps. To achieve this, it will be necessary to model the core as a whole using computational techniques.

Gas gap closure under normal operation is a long term process and the consequences given above will appear only after an extended period of operation with closed gap.

Initially when the gas gaps closes in a limited number of cells, fuel channel pressure tube internal stresses decrease due to restriction of the tube expansion and the pressure tube axial expansion increases. At this early stage it is unlikely that graphite brick cracking, significant column bowing, and the accumulation of other negative consequences of the gas gap closure process will occur.

## 6.4. RBMK-1000 RETUBING EXPERIENCE

The effect of gas gap closure was seen in Leningrad Unit 2 where power production per "average" cell was larger than 500 MW/day than that for Unit 1. During retubing of Leningrad Unit 2, the number of fuel channel withdrawn without rings was equal to 16% (272 as compared to 33 at Unit 1), from the whole set and up to 5 fuel channels were withdrawn with force approx. 15 tons. Because the value of withdrawal force exceeded allowable value of 10 tons, such state of fuel channel pressure tube graphite interaction was considered to be undesirable. In order to ensure that the value of withdrawal force will be within the limit of 10 tons, allowable number of fuel channels withdrawn without rings was set to 10%.

From the experience of retubing Leningrad Units 1 and 2, it was seen that the maximum diameter of fuel channel and minimum diameter of graphite cell were not located on the same axial level. This means that even though the predictions show a closed gap, in reality this will only be observed as excessive withdrawal forces when the tube is being withdrawn.

Retubing experience at Leningrad Units 1 and 2 has shown that local contact should cause only separation of graphite rings from fuel channel tubes during withdrawal of the tube and no damage to the graphite bricks.

It was shown by measurements during the retubing of Leningrad Units 1 and 2 that the prediction of the time to carry out this operation was to a large extent adequate. However, in retubing Unit 2, it was found that some channels required a greater pull-out load than 10 tons to remove the tube. It is therefore recommended that retubing RBMK-1000 reactors should take place earlier than was the case in Leningrad Unit 2.

Prediction made using material test reactor data, has indicated that after retubing of the RBMK-1000 reactor, the gas gap is unlikely to close again. However, calculations indicate that near 30 years of reactor operation some of graphite bricks may be cracked due to high internal stresses. To determine the probability of bricks cracking at any time during life of an RBMK, a probabilistic assessment is required, taking account of material variability and the accuracy of the reactor temperature and irradiation data.

### 6.5. PREDICTION OF GAS GAP CLOSURE IN RBMK-1500

Preliminary assessments carried out in Russia and at the Ignalina plant using linear extrapolation of pressure tube and graphite measurements have shown the time to retube Ignalina Unit 1 is in the range of 11 500 to 16 000 MW/days. Additional data on graphite measurements are required to refine the estimate. After that the prognosis of the gas gap closure time should be defined more precisely taking into account non-linear rate of graphite shrinkage at this stage.

Prediction of gas gap closure in Ignalina Unit 2 indicate that gas gap closure may not occur. Using the results of measurements performed at Ignalina Unit 2 it is possible to expect that the rate of fuel channel tube expansion (for TMO-2) is about 30 to 40 percent smaller than for Ignalina Unit 1 (TMO-1 tubes) but this prediction still needs to be verified with larger numbers of measurements.

The existing regulations require the inside diameter of the tube to be measured and the outside diameter is determined using the original nominal pressure tube wall thickness. For the Ignalina plant an ultrasonic technique has been developed in Lithuania and is now used to determine the outside diameter of the tube. During 1996–1997, measurements of the outside diameter and the wall thickness were carried out for about 400 fuel channel pressure tubes (for the maximum power production cells). These additional measurements are then used in the prediction of the time to gas gap closure thus improving its accuracy. Laboratory evaluations of the measurement device have indicated that the accuracy of this instrument is in the range of ±100 mm. However, it is recommended that the accuracy of this device should be verified using post irradiation measurements of tubes of known dimensions. Such measurements are not carried out at other RBMK plants.

## 7. ACCIDENT CONDITIONS

For cracks that may arise in pressure tubes during normal operation, leak before break behaviour is expected to provide for shutdown of the reactor prior to unstable failure of the fuel channel. The leakage history of the RBMK plants is rather uneven with a significant number of leaks occurring for example at Kursk and Chernobyl plants and zero leakage occurring at the Smolensk plant, Table VIII. The difference in operational experience has apparently resulted from technological improvements in the design, fabrication and materials utilized in the various plants. These technical improvements were documented to ensure application of the appropriate technology for a new set of channels designed for replacement of used channels.

TABLE VIII. FUEL CHANNEL REPLACEMENT DUE TO SYMPTOMS OF LEAKAGES AND RUPTURES OF THE MID-CORE REGION

| Plant | 1973–1975 | 1976–1980 | 1981–1985 | 1986–1993 | Total |
|---|---|---|---|---|---|
| Leningrad | 2 | 0 | 1 | 1 | 4 |
| Kursk | 0 | 1 | 17 | 5 | 23 |
| Chernobyl | 0 | 3 | 39 | 3 | 45 |
| Smolensk | 0 | 0 | 0 | 0 | 0 |
| Ignalina | 0 | 0 | 0 | 2 | 2 |
| Total | 2 | 4 | 57 | 11 | 74 |

In accident situations, such as reactivity insertions or a loss of coolant supply to the fuel, the channels could overheat and rupture.

## 7.1. TUBE RUPTURE EXPERIENCE

Three fuel channel ruptures have occurred during the operational history of the RBMK reactors. The first failure, which occurred in 1975 at the Leningrad plant, was initiated by a localized power excursion with resultant fuel channel overheating. The other failures, Chernobyl 1, 1982, and Leningrad 3, 1992, were the result of flow blockage followed by fuel channel overheating and subsequent failure. Permanent displacement of adjacent channels were observed (Fig. 9), but failure of adjacent pressure tubes did not occur.

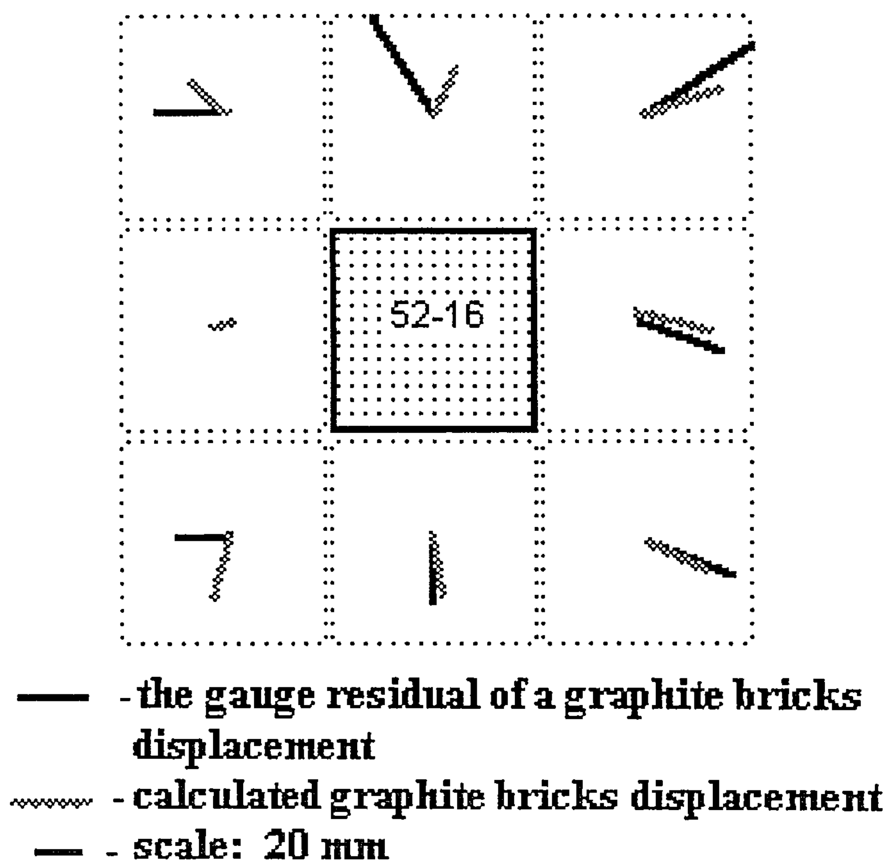


*FIG.9. Leningrad Unit 3 single tube rupture, 1992.*

The potential for propagating fuel channel failures is dependent on the core and material properties as well as the type of initiating event. Many of these initiating events have been analysed earlier in the plants' safety reports by RDIPE using the available computer codes and analysis assumptions in line with the former Soviet Union requirements.

The USA with RDIPE, is supporting development of a data acquisition and analysis effort related to multiple fuel channel ruptures. This effort could be integrated with the TACIS programme 2.8 on reactor cavity overpressure protection. As a part of the US funded effort, a prioritized list of initiating events was developed. Specific analysis of a spectrum of fuel channel failures will be evaluated to determine the potential for propagating fuel channel failures.

## 7.2. PLANT SPECIFIC SAFETY ANALYSIS

The difference in fuel channels manufacturing technology and operating condition results in different rate of gap closure. There is also a difference in fuel channel material properties in RBMK-1000 and RBMK-1500 reactors. These differences must be considered in plant specific safety analyses.

In-depth safety assessments were initiated for the Leningrad Nuclear Power Plant, Unit 2 in late 1996. This in-depth safety analysis is a multilateral cooperative effort to perform probabilistic and deterministic safety assessments for Leningrad Unit 2 following the proposed safety upgrades. The Leningrad Unit 2 safety assessment is a cooperative arrangement with the plant, RDIPE, Sweden, the UK and the USA. The probabilistic safety assessment is being supported by Sweden and the UK with the USA providing funding and technical support for the development of plant specific deterministic analyses, system description and selected system single failure and fault tree analyses.

The Leningrad Unit 2 deterministic analysis includes the evaluation of the potential for multiple pressure tube failures. The evaluations are specific to Leningrad Unit 2 and include appropriate material properties and core configuration for both current and end of life conditions. Since pressure tubes of Leningrad Unit 2 have been replaced, the analysis will not consider plant operation or accidents with closed gas gaps. The analysis is scheduled for completion in mid 1998.

The Leningrad Unit 2 multiple pressure tube rupture evaluations require the identification of the success criteria for pressure tube integrity. Specific criteria are expected to include stress and strain criteria, acceptable flaw sizes, pressure-temperature failure criteria and other criteria required to evaluate pressure tube integrity during normal operations and accident conditions. RDIPE has prepared a draft of the criteria which is currently under review.

An in-depth safety assessment of the Ignalina NPP was completed by the end of 1996. A plant specific safety analysis report (SAR) has been produced, which provided the basis for the safe operation. The responsibility for producing the SAR lay with the operator of the Ignalina NPP. In exercising his responsibility, the operator made use of Vattenfall AB, a western utility, and eastern and western engineering to address the safety issues and help to prepare the SAR. This SAR aimed to:

- assess the current level of safety of the plant through an analysis and its review comparable to that commonly performed for western nuclear power plants;

- identify and evaluate any factors which may limit the safe operation of the plant in the foreseeable future;

- assess the Ignalina NPP safety standards and practices;

- recommend any additional improvements which are reasonably practicable and provide estimates their cost and schedule.

The safety study considered a safety assessment of both units at the Ignalina NPP. The main reference plant for the project is Unit 1, but a survey was included which defined the differences between Units 1 and 2 and assesses their impact.

The safety analysis report was reviewed by a consortium of independent technical safety organizations to eastern and western nuclear safety authorities. The review was performed, with a slight time lag, more or less in parallel with the production of the SAR.

The Ignalina accident analysis covers 23 accidents, which includes:

- loss of coolant accidents;
- reactivity transients;
- loss of electric power supply;
- operational transients;
- reactor cavity venting;
- anticipated transients without scram.

Where possible, western codes were used for the analysis. The analysis is plant specific. The analysis was undertaken in Moscow, primarily by RDIPE with assistance from AECL, Canada and AEA Technology, UK. The review team consisted of experts from GRS, Germany; Lithuanian Energy Institute, Lithuania; RRC Kurchatov Institute, Russia; IPSN, France and Scientech, USA. The review team performed independent analyses for 6 selected accidents.

Failure criteria, including those for fuel cladding, fuel channels and heat transport circuit integrity, were elaborated. If any failure criteria is violated, supplementary analysis should be performed to provide details of certain process and phenomena that are not represented in the integrated analysis of the plant response, or those that are not modeled in detail by the integrated models because they do not impact on the overall plant response (e.g. fuel and pressure tube thermal mechanical response). Supplementary analysis of the response code (e.g. RELAP 5) was carried out to investigate possible variation in fuel channel parameters. Supplementary analysis of the channel thermal mechanical response is performed either to explore variation in channel parameters, or to quantitatively evaluate a potential for fuel cladding or pressure tube failure. Two codes TRANS and RAPTA are used together. The RAPTA code evaluates dynamics of temperature transients in the fuel and cladding, stress-strain relationship for cladding under tension and compression, cladding oxidation in steam, and determines cladding integrity based on strength and oxidation.

Because the Ignalina NPP in-depth safety assessment was conducted for the specific conditions of the existing plant, it does not include an evaluation of the situation when pressure tube fully or partially contacts the surrounding graphite. A specific thermal hydraulic analysis and fracture mechanics analysis is being performed that addresses the following aspects:

- the possible thermal cooldown rates with zero gas gap;
- the most limiting transient scenarios;
- thermal strain of the fuel channel generated by the above mentioned cooldown rates, longitudinal stresses;
- fuel channel failure possibility.

The results of the analysis, sponsored by the USA and performed by the Lithuanian Energy Institute, were scheduled for January 1998.

## 7.3. TRANSIENT HIGH TEMPERATURE FAILURE CRITERIA

The information in this section was provided by Russian specialists, who are now carrying out a testing program on fuel channel integrity under a range of pressures and temperatures. These experiments have recently been summarized in a video by RDIPE specialists. Testing of electrically heated channels with the surrounding graphite rings and blocks provides important information to demonstrate the safety basis for operation. Testing has been carried out for channel overheating and other transients.

This testing programme will provide critical data on mechanistic failures of fuel channels and the response of the fuel channel and graphite block under accident condition.

It is noteworthy that the necessity to complete a large number of experiments in order to obtain good statistics data requires substantial effort.

### 7.3.1. Deformation model and its stress-strain characteristics

During an accident analysis the pressure tube integrity assessment is based on the comparison of the analysis results with a criterion the form of which is appropriate to the code used.

Post accident analysis of pressure tube ruptures has shown that graphite blocks and core components do not prevent the pressure tube from ballooning at full system pressure. There are no significant differences in the test results with and without graphite. So all the data relevant to high-temperature pressure tube behaviour may be used to construct a rupture criterion for RBMK pressure tube integrity assessment.

Pressure tubes which have a middle radius to wall thickness $r/\delta$ ratio of 10.5 may be considered as thin walled, so the data processing uses a plane and axis-symmetric deformation model of a pressure tube with no defect under internal pressure $p$.

The stress and strain intensities are used to characterize large plastic deformation of tubes. The stress intensity is:

$$\sigma_i = \frac{\sqrt{3}}{2}\sigma_\theta ,$$

where

$$\sigma_\theta = p\frac{r}{\delta} .$$

The transversal, radial and axial strains are the main strains (wall thickness change is used as the radial strain). Logarithmic strains are suitable for large plastic deformation:

$$\varepsilon_\theta = ln\frac{L}{L_0}; \quad \varepsilon_\delta = ln\frac{\delta}{\delta_0}; \quad \varepsilon_z = ln\frac{H}{H_0},$$

where $L$, $\delta$, $H$ and $L_o$, $\delta_o$, $H_o$ are the tube cross-sectional perimeter, tube wall thickness and height, for final and initial conditions.

For thin-wall tube the strain intensity is:

$$\varepsilon_i = \frac{2}{\sqrt{3}}\varepsilon_\theta .$$

The strain rate intensity is:

$$\xi_i = \frac{d\varepsilon_i}{d\tau} .$$

### 7.3.2. Test facilities and parameters measured

The pressure tube heating accident was modeled with sections of both regular and scaled one (ratio 1:4) RBMK pressure tube. Tested tubes were vented with steam or argon gas. Some of the tests were carried out with graphite blocks.

The tube wall temperature $T_w(\tau)$, channel pressure $p(\tau)$ and external tube radius increment $\Delta r_2(\tau)$ were measured during the tests. The maximum external tube perimeter $L_2^{max}$ was measured after tube rupture.

During the tests the pressure range was 0.2 to 8.2 MPa, temperature range was 300 to 1300°C and heat-up rate range was 0.1 to 80°C/s.

The rupture temperatures $T_{rup}$ and pressures $p_{rup}$ were determined using $T_w$ and $p$ histories. Rupture strains $\varepsilon_\theta^{max}$ and $\varepsilon_i^{max}$ were determined using measured $L_2^{max}$ values. The rupture stresses $\sigma_\theta^{max}$ and $\sigma_i^{max}$ were calculated. For each test the tube radius increment $\Delta r_2(\tau)$ was transformed into $\varepsilon_i(\tau)$ which was approximated by a cubic spline to calculate $\xi_i(\tau)$.

### 7.3.3. Pressure tube failure criteria and data

The computational assessment of pressure tube failures uses various experimental failure criteria: rupture temperature–channel pressure (temperature criterion), rupture strain–tube temperature (strain criterion), rupture stress–tube temperature (stress criterion) and others.

Figure 10 shows the temperature criterion. Data are presented for different heat-up rates. Tests with graphite blocks are shown as black symbols. The shaded area in Fig. 10 covers all of the experimental $T_{rup}$–$p_{rup}$ combination. At higher heat-up rates $u_T$ (dark gray area) the $T_{rup}$ values are higher than at lower ones (light gray). The $T_{rup}$–$p_{rup}$ combinations above the shaded area are low probability. There are no rupture conditions below the shaded area. Both heat-up rate and strain rate influence the dispersion of $T_{rup}$. The solid lines are the approximation of tube rupture temperature (°C) (channel pressure is in MPa):

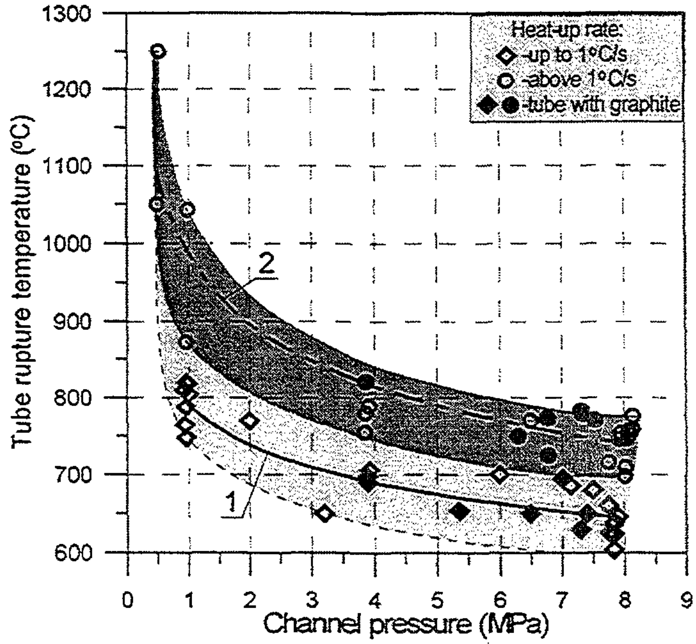


*FIG. 10. Pressure tube rupture temperature versus internal pressure.*

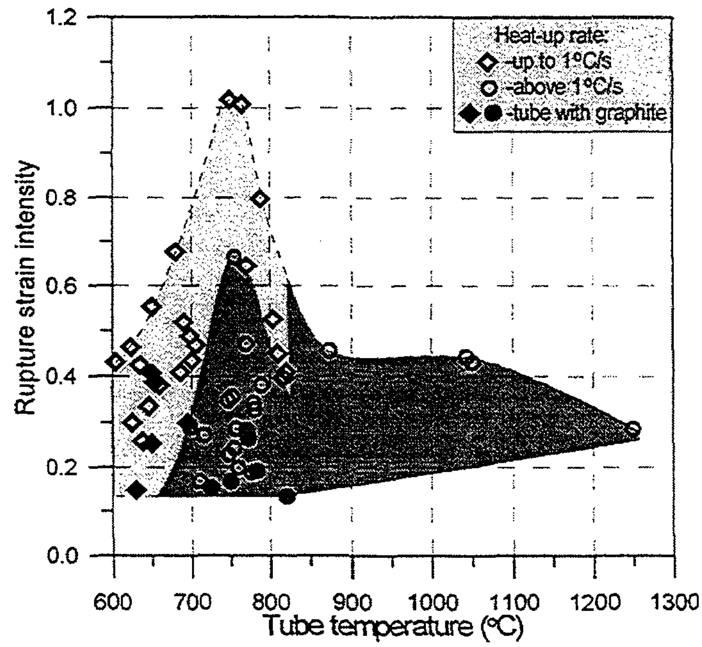


*FIG. 11. Pressure tube rupture strain intensity versus pressure tube temperature.*

***curve 1*** ($u_T \leq 1°C/s$):
$T_{wrup}=790.5 \cdot p^{-0.099}$;

***curve 2*** ($u_T > 1°C/s$):
$T_{wrup}=987.7 \cdot p^{-0.139}$.

These correlations can be used in thermal hydraulic computer codes for pressure tube rupture assessment without calculation of the tube deformation.

Figure 11 shows the strain criterion. The shaded area in Fig. 11 covers all of the experimental $\varepsilon_{irup}$–$T_{rup}$ combinations. At lower heat-up rates (light gray) the $\varepsilon_{irup}$ values are higher than at higher rates (dark gray). At 750–760°C and low heat-up rates the super elasticity of Zr+2.5%Nb alloy can be seen. It is difficult to correlate the $\varepsilon_{irup}$ data due to the large dispersion.

To improve the situation one may introduce the correlation of $\varepsilon_{irup}$ as a function of thermomechanical parameter $K_{tm1}$ (J/kg). $K_{tm1}$ is the combination of thermal energy gained by the tube, stress intensity and internal load:

$$K_{tm1} = \frac{c_w \cdot T_w \cdot \sigma_i}{p},$$

where $c_w$ is the Zr-alloy heat capacity.

Figure 12 shows the $\varepsilon_{irup}$–$K_{tm1}$ correlation. There is smaller dispersion of data than in Fig. 11. Solid lines in Fig. 12 are the approximation of rupture strain as a function of parameter $K_{tm1}$ (J/kg).

To correlate rupture stresses one may use the thermomechanical parameter $K_{tm2}$ (dimensionless) that is the relationship between thermal energy gained by the tube and the tube ballooning mechanical work under internal pressure:

$$K_{tm2} = \frac{c_w \cdot \rho_w \cdot T_w}{p \cdot \varepsilon_i},$$

where $\rho_w$ is the tube material density. Figure 13 shows the $\sigma_{irup}$–$K_{tm2}$ correlation.

The criteria equations for strain and stress intensities have to be expressed as a set of curves for various heat-up rates. The more precise $\varepsilon_i$ and $\sigma_i$ needed the greater the number of curves required in Figs 12 and 13 and in computer codes. To overcome this difficulty more universal criteria have to be used.

The specific rupture strain power (W/kg) is the energy criterion and characterizes the tube deformation work in a time interval:

$$j_i = \frac{\sigma_i \cdot \xi_i}{\rho_w}.$$

Only the data which contained correct measurements of the tube deformation history can be treated to determine tube rupture strain intensity rates.

Figure 14 shows specific rupture strain power as a function of tube temperature. The solid line consists of two approximate curves. One curve has the peak strain power for the super-plasticity temperature (~760°C) of the Zr+2.5%Nb alloy and other curve has the peak at ~1050°C.

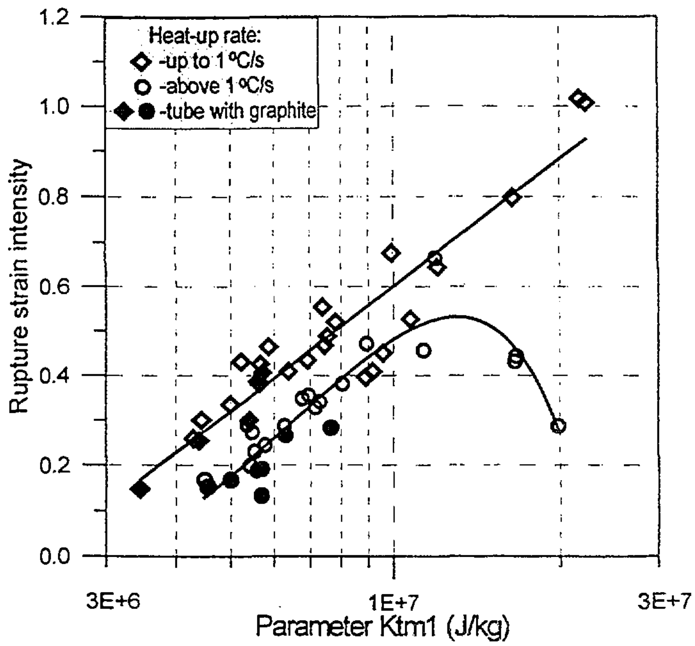


*FIG. 12. Pressure tube rupture strain intensity versus parameter $K_{tm1}$.*

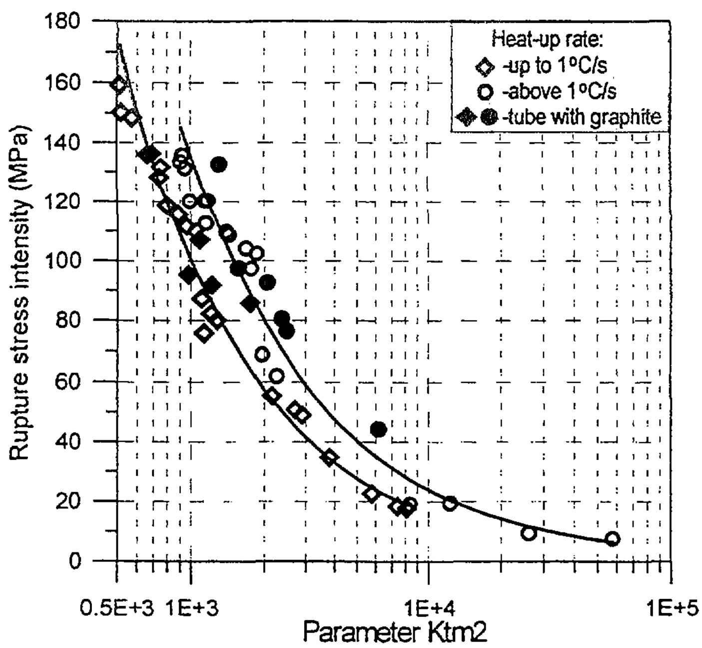


*FIG. 13. Pressure tube rupture stress intensity versus parameter $K_{tm2}$.*

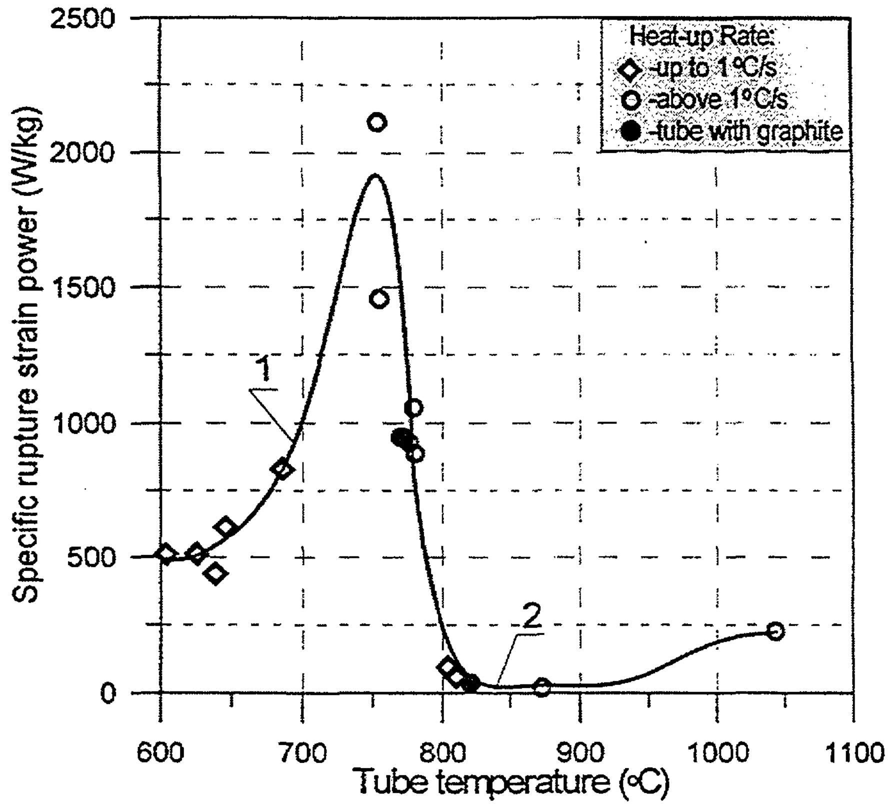


*FIG. 14. Pressure tube rupture strain power versus pressure tube temperature.*

The curves 1 and 2 in Fig. 14 are the polynomial functions of 6th order:

***curve 1*** (600°C $\le T_w \le$ 780°C):

$$j_{irup} = -8.58388\cdot10^7+797061\cdot T_w-3079.41\cdot T_w^2+6.336043\cdot T_w^3-7.322469\cdot10^{-3}\cdot T_w^4 +4.506571\cdot10^{-6}\cdot T_w^5-1.15385\cdot10^{-9}\cdot T_w^6;$$

***curve 2*** (780°C< $T_w \le$ 1050°C):

$$j_{irup} = 9.00264\cdot10^7-581258.59\cdot T_w+1561.12\cdot T_w^2-2.23237\cdot T_w^3+1.7925482\cdot10^{-3}\cdot T_w^4 -7.6633326\cdot10^{-7}\cdot T_w^5+1.362675\cdot10^{-10}\cdot T_w^6.$$

These correlations can be input to computer codes where high-temperature creep equations for Zr+2.5%Nb alloy are used. Values of $j_i$, calculated in the code have to be compared with the limit values calculated using the correlations above.

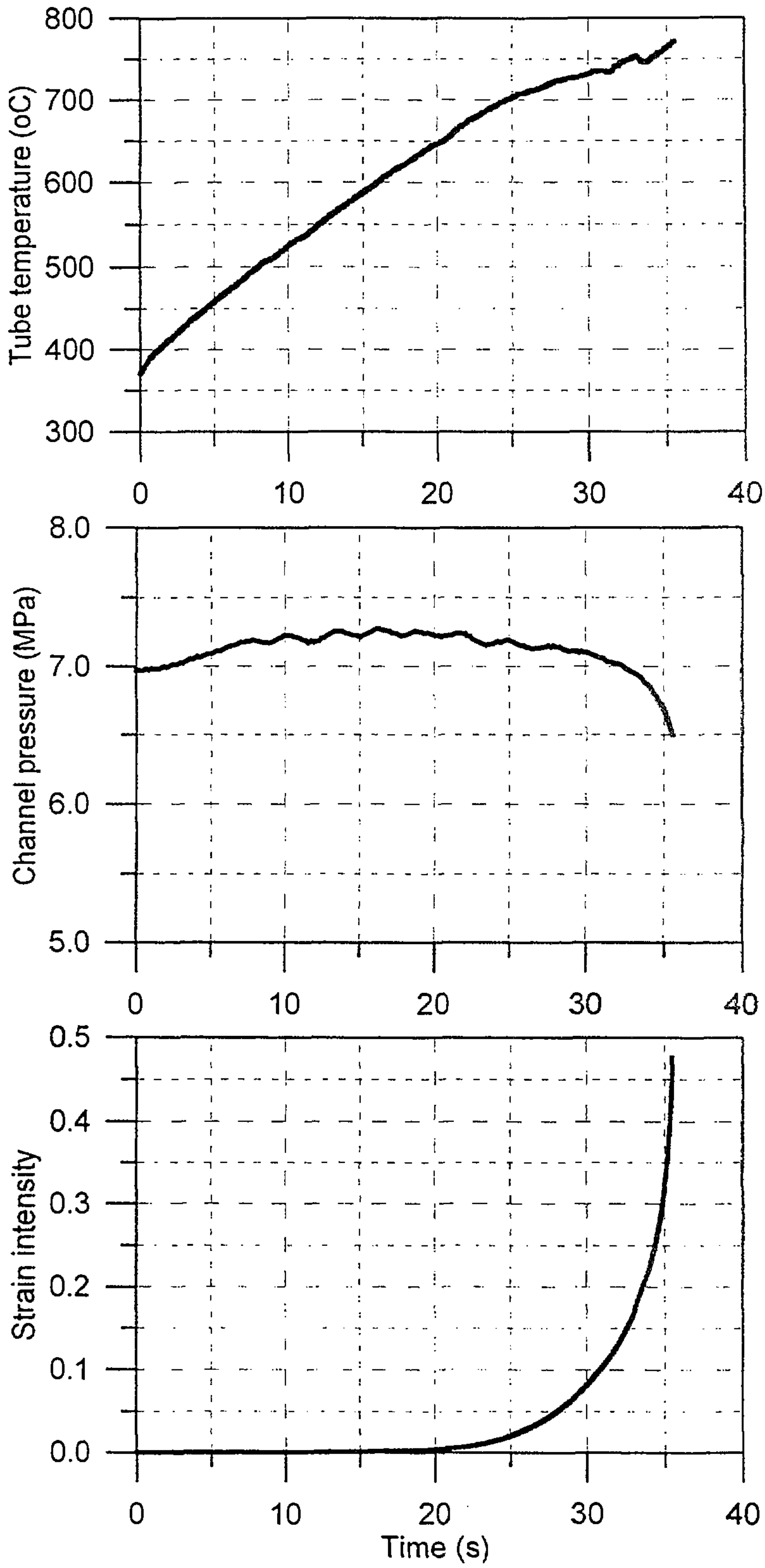


*FIG. 15. Test with section of regular RBMK pressure tube without graphite blocks.*

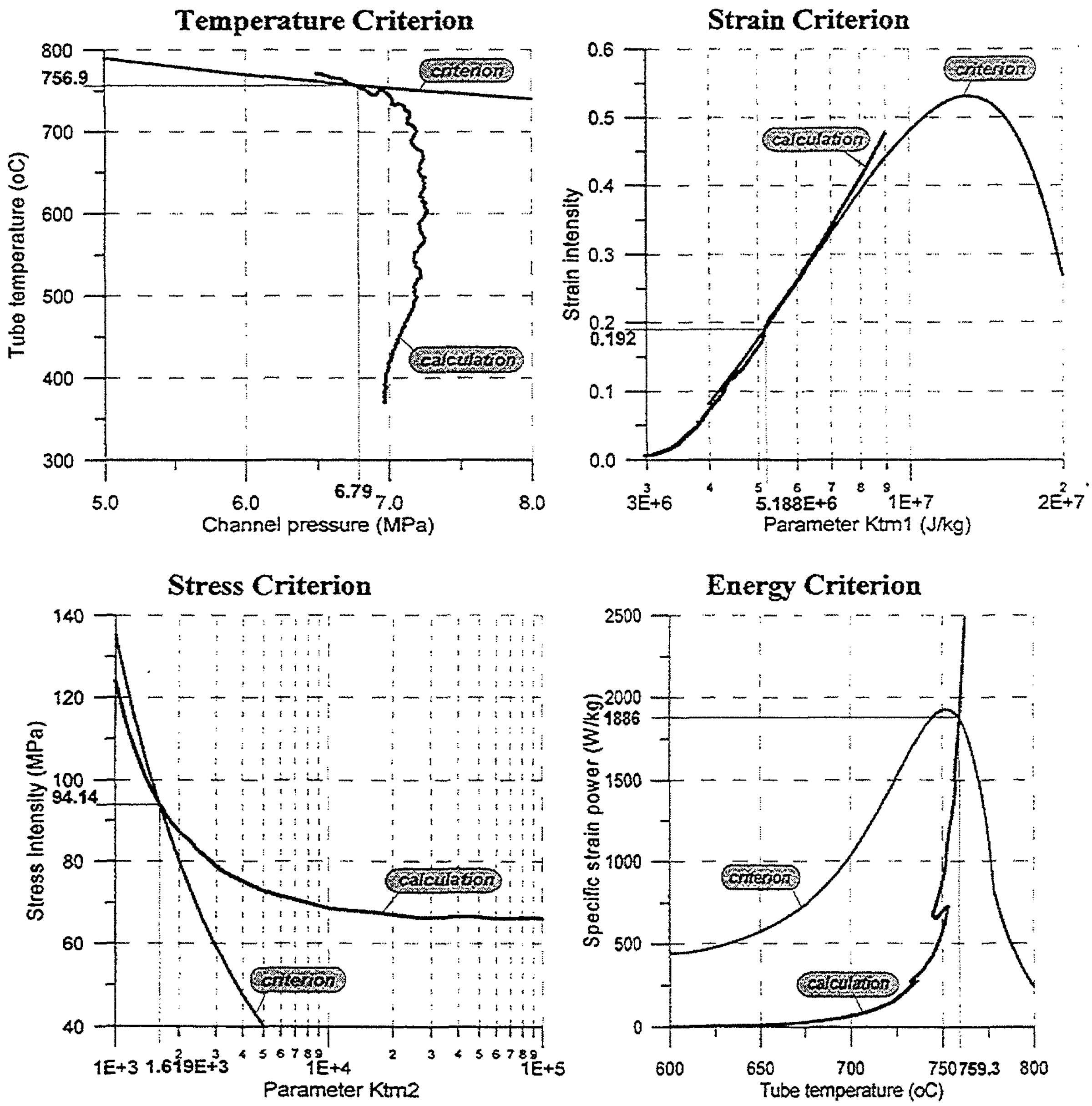


*FIG. 16. Calculation with KATRAN code.*

### 7.3.4. Pressure tube failure prediction

The criteria developed were introduced into thermal mechanical code KATRAN (developed in RDIPE) intended to predict high-temperature pressure tube behaviour.

In Fig. 15 the results of a test are shown. The section of regular RBMK pressure tube was heated with full pressure. Heating rate was ~11°C/s. Thermal-mechanical behaviour of the tested pressure tube was analysed by KATRAN code. Figure 16 shows the results for different criteria. Measured rupture temperature and time were the following: 771°C and 35.5 s. The average values of four criteria are 754°C and 34.2 s.

The Leningrad Unit 3 incident (1992) was analysed using the code KATRAN. The temperature history was calculated by a 2D thermal code taking into account fuel elements deformation and their contact with tube wall. For the four mentioned criteria time and temperature of pressure tube rupture and intensity of deformation were predicted as well. Deviation from the mean of the four criteria for time to rupture and temperature was not more than 3 percent. For intensity of deformation (strain), the minimum estimate was for the temperature criterion (0.105) and the maximum was the deformation criterion (0.28).

The sensitivity of results to temperature history was evaluated as well.

## 8. FUEL CHANNEL DUCTS

The fuel channel is attached to lower and upper ducts. In the top part of the upper duct the fuel channel plug housing is located. The fuel assembly is attached to the plug through a threaded hanger element and a connecting element.

The fuel channel is attached to the upper duct through a welded joint. The fuel channel lower part is connected to the lower standpipe through a bellows, which compensates the thermal expansion as well as axial pressure tube growth resulting from creep and radiation growth and maintains tightness of the reactor cavity.

The principal safety concern related to the ducts is connected with the defects found in the upper duct weld to fuel channel plug housing and its neighboring zone. The full length failure of this duct weld would lead to fuel assembly ejection to the reactor hall. In such a case, the fuel would be damaged, plant personnel exposed and reactor hall contaminated.

In the period 1986 to 1995, defects in the upper ducts were detected in the following sections (Fig. 17):

- in the weld, section A–A;
- in the base metal below the weld, sections B–B, C–C.

According to the results of metallographic examinations, the defects in section A–A of Fig. 17 were caused by the non-compliance with the welding technology requirements during the manufacturing of the ducts in the shop. This led to lack of fusion in the root. The detected defects were repaired by mechanical cutting and re-welding at the site.

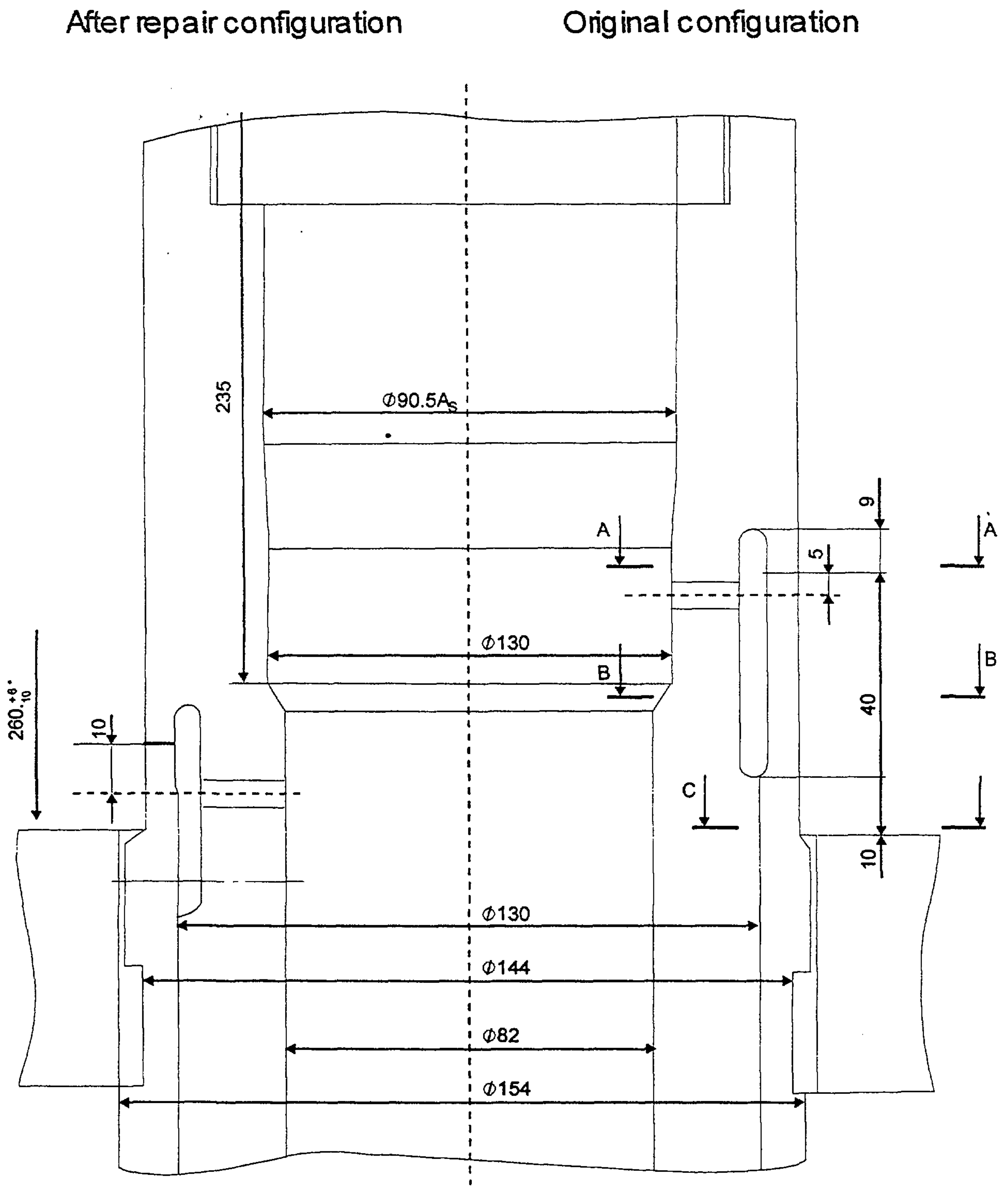


*FIG.17. Upper standpipe to fuel channel plug housing weld; original and after repair configurations.*

In the period 1994 to 1995, during retubing of the Leningrad plant when plug housings were replaced, ultrasonic testing revealed other defects in the ducts base metal (Fig. 17, section B–B). The cracks found were inside surface axial cracks, of depth 0.3 to 1.5 mm, length 2 to 15 mm, located approximately 30 to 40 mm below the weld line. The cracks were transgranular stress corrosion cracks. The cracking occurred due to a high level of residual stresses in combination with a corrosive medium.

Further, in the base metal of the upper ducts (Fig. 17, section C–C), transgranular radial cracks growing from the outer surface were found. These cracks were caused by high cycle thermal fatigue.

Weld defects were found in other RBMK units except Smolensk 3, which had this problem addressed during manufacturing.

The non-destructive examination of upper ducts is carried out using ultrasonic testing, radiography, dye penetrant testing, and visual testing. Some of the ducts, in which defects were revealed, were cut out and investigated (metallography, gamma-spectrometry, fractography). The results obtained indicate that the weld defects (Fig. 17, section A–A) did not grow during operation. The defects found were analysed by fracture mechanics and, when necessary, repaired.

## 9. CONCLUSIONS

(1) For RBMK-1000 reactors and the adopted retubing strategy, limited local gas gap closure occurs at the time of pressure tube replacement. The safety justification for short term operation in this condition has not been documented for normal operation or operational transients.

(2) There is, however, no straightforward indication that the impact of a small percentage of channels operating in a closed gas gaps condition for a "short" period of time will increase the risk of an unstable fuel channel pressure tube failure under normal operating conditions.

(3) The recommendation of the reactor designers not to operate beyond a point when an estimated 10% of the channels would be removed without rings and at a withdrawal force not exceeding 10 tons, is currently, the most useful available criteria for retubing. This does not directly imply that 10% of the tubes are operating with closed gas gaps.

(4) For normal operation and with the retubing approach adopted in Russia as defined in item (3) above for RBMK-1000 reactors, no fuel channel material degradation mechanisms have been identified that could influence the fuel channel integrity.

(5) Analyses to assess the probabilities of fuel channel failure propagation under condition of open gas gaps are underway for Leningrad Unit 2. There are no comparable analyses currently planned for closed gas gap conditions.

(6) There are no analyses to justify operation with a large proportion of channels with gas gaps closed. In addition, the probability of failure propagation under conditions of closed gas gaps has not been assessed.

(7) The acceptance criteria pertinent to fuel channel integrity for normal operating conditions are reportedly available. The development of acceptance criteria pertinent to fuel channel integrity for transient and accident conditions should be continued. Peer review of the acceptance criteria developed should be carried out.

In developing acceptance criteria, account must be taken of plant specific features, plant status, operational states, internal and external events and uncertainties involved. The experimental programmes to provide necessary verification, initiated at RDIPE, should be continued.

(8) For RBMK-1000 reactors the data on fuel channel pressure tube and graphite behaviour under normal operating conditions were generated in Russia. The data available could provide a basis for normal operation analyses over the design life time of the reactor. The data needed to describe fuel channel tube behaviour for accident conditions analyses up to the end of the reactor design lifetime are not complete.

For RBMK-1500 the data to provide a basis for normal operating conditions and accident condition analyses up to the end of the reactor design lifetime are not complete.

## 10. RECOMMENDATIONS

(1) A device for direct measurement of the gas gap should be developed as a high priority task and implemented at RBMK plants. Improvement of the methods and techniques to increase the accuracy of the predictions should continue. Statistical uncertainties associated with the prediction of gas gap closure used should be quantified.

(2) Ultrasonic measurements of the outside diameter of the fuel channel tube, used at present at Ignalina plant, should be extended to all other RBMKs to improve the accuracy of gas gap closure predictions.

(3) The consequences of operating RBMK reactor with the gas gaps closed should be investigated under normal operating conditions and during thermal transients.

(4) The method of prediction of the behaviour of the graphite bricks throughout life using stress analysis should be verified. This can be achieved by analysing the graphite brick behaviour in the oldest RBMK plant, Leningrad Unit 1 or other old graphite moderated reactors.

(5) A probabilistic assessment, based on the Monte Carlo technique, should be used to determine the time dependent probability of graphite bricks cracking.

(6) It is recommended that RBMK experience in post service examination of over 45 fuel channel tubes be consolidated into a report and made openly available.

(7) The experimentation and analysis to develop pressure tube integrity criteria for accident conditions including conditions during reflooding of overheated channels should be continued to increase the data base and improve statistics.

(8) It is recommended that plant specific deterministic analyses regarding MPTR be performed using a prioritized list of initiating events.

(9) Multilateral cooperative in-depth safety assessments are strongly supported. Plant specific safety assessments should be undertaken for RBMK-1000 to provide an updated safety basis for plant operation.

(10) Taking into account that the data concerning behaviour of fuel channel pressure tubes are important for safety analysis of any channel type reactor, it is recommended that the data base be developed which could include all data available in Russia, Canada, Japan and other countries.

(11) That bilateral and international cooperation be used in order to create a sufficient data base of material properties under normal and accident conditions

(12) It is recommended that the number of measurements taken be increased or better statistical analyses be performed, to confirm that there will not be gas gap closure at Ignalina Unit 2.

(13) The evaluation of post irradiation properties of RBMK graphite irradiated in the high flux reactor at Petten should be carried out in order to extend and validate the RBMK graphite database.

(14) Sampling tools to obtain micro samples from tubes for analysis of hydrogen concentration and replication of defects should be developed.

(15) After 10 years of fuel channel operation it is reasonable to review the location of the spacer grids on all new fuel assemblies to avoid concentration (and fretting loss) of materials at the fuel channel pressure tube in this region.

(16) The possibility of changing the water chemistry to a less oxidizing condition to reduce the rate of fuel channel pressure tube corrosion should be considered to the second set of fuel channels.

(17) Additional equipment for ultrasonic flaw detection in fuel channel pressure tubes and transition joints in order to improve in-service inspection at the NPPs should be developed and implemented.

(18) For the upper ducts, non-destructive examination using automated ultrasonic testing should be employed in order to provide for adequate monitoring of defects. Such equipment, developed in Sweden, is being used at Ignalina NPP.

(19) In order to assess the possibility of the upper duct failure with consequent fuel assembly ejection, an integrity assessment, similar to the leak before break concept, should be considered.

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