VVER Reactor

Fuel Performance, Modelling and Experimental Support

Proceedings of an international seminar, held in St. Constantine, Varna, Bulgaria, on 7 - 11 November 1994

Compiled and edited by

S. Stefanova, P. Chantoin and I. G. Kolev

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Bulgarian Academy of Sciences

Institute for Nuclear Research and Nuclear Energy

VVER Reactor Fuel Performance, Modelling and Experimental Support
Proceedings of an international seminar, held in St. Constantine, Varna, Bulgaria, on 7-11 November 1994
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From the Editors

This publication is a compilation of papers presented at the International Seminar on VVER Reactor Fuel Performance, Modelling and Experimental Support, organised by the Institute for Nuclear Research and Nuclear Energy of the Bulgarian Academy of Sciences in co-operation with the International Atomic Energy Agency.

The Seminar took place in F.J. Curie International House of Scientists at the St. Constantine resort near Varna, Bulgaria, from 7 to 11 November 1994 and was attended by 70 specialists from 16 countries, including representatives of all major Russian plants and institutions responsible for VVER reactor fuel manufacturing, design and research.

The Seminar was divided into 6 sessions, covering (1) actual performance of VVER reactor fuel, (2) trends for improved fuel utilisation by introduction of new fuel design and management, (3) fuel economics and licensing, (4) fuel behaviour analysis and corresponding computer codes, (5) experimental support of fuel modelling and (6) problems of spent nuclear fuel of VVER plants - that is, most of VVER fuel life-cycle stages. Four panel discussion sessions were organised and corresponding reports are also included in this publication.

During compilation and editing, all papers have been divided into 4 sections, corresponding to panel discussion sessions, and papers have been re-ordered and grouped according to their main topics for convinience and to facilitate information retrieval. In addition, a comprehensive collection of abstracts of published bibliographic sources of VVER fuel data is included in the Supplement.

In certain cases, the text and captions have been sligthly edited for languige and uniformity reasons. The information contained is, however, in responcibility of the listed authors and neither the seminar organisers, the Publisher, nor any of the institutions supporting the seminar, hold any responsibility for the use of the information contained in the named articles.

The Editors express their gratitude to those authors who presented the text and graphics of their contributions in convenient computer files, as well as to the INRNE staff, responsible for the computer processing of this publication, especially to Ms. Albena Mladenova and to Ms. Lidia Kinova.

Finally, this publication would not have been possible without the overall administrative support from the INRNE management and especially Dr. Milko Milanov, Deputy Director for Applied Research and Head, Dptm. of Radiochemistry and Radioecology.

S. Stefanova
P. Chantoin
I. G. Kolev

International Seminar

VVER Reactor Fuel Performance, Modelling and Experimental Support

7-11 November 1994

F.J. Curie International House of Scientists, St. Constantin, Varna, Bulgaria

Organised by

Institute for Nuclear Research and Nuclear Energy of
the Bulgarian Academy of Sciences
Tsarigradsko shousse 72, Sofia 1784, Bulgaria

in co-operation with

International Atomic Energy Agency
Wagramerstrasse 5, Vienna 1400, Austria

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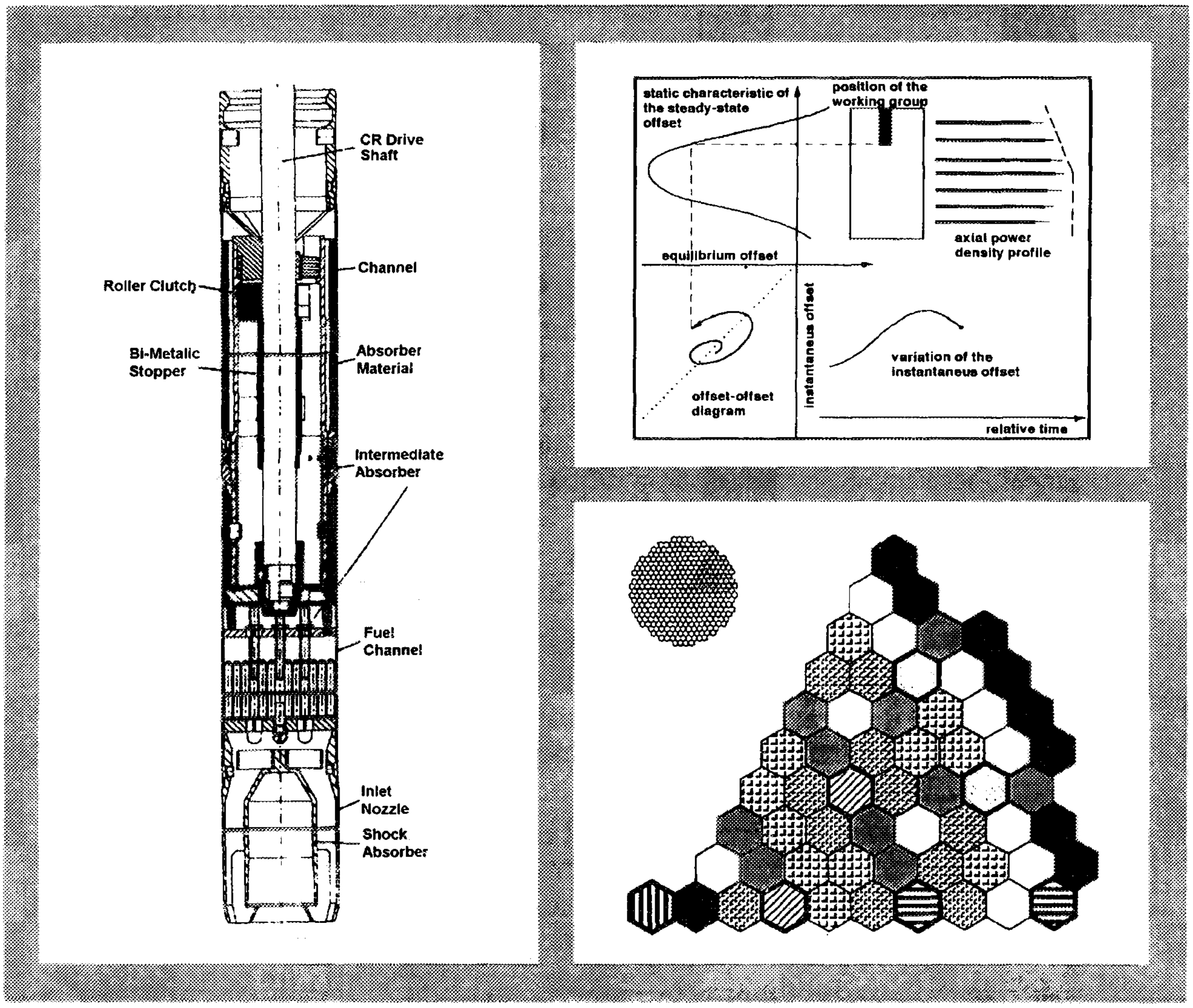
Part 1

VVER Fuel Performance and Economics: Status and Improvement Prospects



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The Institute for Nuclear Research and Nuclear Energy of the Bulgarian Academy of Sciences has been officially announced in 1972 but trace its history back to 1961, when a Scientific and Experimental Nuclear Centre was organised within the former Institute of Physics of the Bulgarian Academy of Sciences.

During these more than 30 years of scientific and applied research in nuclear science and engineering, INRNE grew to become one of the largest institutions in the organisational structure of the Bulgarian Academy of Sciences, as well as a major national institution for scientific support of Bulgarian nuclear generation programme.

THE FUNDAMENTAL **THEORETICAL RESEARCH** IS FOCUSED IN TWO FIELDS:

- *Theory of Elementary Particles*, covering a wide range of approaches: axiomatic, group-theoretical, algebraic, conform-invariant, quasi-potential, etc.
- *Theory of Nuclear Structure and Nuclear Reactions*, based on different model conceptions for shell, unified, local-scale, energy-functional, theoretical group and other approaches.

THE **EXPERIMENTAL RESEARCH** IS CARRIED OUT IN:

- *High and Ultra High Energy Physics*, that solves problems connected with quark-gluon structure, relativistic nuclear physics, cosmic ray physics and modern gamma-astronomy of ultra high energies.
- *Low and Mean Energy Physics*, namely: nuclear spectroscopy of short-lived isotopes, accelerator physics, neutron diffractometry, diffusion and thermalization of neutrons in inhomogeneous media.

THE **APPLIED RESEARCH AND DEVELOPMENT** COVER TWO BASIC FIELDS:

- *Nuclear Energy* - analysis and optimisation of processes in nuclear power reactors, nuclear safety and equipment reliability, nuclear data, nuclear fusion problems and muon catalysis, decontamination and storage of nuclear waste, energy systems analysis.
- *Nuclear Methods* - aimed to performance of complex non-destructive analysis on the basis of neutron activation, beta-refraction, Moessbauer and other methods. X-ray dating of samples, radiochemistry and isotope production services are available, as well as production and application of neutron moisture meters.

VVER Fuel Performance and Economics: Status and Improvement Prospects

Panel Discussion Report

Panel Chairmen: Yu. Bibilashvili[1], *L. Goldstein*[2]

[1] A.A.Bochvar Scientific Research Institute of Inorganic Materials (VNIINM), Moscow, Russian Federation
[2] The S.M. Stoller Corporation, Pleasantville, USA

Several topics were put out for panel discussion, among them:

- low leakage (LL) fuel management;
- higher discharge burnup;
- improved thermal-hydraulic design for more thermal margin (DNB);
- longer fuel cycles.

The idea was to explore future directions for development, however most of the discussion centered on current approaches.

1 Fuel for VVER-440 Plants

1.1 Advanced Fuel Cycles

The VVER-440 operators expressed an interest in changing all their units to the 4 year cycle, using Zr alloy spacer grids, assembly enrichment shaping and low leakage loading pattern (LLLP). It is considered a high priority task.

As was noticed during the meeting, a few of the above mentioned improvements have already been realized in several countries or are currently being implemented (particularly 4 year cycle, Zr-alloy spacer grids, partial or full LLLP).

It was suggested by the VVER fuel users that implementation of advanced fuel cycles should start with all the above mentioned features. Russia will begin this process in 1995 as soon as these fuel management schemes are available for Russian fuel customers. Representatives of all present VVER utilities also expressed a wish to change to 5 year cycle in the future.

1.2 Fuel Cycle Flexibility

During discussion, particular emphasis was given to the need to improve fuel cycle flexibility. Some users of Russian fuel noticed that they had a need to increase the power and the irradiation time by 20-30 EFPDs.

It was noticed that this modernization could be solved by an extension of a number of enrichment levels in FA cross-section and without the introduction of integrated burnable absorber fuel (UO_2-Gd_2O_3). When an 18 month cycle is requested, Russia can supply Gadolinia fuel and related reloading schemes.

1.3 Coolant Mixing Issue: the DNB Margin

In discussion, serious attention was paid to coolant mixing issues in VVER-440 FAs. The representatives of VVER fuel designers emphasized that due to the small distance between adjacent SGs (220 *mm*) good mixing was provided as necessary to minimize the DNB margin even for 4.4% ^{235}U enriched fuel and there was no need for coolant deflectors. This is also true of fuel with higher enrichment even loaded in the reactor central position.

Mr. Silberstein (EVF) mentioned that Framatome has improved the DNB correlation over time with improved spacer design (of particular importance in higher power density cores) even under transient conditions.

2 Fuel for VVER-1000 Plants

2.1 Hihg Priority Tasks

On the basis of discussions, the high priority tasks for the next 2 - 3 years have been formulated as follows:

- Use of integrated fuel burnable absorber UO_2-Gd_2O_3 which permits a more negative TMC for low power levels (J10% Pnom], better cycle flexibility and a time increase between refuelling up to 330 - 350 EFPDs;
- Provision of additional capabilities to obtain better effectiveness of emergency protection systems and enable the implementation of LLLP reloading schemes.

- Exclusion of lumped burnable absorbers;
- Change of spacer grids and guide tube material from stainless steel to Zirconium alloy
- Increase the fuel residence time from 3 to 4 years;
- Change the design of the VVER 1000 assembly by making it dismountable with a more rigid skeleton and increased guide tube diameter,
- Uilization of control rods with increased life-time, effectiveness and weight.

Very important, but less urgent, is the provision of a load follow mode of operation.

2.2 Reduction of Control Rod Insertion Time

The situation of control rod reduced insertion time in VVER 1000 was discussed. The representative of the VVER-1000 designer (OKB "Hydropress", Russia) stated that the cause of the CR drop time exceeding the design value had been established. The drop time increase observed on several reactors is due to assembly bowing.

After extensive study, the following counter-measures were implemented at five VVER units in Russia and these modifications are underway at all other VVER-1000 units in Bulgaria, Russia and Ukraine. These counter-measures are:

A) Implemented measures:
   i) optimization of the fuel assembly axial loads in VVER-1000;
   ii) improvement of the power distribution by profiling the core and fuel assemblies;
   iii) implementation of a stricter control of water chemistry.

B) Measures which are under way:
   i) modification of the assembly to make it more rigid;
   ii) utilization of new fuel cladding and guide channel materials providing a better dimensional stability under irradiation (Zr alloy);
   iii) increasing the control rod gravitational weight.

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Nuclear Fuel Utilisation in Kozloduy NPP

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Fuel utilisation in Kozloduy NPP as a matter of principle strives after efficiency and economy with safe reactor operation criteria met.

In compliance with this principle the core loadings of units 1 - 6 are designed so that at optimum quantity of fresh fuel and fuel cycle duration the basic operation parameters should be within the admissible limits, providing for nuclear safety. For this purpose in Kozloduy NPP the following reasearch and developments are being carried out in terms of:

- extended reactor physics calculations and analyses;
- improving the accuracy of the computer codes used to model and calculate the processes occurring in the core;
- study the fuel behaviour under operation conditions;
- evaluation of the fuel assemlies condition by applying the methods for checking fuel cladding integrity at the end of each fuel cycle
- actions to reduce the irradiation of the reactor pressure vessels;
- improvement and development of the in-core monitoring systems;
- working-out of an optimized programme for refuellings and outages, etc.;

Up to the present moment the nuclear power reactors in Kozloduy NPP have been operated for a total of 62 fuel cycles.

Units 1 and 2 utilizing VVER-440/V-230 reactors, which in 1991 were shut down for upgrading and modernization, are presently running their 17th and 18th fuel cycles. Tables 1 and 2 show some basic characteristics of core loading and operation conditions during past fuel cycles, for Unit 1 and Unit 2, respectively. Tabulated are:

- fuel enrichment and number of fresh fuel assemblies;
- fuel cycles duration in effective days;
- maximum fuel power peaking factors at design calculations Kqcalc and during reactor operation at Kqmeas, taken for one and the same moments from the fuel cycles;
- reactor capacity factor;
- average discharge burn-up in *MWd/kgU*.

The tables show the basic variations of the core loading design [1] approved by the respective cognizant authorities and the fuel supplier:

- core design change principles amounting to placing more heavily burned-up assemblies in the periphery of the core, i.e. low-leakage core loading pattern;
- deployment of 36 dummy assemblies in the core periphery, protecting the reactor pressure vessel from the damaging impact of the fast neuton flux.

The average design value of the discharge fuel burn-up has been reached (31 *MWd/kgU*).

Unit 3 was commissioned in 1981. It has been operated for 12 fuel cycles, the characteristics of which are given in Table 3. Also tabulated is the specific burn-up of ^{235}U in the fresh fuel, representing the ratio of ^{235}U mass to the number of effective days in the design fuel cycle duration. Since the 7th fuel cycle, aiming at reduction of the reactor pressure vessel irradiation, thirty six dummy assemblies have been placed in the core in combination with a low-leakage core loading pattern. The rest of the core periphery has been supplied with 24 or 30 fuel assemlies with considerable burn-up (Fig. 1).

Since the 10th fuel cycle, the enrichment of a number of the fuel portions of the control assemblies have been increased from 2.4% of ^{235}U (design) to 3.6%.

The core loading selection method adopted since the 10th fuel cycle includes the following actions:

- to observe, if possible, core loading pattern by alternating 90,90,96 fresh fuel assemblies of 3.6% enrichment in the course of 3 successive fuel cycles;
- to utilize 12 fresh control assemblies of 3.6% enrichment in every cycle and every second year 1 control assembly of 2.4% enrichment in central position;
- extension, if necessary, of the design duration of the fuel cycles utilizing a power effect.

By meeting the above mentioned conditions provision is made for 300 effective days of the fuel cycles compared to 260 - 270 by design if dummies are utilized. The average discharge fuel burn-up is approximately 33 - 34 *MWd/kgU*.

Unit 4 of Kozloduy NPP was commissioned in 1982. It has got an improved technical design compared to that of Units 1 and 2 and to a greater extent meets the raised safety requirements to nuclear power units. Up to the present moment the reactor has completed 11 fuel cycles. The number of assemblies in the core corresponds to design - 349, of them 312 fuel assemblies and 37 control elements.

Table 4 shows the basic core loading and operation characteristics over the past 11 fuel cycles.

Since the 4th fuel cycle low-leakage core loading patterns have been used by placing more heavily burned-up assemblies in the peripherey of the core. At the same time, part of the fresh fuel assemblies are moved to the centre of the core (Fig. 2).

The adoption of the low-leakage core loading patterns increases the fuel cycles duration by approximately 10% compared to design, while for steady-state fuel cycles the increase is 3 - 4%, the amount of fresh fuel being the same.

Since the 7th fuel cycle of Unit 4 the core loading design has been based on the following fundamental principles:

- forty-eight assemlies of considerable burn-up are deployed in the core periphery;
- part of these periphery assemblies have been used in the core for 3 fuel cycles and their integrity has been checked;
- gradually adopt the practice of utilizing control assemblies the fuel portion of which is enriched to 3.6%;
- the height of the working group of control assemblies in the core with the reactor at power is 200 *cm*;
- it is possible to use a power effect to extend the fuel cycle in compliance with grid demand and nuclear safety principles.

As can be seen from Table 4, by means of the core loading described above the average duration of the fuel cycles achieved under operation at nominal parameters is 300 - 310 effective days and the average number of fresh fuel assemblies (both working and control) is about 102 - 108 compared to design number 114 - 120.

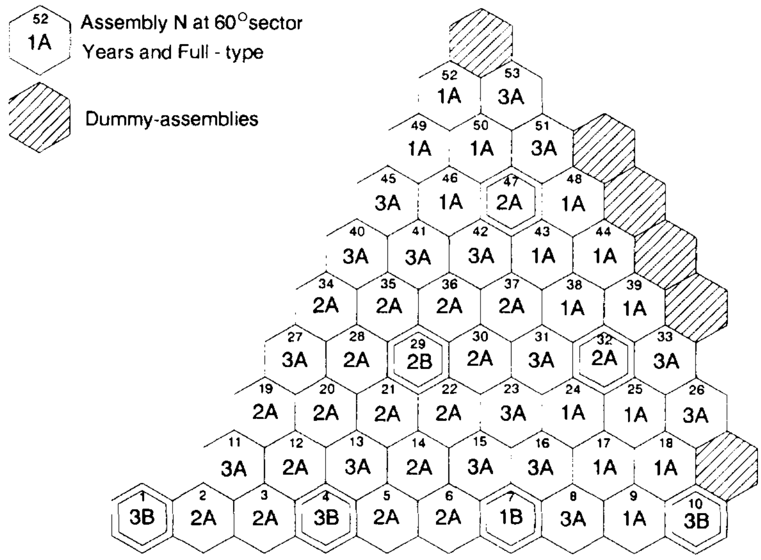


Figure 1 Unit 3, Cycle 11

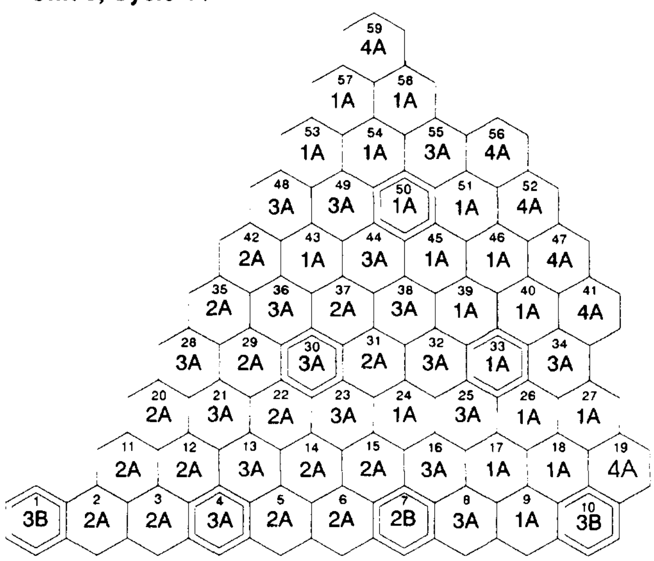


Figure 2 Unit 4, Cycle 12

The average discharge fuel burn-up achieved is within 33 - 36.5% *MWd/kgU*.

The average burn-up of 3.6% enrichment control assemblies discharged upon completion of their 3rd fuel cycle is within 32 - 35.7 *MWd/kgU*. Since 1986 (Unit 4's 6th fuel cycle) core loading designs are performed by means of computations using the SPPS-1 code.

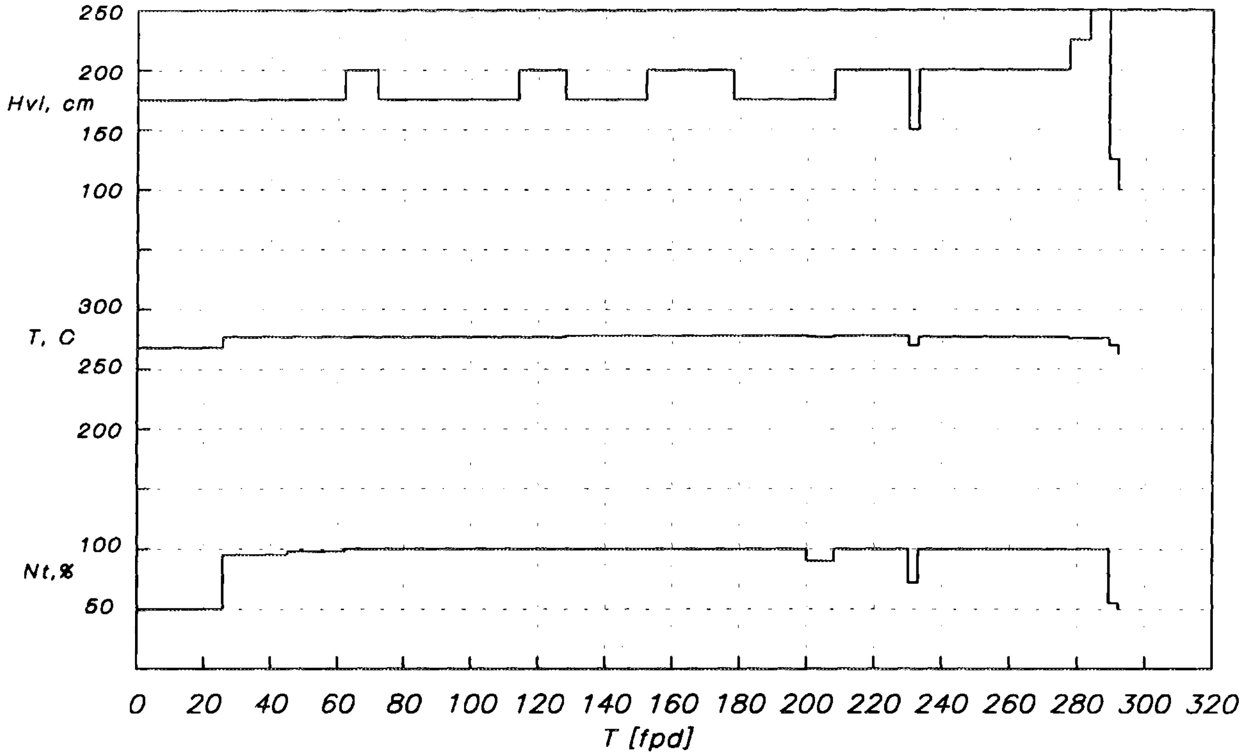
The multiple comparisons made on the basis of operational data from Loviisa NPP and Kozloduy NPP show that the code is precise and adequate to compute both low-leakage core loading patterns and cores utilizing dummies. Fig. 3 shows the basic core parameters corresponding to the operating conditions of Unit 3 during its 10th fuel cycle.

Plotted are also the dependence of boron critical concentrations and the fuel burn-up obtained from computations and they have been compared to the operational data. Results show that in rated power reactor operation modelling the measured and computed boron concentrations values concur very well.

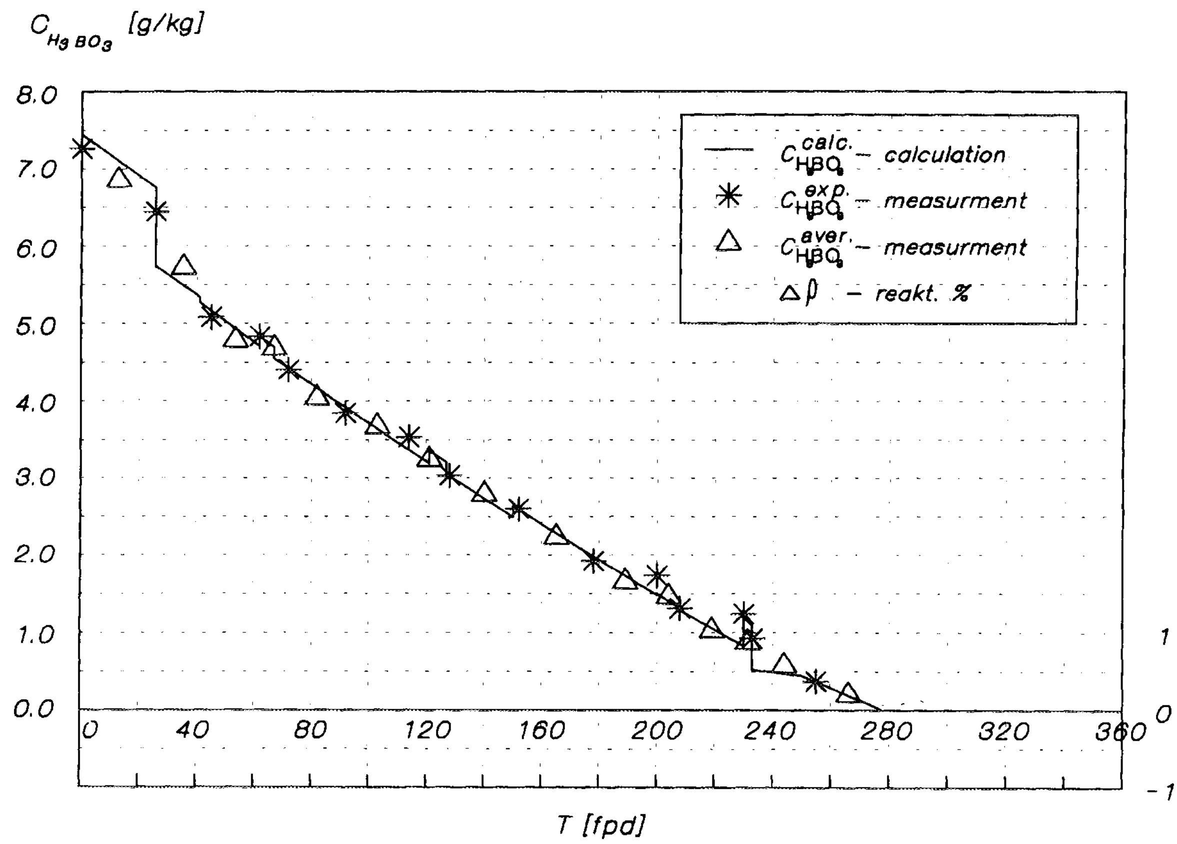
To obtain the measured relative power K_q^{meas} the data from the regular temperature control system have been processed according to the procedure adopted in the plant [4]. Since units 1 - 4 have not got an automatic data acquisition and processing system yet, these have been used without being statistically processed.

Tables 5 and 6 and Fig. 4 show the basic results of the computed K_q^{calc} and the measured K_q^{meas} peaking power factors for Unit 3 and Unit 4. A lot of summarised results are also shown concerning the errors (deviations) of the measurement and the computation. The mean square errors are within 2 - 3% and they can be higher only when the reactor is operated at a power level lower than rated. The maximum positive and negative deviations are within ±7%.

The maximum measured values of the power peaking factors in fuel are predicted by computation with precision better than 4%.



Thermal power, coolant average temperature and requlating control rod group position as a functions of energy production in full power days



Boric acid concentration changes (calculated and measured)

Figure 3 Reactor parameters during 10 cycle of Unit 3.

1A/52	3A/53	Years and Fuel-type / Assembly N at 60° sector
0.825	0.390	K_q^{calc}
-1.49	-8.07	$(K_q^{calc} - K_q^{meas}) \cdot 100$ %

1A/49	1A/50	1A/51
1.152	0.968	0.656
1.46	3.18	-0.14

2A/45	3A/46	2B/47	1A/48
1.153	0.959	0.801	0.721
4.36	-0.93	0.00	-1.03

2A/40	2A/41	1A/42	1A/43	2A/44
1.177	1.129	1.227	1.103	0.658
1.05	-0.28	-1.48	-0.71	-4.34

3A/34	3A/35	3A/36	2A/37	1A/38	1A/39
1.069	1.073	1.009	1.116	1.109	0.730
1.46	0.19	-0.54	-1.43	-0.72	-1.04

2A/27	2A/28	1B/29	3A/30	1A/31	2B/32	1A/33
1.201	1.205	1.139	1.010	1.235	0.841	0.666
2.73	1.81	0.00	-0.54	-1.49	0.00	-0.14

3A/19	3A/20	2A/21	3A/22	2A/23	3A/24	1A/25	3A/26
1.069	1.051	1.205	1.074	1.132	0.967	0.978	0.393
0.53	-0.88	1.81	0.19	-0.28	-0.93	3.18	-8.07

2A/11	3A/12	2A/13	3A/14	2A/15	2A/16	1A/17	1A/18
1.164	1.069	1.201	1.076	1.178	1.155	1.156	0.828
-0.52	0.53	2.73	1.46	1.05	4.36	1.46	-1.49

1B/01	2A/02	3A/03	1B/04	2A/05	2A/06	1B/07	3A/08	1A/09	2B/10
0.875	1.143	1.039	1.159	1.196	1.161	0.873	0.953	1.047	0.432
0.00	0.82	0.29	0.00	1.58	2.26	0.00	0.30	-2.63	0.00

K_{eff}= 0.99516 Power=-100.0 T_{in}= 262.7 C_B= 0.542 H_6=175.00 G= 35000.0

Max + 4.4 / 16 / A2 Disp = 2.4 Max K_q^{meas} & K_q^{calc} = 1.250 1.235 / 31

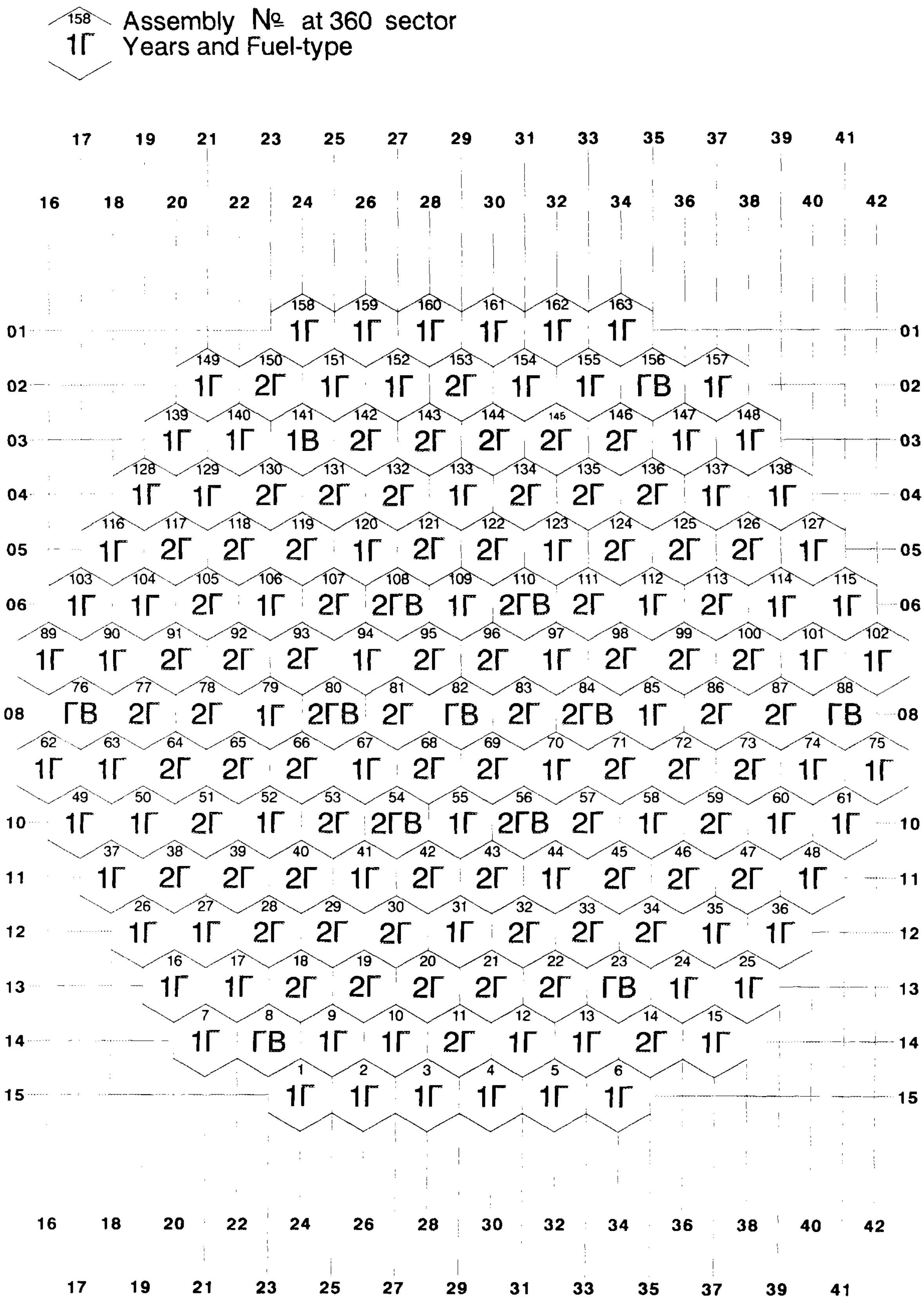
Max - -8.1 / 26 / A3 E-ARK= 0.1 Max K_q^{calc} & K_q^{meas} = 1.235 1.250 / 31

E-ster= 2.0 E-Rad= -2.9 E-Rad 1 2 3 = -1.5 -1.3 -8.1

Err & Disp by Fuel-type and Years

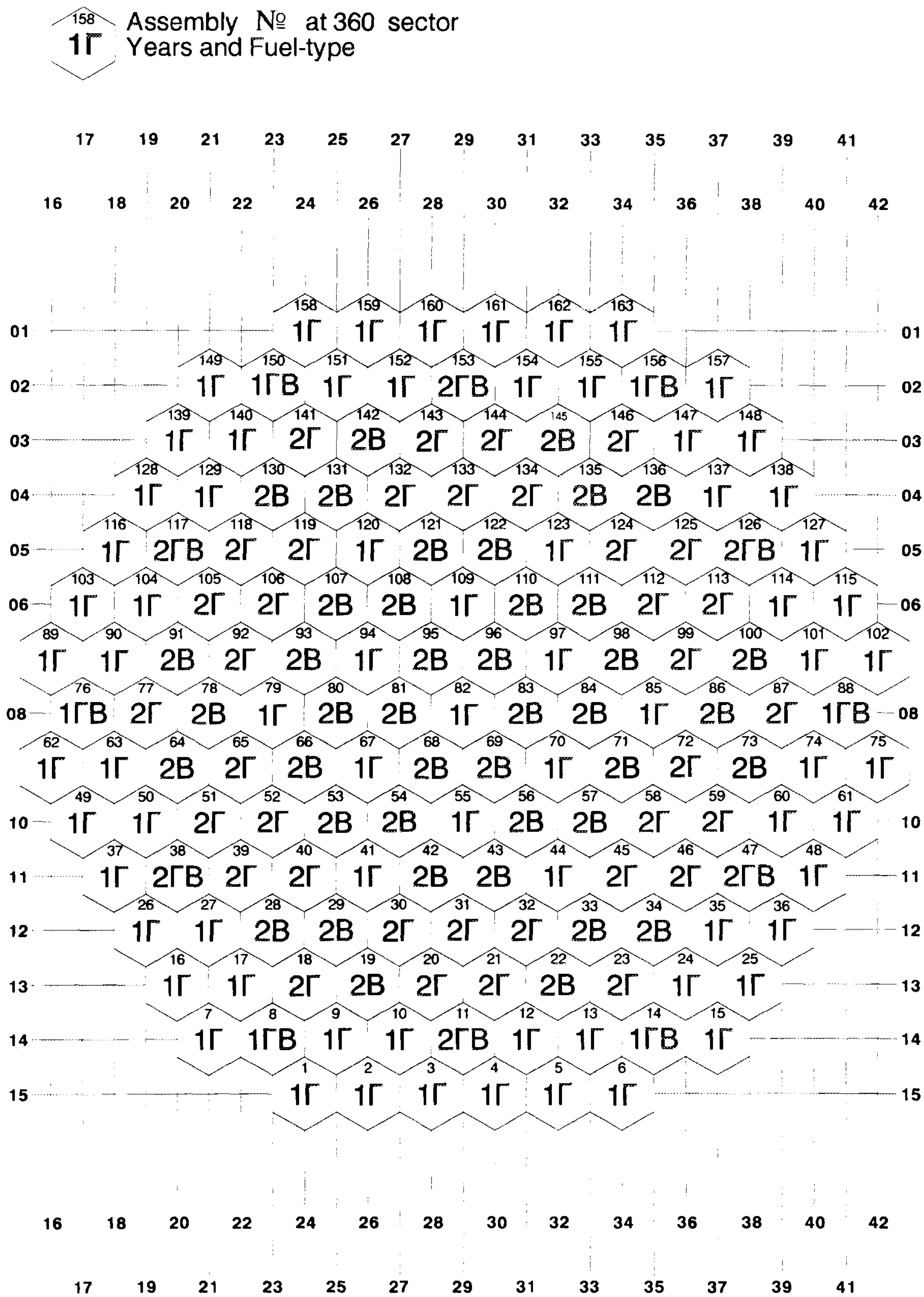
A1: 90.0x -0.2 A2: 96.0x 1.1 A3: 90.0x -1.0 A4: 0.0x 0.0

Figure 4 Distributions of assembly-wise relative power K_q^{calc} calculated by SPPS-1 and absolute error from measurments $(K_q^{calc} - K_q^{meas}) \cdot 100$ % (Cycle 7, Unit 3, FPD=152)



A - enrichment 2.0%
B - enrichment 3.0%
Г - enrichment 3.3%
ГВ - enrichment 3.3% prof. with 3.0%

Figure 5 Unit 5, Cycle 3



A - enrichment 2.0%
B - enrichment 3.0%
Г - enrichment 3.3%
ГВ - enrichment 3.3% prof. with 3.0%

Figure 6 Unit 6, Cycle 2

Table 1 Basic data on the previous 16 fuel cycles carried out on Unit 1

Cycle N	Loaded assemblies: ACA enrich	ACA No.	FWA enrich	FWA No.	Cycle life time FPD	TPAF %	Spec. consumption of ^{235}U	K_q^{calc} / FPD	K_q^{meas} /FPD	Core periph. spec. features	Average burnup of withdrawn assemblies with corresponding enrichment *MWd/kgU* 1.6%	2.4%	3.6%
1	1B 1C	25 12	1A 1B 1C	102 108 102	287.79	0.59		1.300	1.31	1A - 60 1B - 6	9.33	8.73	
2	1B 1C	12 25	1A	102	357.40	0.77		1.324 / 27	1.34 / 30	1A - 60 1C - 6	10.24	22.39	
3	1B	13	1A 1B	102 6	303.60	0.87	1.61	1.255 / 7	1.27 / 16	1A - 60 2B - 6	19.13		32.98
4	1B	12	1A	102	311.70	0.95	1.50	1.268 / 2	1.26 / 12	1A - 60 3B - 6		23.53	32.37
5	1B 1C	13 1	1A 1B	102 6	317.70	0.92	1.54	1.311 / 3	1.24 / 2	1A - 60 2B - 5 1C - 1		28.22	32.01
6	1B	12	1A 1B 1C	96 3 3	308.30	0.95	1.48	1.310 / 2	1.29 / 6	1A - 60 1B - 6	15.58	27.62	31.68
7	1B	13	1A 1B 1C	90 6 6	297.65	0.92	1.51	1.275 / 3	1.28 / 3	1A - 60 1B - 6	19.16	27.97	31.16
8	1B	12	1A 1B 1C	72 36 5	302.70	0.95	1.50	1.278 / 3	1.292 / 6	1A - 60 1B - 6	18.09	26.34	30.70
9	1B	13	1A	102	312.87	0.93	1.51	1.272 / 5		1A - 60 1B - 6	19.04	27.08	30.68
10	1B	13	1A 1B	102 6	329.23	0.94	1.48	1.23 / 25	1.27 / 28	1A - 60 2B - 1 3B - 5		29.16	32.29
11	1B	12	1A	90	310.30	0.95	1.35	1.255 / 8	1.35 /7	1A - 36 2A - 12 4A - 12 2B - 6		27.61	32.67
12	1B	13	1A 1C	90 1	293.70	0.95	1.44	1.278 / 1	1.31 / 1	1A - 24 3A - 12 4A - 24 3B - 6	10.24	28.76	31.56
13	1B	12	1A	84	270.60	0.96	1.45	1.266 / 14	1.27 / 14	KE - 36 1A - 12 4A - 12 3B - 6		24.23	32.80
14	1B	18	1A	90	304.50	0.95	1.43	1.245	1.286	KE - 36 3A - 12 4A - 12 3B - 6		23.10	31.97
15	1B	7	1A 1B	84 3	277.00	0.92	1.40	1.225 / 16.5	1.220 / 16.5	KE - 36 1A - 12 3A - 1 4A - 11 3B - 6		26.63	33.79
16	1B	12	1A 1B	96 6	274.00	0.85	1.68	1.228 / 28	1.256 / 28	KE - 36 1A - 24 3B - 6		27.55	32.31

Key to abbreviations: ACA - Automatic Control Assemblies
WFA - Working Fuel Assemblies
TPAF - Thermal Power Availability Factor
FPD - Full Power Days

A - enrichment 3.6%
B - enrichment 2.4%
C - enrichment 1.6%

Table 2 Basic data on the previous 17 fuel cycles, carried out on Unit 2

cycle N	Loaded assemblies ACA enrich	ACA No.	FWA enrich	FWA No.	Cycle life time FPD	TPAF %	Spec. consumption of ^{235}U	K_q^{calc} / FPD	K_q^{meas} /FPD	Core periph. spec. features	Average burnup of withdrawn assemblies with corresponding enrichment MWd/kgU 1.6%	2.4%	3.6%
1	1A 1C	25 12	1A 1B 1C	102 108 102	290.2	0.82		1.268 / 15	1.260 / 15	1A - 60 1B - 6	9.57		
2	1B	12	1A	72	284.8	0.89		1.237 / 40	1.290 / 40	1A - 60 2B - 6	17.83	20.34	
3	1B	12	1A 1B	96 42	345	0.99	1.63	1.228 / 5	1.252 / 5	1A - 0 1B - 6		26.14	31.37
4	1B	13	1A	102	312.2	0.92	1.51	1.237 / 2	1.264 / 2	1A - 60 1B - 6		29.19	31.68
5	1B	12	1A 1B	66 6	213.5	0.98	1.56	1.278 / 5	1.287 / 5	1A - 60 2B - 6		28.10	30.77
6	1B	13	1A 1B	84 6	290.4	0.97	1.42	1.262 / 2.7		1A - 60 1B - 6		23.76	32.79
7	1B	12	1A 1B	90 12	328.7	0.88	1.38	1.240 / 8.0	1.282 / 8.0	1A - 60 2B - 6		26.39	33.04
8	1B	13	1A 1B	90 6	278	0.94	1.58	1.197 / 6.4	1.203 / 6.4	1A - 60 1B - 6		26.81	33.01
9	1B	12	1A 1B	84 6	298.9	0.92	1.37	1.258 / 8.7	1.250 / 8.7	1A - 60 2B - 6		28.40	32.41
10	1B	13	1A 1B	90 14	339.3	0.95	1.36	1.269 / 3.0	1.269 / 3	1A - 36 4A - 24 3B - 6		25.47	32.10
11	1B	12	1A	102	310.5	0.94	1.51	1.283 / 2	1.313 / 2	1A - 36 4A - 24 2B - 6		25.52	32.73
12	1B	18	1A	96	322	0.91	1.43	1.301 / 10	1.277 / 10	1A - 24 3A - 12 4A - 23 3B - 7		21.89	34.27
13	1B	19	1A	90	320.2	0.79	1.37	1.320 / 8	1.340 / 8	1A - 12 3A - 24 4A - 24 2B - 6		21.23	32.48
14	1B	12	1A 1B	90 4	303.1	0.91	1.42	1.261 / 15	1.29 / 15	KE - 36 1A - 12 3A - 6 4A - 6 3B - 6		23.52	34.82
15	1B	6	1A	90	280.6	0.92	1.43	1.270 / 4	1.250 / 4	KE - 36 1A - 12 3A - 6 4A - 6 3B - 6		28.30	34.29
16	1B	19	1A 1B	90 2	300	0.87	1.48	1.282 / 7.8	1.270 / 7.8	KE - 36 1A - 12 3A - 11 4A - 1 3B - 6		27.65	31.75
17	1B	12	1A 1B	96 6	293.4	0.73	1.57	1.236 / 12.4	1.23 / 12.4	KE - 36 1A - 12 3A - 6 4A - 4 2B - 2 3B - 6		26.87	33.72

Key to abbreviations: see Table 1.

Table 3 Basic data on the previous 12 fuel cycles, carried out on Unit 3

cycle	Loaded assemblies				Cycle life time	TPAF	Spec. consumption of ^{235}U	K_q^{calc} / FPD	K_q^{meas} /FPD	Core periph. spec. features	Average burnup of withdrawn assemblies with corresponding enrichment *MWd/kgU*		
	ACA		FWA										
N	enrich	No.	enrich	No.	FPD	%					1.6%	2.4%	3.6%
1	1C 1B	12 25	1A 1B 1C	102 108 102	406	82		1.340 / 2	1.364 / 2	1A - 60 1B - 6	13.3	-	
2	1B	12	1A 1B	102 6	270.2	92		1.262 / 9	1.211 / 9	1A - 60 2B - 6	-	23.8	
3	1B	13	1A 1B	84 12	289.8	96	1.482	1.240 / 14	1.266 / 14	1A - 60 1B - 6	-	29.1	32.7
4	1B	12	1A 1B	84 7	330.9	97	1.246	1.265 / 7	1.28 / 7	1A - 36 4A - 24 2B - 6	-	26.2	31.4
5	1B	13	1A	102	335.8	95	1.406	-	-	1A - 36 4A - 24 2B - 6	-	26.6	33.8
6	1B	18	1A	96	293.8	86	1.568	1.278 / 58	1.296 / 58	1A - 24 3A - 12 4A - 24 3B - 6	-	21.9	33.1
7	1B	19	1A	90	300.5	95	1.457	1.258 / 5	1.270 / 5	1A - 36 2A - 6 3A - 12 2B - 6	-	19.5	32.8
8	1B	12	1A	90	326.4	96	1.281	1.265 / 5	1.282 / 5	1A - 24 3A - 30 3B - 6	-	22.1	32.9
9	1B	19	1A	96	322.5	95	1.437	1.264 / 5	1.280 / 5	1A - 30 3A - 24 3B - 6		24.0	34.8
10	1B 1A	6 12	1A	96	292.1	88	1.636	1.289 / 26	1.280 / 26	1A - 30 3A - 24 2B - 6	-	24.7	35.1
11	1B	6	1A	84	310.4	82	1.209	1.259 / 9	1.242 / 9	1A - 30 3A - 24 3B - 6	-	26.0	34.1
12	1B 1A	7 12	1A 1B	90 7	330.5	69	1.437	1.275 / 6	1.270 / 6	1A - 24 3A - 33 4B - 3	-	-	34.7

Key to abbreviations: see Table 1.

The maximum measured values of the power peaking factors in fuel are predicted by computation with precision better than 4%.

The systematic analyses conducted on the power distribution in the cores of Units 1 - 4 show that it is possible to ensure efficient operation without violation of the peaking factors permissible limits.

We came to the conclusion, however, that the accuracy of the computation is commensurate with the operation measurements accuracy. The forthcoming implementation of the automatic parameter monitoring systems for the VVER-440 reactor cores in Kozloduy NPP will intensify the reactor physics analyses and will provide a better systemizaton of operational data.

Units 5 and 6 of Kozloduy NPP utilize VVER-1000 reactors of the third generation which have incorporated higher techical-economical indicators. Compared to the VVER-440 they have increased fuel power density and core sizes, the electric power has also been increased to 1000 *MW*, the mechanical reactivity control system makes use of cluster assemblies, the in-core monitoring system has been improved.

For the first core loadings of VVER-1000 a two-year fuel cycle has been applied, the fuel having an average fuel enrichment level of 3.3% and average discharge burn-up 28.5 *MWd/kgU*. The same was applied for Unit 5 and 6 reactors.

Unit 5 was put into service in November 1987. At present it has been running its 4th fuel cycle. The basic operational characteristics of the core loadings of Unit 5 are shown in Table 7. For 4th fuel cycle the computed data for the cycle life time and the average burn-up are given. The first loading was carried out according to design. At the end of the 2nd fuel cycle a power effect was used in the course of 33 effective days. For the 3rd fuel cycle (Fig. 5) a low-leakage core loading pattern has been partially applied and provision was made for a negative temperature effect at the beginning of

Table 4 Basic data on the previous 12 fuel cycles, carried out on Unit 4

cycle	Loaded assemblies				Cycle life time	TPAF	Spec. consumption of ^{235}U	K_q^{calc} / FPD	K_q^{meas} /FPD	Core periph. spec. features	Average burnup of withdrawn assemblies with corresponding enrichment *MWd/kgU*		
	ACA		FWA										
N	enrich	No.	enrich	No.	FPD	%					1.6%	2.4%	3.6%
1	1B 1C	25 12	1A 1B 1C	102 108 102	390.0	0.90		1.276 / 11.4	1.298 / 11.4	1A-60 1B- 6	12.8		
2	1B	12	1A	102	241.2	0.95		1.271 / 16.8	1.300 / 16.8	1A-60 2B - 6		22.2	
3	1B	13	1A 1B	90 6	348.9	0.97	1.26	1.358 / 7.5	1.346 / 7.5	1A-36 2A-12 1B - 6 3B -12		26.5	33.2
4	1B	12	1A	102	313.7	0.96	1.50	1.361 / 9	1.270 / 9	1A-36 2A-12 4A-12 2B - 6		25.6	32.7
5	1B	19	1A	96	354.4	0.96	1.31	1.306 / 3.9	1.240 / 3.9	1A-24 3A-12 4A-24 3B- 6		21.3	33.9
6	1B	18	1A	96	309.5	0.96	1.49	1.307 / 19	1.301 / 19	1A-12 3A-30 4A-18 2B - 6		21.1	34.1
7	1B 1A	7	1A	90	244.0	0.90	1.66	1.316 / 20.9	1.288 / 20.9	1A-12 3A-18 4A-30 2B - 6		20.3	33.2
8	1B	6	1A	78	350.4	0.91	1.00	1.292 / 19	1.301 / 19	1A-12 4A-48 3B - 6		26.4	34.4
9	1A	12	1C 1B 1A	1 6 90	295.0	0.95	1.54	1.338 / 2.7	1.339 / 2.7	1A-12 4A-48 3B - 6		25.8	35.1
10	1B 1A	7 12	1A	90	309.3	0.87	1.47	1.335 / 3.8	1.270 / 3.8	1A-12 4A-48 3B - 6	13.9	24.2	36.1
11	1B	6	1A	90	332.0	0.64	1.21	1.326 / 5.4	1.310 / 5.4	1A-12 3A-12 4A-36 2B - 6		29.7	36.5
12	1A	12	1A	96	297.4		1.49	1.337		1A-12 3A-12 4A-36 3B - 6		21.7	34.42

Key to abbreviations: see Table 1.

the cycle by changing the places of 6th and 8th group of the control cluster assemblies.

The reactor of Unit 6 reached criticality level on 29th May 1991. At present it has been running its 2nd fuel cycle (Fig. 6). The basic core loading characteristics of Unit 6 are shown in Table 8.

The core loading option of VVER-1000 (Unit 5 and 6) are chosen by means of the computer code developed by Kurchatov Nuclear Research Institute, Russia (BIPR-7) [5] and by VNIIAES (ALBOM, PROROC) [6, 7]

As can be seen on Tables 8 and 9, the fuel cycles of Units 5 and 6 do not exceed the admissible peaking factors values and the burn-up values.

Table 9 and 10 show the results of comparisons made between computed and operation values of the relative fuel power distributions for some moments of the fuel cycles. The operation values are obtained by processing the current of the sensors located in the neutron measurement channels. The VMPO system, performing this processing, shows greater errors for these fuel assemblies that have no neutron measurement channels. The maximum error in the shown comparisons apply to such assemblies. The errors in fuel assemblies having maximum energy release do not exceed 6%.

The three-year fuel cycle proved to be more efficient for VVER-1000. The average enrichment level in this type of cycle is 4.4% and the average computed burn-up - 43 *MWd/kgU*. It is foreseen Units 5 and 6 also to adopt the 3-year fuel cycle. The latter is more economical and allows better utilization of fuel. The fuel cost within the energy cost is reduced approximately by 15% compared to that relative of the 2-year cycle. The Research and Development department of Kozloduy NPP -2 carried out computations and analyses to provide the basis for the adoption of the 3-year fuel cycle.

Conclusions

The given results enable us to make some conclusions and suggestions, as follows.

1. The core loading option chosen for VVERs in Kozloduy NPP (Units 1 - 6) has led to efficient utilization without violation of nuclear safety criteria.
2. By utilizing dummies and low-leakage core-loading patterns for VVER-440 cores (Units 1- 4) we can reach effective reduction of the reactor presure vessel irradiation. The increased enrichment level of the fuel portion of the control rods and the improved characteristics of Unit 3 and 4 fuel cycles have lead to maintaining the design duration of the fuel cycles at a reduced number of assemblies by a factor 5 - 10%
3. The improvement of the in-core monitoring system for VVERs-440 (V-230), the implementation of on-line simulation of the processes occurring in the core together with the intended implementation of new more accurate computer codes will enables us to utilize fuel with higher enrichment and implement the 4-year fuel cycle.
4. For the VVER-1000s the adoption of the 3-year fuel cycle utilizing fuel of 4.4% initial enrichment will ensure better core loading characteristics and more effective fuel utilization.
5. The improvement of the in-core monitoring system for the VVER-1000 will allow to better the accuracy of the operation data and will considerably increase the reliability of nuclear fuel control. This will create conditions to implement improved fuel with a new type of absorbers and the more effective low-leakage core loading patterns.

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Table 5 Main results of SPPS-1 calculation, compared with the measured assembly-wise power distributions $K_q(n)$ for different moments [FPD] of the Cycle 11, Unit 3. Error $K_q(n) = (K_q^{calc}(n) - K_q^{meas}(n)) \cdot 100$ %; n - location number in 360° sector.

Time, FPD	3.5	5.4	209.9
Parameters of conditions in the core	H_{VI}=197*cm* T_{in}=264.2°C N_T = 90 % C_B=1.007*g/kg*	H_{VI}=193*cm* T_{in}=263°C N_T = 100% C_B= 0.963*g/kg*	H_{VI}=200*cm* T_{in}=262.3°C N_T = 100% C_B=0.180*g/kg*
K_{eff}	1.0018	1.0020	0.9970
ΔK_{eff} - calcul error [%]	0.0018	0.0020	-0.0030
Max. K_q			
K_q^{meas} / K_q^{calc} (N_{ass})	1.260 / 1.267 (22)	1.253 / 1.260 (22)	1.212 /1.208 (51)
K_q^{calc} / K_q^{meas} (N_{ass})	1.267 / 1.260 (22)	1.260 / 1.253 (22)	1.213 / 1.166 (228)
Maximum positive error	6.0	7.3	4.7
N of assembly (type)	82 (2A)	82 (2A)	228 (2A)
Maximum negative error	- 6.4	- 5.2	- 3.1
N of assembly (type)	1 (3A)	20 (1A)	230 (2A)
σ mean-square error	2.7	2.7	1.8
mean error 1A (84)	0.1	- 0.4	1.1
mean error 2A (96)	1.8	2.1	0.4
mean error 3A (95)	- 1.9	- 1.7	- 1.4
error peripheral assemblies	-2.4	-2.4	-0.2

Table 6 **Main results of SPPS-1 calculation, compared with the measured assembly-wise power distributions $K_q(n)$ for different moments [FPD] of the Cycle 11, Unit 4. Error $K_q(n) = (K_q^{calc}(n) - K_q^{meas}(n)) \cdot 100$ %; n - location number in 360° sector.**

Time, FPD	5.44	33.5	67.5	275
Parameters of conditions in the core	H_{VI}=181cm T_{in}=262.1°C N_T = 100% C_B=1.06 *g/kg*	H_{VI}=191cm T_{in}=261.7°C N_T = 100% C_B=0.91 *g/kg*	H_{VI}=193cm T_{in}=262.4°C N_T = 100% C_B=0.79 *g/kg*	H_{VI}=180cm T_{in}=264.4°C N_T = 100% C_B=0.068 *g/kg*
K_{eff}	0.9989	1.0011	0.9995	0.9952
ΔK_{eff} - calcul error [%]	-0.11	0.11	-0.05	-0.48
Max. K_q				
K_q^{meas} / K_q^{calc} (N_{ass})	1.314 / 1.327 (48)	1.315 / 1.318 (305)	1.304 / 1.312 (305)	1.295 / 1.276 (305)
K_q^{calc} / K_q^{meas} (N_{ass})	1.327 / 1.314 (48)	1.320 / 1.315 (48)	1.314 / 1.304 (48)	1.277 / 1.295 (142)
Maximum positive error	4.8	6.2	5.4	5.8
N of assembly (type)	52 (1A)	72 (4A)	90 (4A)	91 (1A)
Maximum negative error	- 3.6	- 6.1	- 4.7	- 6.8
N of assembly (type)	78 (3A)	257 (3A)	272 (3A)	272 (3A)
σ mean-square error	1.8	2.5	2.3	2.8
mean error 1A (78)	0.7	0.0	0.4	1.4
mean error 2A (90)	0.5	0.1	0.0	0.5
mean error 3A (96)	-1.1	-1.7	-1.8	-2.7
mean error 4A (48)	-1.3	3.6	3.4	2.4
mean error 3B (6)	-1.3	-0.5	-0.9	-3.4
error peripherial assemblies	0.6	3.1	3.0	1.6

Table 7 **Main cycle characteristics, Unit 5**

Cycle	Loaded assemblies		Cycle lifetime	Crit. C_{H3BO3}	Relative power distribution factors				Average burnup of withdrawn assemblies
					BOC		EOC		
	type	No.	FPD	*g/kg*	Kq	Kv	Kq	Kv	*[MWd/kgU]*
1	A B Г ГВ	79 42 36 6	298.6	6.53	1.31	1.81	1.20	1.35	12.670 15.059
2*	A B Г ГВ	- - 73 6	325.0	6.95	1.25	1.56	1.20	1.33	23.786 28.180 24.742 26.806
3	B Г ГВ	1 78 6	317.7	7.77	1.25	1.63	1.18	1.31	28.180 24.742 26.806
4	B Г ГВ	- 72 6	308.6	7.49	1.27	1.62	1.20	1.33	27.170 27.040 24.540

* stretch-out

Key to abbreviations: A - enrichment 2.0%; B - enrichment 3.0%; Г - enrichment 3.3%; ГВ - enrichment 3.3% prof. with 3.0%

Table 8 **Main cycle characteristics, Unit 6** (see abbreviations in Table 7)

Cycle	Loaded assemblies		Cycle lifetime FPD	Crit. C_{H3BO3} *g/kg*	Relative power distribution factors				Average burnup of withdrawn assemblies *[MWd/kgU]*
					BOC		EOC		
	type	No.			Kq	Kv	Kq	Kv	
1	A B Г ГВ	79 42 36 6	301.6	6.53	1.31	1.81	1.20	1.35	12.865 15.360
2	B Г ГВ	- 73 6	298.2	7.13	1.29	1.65	1.19	1.31	27.280 23.505 26.230

Table 9 Main results of BIPR-7 calculation, compared with measured assembly-wise power distributions Kq(n) for different moments (FPD) of the Cycle 3, Unit 5.

Parameters of conditions in the core	T, FPD			
	75	153	208	275
H_{10}, *cm*	280	273	280	278
T_{in}, °C	283.2	285.5	285.2	283.5
N_T, %	79	94	95	77
C_{H3BO3}, meas., *g/kg*	6.01	3.72	2.54	1.24
C_{H3BO3}, calc., *g/kg*	6.00	3.88	2.54	1.32
$Kq_{max}^{meas}/Kq^{calc}$	1.26 / 1.23	1.23 / 1.20	1.19 / 1.19	1.19 / 1.18
at assembly No.	60	129	129	10
$Kq_{max}^{calc}/Kq^{meas}$	1.23 / 1.26	1.20 / 1.23	1.19 / 1.19	1.18 / 1.19
at assembly No.	60	129	129	10
Max. (+) difference	8.70	9.74	6.80	7.33
N of assembly (type)	123 (Г)	44 (Г)	1 (Г)	123 (Г)
Max. (-) difference	5.43	4.63	5.83	4.57
N of assembly (type)	11 (Г)	153 (Г)	14 (Г)	11 (Г)
Root mean square (all assemblies)	2.79	2.80	2.30	2.89
Root mean square (FA with SPD)	1.43	1.57	2.54	2.97
Root mean square (FA w/o SPD)	3.10	3.11	2.11	2.74

Here and in Table 10 below:

$$\text{Difference} = \frac{Kq_{(n)}^{calc} - Kq_{(n)}^{meas}}{Kq_{(n)}^{meas}} \, 100\,[\%]$$

n - location number in 360° sector

Table 10 Main results of BIPR-7 calculation, compared with measured assembly-wise power distributions Kq(n) for different moments (FPD) of the Cycle 2, Unit 6.

Parameters of conditions in the core	T, FPD			
	24	42.5	123.8	221.7
H_{10}, *cm*	279	278	274	268
T_{in}, °C	287.5	286.9	285.1	282.5
N_T, %	99	93	79	51
C_{H3BO3}, meas., *g/kg*	6.14	5.82	4.22	2.42
C_{H3BO3}, calc., *g/kg*	6.32	5.97	4.02	2.51
$Kq_{max}^{meas}/Kq^{calc}$	1.322 / 1.251	1.316 / 1.251	1.260 / 1.241	1.230 / 1.226
at assembly No.	129	129	35	152
$Kq_{max}^{calc}/Kq^{meas}$	1.251 / 1.322	1.251 / 1.316	1.241 / 1.260	1.226 / 1.230
at assembly No.	129	129	35	152
Max. (+) difference	7.23	8.25	5.97	11.01
N of assembliy (type)	20 (Г)	41 (Г)	120 (Г)	44 (Г)
Max. (-) difference	6.58	6.52	6.81	8.08
N of assembly (type)	4 (Г)	4 (Г)	56 (B)	93 (B)
Root mean square (all assemblies)	3.68	3.48	2.60	4.37
Root mean square (FA with SPD)	2.80	2.08	1.98	4.02
Root mean square (FA w/o SPD)	3.49	3.26	2.30	4.16

The Evolution of the Fuel Cycle in the Dukovany NPP

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There are 4 units with VVER-440, type V-213 (modernised design) reactors, operated at the Dukovany NPP. Thirty five cycles were realised during 10 years of operation (see Table 1).

The strategy of fuel reloadings has been changed from out-in schemes to low leakage patterns (LLP). Our reloading patterns are limited by the primary circuit flow rate (see Table 2) and by other phisical limitations, listed below. We know that the limitation (h) is very strict and we have been trying to change this one. Our global goal is to introduce the linear pin power limitation, but it is conditioned by new core monitoring system instalation.

Table 1 NPP Dukovany (4 units VVER-440)

Unit	First start-up	Number of realised cycles
1	1985	10
2	1986	9
3	1986	8
4	1987	8

Table 2 Primary Circuit Flow Rates and Mean Reactor Heatings

Unit	Flow rate, m^3/h	Mean reactor heat-ups
1	40 397	30.2
2	39 366	30.2
3	40 680	29.8
4	41 200	29.5

Table 3 Heat-Up of the Assemblies

Assembly type	Assembly heat-up, *K*	Number of neighbour assemblies
central	42	6
A	33	5
B	30	4
C	25	3

The Main Physical Limits

The main limits and conditions for the Dukovany NPP operation are listed below:

(a) $k_q < 1.35$;

(b) $k_v < 1.9$ for Monitoring System HINDUKUS,
$k_v < 2.1$ for Monitoring System VK-3;

(c) the temperature reactivity coefficient < 0;

(d) the undercriticality after scram with 1 maximal worth CR in upper position and with decreasing of average core temparature by 40 K - 2%;

(e) limit of rod ejection reactivity - 1.1 k_{ef};

(f) the reactor inlet temperature < 269° *C*;

(g) the assembly oulet temperature < 312° *C*;

(h) The heat-up of assemblies is shown in Table 3.

There are other requirements during loading pattern design process:

a) Refuelings must not be realized during winter and they must not overlap => the length of cycle with outage has to be about 1 year (it is about 310 FPD).

b) To produce as a small number of spent fuel assemblies as it is possible.

The burnup fuel production in the Dukovany NPP is shown in Tables 4 and 5. There are our past results and our plans to the future presented. Our main goal is to go step by step to full 4-years fuel cycle. The mean burnup of 4-years fuel will be about 40 *MWd/kgU*.

Table 4 Numbers of assemblies which were used 4 cycles in the core

Number of cycle	EDU-1	EDU-2	EDU-3	EDU-4
4	0	0	12	12
5	12	0	18	18
6	12	6	18	30
7	24	18	30	36
8	24	30	42	42
9	36	36	54/48	54/48
10	42	48/42	66/54	66/54
11	54/48	60/48	78/66	78/66
12	66/54	72/54	78	78
13	78/66	78/66	78	78
14	78	78	78	78
15	78	78	78	78

Table 5 **Numbers of assemblies which were discharged during refuelings**

Number of cycle	EDU-1	EDU-2	EDU-3	EDU-4
1	114	114	114	114
2	114	114	114	114
3	121	121	114	109
4	102	114	109	108
5	114	108	108	102
6	109	109	102	103
7	102	102	103	102
8	102	102	102	90/96
9	102	96/102	90/96	90/96
10	96/90	96/90	96/90	96/90
11	96/90	96/90	96/90	90
12	96/90	96/90	90	90
13	90	90	90	90
14	90	90	90	90

Table 6 **Average burnup of assemblies with 3.6% enrichment after 3/4 years of operation, MWd/kgU**

Number of cycle	EDU-1	EDU-2	EDU-3	EDU-4
3	29.8/ -	29.8/ -	29.7/ -	30.3/ -
4	29.1/ -	29.2/ -	29.4/28.2	30.9/31.9
5	28.1/30.7	27.9/ -	29.3/30.2	32.2/34.1
6	28.8/29.6	31.2/29.5	30.8/32.7	32.6/33.9
7	31.2/32.8	32.2/33.6	31.7/34.5	33.1/36.1
8	31.5/32.5	32.8/34.1	32.7/37.3	32.0/37.5
9	31.8/35.6	32.0/36.8		
10	32.0/35.8			
11	31.8/38.1			
12	32.2/38.5			
13	31.7/39.3			
14	32.0/39.4			

Practical Experience with the Fuel Monitoring at Ducovany NPP

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1 Fuel State Monitoring During Power Operation

The basic means for the fuel state monitoring are concentrated at the operational chemistry department. Their activities are based especialy on following the primary coolant activity during cycles in view of the fact that its value is limited by the main opertional regulation - "Limits and Conditions" [1]. Other tasks are to identify the fuel damage initiation phase, and to find a suitable reactor power mode to restrict the defect development and fission products release to the primary coolant for the rest of the cycle as well as provide with data necessary for a correct decision towards a fuel inspection during the refueling.

1.1 Primary Coolant Gamma Spectrometry Monitoring

All reactor units are equipped by primary coolant continuos monitors which can be described as follows.

The on-line primary coolant gamma spectrometry monitor is located under the by-pass pipelines of the water cleaning system (SVO-1). The spectrometry channel consists of a high purity Ge semiconductor detector, a multichannel analyzer and a collimator facility. The collimator was developed with respect to the water purification technology. It is adjusted to its both operation modes, i.e. both with one-loop operation as well as with two loops of SVO-1 in operation. The collimator has adjustable widths of the slot and a possibility to monitor separately a single SVO loop. The control of the facility is done remotely and automatically by a computer program, with the aim to set such a slot that the dead time is optimal with regard to the instantaneous primary coolant activity. The program is implemented directly in the software of the chemistry department superior computer network. The control can be done also manually by the operator, e.g. in case of periodic recalibration.

The system enables to carry out the gamma spectra measurements within the whole range of conditions set up by the design, i.e. comfortable determination of the equilibrium activities of practically all significant gaseous fission products for energies from 80 *keV* to 2 *MeV* and reaching even up the level of "Limits and Conditions" for operation. The sensitivity and accuracy of the measurements are acceptable.

The measurement of volatile, chemically reactive nuclides is at the steady-state strongly distorted by the sorption on the tube surface at the measuring point and thus cannot be used to evaluate tightness of fuel elements during the reactor operation. However, the system enables to perform comfortable and reliable measurements of transients. The corresponded effects have direct impact on a short-time release of the fission product cumulated activity into the primary coolant, when activity can arrise by several orders of magnitude (so-called spikes, typically ^{133}Xe, ^{135}Xe, ^{131}I, ^{132}I) and the above mentioned distortion can be considered negligible. Obtain rates of activity reflect the relations inside the damaged fuel element.

On-line activity measurements give reliably indication of a defect origin with the ^{133}Xe activity level of the order of tens *kBq/l*. The coolant activity in case of a fully developed deffect can reach up to 3000 *kBq/l* of ^{133}Xe. All measured activities are corrected to the transport time from the core to the measuring point (about 300*s*).

1.2 Gamma Spectrometry Monitoring Software Package

A special software package was developed at NRI Rez and CHEMCOMEX Ltd. for processing primary coolant gamma spectrometry data. Nowadays it is created by C++ language object-oriented versions of two programs - KGO and PEPA, which have been installed into the chemistry information system.

The aim of KGO is estimation the number of the damaged fuel elements and the extent of their damage based on the instantaneous noble gases activities during the steady-state reactor operation. The evaluation is done under these conditions: the power level is higher than 50% the nominal power and the reactor has been operated on a constant power level for at least 5 half-times of ^{133}Xe. The activities of Xe and Kr, namely ^{133}Xe, ^{135}Xe, ^{137}Xe, ^{138}Xe, ^{87}Kr, ^{88}Kr and ^{89}Kr are used for evaluation of the number of "leakers". The independence of the gaseous fission products on local physical-chemical processes in the primary curcuit, as well as the variety o half-lives of the measured nuclides enables to identify defects with various release rate of cumulated activity into the coolant through a defect in the cladding. The model considers 3 types of defect (small, medium and large) that are characterized by the rate release constants. The code uses the nonlinear regression method to find such a combination of single types of damaged fuel elements, that gives the minimum sum of the squares of the differences between the predicted and measured activity levels. The program estimates the total number of damaged fuel elements which is relatively stable. The program performs the classification to individual groups, too. KGO is activated at each finished measurement (at least once the shift).

The aim of PEPA is to predict the radiation set-up development, i.e. the activity levels of cca 20 radiologicaly significant nuclides in the primary coolant for the assumed reactor power mode. As an input serve modified (due to their stochastic character) outputs of KGO in the last week period of time, and supposed time dependence of power ramps, coolant pressure and water exchange. The code solves a system of balance equations for the selected nuclides. The real time dependence of the release rate during the transient is approximated by the set time intervals with constant course. Unacceptable predicted release rates can cause some modification of the planned power mode (usually the speed of power increase is lower).

The burnup of damaged fuel element is estimated based on a simple relation between burnup and the ^{134}Cs and ^{137}Cs activity ratio at a power ramp. This method is usable only in case of one large defect or more defects with the same burnup.

2 Fuel State Inspections During Outages

Based on spike occurences and stabilized activity level at least 1000 *kBq/l* for ^{133}Xe or ^{135}Xe, as well as on by the KGO program fuel leaking element indications during a fuel cycle the chemistry department estimates the chance to discover the untight fuel assemblies after the shutdown. Some safety and economical aspects are also taken in account for making decision concerning a fuel inspection during the outage.

As usual a two-step inspection procedure is performed:

In the first step the SIEMENS equipment is used (in-core sipping test). The inspection is done in the core itself, without moving the fuel. The whole groups of 8 assemblies, to which the bell is fixed, are checked step by step. The force causing the transfer of mass (activity release from the defect) is the increase of fuel temperature and a small pressure ramp. The presence of a defect is indicated based on the removed coolant sample activity increase (activity of ^{131}I, ^{137}Cs, ^{134}Cs; alpha activity of ^{239}Np). The concrete assembly, causing the significant activity increase, is then found with the help of certain sipping test step combination. The sipping test enables the inspection of all assemblies of the core during cca 2 days and to find a group of "suspected" assemblies.

In the second step a group of "suspected" assemblies together with a few unambiguously non-damaged assemblies are checked by the well-known standard individual gas-tight can test in the spent fuel pool. An assembly is excluded from further operation when ^{131}I sample activity violates the limit $3.7 \cdot 10^6$ *Bq/l* [1] or it meets 3σ statistical test.

3 Fuel Damage Predictions and Reality

The success of fuel damage predictions during cycles can be illustrated on two examples from the operational history of the Dukovany units.

During 5th and 6th fuel cycle of 1st unit the first signs of untightness occured. On-line activity monitoring showed microspikes during power changes. However the fission products activity levels at the stabilized reactor operation were deeply under the level quantifiable by the KGO program. The effect was opened during 7th cycle when a slow increase of activities of ^{133}Xe and ^{135}Xe was observed and after the following power change the spike became more distinct. After further power changes the defect became completely open and the gaseous fission products activity was stabilized at the level of 2000 to 3000 *kBq/l* for ^{133}Xe. At that time the automatic evaluation of the number of damaged fuel elements varied between 1 and 3, and the burnup estimates based on ^{134}Cs/^{137}Cs activity ratio at the moments of the spike maximums corresponded to the values characteristic for the assemblies serving for the second cycle.

The in-core sipping test performed during the shut-down before 8th fuel cycle indicated 3 "suspected" assemblies which were situated in neighbouring positions. Only one of them was confirmed as damaged after the inspection in the gas-tight can. Its burnup complied quite well with the predicted one (the error was less than 10%). This assembly was put out of futher operation because the ^{131}I sample activity was close to the above mentioned limit. The higher sipping test results of the other two assemblies were caused by the influence of the damaged assembly only.

During 6th cycle of 2nd unit an increase of some isotopes activities signalized probably a microdefect in a fuel element cladding which could not be surely found by any available method. During 7th cycle the KGO program evalueted 1 small defect time to time, the probable burnup of the damaged assembly corresponded to the third cycle in operation. The sipping test after 7th cycle discovered no "suspected" assembly, but in spite of this fact 16 assemblies with the highest activity results were tested in gas-tight can. Only 2 of them showed the ^{131}I sample activities higher then 10^4 *Bq/l*, i.e. deeply under the limit value. During 8th and 9th cycles analysis of radiation situation shows that the defect character remains unchanged at the level of a small gaseous untightness.

We can state on this occasion that the above described case of the 1st unit was the only one during the whole plant history when an assembly was elimitated from further operation due to its untightness. This fact could be explained by a relatively mild and steady-state mode from point of view of fuel straining and by high fuel production quality.

4 Prevention Against Fuel Danage Caused by PCI

Another type of software has been developed at NRI and CHEMCOMEX Ltd. for on-line resp. preliminary analyses of transients that could have influence on fuel cladding integrity.

A program module PES-Bu is installed in the in-core monitoring systems at all units. It serves for on-line evaluation of assemblywise power margins to the dehermetization of fuel elements as a result of pellet-cladding interaction during the whole fuel cycle. It calculates maximal permissible reactor power increase at the moment. The in-core monitoring system provides with all necessary data (power and burnup distribution). As an output is obtained a list of power margins in all symmetry groups of the core. It can be printed for the personnel on request or automatically when the critical power level is reached in any symmetry group. The outputs of PES-Bu are relevant especially during the hot start-up after the outage for refueling.

An off-line (on PC/AT at the reactor physics department) version of the program PES is used for verification of planned power manoevreings from the same point of view, i.e. PCl. The verification with this version of PES is relevant in some special cases of transients. As an input serve data generated by the in-core monitoring system (power and burnup distributions) and its above mentioned on-line segment PES-Bu (conditioned power distribution) related to a time point before the transient on the one hand, and the planned course of reactor power during the transient on the other hand. However, such cases of transients don't often occur at our plant. The results of the verification show in rare cases a necessity to make a correction of the planned course of reactor power.

However, we are aware of a decreased reliability of gained results due to some uncertainties, especially of the accuracy of fuel element "conditioning" and "deconditioning" kinetics description, the determination of maximal permissible local power ramp as well as power and burnup 3D distribution.

Conclusion

In spite of the above mentioned aspects and experience the utilization of the SW package KGO-PES-PEPA [2] has been standardized and legalized in a special operational regulation [3]. It defines obligatory conditions and rules for activities and cooperation of interested plant sections in the process of fuel state monitoring and fuel damage prevention. We believe it contributes to a more cultural exploitation of the core from point of view of fuel integrity maintainance.

We are in our belief that the significance of the above described means will increase in respect of the possibility to new modern fuel from different fuel vendors under strong requirement for high fuel burn up and of the adaption to a not steady-state power mode in the future.

References

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VVER Fuel Performance, Development, QA and Future Prospects at Loviisa NPS

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1 Introduction

Imatran Voima Oy operates two VVER-440 type (Russian PWR) nuclear power units located in Loviisa, Finland. These two units have some major differences compared to Western PWR's or to the other VVER-440 plants. The primary coolant circuit of VVER-440 plants comprises of six loops with one horizontally oriented steam generator in each loop. The Western design instrumentation and control system and containment contribute to the fact, that Loviisa plant fulfills the rigid internationally accepted safety requirements. The units of Loviisa NPS were put into operation in 1977 and in 1980, currently 18 and 15 cycle are passing.

The fuel for Loviisa NPS is manufactured by A/O Mashinostroitelny Zavod and supplied by Techsnabexport (TENEX). The main differences of the fuel compared to Western PWR fuel are application of triangular lattice, hexagonal shroud tube around the assembly and particular cladding material (Zr1%Nb-alloy). The types of fuel assemblies are divided into two classes, fixed fuel assemblies and fuel followers of control rods.

The essential points of Loviisa fuel service are presented in this paper. The steps in developing the performance values together with supporting research work and analyses as well as the performed design changes and follow up of the in-reactor behaviour of the lead fuel assemblies are described.

The quality assurance of the fuel procurement and operation are explained in chapter 4. Finally the future plans in the fuel area are visualized.

2 Performance

2.1 Power and Burnup

The maximum linear heat rate in Loviisa reactors is 325 *W/cm* for fresh fuel rods. This is set by the loss of coolant accident limit. This limit has not been re-evaluated although the latest LOCA-analyses suggest that it might be somewhat over-conservative. The maximum linear heat is dependent on burnup and decreases according to curve illustrated in Figure 1. The burnup dependent upper limit for linear heat rate is based on code calculations indicating the start up of thermal fission gas release and positive feedback of the release above the allowed linear power. The code analysis was made with Gapcon-Thermal-2 (GT2) program in 1987. The GT2 code is not specially validated against VVER-data, but the cladding creep correlation in the code is specially developed for Zr1%Nb material and the fuel centreline temperature predictions are compared to some measurements in low burnup test rods of VVER-type.

In 1993 a new fuel performance code was available for evaluating the fuel performance. The validation of the code to VVER fuel was made at Technical Research Centre of Finland, based on data from irradiations of test rods in research reactor MR in Kurchatov Institute and on PIE data from two precharacterized fuel assemblies irradiated to high burnup in Loviisa-2 reactor in 1987-1991. An example of comparison between ENIGMA code calculations and measured data is shown in Figure 2. Specific cladding creep model and some material properties of VVER fuel were implemented into the code. Although ENIGMA is written for Western fuels it offers an opportunity to analyze VVER fuel behaviour better than GT2-code, since in ENIGMA some models are of more mechanistic nature than in GT2. The ENIGMA calculations show that there is 8% margin in maximum allowed linear power before the fuel temperature reaches the range where positive feedback from enhanced fission gas release starts, even if conservative rod parameters were used.

The design average linear heat rate of VVER-440 plant is 133 *W/cm*. In Loviisa the reactor core was reduced in 1980 because of the need to slow down the irradiation induced embrittlement of the pressure vessel. The degradation of fast neutron flux to the vessel was performed by mounting 36 dummy assemblies in the periphery of the core,

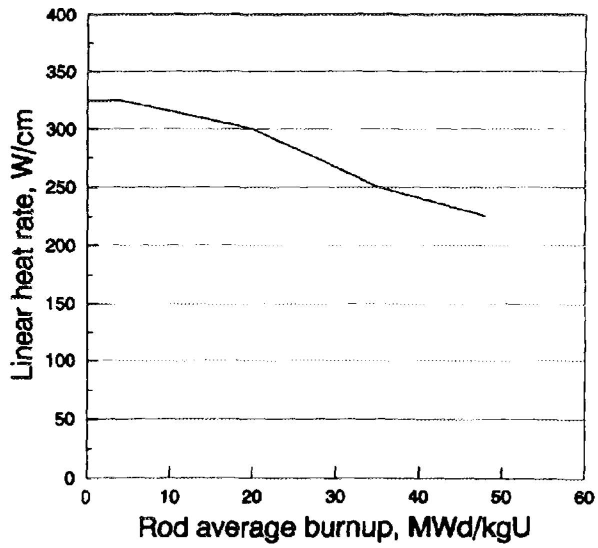

Figure 1 Burnup dependent linear heat rate for Loviisa fuel rods

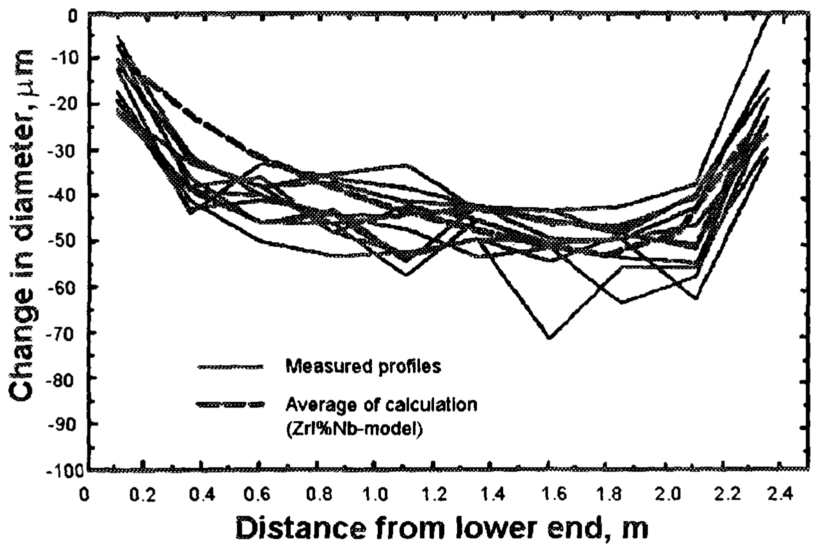

Figure 2 **Comparison of measured and calculated (ENIGMA) cladding creep down of some Loviisa rods examined with pool side inspection facility at Loviisa NPS**

acting as neutron absorbers. The core average linear heat rate was increased up to 144 *W/cm* due to this operation.

The average discharge burnup in Loviisa is about 35 *MWd/kgU*, which is almost 30% higher than the initial design burnup. The burnup has been increased because of improved fuel utilization, the reduced core created a "natural" need for increasing the burnup. The development of discharge burnup of Loviisa reactors is shown in Figure 3.

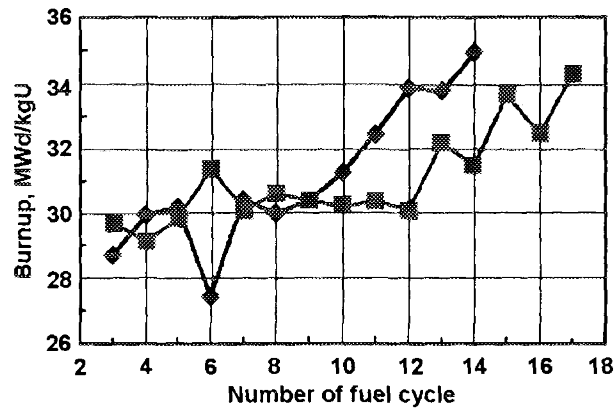

Figure 3 **Development of average discharge burnup of Loviisa fuel: unit 1 (■) and unit 2 (◆)**

2.2 Fuel Leakages

Generally the fuel in Loviisa has shown good behaviour. The cladding corrosion has not been a problem and mechanically the assembly is of firm design.

There has been observed all together 20 fuel leakages in Loviisa plant so far, which makes ~ 0.6 leaks per fuel cycle. If it is assumed that there is only one leaking rod in each leaking assembly, the frequency of rod failures is about $4.5 \cdot 10^{-5}$. Ten leaking assemblies that failed before 1990 have been disassembled and visually inspected. In two of those assemblies a clear fuel-to-coolant contact has been verified by the pool side examinations. In most of the failed assemblies the defects are of micro size. The difficulties to find out the features of the defects arises from the mechanical structure of the assembly. It is very difficult to identify the leaking rod if it is located in the middle of the bundle.

In one of the assemblies with fuel-to-coolant contact one rod was almost cut about 30 *cm* below the top of the fuel column. Below the defect, slightly increased diameter of the cladding was seen in three locations and one tight axial crack with length of 15 *mm* existed below the region of the enlargened diameter [1].

The other assembly with observed fuel-to-coolant contact had a hole of 70 mm^2 in plenum region of one rod. The reason of this failure is assumed to be water logging in the beginning of its first cycle [1]. The rod has obviously been leaking already from the start up.

The dismantling and visual inspection of the four leaking assemblies that failed after 1990 is currently being accomplished. For those assemblies the activity release was small, however, in one assembly some release of solid actinide ^{239}Np was detected indicating possible direct fuel to coolant contact.

Table 1 summarizes the key data of leaking fuel assemblies in Loviisa.

2.3 Experimental Support

The verification of fuel performance at increasing burnup has been performed not only through code calculations, but also experimental work has been carried out in the form of irradiations of test rods in research reactors, non-destructive post irradiation studies of high burnup fuel rods and destructive PIE in hot cells.

Table 1 Status of leaking fuel assemblies in Loviisa NPS.

Assembly identification	Fabrication date	Unit	Failure date/ cycle	Burnup, *MWd/tU*	Notes
AA0664	12/1976	LO1	1981 / 3	31297	Sipping done
13601255	11/1979	LO2	1983 / 2	19889	"
13601419	12/1979	LO2	1983 / 2	20813	"
13601428	12/1979	LO2	1983 / 2	20158	"
13602348	04/1980	LO1	1985 / 3	32859	"
13602446	05/1980	LO1	1983 / 2	19028	"
13614735	03/1984	LO2	1987 / 2	27459	PIE done (in pool)
13614748	03/1984	LO2	1989 / 3	32831	"
13617619	02/1985	LO1	1988 / 2	23937	"
13617624	02/1985	LO1	1989 / 3	30837	"
13617628	02/1985	LO1	1989 / 3	30629	"
13617649	02/1985	LO1	1988 / 2	26367	"
13617667	02/1985	LO1	1989 / 3	30790	"
13617735	03/1985	LO2	1988 / 2	28256	"
13617912	03/1985	LO2	1989 / 1	28256	"
13624564	02/1987	LO2	1989 / 1	16676	"
13628285	01/1988	LO1	1991 / 2	23062	PIE under way
13628400	01/1988	LO2	1992 / 3	35193	"
13632468	01/1989	LO1	1993 / 3	33603	"
13637285	02/1990	LO2	1993 / 2	22456	"

In joint Finnish - Russian research project "SOFIT" a number of instrumented and well characterized short test rods have been irradiated to burnup levels from 1 - 17 *MWd/kgU*. Qualified on-line information has been obtained from fuel densification-swelling behaviour and the start up of the pellet cladding mechanical interaction. The fuel temperatures were measured and their dependence on linear heat rate is evaluated. After the irradiation the rods are examined in hot cells [Ref. 2, 3,4].

The structure of VVER-fuel assembly in Loviisa does not allow its easy dismantling. The removal of the hexagonal shroud tube requires drilling operations, and the lower end caps of the rods have to be cut off until the rods can be pulled out of the bundle. At Loviisa NPS site there has been installed a pool side examination equipment which is capable to get through all this. The equipment, "ATULA" is designed by IVO for investigating discharged fuel assemblies and rods [5]. The outer dimensions of the assembly (length, bow, twist, wrench...) and the rods (length, diameter, ovality) can be measured by ATULA. The pellet cladding gap size is obtained through squeezing the rod and evaluating the shift of the compressing jaws versus compressing force. The width of both the open and closed gap are defined. A subjoined facility enables the determination of the fraction of released fission gases by measuring the activity of ^{85}Kr in the plenum region. The axial gamma scanning curves can also be taken.

Rods from more than 30 fuel assemblies have been investigated with the pool side examination facility at spent fuel storage in Loviisa. Additionally the length of fuel rods have been measured from about 50 assemblies non destructively in reactor hall between irradiation cycles during refuelling outages. The burnup of the inspected rods varies between 10 and 50 *MWd/kgU*. The results of the pool-side examinations have been reported for example in [6] and [7].

One important part of the verification of the fuel performance have been the examinations of 14 high burnup fuel rods in hot cells of Studsvik Nuclear. Ten rods were examined non-destructively in 1982-1983. One rod out of those 10 was destructively examined in 1985. Thorough investigations of 4 rods from three high burnup assemblies were done in 1988-89, [8].

3 Main Design Changes

Some noticeable design changes have been carried out since the first delivered fuel batches. Some of the alterations have been performed because of the interest in increasing the burnup, while the reasons to others are peculiarities or malfunctions encountered in the irradiation behaviour or demand to improve the economy of the fuel utilization. The modifications to the fuel design have always been done in good cooperation between IVO and the fuel designer/manufacturer. The initiatives for the changes have come from either IVO's or designer's side and careful considerations and calculations have preceded the loading of lead fuel assemblies of the new design. The follow-up of the behaviour of new constructional solutions has been a prerequisite for the approval of them by the authority. Inspections after discharge or even between the cycles during refueling outages are the means of controlling the performance of fuel having new design features.

In 1987 two assemblies with increased rod prepressurization from 1 to 6 *bar* were loaded into Loviisa-2 reactor. The pressure was optimized for improving the resistance of the fuel pins against thermal feedback effect of fission gas release. The cladding creep down was also slowed through the decreased pressure difference across the clad wall. The pellets had small chamfering and they were manufactured by common pressing-sintering technique replacing the former extruding-sintering method. These design changes were aimed for improving the thermal-mechanical properties of the fuel rods by decreasing pellet cladding interaction (PCI) and for achieving more stable quality of the pellets. The rod length measurements of spent fuel showed clearly the positive effect of these design modifications: the rod elongation decreased by about 40% indicating lower PCI.

Some irregular deformations of upper grid were observed at Loviisa in the visual inspections of the spent fuel assemblies. The reason for this was jamming of the growing fuel rods to the grid. The upper grid was rigidly fixed in the upper nozzle of the assembly. A new upper grid construction was introduced to allow axial movement of the grid compared to the assembly cap. Six assemblies with this new upper grid were loaded in 1992. The axial position of the upper grid was measured in refuelling outages in 1993 and 1994. The measurements showed good performance of the grids, but concluding remarks will be presented after visual inspection for the discharged and dismantled assemblies.

The reduced core and requirement for yearly cycles forced to discharge part of the assemblies after two cycles only, despite of the low leakage loading pattern and introducing 3.6% enrichment also for the fuel followers. Zirconium spacer grids and thinner assembly shroud are measures taken to improve the neutron economy of the core. In 1994 the first 6 assemblies equipped with Zr-spacers were loaded into Loviisa-2 reactor. The reduction of wall thickness of the assembly shroud will be reality in fuel batches delivered to Loviisa in 1996. These items together will improve the neutron economy by 3 - 4 %.

4 Quality Assurance

The quality assurance (QA) program of Loviisa nuclear fuel is part of the Loviisa NPS QA program. The purpose for the QA program of Loviisa fuel are to ensure that:

- correct quality requirements are directed to nuclear fuel;
- the actions for obtaining the quality targets laid for nuclear fuel are based on procedures planned in advance;
- the actions affecting the quality of nuclear fuel are carried out correctly;
- anomalies are identified and measures for remedies are initiated;
- the requirements of the QA program are known and they are followed and their compliance is supervised and;
- the efficiency of the entire QA program is evaluated and the required corrective actions are planned and taken into use.

Several parts of Imatran Voima are involved in the fuel supplies for Loviisa. In addition to the Loviisa plant itself the Energy Production division, R&D division and IVO International Ltd. participate in implementation of QA of Loviisa fuel. In Figure 4 the organization units and their tasks in fuel QA are presented.

The director of Loviisa NPS is responsible for the compilation and implementation of the QA program of Loviisa plant. Concerning the QA of nuclear fuel, the co-ordination and responsibility of activities has been delegated to IVO International Ltd, Nuclear Power Engineering, whose director is responsible for fuel QA.

Quality assurance engineer of Nuclear Power Engineering in IVO International Ltd supervises the implementation of QA and participates in follow-up and updates the QA manual of nuclear fuel.

During the first twelve years of operation of Loviisa NPS foreign companies had no access to the nuclear fuel manufacturing plants of the former Soviet Union. For this reason complementary QA measures were worked out. An independent Soviet (now Russian) inspection service company Soyuzexpertiza (SOEX) was awarded with a control contract for quality audits and QC inspections on behalf of IVO. In addition, one fuel assembly from each year's fuel delivery was destructively investigated to compare the measured spot check data with the QC data from the fuel manufacturing.

Since 1989 IVO itself has had the access to the fuel factory and regular quality auditing by IVO, although still complemented by SOEX, has been established. The destructive investigations of delivered fuel assemblies have been stopped at the time being.

5 Future Prospects

The factors that may affect the fuel supply to Loviisa in the near future are:

- possible power raise of Loviisa reactors;
- stopping of spent fuel transportation to Russia for final disposal and
- new VVER-fuel deliverers.

Preliminary studies to increase the electrical output of Loviisa NPS have been carried out by IVO International Ltd. The thermal power of Loviisa reactors can be increased without changing the current design limits of fuel or the core. However, in case of increasing the power the fuel economy would deteriorate, since some part of the fuel assemblies would have to be discharged after two years irra-

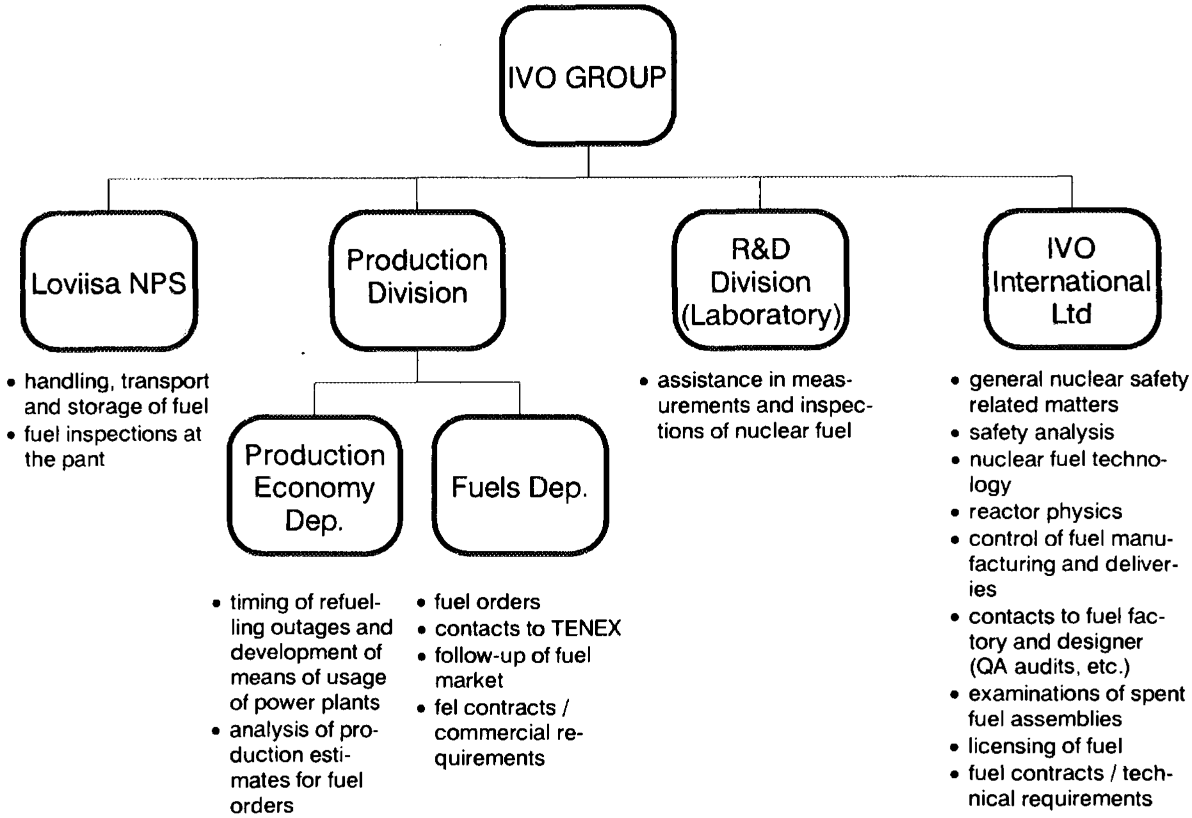

Figure 4 Organisations of IVO and their tasks in Quality Assurance of Loviisa fuel

diation only and the low leakage loading strategy would have to be abandoned. This would result in requirements of improved fuel performance, concerning both linear heat rate and burnup. The possibilities to increase the thermal output depend mostly on secondary side and electrical equipment. So far, no decision has been made.

Some preliminary considerations have been made to further extend the burnup by increasing the enrichment and at least partially turning into four cycle fuel. The four cycle fuel has been used for some years at least in Kola NPS in Russia. Therefore operating experience already exists. However, research work is required for considerable extension of burnup to clear out the fission gas release and PCI properties of the fuel and behaviour of the rods in abnormal transients, especially in reactivity initiated accidents.

Up to now the spent fuel management of Loviisa NPS has totally rested on returning the irradiated assemblies back to Russia for final disposal. At the moment it is evident that it will not be possible to return spent fuel to Russia after 1996 due to new legislation under preparation in Finland. One important advantage of the present fuel supplier is thus removed at the same time. This fact together with the new (Western) fuel vendors emerging in the market may open new possibilities to arrange the fuel cycle services at Loviisa NPS.

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State-of-the-Art and Perspectives of the Fuel Rod and Material Development Activities in Russia

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1 Introduction

The work recently accomplished in Russia to improve fuel and fuel cycles promoted the reliability of VVER-1000 operation in a 3 year mode and VVER-440 in a 4 year mode (Kola and Novovoronezh NPPs) at the mean fuel discharge burnup ~ 42 *MWd/kgU*. The fuel rod operational reliability is at a high level. During 1991 - 1993 in VVER in Russia and Ukraine the frequency of fuel rod leakage did not exceed on the average ~ $2 \cdot 10^{-5}$ (the number of leakers in relation to the total number of discharged fuel rods). In some units no failures of fuel rods in discharged assemblies were observed.

The high reliability of fuel operation is also attested by the low activity of the primary circuit coolant. In 1991-1993 in VVER-1000 under operation in Russia it was from 10^{-6} *Ci/l* to $2 \cdot 10^{-5}$ *Ci/l* of total iodine isotopes. It is clear that high fuel performance is also promoted by the high quality of products and correct design and engineering solutions, parameters and operational conditions.

However, we understand that at the burnup of ~ 50 *MWd/kgU* which is presently traditional in world practice the operational reliability if one compares fuel supplied by different customers, is not a decisive criterion since to-day essentially all companies-fuel supplies produce fuel of very high quality that assures the needed operational and life-time characteristics at an adequately low level of fuel failures ~ $(1 - 2) \cdot 10^{-5}$. Therefore, the efforts made in Russia to improve nuclear fuel are aimed at both further increasing the fuel operational reliability and improving the technico-economical parameters of fuel cycles primarily through increasing endurance characteristics of structural materials and fuel rods to reach the mean fuel burnup ~ 55 - 60 *MWd/kgU* and higher.

2 Development of Work in Russia to Improve Fuel Rods and Materials

By now a large scope of experimental, design and pilot work has been accomplished to further improve the life-time of fuel. U-235 enrichment of VVER-440 fuel increased to 4.4% mass allowed at the 4 year mode of operation a 25% increase of both the mean burnup and fuel cycle duration to 42 *MWd/kgU* and 8000 hours, respectively. In this case fresh fuel was essentially loaded to the core circumference, i.e., the low leakage loading pattern was realized. In 1990 the core of Kola NPP unit III was fully loaded in this way 12 fuel assemblies have successfully operated in this unit for 5 years to reach the burnup of 48 *MWd/kgU* [1]. The programme of work under way in Russia aimed at new improved fuel envisages creation of VVER type cores having the following basic characteristics:

- the mean fuel burnup of 55 - 60 *MWd/kgU*;
- 5-6 year fuel cycle;
- time between reloads increased to 18 months;
- use of integral fuel burnable absorber ($UO_2+Cd_2O_3$);
- optimized enrichment and UO_2 fuel charge in a fuel rod (through increased weight per fuel rod unit length);
- use of high irradiation and corrosion resistant Zr-Sn-Nb-Fe alloy as a structural material of claddings and guide thimbles;
- use of Zr alloys containing Hf less than 0.01 % mass as structural material spacer grids and guide thimbles included;
- axial blankets used in fuel rods;
- higher operational reliability of fuel rods, failure level ~ $5 \cdot 10^{-6}$;
- provision of load follow characteristics of NPPs.

Introduction of new fuel into VVER reactors will significantly improve the economic parameters of VVER fuel cycles. Starting from 1995-1996 it is planned to commercially realize improvements after they have been mastered, experimental and design studies have been completed and data have been generated needed for licensing: Fig. 1.

3 Some Operational Specificity of Russian VVER Fuel

In principle, the conditions of VVER fuel operation are not different from those of PWR (see Tables 1 and 2). However, the Russian fuel has several specific features, the major of them are:

- use of Zr-Nb alloys;
- hexahedron shape of fuel assembly instead of a square one;
- use of spacer grids of honeycomb type the cells of which are manufactured from tube fragments;
- pellets having plane ends and central hole instead of solid dished pellets.

Aside from these design features there are also distinctions in technological processes of fuel and component production (fuel pellets, cladding tubes, etc.). Therefore, it is of interest to make comparisons whenever it is possible between individual operational characteristics of Russian and standard PWR fuel. Of special interest are characteristics that are capable of limiting fuel service life with a

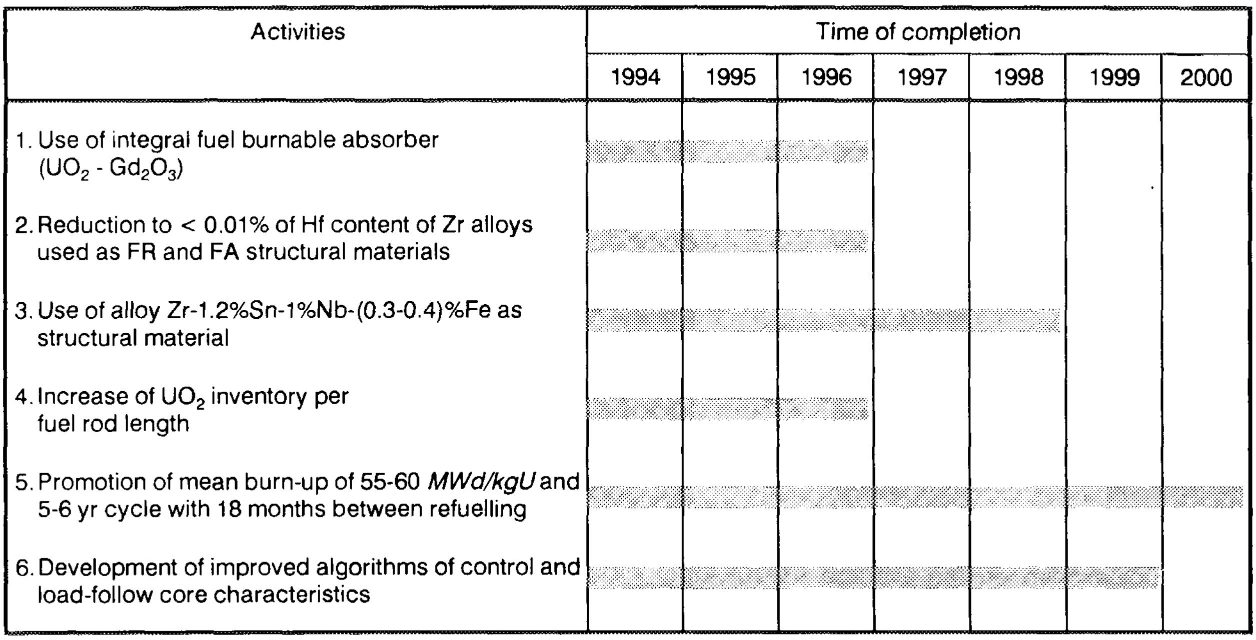

Figure 1 Milestones in improvement of VVER fuel rods and structural materials

further increase of fuel burnup in commercial reactors (> 50 *MWd/kgU*): water-side corrosion and hydrogen pick-up by fuel claddings; corrosion induced cracking; fuel behaviour in power ramps; initial characteristics, microstructure included, and irradiation behaviour of fuel pellets.

3.1 Water-Side Corrosion and Hydrogen Uptake

Fig. 2 and Table 3 contain experimental data on commercial fuel operation in VVER-1000 and pilot fuels in a loop facility of MR reactor.

From the presented experimental results, it follows that corrosion and hydrogen pick-up by VVER fuel claddings are not factors that fundamentally affect the life-time of fuels clad in (Zr-1%Nb) alloy in VVER type reactors. The following pattern is typical:

- no nodular corrosion;
- outer surface of fuel claddings has a uniform dark oxide film from 3 to 8 μm thick;
- insignificant amount of platelet hydrides not more than 100 μm in size are observed in claddings. Hydrides have random or tangentional orientation, the hydrogen content of claddings is within 30 - 80 *ppm*;
- fuel claddings retain high strength and significant ductility margin (see Figs. 3 and 4).

3.2 Influence of Power Cycling on VVER Fuel Cladding Corrosion

Under load follow conditions at the stage of power rising tensile stresses can develop in fuel claddings.

The time during which fuel claddings are subject to tensile stresses can totally reach 5 - $7 \cdot 10^3$ *h* per fuel cycle for cores designed for the mean fuel burnup of 55 - 60 *MWd/kgU*.

The level of tensile stresses depends on a number of factors, namely, heat generation rates, power variation rates, algorithm of power rating control on power changes, burnup, etc. Tensile stresses are capable of accelerating corrosion.

The influence of this factor on Zr-1%Nb cladding corrosion was studied in MR loop. Two dummy fuel assemblies were tested under the following conditions:

- number of cycles of power variations is 778 and 794, respectively;

Table 1 Characteristics of VVER-440 (4-Year Cycle) and VVER-1000 (3-Year Cycle)

Characteristics	VVER 440	VVER 1000
Number of FAs per core	349	163
Number of FRs in FA	126	312
Number of CPS subassemblies	37	61
Core height, *mm*	2500	3530
Total UO_2 fuel charge, *t*	44	74
Spacing between FAs, *mm*	144	236
Spacing between fuel rods in FAs, *mm*	12.2	12.6
Fuel management scheme	partial LLLP	partial LLLP
Maximum make-up fuel enrichment, % U-235	4.4	4.4
No. of FAs reloaded per year	~ 90	~ 55

- maximum fuel heat generation rate is 440 *W/cm*; amount of power variation (depth of load follow) is 70-80% in a pattern like: (100%*N* - 20-30%*N* - 100%*N*);
- maximum fuel rod burnup is from 29.3 to 36.5 *MWd/kgU*.

Hot-laboratory examination of fuel rods showed:

- oxide film thickness was less than 10 μm;
- nodules up to 40 μm were observed at fuel rod surfaces; the cause of nodules was not conclusively established since during experiment the free oxygen content of coolant increased from 5 to 20 *mg/l* (compared to specs).

3.3 Corrosion Induced Cracking

Fig. 5 shows the crack propagation rate vs. stress intensity factor (K_{SCC}) for claddings of Zry-4 as stress relieved (SR) [2] and of Zr-1%Nb as recrystallized, i.e. the states in which the alloys are used as fuel claddings in commercial reactors. In both cases internally pressurized claddings with purposely made fatigue cracks at the inner surface were used for testing. Comparison between crack resistances of two alloys in iodine shows that the

Table 2 Main Geometrical Characteristics and Parameters of Fuel Operation in VVER-440 and VVER-1000

Characteristics	VVER 440	VVER 1000
Fuel rod diameter, *mm*	9.1	9.1
Fuel cladding thickness, *mm*	0.7	0.7
Cladding material	Zr-1%Nb	Zr-1%Nb
Coolant pressure, *MPa*	12.7	16.7
Coolant flow rate in bundle, *m/s*	3.5	6.0
Linear heat gener. rate, *W/cm*: mean	128	167
maximum	325	448
Average coolant temper., °C	283	306
Maximum temperature of fuel rod surface, °C	335	352
Mean (design) discharge fuel burnup, *MWd/kgU*	40-43	40-45
Time of fuel cycle, *h*	$(7\text{-}8)\cdot 10^3$	$(7\text{-}8)\cdot 10^3$
Water chemistry	ammonia-boron-potassium	

Table 3 Experiment Conditions and Results of Cladding Corrosion and Hydrogen Pick-Up Tests

Parameter	Value	Oxide film, μm	H_2 content *ppm*
Coolant pressure, *MPa*	16.0		
Coolant flow rate, *m/s*	5.5		
Coolant temperature, °C: inlet	305		
outlet	320	3	30 - 60
Maximum LHGR, *W/cm*: at start of irradiation	490	(15 at one test)	
at end of irradiation	200		
Fuel burnup, *MWd/kgU*: average	70.0		
maximum	92.9		
Testing time at different power levels, *h*	40164		

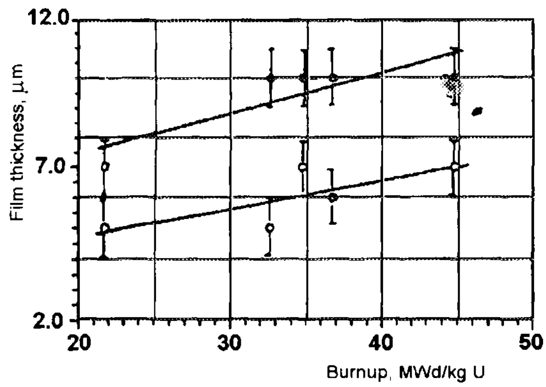

Figure 2 Dependence of oxide film thickness on inner (□) and outer (O) fuel claddings for VVER-1000 FAs on fuel burnup

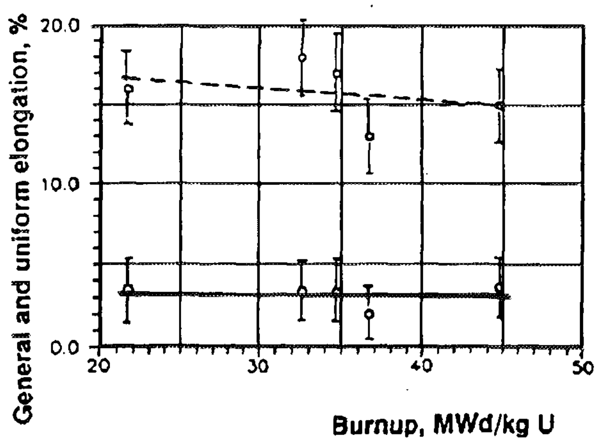

Figure 3 Dependence of general (□) and uniform (O) material cladding elongation on fuel burnup for VVER-1000 Fas (T_{ts}=380°C)

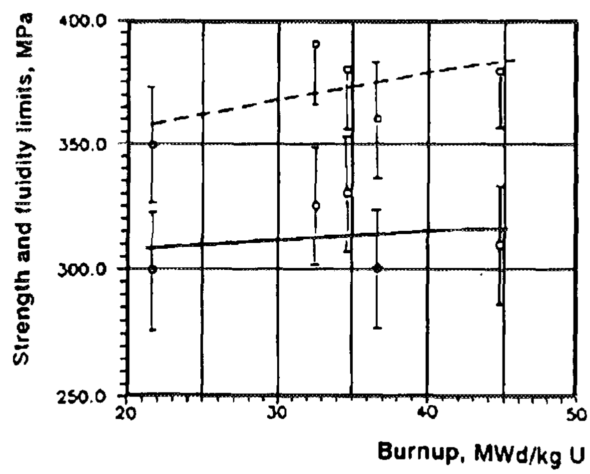

Figure 4 Dependence of strength (□) and fluidity (O) limits material cladding on fuel burnup for VVER-1000 Fas (T_{ts}=380°C)

threshold stress intensity factor of Zr-1%Nb is much higher which indicates the advantage of this alloy over Zry-4 (SR).

3.4 Fuel Rod Behaviour in Power Ramps

Fig. 6 presents the results of experimental studies carried out in MIR reactor using specially fabricated experimental fuel rods and the ones refabricated from commercial VVER-1000 fuel rods. In the same figure the dashed line shows the damageability threshold accepted for PWR fuels (Siemens, CEP, Japan) [3, 4]. The advantages shown by the VVER fuel rods can be possibly explained by two factors, namely, higher ductility of Zr-1%Nb and use of pellets with central holes.

3.5 Initial Characteristics and Irradiation Behavior of Pellets Produced in Russia

The specimens to be tested were either commercial ones (VVER-1000) or manufactured by the technology corresponding to the technology of VVER-1000 pellets manufacture. The initial characteristics of the specimens are listed in Table 4.

Fig. 7 illustrates the stress dependence of the rate of irradiation induced secondary creep at different temperatures and fission densities. In the experiments the equilibrium stresses were determined at which the irradiation induced creep and swelling rate are the same and equal to 5 - 6 *MPa* and are essentially independent of fission density. The equilibrium stress is an important characteristic that enables one to assess the level of stresses originating in fuel claddings as a result of UO_2 swelling. Fig. 8, 9 show experimental data on UO_2 volume changes under irradiation. These and similar experimental data allowed the dependence to be derived of irradiation effected densification on the initial porosity (see Fig. 10) and temperature (see Fig. 11).

The specific behaviour of VVER fuel rods in LOCA (using as an example VVER-1000 fuel rods) is dealt with in a special report that will be presented by Mr. Sokolov at this meeting.

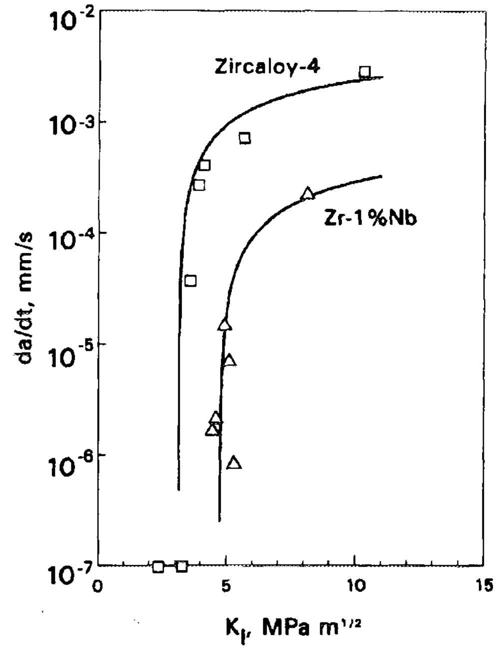

Figure 5 Crack propagation rate as a function of stress intensity coefficient

4 Fuel Rod Design and Structural Material Improvements

As it is shown above the VVER fuel fabricated in compliance with the specifications now in action has high serviceability and meets the present-day operational requirements. The more so, as far individual parameters are concerned that are important for the life-time characteristics (corrosion and hydrogen uptake, corrosion cracking, damageability in power ramps) the VVER fuel is superior to that of PWR which evidences its higher potential ability for increasing its life-time.

Table 4 Characteristics of samples and testing conditions

Sample No.	Oxygen - Metal ratio	Grain size, μm	Total porosity %	Porosity fraction, %				T, °C	Φ, 10^{15} $1/cm^3s$	BU Φ_t, 10^{20} $1/cm^3s$
				d<1 μm	1<d<2 μm	2<d<4 μm	d<4 μm			
1	2.003	10	4.38					270	3.6	1.0
2	2.006	8.8	4.2					400	2.0	0.6
3	2.005	10.4	5.93	0.09	0.94	3.60	1.38	540	6.6	2.0
4	2.003	8.8	4.2	0.31	0.71	1.80	1.40	630	4.5	1.2
5	2.005	10	6.94	0.04	0.40	4.98	1.52	790	8.2	2.4
6	2.006	8	4.6					900	2.6	0.8
7	2.003	10	4.2					930	4.0	2.0
8	2.003	10.1	3.28	0.02	0.20	1.89	1.17	950	9.1	2.0
9	2.006	8.7	5.2	0.10	2.00	1.50	1.60	950	2.8	0.5
10	2.003	8.8	4.38	0.48	1.51	2.39	-	970	8.2	2.0
11	2.003	10	7.9					990	4.6	1.4
12	2.006	10	2.65	0.08	1.02	1.55	-	1000	2.8	0.8
13	2.003	10	2.4	0.07	0.09	1.41	-	1010	4.6	1.6
14	2.005	8	4.4					1040	4.6	1.4
15	2.003	8.7	5.1	0.08	0.62	1.94	2.48	1200	4.6	1.5

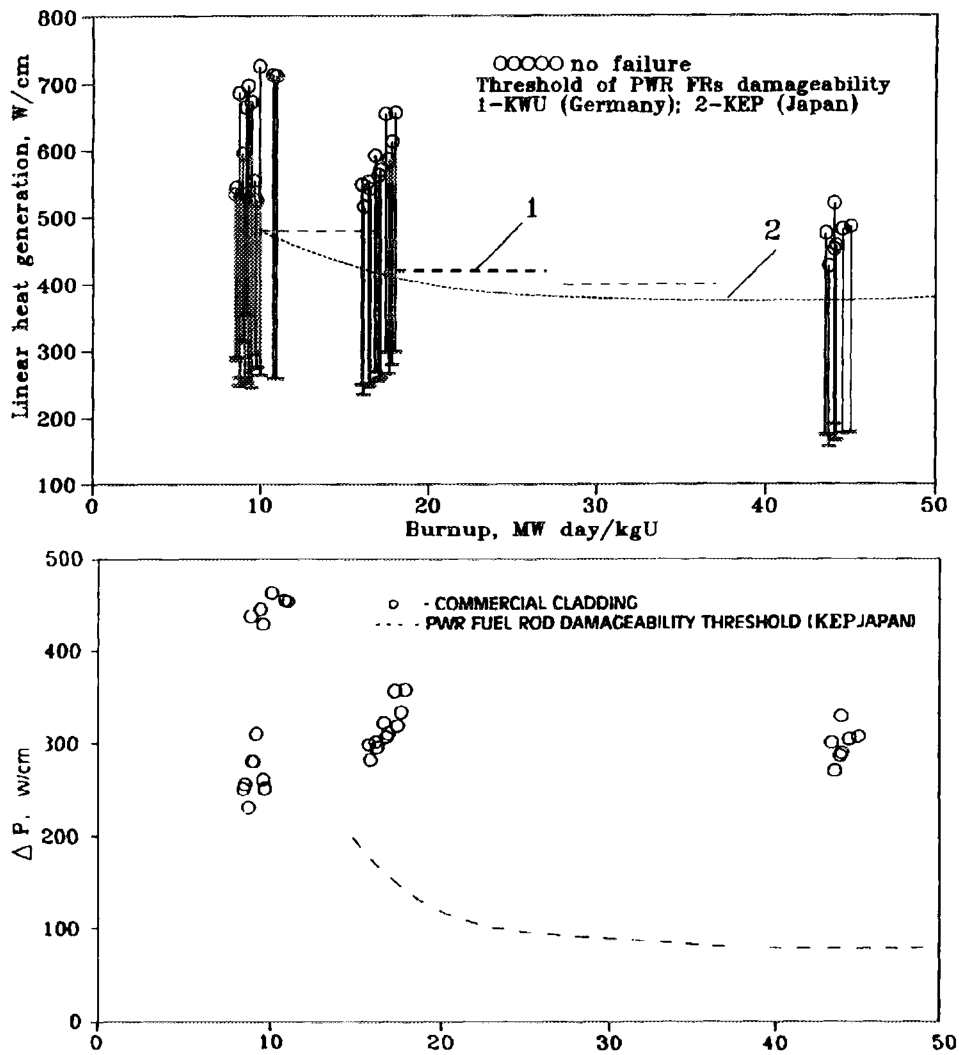

Figure 6 Experimental results of VVER-1000 fuel behaviour in power ramps (MIR reactor) study

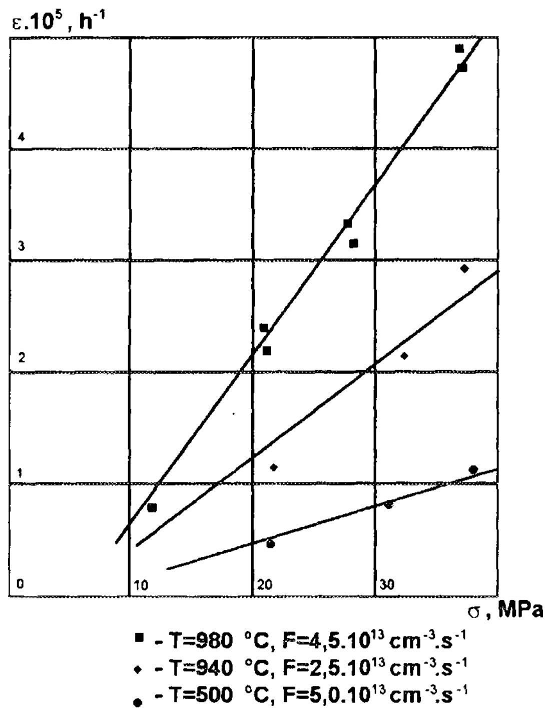

Figure 7 Stress dependence of low temperature irradiation creep of UO_2

4.1 Use of Integral Fuel Burnable Absorbers

In 1990-1992 VNIINM together with PC UMP in Ust-Kamenogorsk have mastered and commercially executed the technology of fabrication of uranium-gadolinium ($UO_2 + Gd_2O_3$) fuel pellets.

A batch of pellets has been manufactured containing 8% mass Gd_2O_3. At NZKK plant in Novosibirsk, 12 pilot fuel assemblies were fabricated, each containing 18 fuel rods fuelled with uranium-gadolinium oxide pellets. From 1993 such fuel assemblies are under test at unit 3 of the Balakovo NPP. Presently the work has started to fabricate the next larger batch of fuel assemblies with uranium-gadolinium fuel rods.

Through the use of integral fuel burnable absorbers the VVER-1000 core will be arranged at the lower radial neutron leakage pattern (LLLP) and it will be easier to provide negative values of the temperature coefficient of reactor reactivity [1]. Now the work is under way to use integral fuel burnable absorbers in VVER-440 to attain the fuel LLLP.

4.2 Reduction of Hf Content of Zr Alloys Used as Fuel Claddings

Beginning from 1993, commercial production of components from Zr-alloys containing less than 100 *ppm* of Hf has started in Russia. Until that time the Hf content was limited to 500 *ppm*.

The need for reducing Hf content arose in connection with the replacement of steel spacer grids and guide thimble tubes by Zr ones. With large amounts of steel available in the core the above mentioned Hf content of Zr did not in principle affect the core characteristics.

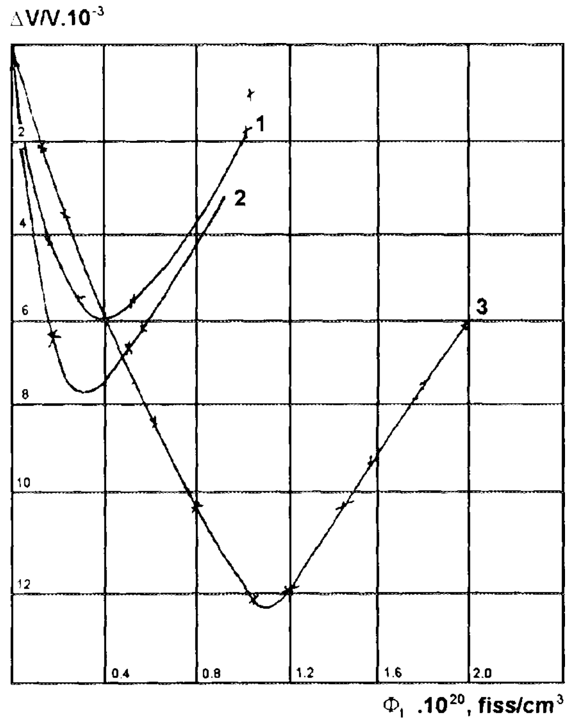

Figure 8 Change in volume of uranium dioxide under irradiation:
1: T=270°C, P_0=95.6%;
2: T=400°C, P_0=95.8%, $\Phi = 2 \cdot 10^{13}$;
3: T=450°C, P_0=94.07%.

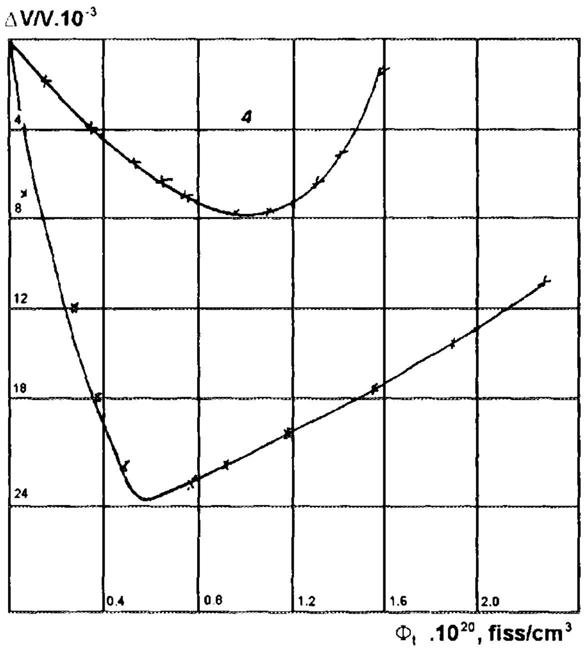

Figure 9 Change in volume of uranium dioxide under irradiation:
4: T=630°C, P_0=95.8%;
5: T=790°C, P_0=93.16%.

4.3 Use of Zr-Nb-Sn-Fe Alloy as Structural Material

Special attention is paid to the choice of an adequate fuel cladding material for operation at a higher fuel burnup (55 - 60 *MWd/kgU*). The fact is, that despite high operational characteristics, alloy Zr-1%Nb has some disadvantages, namely:

- sharp increase of irradiation growth after a fluence of $2 \cdot 10^{22}$ N/cm^2 (E > 0.1 *MeV*) is reached;

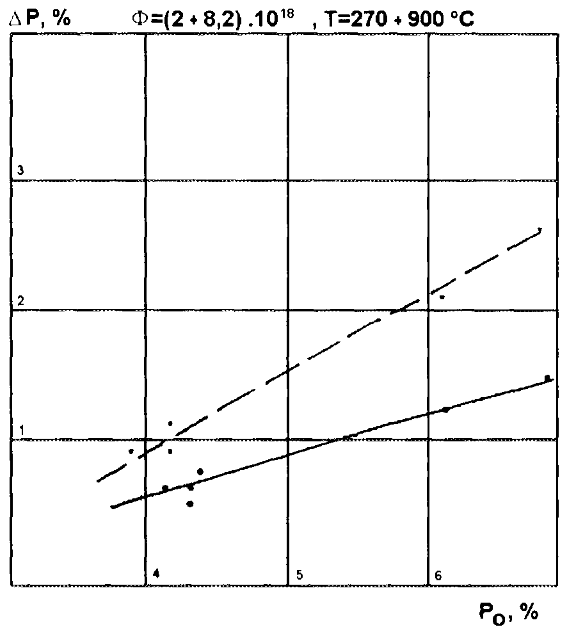

Figure 10 Dependence of radiation densification on initial porosity (dotted line - with deduction of radiation swelling)

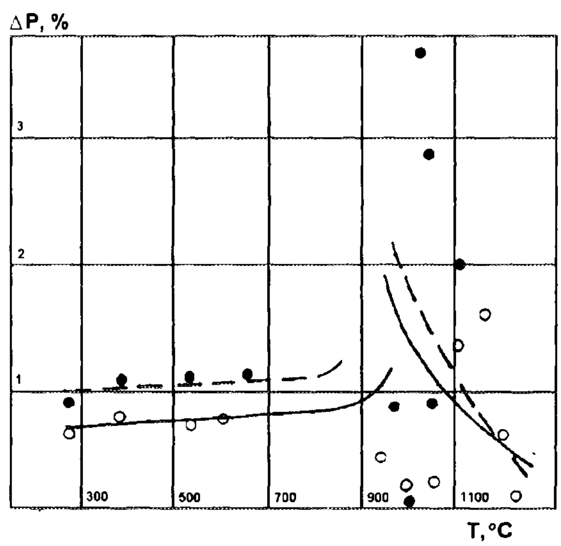

Figure 11 Dependence of uranium dioxide densification on temperature:
● - with deduction of radiation swelling;
○ - accounting for radiation swelling.

- high thermal and irradiation creep;
- rather high sensitivity of corrosion resistance to oxygen content in water above 10 *mg/l*.

Late in the 60-s another Nb-containing alloy Zr-Sn-Nb-Fe has been developed in Russia. To-date it is thoroughly studied in terms of both technology and performance. The optimal composition of the alloy and its metallurgical condition have been identified. While having all the advantages of Zr-1%Nb alloy, it is superior to the latter in its irradiation resistance (see Figs. 12 and 13).

There are also some other operational advantages of this alloy:

- The alloy has a high corrosion resistance not only in pressurized water but also in boiling reactors.

  A large amount of pilot FAs (~ 40 items) with fuels clad in this alloy were successfully tested for 3 years at the Leningrad NPP. At the full absence of nodular corrosion the oxide film thickness did not exceed 10 - 15 μm.
- The corrosion resistance of the alloy is essentially not sensitive to a higher oxygen content of a coolant under VVER service conditions.

  When comparison tested in the same loop and during the same period of time with the oxygen content of water increased to 20 *mg/l* Zr-1%Nb claddings showed individual nodules up to ~ 40 μm in size while no nodular corrosion was observed in Zr-Sn-Nb-Fe claddings.
- Zr-Sn-Nb-Fe alloy shows a high corrosion resistance under water the chemistry conditions of PWR reactors (Fig. 14).

One more important circumstance is to be also noted. Use of FA structural materials having high irradiation resistance such as the Russian Zr-Sn-Nb-Fe alloy (very low irradiation growth, high resistance to thermal and irradiation creep) allows to avoid to a significant extent the problem relevant to geometrical instability of FAs, particularly at high burnup and high neutron fluence. Today Zr-Sn-Nb-Fe tubes are produced at the plant in Glasov, Russia.

4.4 Increase of UO_2 Inventory per Fuel Rod Length: Optimization of Geometrical Parameters and Allowances of Fuel Pellets and Cladding Tubes

It is known that UO_2 inventory per fuel rod length in VVER-440 does not essentially differ from that in PWR. In VVER-1000 it is somewhat less which although slightly but still affects the economic parameters of fuel cycle.

To eliminate this, complex design, experimental and technological investigations, industry included, were carried out and have been completed by now; as a result the following recommendations were issued:

- The central hole diameter that was specified on the basis of adequately low bounds on FGP

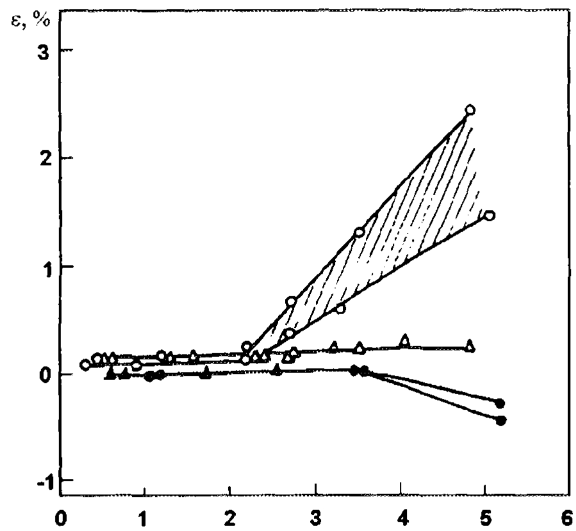

Figure 8 **Irradiation growth strain of 2H annealed at 580°C fuel rod tubes of Zr1%Nb (circles) and ZrNbSnFe (triangles) as a function of fluence in axial (white symbols) and transverse (black symbols) directions at T_{irr}=330-350°C**

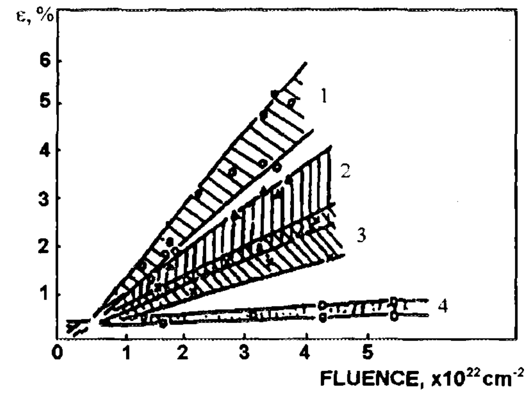

Figure 13 **Creep strain of Zr-alloys as function of neutron fluence irradiation at 300 -350°C and stress of 130 MPa: 1 - Zr1%Nb; 2 - Zr-1%Nb-O; 3 - Zircaloy-4F; 4 - Zr-1%Nb-1.2%Sn-0.3%Fe**

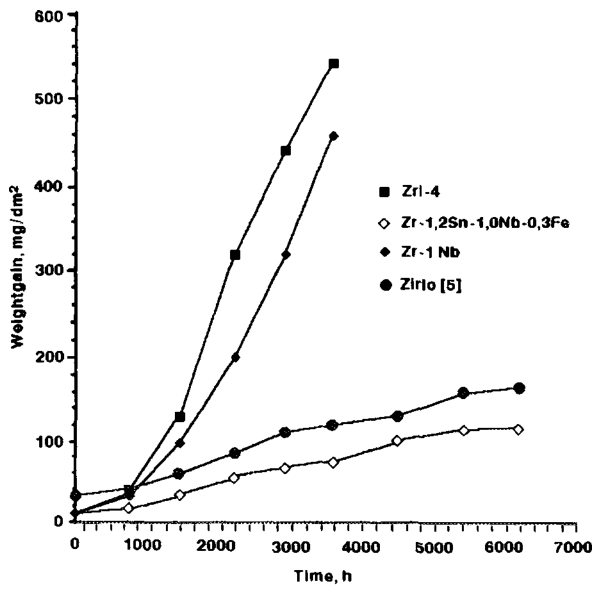

Figure 14 **Corrosion tests in water with Li. Test conditions: H_2O, Li content 70 mg/l, T=360°C, P=18.6 MPa.**

release from fuel can be decreased. Its optimized value is 1.5 - 1.6 *mm* (see Table 5).

- A fuel pellet having plane ends and a central hole is a preferable option compared to a dished one. Under conditions typical of VVER (reactor MR loop) fuel rods stacked with pellets having central holes promoted the fuel burnup ~ 100 *MWd/kgU* while retaining some margin to prolong the life-time.

A decrease in the maximum manufactured defect permissible in Zr tubes from 50 to 35 μm achieved by our industry in 1993 as well as a further improvement of technological processes and control of final products promote optimization of the nominal value and allowance for the inner diameter of a tube at the similar changes in the geometrical parameters and a fuel pellet.

In the nearest future the above recommendations will be introduced into the practice of the commercial production of VVER fuel (early in 1995).

Table 5 Some initial parameters of tested fuel rods

Parameter	Value
Coolant pressure, *MPa*	16.0
Coolant rate, *m/s*	5.5
Coolant temperature, °C	
inlet	305
outlet	320
Maximum linear heat generation rate, *W/cm*	490
Initial coefficients of power density non-uniformity	
in height	1.37
in section	1.20
Time of testing at different power levels, h	40164
Power generation, *MWd*	317.9
Fuel burn-up, *MWd/kgU*	
mean	70.0
maximum	92.9
Coefficients of fuel burn-up non-uniformity	
in height	1.17
in section	1.14
Number of reactor trips	
during testing of SS signals	273
for scheduled maintenance	92

It is certain that if the fuel mass per unit length is increased the feasibility will be considered of decreasing the fuel enrichment although an increase in the specific inventory of UO_2 per fuel rod will not result in an increase of the maximum design linear heat generation rate or a rise of the fuel temperature due to a decrease of the non-uniformity coefficients (decrease of K_{eng} through a decrease of allowances for geometry, density, enrichment, etc.).

4.5 Profiled Fuel Enrichment in Cross Section of Fuel Bundle

VVER fuel is produced with six ^{235}U enrichment values, namely, in %: 1.6; 2.0; 3.0; 3.3; 3.6; 4.4. The Profiled enrichment of fuel in FAs promotes lower non-uniformity coefficients in the distribution of power density within a core and optimization of fuel inventory. Presently VVER-1000 fuel assemblies have two levels of enrichment. In accordance with customers needs, the number of enrichment levers can be increased also through increasing the nomenclature of enrichment.

5 Investigations to Validate Extended Burn-Up in VVER Fuels

The work in this area is under way in the following main directions:

- Research reactor experimental studies of fuel-simulator behaviour to reach a burnup significantly (a factor of 1.5 - 2.0) higher than the design fuel burnup in commercial reactors.
- Extension of life-time of a specific number of commercial FAs by one and in some cases by two fuel cycles promoting the burnup slightly higher than the design one with the simultaneous resolution of other important tasks: corrosion, fretting corrosion, geometrical stability of FAs, etc.
- Improvement of existent and development of new thoroughly verified design methods and codes with the provision of a full set of the needed properties enabling a reliable prediction of the fuel life-time and performance and foundation of licensing criteria.

Table 6 Major results of calculations

Parameter		Standard fuel rod design: dia 2.3 *mm*[1]	Hole diameter 1.2 *mm*	Hole diameter 0.8 *mm*
Max. stress at steady-state conditions, *MPa*	3 years	50	95	101
	4 years	74	115	125
Change of clad diameter in steady-state condition, μm	3 years	-13	20.6	32.2
	4 years	4	70.1	94
Max. stress at power ramp, *MPa*	3 years	80	138	147
	4 years	102	145	157[2]

[1] Limit condition is achieved.

[2] For 2-year cycle fuel hole dia was 1.5 *mm*. It was increased to 2.3 mm in order to increase free volume for fission gases. At present on the basis of FGR results, hole dia is in stage of optimization (some reduction to 1.5 - 1.6 *mm*).

It should be noted that thoroughly verified codes with a full set of fuel and structural material properties are very essential since they enable one to quickly execute and introduce new developments into the practice of the operation of commercial reactors at NPPs.

- Refinement of the existent licensing criteria or development of new ones for fuel of extended burnup, longer time between reloads and lifetime of operation (to 5 - 6 years).
- Extended research reactor MIR loop channel irradiation (conditions typical of VVER) of refabricated or individual full-scale fuel rods specially sifted after dismantling commercial FA and preliminary non-destructive examinations of fuel rods in a hot laboratory.

At this stage our efforts are primarily directed to research activities to study the re-irradiation of full-scale fuel rods. In this case one can obtain adequately unbiased experimental data on the life-time characteristics including the maximum design abilities. Besides, by performing in the process of re-irradiation some experiments (power ramps) or non-destructive assays in a cooling pond or hot cells (non-destructive measurements of mixed gas pressure inside a fuel rod, eddy - current control, etc.) data can be obtained that are very important for code verification and refinement of licensing criteria.

Very soon at SRIAR the work will be completed aimed at mastering the procedure and fabrication of equipment designed for this kind of investigations. Experiments with re-irradiation of commercial full-scale fuel rods are planned for 1995. The organization and performance of this type of activities enable a speedy generation of data on the behaviour of commercial fuel rods at burnups significantly higher than those in commercial reactors.

To-date in Russia a large scope of experimental work has been carried out to study the research reactor behaviour of prototype fuel rods to burnup a factor of 1.5 - 2.0 higher than the design one in commercial reactors. Post-irradiation examinations of a rather large number of such fuel rods show that fuel rod relevant design and technological solutions developed for commercial reactors are optimal for promotion of fuel burnup ~ 60 *MWd/kgU* and higher (mean discharged one). Tables 6 and 7 list the initial parameters and service conditions for some pilot fuel rods that were successfully MR reactor tested up to extended burnup. The detailed results of their post-irradiation examinations are discussed in [1]. However, it should be noted that several features relevant to the specific conditions of fuel rod operation within full-scale fuel assemblies in commercial reactor cores (high neutron fluence, substantial temperature gradients in radial and azimuthal directions, etc) will require a new structural material that compared to Zr-1%Nb should have a higher irradiation resistance, a higher corrosion resistance under conditions under conditions of cladding wall temperature rise and low boiling as well as low corrosion sensitivity to variations in the water chemistry, including free oxygen generated in the coolant above permissible amounts. This structural material is available in Russia, it is Zr-Sn-Nb-Fe.

6 Conclusions

The main conclusions are listed below:

1. The work recently performed in Russia to improve fuel and fuel cycles promoted the reliable operation of VVER-1000 in a three year mode and VVER-440 in a four year mode at the mean discharge fuel burnup ~ 42 - 43 *MWd/kgU* in FA sets. In 1991 - 93 in Russia and Ukraine, the frequency of VVER fuel rod leakage did not exceed on the average $2 \cdot 10^{-5}$.
2. Comparison between the individual characteristics of Russian VVER fuel performance and those of the standard PWR fuel shows that in a whole series of characteristics important for further increasing the life-time, such as: water-side corrosion and hydrogen uptake by fuel claddings, corrosion induced cracking, fuel behaviour in power ramps - the Russian fuel is much superior to the standard PWR fuel which evidences the higher potential abilities of the Russian fuel for further extension of burnup in VVER reactors.
3. The programme of work now under way in Russia to create new improved fuel is aimed, on the one hand, at higher fuel utilization and, on the other, at further improvement of technico-economic parameters of fuel cycles through extension of life-time of structural materials and fuel rods to promote the mean fuel burnup up to 55 - 60 *MWd/kgU* and higher.
4. A large scope of experimental work has been carried out in Russia to study the behaviour of prototype fuels in research reactors up to burnup significantly a factor of 1.5 - 2.0 higher than the design burnup of commercial reactor fuel. The investigations show that the design and technological solutions on fuels developed for commercial reactors are optimal. However, some specific features that are relevant to the operation of fuel rods within a full-scale fuel assembly in commercial reactor cores will need a new structural materials that compared to Zr-1%Nb alloy should have higher irradiation and corrosion resistances for operation at higher temperature of a cladding wall and low boiling of coolant as well as a weak corrosion sensitivity to water chemistry variations, including free oxygen in the coolant. This alloy is already being produced commercially in Russia: Zr-1.2%Sn-1%Nb-(0.3 - 0.4)%Fe.

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Operational Indices of VVER-1000 Fuel Assemblies and Their Improvements

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1 Introduction

This paper deals with the most general design features of VVER-1000 fuel assembly as compared with the other prototypes. The advantages of design are stated as well as their operational confirmation and the main operational occurrences. The essence of anomalies occurred in 1993-1994 concerning the time of emergency protection (EP) actuation. Measures to eliminate these occurrences are described. The main activity to enhance the design and to provide high economic indices of VVER-1000 fuel cycle are outlined.

2 Design Features

As it is known, the VVER-1000 fuel assembly design differs greatly from the western ones. It is predetermined by the original solutions including the peculiarities of materials, technologies, reactor design whose development was performed under the conditions of information isolation from the Western countries as well as by the continuity of VVER-based developments. Objective comparison indicates a number of significant advantages in the VVER-1000 fuel assembly designs:

a) "packing" density (tight-lattice) of fuel rods within the fuel assemblies and the latter within the core;
b) simple handling with the fuel assemblies, its small vulnerability during fuel handling procedures;
c) good conditions for coolant mixing;
d) protection of the absorber rods against coolant effect;
e) adaptability to manufacture that provides stable quality.

3 Main Operational Indices

During 10 years (from 1982 till 1992 including the period of transition for 3-year cycle) the fuel assembly structure operated successfully at 17 units with VVER-1000. The results of it were as follows:

- average burnup was reached to be about 45 *MWd/kgU*;
- out of 8500 fuel assemblies being examined only 193 ones were recognized to be leaky (it is 0.0072%), 8 fuel assemblies were unloaded ahead of time;
- about 170 fuel assemblies were in successful operation for 4 years;
- average activity of coolant was decreasing as shown in Fig.1.

Application of 18 guide channels of stainless steel have provided no radiation growth of the fuel assembly as a whole and cancelled the question on preserving the channel geometry during LOCA.

Minimum corrosion of all the fuel assembly components has been provided.

Radiation growth of the fuel assemblies is stable and is provided by the fuel assembly design.

Behaviour of the fuel rods, spacer grids, fuel as-

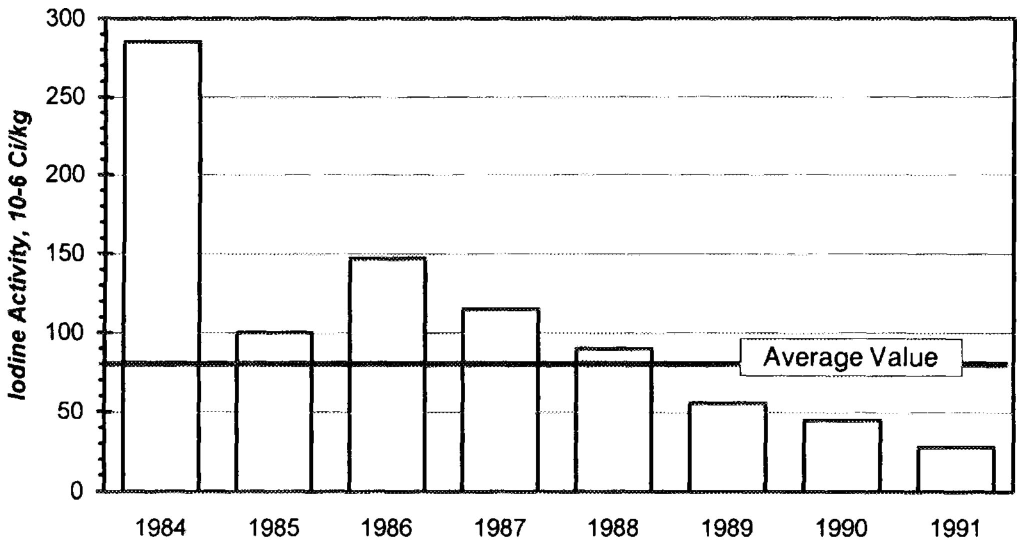

Figure 1 Total average activity of Iodine isotops for all units at the moment of unloading

sembly cap springs have been studied and found to be, mainly, in conformity with the design prerequisites.

Worth and burnup rate of the absorbing elements and burnable poison rods have been revised.

For this period some operational occurrences were observed, among those we should highlight the following:

- Till 1986 delay in emergency protection actuation was practically at all units. In 1986-1987 it was eliminated by replacing the CRDM screw-type detectors by the linear ones.
- In 1983 the South Ukraine NPP experienced failure of the reactor coolant pump (RCP) bearings that resulted in carbon-containing deposits on the fuel rods and caused increased depressurization of the fuel assemblies
- In 1991 Zaporozhye NPP experienced partial degradation of the RCP thermal shield, ingress of metal chip into the core, increased depressurization of the fuel assemblies
- In 1991 Kalinin NPP after the preventive maintenance experienced ingress of organic materials.
- In 1987 undetected failure of the fuel assembly by the mast of fuel handling machine was observed at Kozloduy NPP during fuel assembly loading into the core. This and further deviation from the technological schedule during circulation loop connection resulted in increased depressurization of the fuel assemblies in 1988. A value of fuel assembly depressurization exceeded the normal operational indices (10^{-5} - 10^{-6} *Ci/l*) but did not exceed the allowable ones.

4 Provision for Emergency Protection Reliability

Since 1993, with other indices of VVER-1000 NPPs being favorable, the time delays in EP actuation (> 4 *sec*) have been observed. In June 1994 this question was reviewed at IAEA experts' meeting.

An ad-hoc interbranch commission of Russian and Ukraine specialists have been formed to study and to eliminate the reasons for such occurrences. Under review were the following versions:

a) influence of water chemistry including formation of deposits during boiling;

b) fuel assembly and guide channel distortion as a result of radiation and thermomechanical effects;

c) fuel assembly and guide channel distortion as a result of axial loads upon the fuel assemblies.

Apart from it the influence of observed disturbances upon safety was under investigation. In accordance with these versions the investigations have been performed in the cooling pond and reactor at Unit 2 of Balakovo NPP and in NIIAR hot cell.

The main works performed at Balakovo NPP are as follows:

- visual examination by means of TV-camera;
- corrosion products sampling in the guide channel;
- measuring the guide channel diameter;
- measuring the guide channel curvature in the separate fuel assemblies and in the core;
- measuring the spring performances;
- measuring the fuel assembly cap elevation and the forces of their preliminary compression;
- measuring over the reactor components.

A set of special equipment have been designed and fabricated to perform these works.

Two faulty fuel assemblies from Zaporozhye NPP were disassembled in NIIAR hot cell and subject to the detailed examination. Among other results it was confirmed that no deposits and foulings were available on the channel walls and inlet holes as well as no radiation growth of the guide channel and uniform radiation growth of the fuel rod.

As a result of all activities performed it was set up that the main reason for delay in EP actuation was increase of friction between the guide channel and the absorber element due to fuel assembly distortion caused by the axial loads exceeding the design value.

To provide power unit startup the reactor internals modification have been performed as well as the calculation analysis of safety. This calculation includes the analysis of accident progression with EP actuation time delay up to 10 *sec* as well as the thermohydraulic analysis of fuel rod heat engineering reliability within the fuel assembly with the measured curvature and respectively increased water gaps between the fuel assemblies.

To cope with operational occurrences the standard set of measures have been elaborated to retrofit the units to the design state. These measures have been already implemented at 5 units.

The number of occurrences prior to modification amounted to the following: 36 for Balakovo - 2 NPP, 29 for Balakovo - 3 NPP, 3 for South Ukraine - 2 NPP and 5 for Kalinin NPP.

After modification no disturbances were observed.

Such experience proves the absolute necessity for these measures to be implemented. However, some NPPs when linking the quantity of disturbances with duration of fuel assembly irradiation make decision on usage of the shortened fuel cycle. Nowadays, it is possible to say that economical losses from it can't be compared with losses for unit modification.

The basic conclusion is that references for the negative influence of the fuel assembly inherent properties, incapability thereof to operate adequately during all the design life are not correct.

At the same time the experience gained gives a lot of information for further improvement of the fuel assembly design. And one of the main conditions to be accounted for should be the detailed study of reactor properties, its geometry and heat engineering conditions.

The said disturbances and methods of their elimination have no influence on the program of fuel improvement as planned by Russian Minatom. Moreover, its implementation will increase the operational inventories of fuel. The main arrangements of this program are published in the proceedings of IAEA experts' meeting in April 1994. Some easily implemented solutions will be realized in 1995 fuel delivery without changes made in the terms and conditions of delivery.

Other solutions, in particular, replacement of the guide channel and spacer grid materials will be implemented since 1997. For this period of time the thermomechanical properties of fuel rod bundle will be investigated and the operating experience will be gained at NPP.

5 Direction and Status of Improvements

5.1 Improvements to Increase the Fuel Economy

Certain improvements aim to incease the efficiency of fuel utilisation:

1. Replacement of the guide channel and spacer grid materials by the zirconium alloy. It has 5-year operating experience with VVER-440. Pilot operation with VVER-1000 is going on since 1993.
2. Provision for maintainability of the fuel assemblies. It is reached owing to the cap being remotely detached and the fuel rods being remotely withdrawn without any mechanical failure. Pilot operation is under way since 1993.
3. Application of Gadolinium burnable absorber incorporated into the fuel instead of boron burnable poison rods. Pilot operation is going on since 1993.
4. Application of the new materials $Dy_2O_3TiO_2$ and Hf for absorbers in the region of intensive neutron flux. There is a positive operating experience of $Dy_2O_3TiO_2$ in VVER-1000 during 4 years in the automatic control group. At stage 1 the extended service life is achieved as well as the gravitational weight. At stage 2 the service life is above 10 years, physical and gravitational weights are increased. Materials testing has been completed whereas mechanical life tests together with control rod drive mechanism are coming to the end.
5. Fuel cycle optimization (implementation of 4-year cycle, application of "in-out" refuelling schemes.
6. Implementation of pitch-by-pitch detectors (pitch is 20 *mm*) to monitor the control rod movement.

5.2 Modifications to Improve Control Rod Operability

Some modifications aim to improve dimensional stability and to enhance reliability of control rod insertion into the core:

1. Optimization of the fuel assembly axial loads in VVER-1000.
2. Core layout with low non-uniformity of power distribution due to better profiling over the core and fuel assemblies.
3. Implementation of the more strict control for water chemistry.
4. More rigid fuel assembly structure.
5. Application of fuel cladding and guide channel materials providing the better dimensional stability under irradiation.
6. Increase in the control rod gravitational weight.
7. Implementation of the regulatory restrictions for heatup conditions.

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Status and Prospects of VVER In-Core Fuel Management Activities

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In Russia extensive calculations and experimental studies on physics and fuel has been recently carried out to substantiate the possibilities of introducing the modernized VVER fuel cycles into practice:

- using the results of physical experiments and the operation data, a complex of codes for physical calculations of the VVER core characteristics has been developed and verified [1];
- based on the data of loop tests and post-reactor studies, codes of thermal and physical and mechanical calculations of the fuel elements (PIN-micro, PIN-mode 2, PIN-mod2 (Gd), RET (R), START, PULSAR) have been developed and verified;
- an extensive series of calculations and experimental studies on optimization of the VVER core neutronic characteristics has been carried out;
- a large volume of works on improvement of process and operation characteristics of fuel elements and fuel assemblies has been accomplished (fuel pellet fabrication quality has been increased [5], initial He-backfill pressure in VVER-440 fuel rode has been optimized [6], fuel assembly shroud tube thickness has been reduced and thus the water-uranium ratio increased;
- four-five-year positive operation experience with the VVER-440 fuel assemblies has been gained [7], transition to the use of ARK SUZ (CPS) fuel assemblies with 3.6% fuel enrichment has been substantiated and realized [8], at the Kola NPP a four-year fuel cycle using 4.4% enriched fuel assemblies has been successfully employed, with an average burnup of 48.2 *MWd/kg* being reached [9] (Fig. 1); twenty four 4.4% enriched fuel assemblies had been operated in the Kola NPP-3 reactor for five years, which made it possible to substantiate the possibility of implementing a five-year fuel cycle; at the Novo-Voronezh NPP the fuel cycle based on the fuel assemblies with zirconium spacers has been implemented [10], some of these fuel assemblies had worked for four years; in some units loading schemes with partially reduced neutron leaks are used [11];
- on enhance reactor power positive operation experience with the VVER-440 fuel has been gained (107% P_{nom}) [12];
- in order to reduce the non-uniformity of specific loads in the fuel assemblies and to facilitate realization of refueling schemes with reduced neutron leaks (LLLP) the technology of profiling the enrichment over the fuel rod bundle cross section has been mastered;

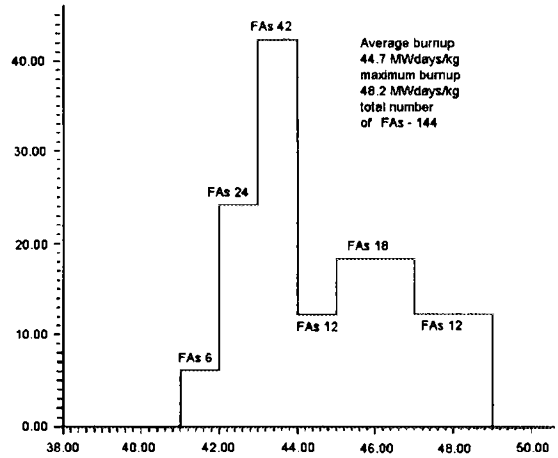

Figure 1 **Distribution of fuel burnup in VVER-440 Fas (4.4% enrichment), operating 4 years (Kola NPP, Unit 3)**

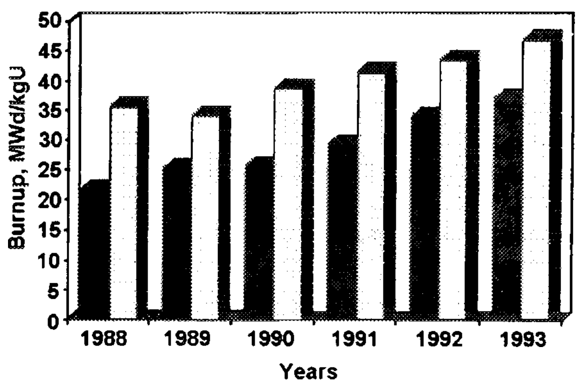

Figure 2 **Variation of average (dark columns) and maximum fuel burnup within last six years**

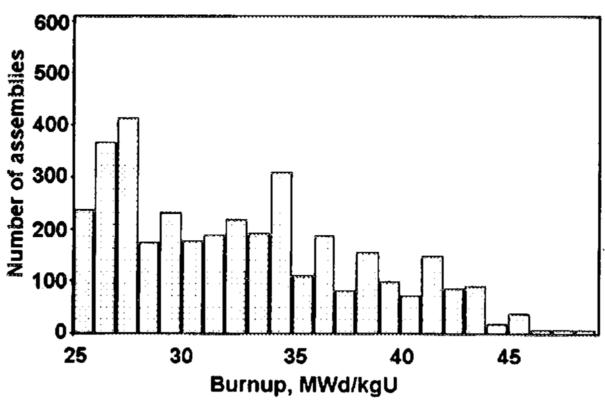

Figure 3 **Distribution of fuel assemblies over the fuel burnup**

- all VVER-1000 reactors (except for units 5 and 6 of the Kozloduy NPP) are changed over to the three-year fuel cycle with reaching average fuel burnup of about 42 *MWd/kg* and an average burnup of about 48.5 *MWd/kg* in the maximum burn-out fuel assembly (Fig. 2); a part of the fuel assemblies (about 190 *pc*) had been operated for four years; the increase in the fuel burnup with gaining the operation experience is shown in Fig. 3;
- at the Balakovo NPP experimental operation of six fuel assemblies with zirconium spacers and zirconium guiding channels has been accomplished;
- positive experience with the use of refueling schemes with reduced neutron leaks has been gained [13];
- a technology of fabricating a fuel with integrated burnable poison (uranium-gadolinium fuel) has been developed [1]; 12 fuel assemblies with uranium-gadolinium fuel (UGF) have been fabricated and installed into the Balakovo NPP-3 reactor for experimental operation, each fuel assembly containing 18 fuel elements with gadolinium dioxide (Gd_2O_3) integrated into the fuel; fuel enrichment 3.6%, Gd_2O_3 concentration - 8%.

Based on the experience gained, modernized fuel cycles have been developed. The following improvements have been introduced into the VVER fuel cycle characteristics:

- increased fuel burnup;
- reduced natural uranium consumption and decreased amount of separation work per energy output unit;
- increased efficiency of the reactor emergency protection;
- reduced fast neutron flux onto the reactor vessel.

The main comparative economic indices are listed in Tables 1 and 2.

Main Characteristics of Modernized Fuel Cycles

A. Reactor VVER-440

At present development of advanced four- and five-years fuel cycles with the use of Fas with enrichment profile in cross section of fuel rods bundle and with zirconium spacer grids has been completed.

The main characteristics of the developed fuel cycle are given in Table 3.

B. Reactor VVER-1000.

At the present development of three- and four-year fuel cycles with the use of boric rods and gadolinium dioxide as burnable poison (BPR) has been completed [13].

Pellet central hole diameter in improved fuel cycle is 1.4 *mm*.

The main characteristics of the developed fuel cycles are listed in Table 4.

Experience of operating and the fuel rod calculations show that they maintain reliability up to extended burnup.

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Table 1 Specific fuel consumption in design and modernized VVER-440 fuel cycles (estimates)

No	Characteristics	Design	Requirement	Effect of specific fuel consumption decrease, %	State
1	Number of refueling	3	3	- 7.8	yes
			4	~6.3 ~14	yes
			5	~9.2 ~16	
			3 (in 18 months)	~3.0 ~4.7	4 - 5 years
2	Structural material of spacing grids	steel	zirconium	2.0 - 2.5	yes
3	Loading pattern	out-in-in	in-in-out (LLLP)	4.0 - 5.0	yes
4	Type of burnable absorber	no	no		yes
			Gd_2O_3	18-month f.c.	
5	Profiling of fuel enrichment in cross section	no	yes	0.5 - 1.0	1 year
6	Profiling of fuel enrichment in high of fuel rod	no	yes	0.5	

Table 2 Specific fuel consumption in design and modernized VVER-1000 fuel cycles (estimates)

No.	Characteristics	Design	Advanced fuel cycle	Effect of specific fuel consumption decrease, %	State
1	Number of refueling	2	3	13	yes
2	Structural material of spacing grids, GTs	steel	zirconium	8	~3 - 4 years
3	Loading pattern	out-in	in-in-out (LLLP)	4.0	yes
4	Type of burnable absorber	Boron rods	Gd_2O_3	2.0	~3 - 4 years
5	Material of absorber rod	B_4C (natural)	B_4C enriched Ag-In-Cd		
6	Profiling of fuel enrichment in high of fuel rod	no	• yes	0.5	

Table 3 Refined neutron physical characteristics of reference four year fuel cycle (equilibrium fuel reloading regime) for VVER-440

Serial No.	Characteristics	Value		
		Odd year of operation	Even year of operation	Design fuel cycle
1	Reactor power, *MW (th)*	1375		1375
2	Number of fuel assemblies in the core	349		349
3	Number of CPS members	37		37
4	Geometry of fuel assemblies	standard, thickness of independent FA shroud is 1.5 *mm*		stainl
5	Structural material of spacing greeds	zirconium		stainl
6	Profiling of fuel enrichment in FA	used		not used
7	Type of burnable absorber	not used		not used
8	Total weight of uranium in the core, *t*	41.8		41.8
9	Time between refuelings, months	12		12

Table 3 (cont.)

Serial No.	Characteristics	Value		
		Odd year of operation	Even year of operation	Design fuel cycle
10	Loading patten	L^3P		out-in^2
11	Boron concentration in the coolant during refuelings, *ppm*	2450		2100
12	Number of fuel assemblies to be replaced at refueling, including types	90 78A, 12H	91 78A, 12H, 1I	117.5 12.5E, 84C, 21B
13	Operation time of fuel assemblies in the reactor, *years* average maximum	 3.87 4	 3.85 4	 2.98 3
14	Average enrichment of make-up fuel, *weight* %	3.663	3.611	3.26
15	Average burnup depth of inloaded fuel, *MWd/kg* for all FA for independent FA for control FA	 38.22 39.95 (A) 26.98 (H)	 38.02 40.17 (A) 25.86 (H), 16.32 (I)	 27.9
16	Duration of reactor operation between refuelings, eff. days	303.7	301.8	286
17	Water temperature coefficient of reactivity (TH2O=260°C, zero power, unpoisoned state, all members of CPS are in upper position), 10^{-5} °C^{-1} BOC EOC	 -0.63 -22.36	 -0.65 -22.35	 -2.95 -23.2
18	Uranium temperature coefficiet of reactivity (see p.17 above), 10^{-5} °C^{-1} BOC EOC	 -2.43 -2.43	 -2.43 -2.44	 -2.52 -2.51
19	Effective delayed - neutron fraction, % BOC EOC	 0.696 0.613	 0.697 0.613	 0.70 0.69
20	Prompt neutron lifetime, μs BOC EOC	 21.08 23.46	 20.98 23.44	 20.9 22.7
21	Efficiency of control rod group at TH_2O=260°C, % BOC EOC	 1.869 1.918	 1.813 1.867	 1.83 1.85
22	Power coefficient of reactivity (at constant average temperature of coolant), 10^{-5} MW^{-1} BOC EOC	 -0.62 -0.60	 -0.61 -0.60	 -0.62 -0.62
23	Power effect, % BOC EOC	 1.31 1.46	 1.31 1.46	 1.21 1.44
24	Xe-135 poisoning effect, % BOC EOC	 2.87 2.97	 2.87 2.97	 2.87 2.82
25	Reactivity margin compensating fuel burnup, %	8.77	8.73	8.04

Table 3 (cont.)

Serial No.	Characteristics	Value		
		Odd year of operation	Even year of operation	Design fuel cycle
26	Total efficiency of emergency protection, %			
	BOC	10.727	10.496	9.41
	EOC	11.015	10.790	10.95
	under conditions of stricking of more effective CPS member in upper position, %			
	BOC	8.125	7.617	7.02
	EOC	8.324	7.796	6.79
27	Temperature of repeated critically at the end of fuel loading when boron in coolant is absent, °C			
	Xe-135 poisoned state			
	CPS all members in lower position	<20	<20	<100
	CPS all members except for more effective one in lower position	<100	<100	150
	Xe-135 unpoisoned state			
	CPS all members in lower position	20	20	<100
	CPS all members except for more effective one in lower position	<200	<200	230
28	Initial value of critical boron concentration in the coolant (poisoned state of reactor at full power), *ppm*	1041	1036	935
29	Power peaking of fuel assemblies, K_q			
	BOC	1.349 (53)	1.343 (15)	1.24
	EOC	1.344 (42)	1.341 (15)	1.26
30	Maximum value of average coolant temperature at fuel assembly outlet, °C			
	BOC	311.9	312	308.1
	EOC	311.8	311.9	308.5
31	Core volume power peaking (for 3490 nodes), K_v			
	BOC	1.720	1.682	1.66
	EOC	1.617	1.613	1.75
32	Power peaking of fuel pins in cross-section of fuel assembly fuel bundle having the greatest power, K_k			
	BOC	1.06	1.08	1.15
	EOC	1.05	1.05	1.12
33	Maximum value of specific linear load of fuel pins (with regard for engineering safety factor), *W/cm*			
	BOC	292	286	287
	EOC	275	274	285
34	Effective specific natural uranium consumption, *kg/MWd*	0.212	0.212	0.258
35	Effective specific separated work, *SWU/MWd*	0.120	0.120	0.142
36	Average power of periphery FA's (K_q), relative unit			
	BOC	0.379	0.387	0.74
	EOC	0.440	0.442	0.74

Table 4 VVER-100 neutronic characteristics in various methods of fuel cycle organization (steady-state refueling conditions)

No.	Characteristic	Version of fuel cycle organization				
		Designed 3-y cycle	Modernized 3-y cycle	Modernized 4-y cycle	Modernized 3-y cycle	Modernized 4-y cycle
1	Thermal power, *MW*	3000	3000	3000	3000	3000
2	Number of fuel assemblies in the core, *pcs*	163	163	163	163	163
3	Number of control rods (SUZ)	61	61	61	61	61
4	Fuel assembly geometry, fuel rod design	normal	Increased diametr of control rod guide tubes			
5	Structural material of spacing grids and guiding tube of control rods	Stainless steel	Zirconium alloy	Zirconium alloy	Zirconium alloy	Zirconium alloy
6	Profiling of fuel enrichment overthe fuel assembly cross section	used	used	used	not used	not used
7	Burnable absorber	Boric BAR	Boric BAR	Boric BAR	Gd_2O_3	Gd_2O_3
8	Weight of uranium in the core, *t*	65.	71.1	71.1	70.9	71.0
9	Refueling scheme	out-in-in	LLLP	LLLP	LLLP	LLLP
10	No. of fuel assemblies changed in refueling	54	54	42	54	42
11	Average enrichment of make-up fuel, *wt%*	4.31	3.52	4.23	3.53	4.16
12	Average burnup of discharged fuel *MWd/kg*	40.3	37.1	48.1	37.9	47.6
13	Time of reactor operation between refueling, eff.day	291	292	294	297	290
14	Efficiency of emergency protection, %					
	total, BOC	6.6	8.8	8.0	8.5	8.5
	total, EOC	6.6	8.7	8.1	8.4	8.6
	with one most efficient control rod stuck in the upper position					
	BOC	6.0	7.9	7.2	7.5	7.5
	EOC	6.0	7.7	7.3	7.6	7.4
15	Fuel assembly power peaking factor, K_q					
	BOC	1.28	1.31	1.32	1.31	1.33
	EOC	1.27	1.30	1.31	1.30	1.32
16	Volume power peaking factor, K_v					
	BOC	1.44	1.50	1.46	1.39	1.50
	EOC	1.34	1.42	1.40	1.41	1.46
	maximum	1.44	1.50	1.46	1.47	1.50
18	Max. linear power of fuel rod (with allowance for engineering hot channel factor for heat flux K_{eng}=1.20) *W/cm*					
	BOC	340	365	343	344	378
	EOC	300	313	309	315	324
19	Specific consumption of natural uranium, *kg/MWd*	0.243	0.211	0.199	0.208	0.198
20	Specific volume of separated works, *SWU/MWd*	0.146	0.118	0.119	0.116	0.118
21	Average power of periphery fuel assemblies, relative unit					
	BOC	0.78	0.40	0.57	0.58	0.58
	EOC	0.90	0.53	0.62	0.62	0.63

Status and Prospects of Activities on Algorithms and Methods in VVER-1000 Core Control

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Introduction

VVER-1000 reactors, in common with analogous PWRs, possess anaxially instable power density field because of the non-equilibrium processes of redistribution of Xe-135 nuclei in the volume of the reactor core. This instability results in so-called xenon oscillations of the power density distribution. To prevent and suppress such oscillations is one of the reactor control problems. For VVER-1000s the period of the xenon oscillations is about 300 hours; the amplitude depends on the mechanism of their excitation,and initial deformation of the power density distribution; the convergency of the oscillationsis maximum at the beginning of the campaign to decrease with fuel burnup. To describe the xenon oscillations use is made of an axial offset (AO). This quantity characterizes an axial non-uniformity of the power density distribution to be equal to the percent ratio of the difference between the released powers in theupper and lower halves ofthecore to the total released power. In the "ascending phase" of the xenon oscillations where the power is redistributed from the lower half of the core to its upper half the offset increases; in the "descending phase" where the power is redistributed from the upper to the lower half of the core the offset decreases.

The problem of the xenon oscillations was worked out in Russia- 3 -even before the appearance of large-size, high-capacity VVERs for which this problem would be actual [1, 2].

In the process of designing a pilot VVER-1000 unit (Novovoronezh-5) the generation of the xenon oscillations was predicted for this type of reactors and the basic parameters of the oscillations were determined by computer simulation. After commissioning of the pilot unit and the first units with commercial VVER-1000s, in 1980-1985 a number of experiments were made to study the xenon oscillations [3, 4]. Figure 1 shows the plots of "free" xenon oscillations obtained during the investigations in the first campaign of Zaporozhye-1 NPP. The oscillations were excited by lowering the control rods (CRs) of a working group down to a position of 40 - 50% core height with the reactor power unchanged. The first curve was plotted after the operation for 40 eff.d at 73% of rated power; the second curve corresponds to 175 eff.d and 97% of rated power.

As seen in the figure, the xenon oscillations become divergent in the second half of the campaign at full-power operation. Nevertheless, as the practice shows, in the steady-state operation of the reactor the power density distribution is sufficiently stable. This is explained by an interesting side effect arising from the operation of an automatic power controller (APC). The fact is that, as the burnup increases, the effect of the axial oscillations of the power density distribution on the total power of the reactor begins to manifest itself. In stabilizing the power the APC moves the CRs in a way to ensure a negative feedback to a change in the AO and suppresses the xenon oscillations. Thus, the APC serves also as a regulator of the axial power density profile, though it was not designed for this purpose. Figure 2 gives the mechanism of APCs operation in that case and an example of the automatic suppression of the xenon oscillations by the APC at Novovoronezh-5 NPP (12th campaign, 244 eff.d).

In the case of strong disturbance of the steady-state operation of the reactor resulting, as a rule, from variations in its power, there appears a need for controlling the power density distribution. In the course of the above investigations of the xenon oscillations algorithms for their suppression were developed and appropriate instructions for NPP personnel were worked out. The algorithms are of a discrete character and resemble Westinghouse's BANG-BANG CONTROL algorithms. As a rule, a single group of control rods is used for the control. Provision was also made for an auxiliary group of half-length control rods which should be dropped into the lower half of the core, if the descendent phase of xenon oscillations was intensively developed.

The experience in operating VVER-1000s had demonstrated that the available control algorithms allow, with a fair degree of assurance, the stabilisation of the axial offset and the prevention of intensive xenon oscillations. In this case there is no need

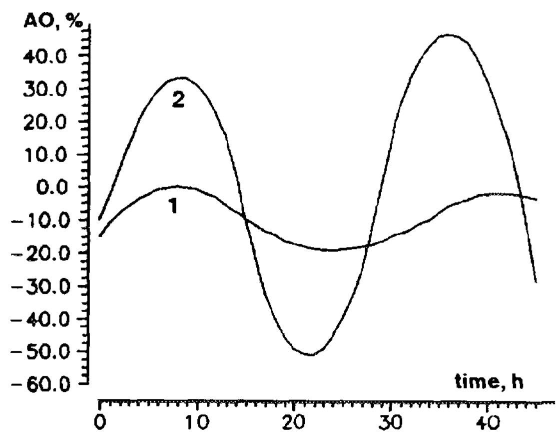

Figure 1 Free axial xenon oscillations in the core of commercial VVER-1000: (1) 40 eff.d.; (2) 175 eff.d.

to use the half-length control rods which may be replayed by full-length ones. The corresponding works are under way at most of NPPs with VVER-1000s.

At present the further works on improving the control of the reactor power and power density distribution in VVER-1000s are oriented to the following problems:

- Even though present-day VVER-1000s operate under steady-state conditions, there may occur situations (equipment failures or an accident to the power supply system) involving a sufficiently deep and complex power cycling. In these cases the NPP personnel must be provided with efficient means of monitoring and predicting the state of the reactor;
- The abandonment of the half-length CRs imposes more strict limits on the permissible deformation of the power density distribution. This requires modification of the control algorithms;
- The development of power-cycling regimes demands the appropriate algorithms, the automation of the control and the computation of fuel power density.

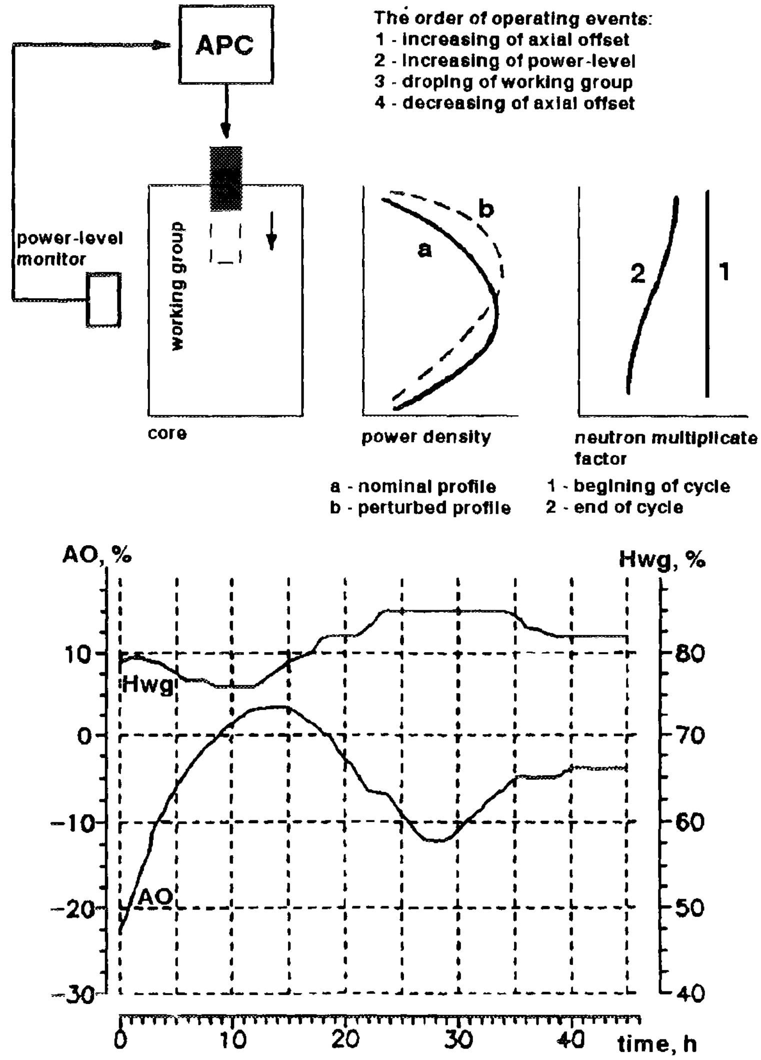

Figure 2 Suppression of the xenon oscillations by the automatic power regulator (Novovoronezh NPP, unit 5)

1 Transients Simulation and Monitoring, Operators Support and Control Automation

Of great importance for the control of the power density distribution is an adequate assessment of the current state of the reactor and the prediction of its response to various control actions. A computerised operators' adviser (OA) may provide a solution to this problem. In particular, such work is in progress now as applied to Novoronezh-5.

One of the OA elements is planned to be a computer code reactor simulator (RS) realising a physical reactor model based on a BIPR-7 code commonly used for design and operational calculations of VVERs [5]. According to the present-day views, the RS must be capable, in the on-line application, based on the data of the conventional monitoring system, to reconstruct the current state of the reactor, to predict the behaviour of the reactor as well as to allow the operator to follow some variants of the control and to seek advise.

One of the basic problems in controlling the axial power density profile is to determine the current phase of the xenon oscillations and to present the control process as a continuum of the reactor states in the form convenient for the assessment of the current situation, the prediction of the subsequent events and the selection of control actions. To solve this problem in the OA it is planned to use an "offset-offset" diagram [6]. This is a two-dimensional phase diagram plotted against the instantaneous, AO, and the equilibrium, AO*, offsets. In this case the instantaneous offset corresponds to the current power density distribution. The equilibrium offset is related to the power density distribution in the equilibrium state (from the viewpoint of the relationship between the concentrations of iodine and xenon nuclei in the core) at the current values of the reactor thermal power, the fuel burnup and the CR positions.

Figure 3 schematically shows an offset-offset diagram. The rectangle bounded by the values AO_{max}, AO_{min}, AO^*_{max} and AO^*_{min} encloses a permissible area. The points on diagonal AA correspond to the steady states. The free xenon oscillations will show up as oscillations of a point along a vertical with respect to its intersection with diagonal AA, that is, with respect to a steady-state point (determined by CR positions) to which the current point will tend as the oscillations decay. The control actions (the movement of the CRs) form a trajectory of the process. The trajectory may be specified in advance or determined starting from the situation at each instant of time. The position of the current point with respect to the boundaries of the permissible area and diagonal AA allows one to estimate the degree of instability of the current reactor state as well as the amplitude of the xenon oscillations and the scope of controllability.

The first trajectory in Fig. 3 corresponds to the free damped oscillations; the second trajectory is representative to the control process holding the offset within the permissible area. Here the control actions consist in the movement of the control rods in the upper half of the core. In so doing the values AO and AO* change simultaneously, and in the

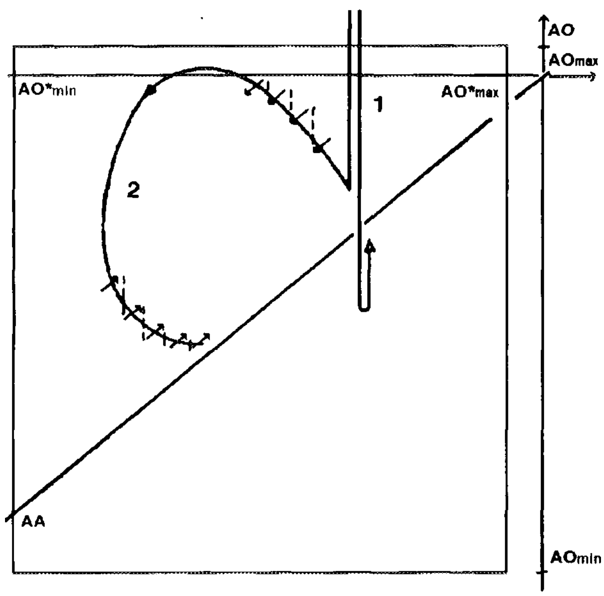

Figure 3 Offset-offset diagram

"instantaneous" movement of the rods the point on the diagram moves along diagonal AA.

In practice the movement in a specified trajectory can be performed in steps when the "instantaneous" control action and the development of free oscillations alternate. In this case a sawtooth trajectory takes place. Such an algorithm is particularly convenient for an automated control, because it can be simply interpreted technologically: the intersection of the specified trajectory by the current point represents an on or off signal for the CR drive. The feasibility of an automatic offset controller on this basis is now under study.

The visualisation of the information required for the reactor control is one of the functions of the OA. Figure 4 schematically shows the format of visualisation of the processes of control of the axial power density distribution in the reactor on the PC display in the OA system. Use is made of the values directly measured in the reactor as well as those calculated with the use of the RS. The display screen is divided into four zones:

- the static characteristic of the equilibrium offset at the current instant of the campaign and the current reactor power is displayed in the left top corner;
- the offset-offset diagram is plotted in the left bottom corner;
- the current position of the working group of the CRs is shown in the right top corner, where the histogram of the power density distribution along the core height is also plotted;
- the time variation of the instantaneous offset is plotted in the right bottom corner.

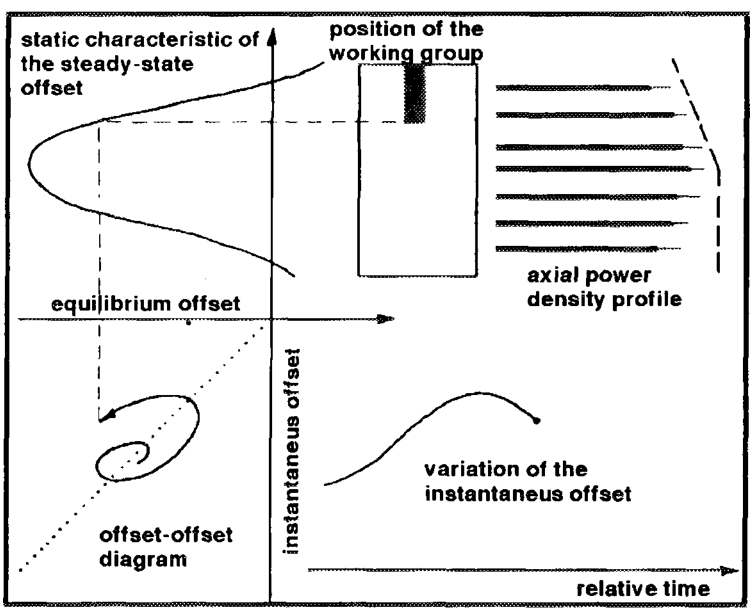

Figure 4 Screen format to visualize the control of the axial power distribution in the graphic module of the reactor simulator.

2 Development of Control Algorithms

In connection with the renunciation of half-length CRs the challenge is to suppress the intensive xenon oscillations in the descending phase with the use of full-length CRs.

An algorithm is suggested based on the following principles:

- the control is realised by the movement of two CR groups: a working group (WG) and a special control "grey" group (CG).These-8 -CR groups compensate simultaneously for changes in reactivity. The CG is chosen so that, when the WG moves up to the top core boundary, the CG drops down to the core bottom;
- when the CG moves in the upper half of the core, the effects of the two groups on the axial power

density distribution are mutually substracted - the value of the offset varies only slightly;

- when the CG moves in the lower half of the core the effects of both groups on the axial power density distribution are summed - the offset changes in the positive direction.

Figure 5 shows qualitatively the dependence of the offset on the WG position in the case of the reactivity compensation by liquid boron (method 1) as well as the dependencies of the CG position and the offset (AO) on the WG position in the case of the movement of the groups for the purpose of the reactivity compensation (method 2). Point A corresponds to an initial WG position, point B to the CG position at the half height of the core and point C to the CG position at the bottom of the core. The integral efficiency of method 1 is higher by DAO than that of method 2. This is explained by the effect of the boron concentration in the coolant on the temperature coefficient of reactivity. The differential efficiencyfor method 2 is two times higher when the CG moves in the lower half of the core.

The same dependences calculated by the BIPR-7 code for O and 240 eff.days of the 12th campaign at Novovoronezh-5 are shown in Figs. 6 and 7.

The computer simulation of the control processes supports the efficiency of the algorithm proposed in suppressing the xenono scillations.

The algorithms currently used for the control of the VVER-1000 reactor in power cycling are oriented to the use of the "black" WG and the sequential movement of the CR groups with an interval equal approximately to the full height of the core (the group begins to move at the instant when the preceding group reaches the boundary of the core).This leads to excessive deformations of the power density distribution which excite the xenon oscillations and restricts the controllability. Figure 8(a) shows a variant of the computer simulation of a real episode of the reactor control when the power was reduced from 100% to 50% (Novovoronezh-5, 12th campaign, 123 eff.d).

In the course of works on the replacement of the half-length CRs and the implementation of the power cycling regimes, under consideration are control algorithms involving less "black" CR groups and a shorter interval between the groups. Figure 8(b) shows a variant of computer simulation of the case considered above where the power is reduced with the use of two CR groups: a working group and a control "grey" group.The interval between the groups is 50 - 60% of the core height. The time dependence of the liquid waste volume in the boron control system is given in Fig. 8(c). The advantages of the modified variant are evident.

A program for testing the modified control algorithms at Novovoronezh-5 is in the preparation stage.

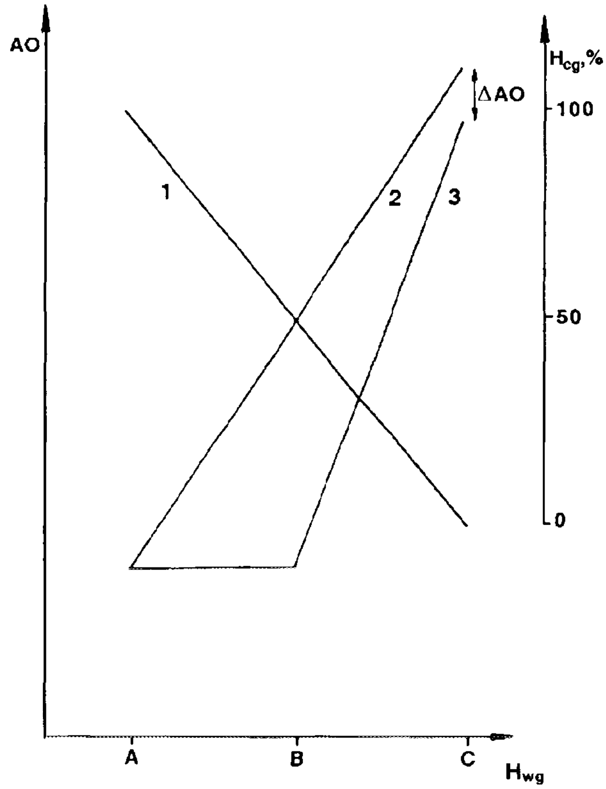

Figure 5 **The offset control:**
(1) $H_{cg}(H_{wg})$ overcompensation by rods;
(2) $AO(H_{wg},C_B)$ overcompens. by boron;
(3) $AO(H_{wg},H_{cg})$ overcompens. by rods.

3 Estimation of the Fuel Element Thermal Load in Reactor Power Changes

A method of estimating the value and rate of change o fthe linear power rating in the fuel element in power cycling is under development.The method is based on the BIPR-7 and PERMAK calculations [7].

The BIPR-7 code allows the thorough simulation of the reactor control as well as the calculation of the accompanying xenon-related transients and the redistribution of the released power in the volume of the reactor core. The value of the relative released power K_v relating to an elementary calculation cell - a fuel assembly fragment - and not accounting for the power distribution among the fuel elements yields the most detailed power density field.

The PERMAK program allows one to calculate the value of K_k -the relative released power of an individual fuel element in a certain layer of the core, but does not permit the simulation of the xenon-related transients and the axial deformation of the power density distribution accompanying the reactor control in the process of power cycling.

The linear power rating (*W/cm*) of a given fuel element in a given segment of the core can be estimated as follows:

$$q_l = \overline{q_l}\, K_v\, K_k$$

where $\overline{q_l}$ is the average linear power rate.

The following assumptions are made:

- the relative power density distribution throughout the fuel elements in the core is axially independent;
- the xenon-related transients do not distort the relative power density distribution throughout the fuel elements inside a fuel assembly;
- when the CRs move, the maximum deformation of the power density distribution in the fuel occurs in the fuel element sections nearest to the ends of the moving absorber rods. In this case the degree of deformation is axially independent.

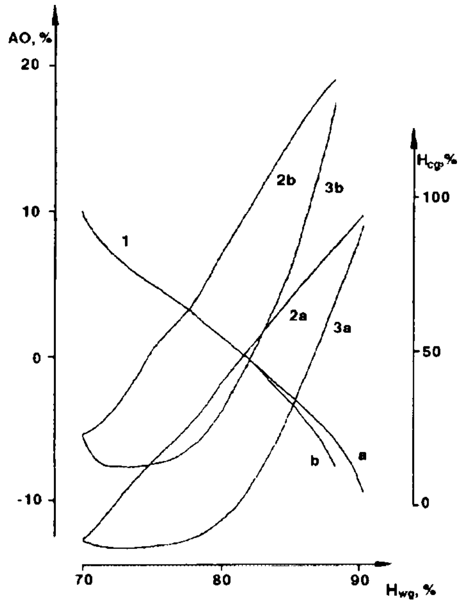

Figure 6 **Offset control at cycle beginning; (1), (2), (3) as in Fig. 5; power level: (a) 100 %, (b) 50 %.**

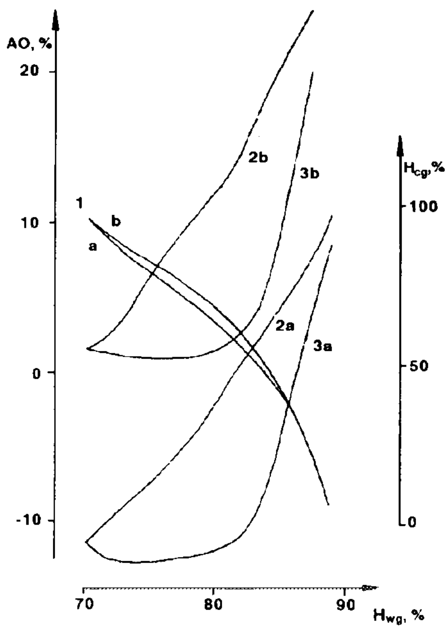

Figure 7 **Offset control at 240 eff. days; (1), (2), (3) as in Figs. 5&6; power level: (a) 100 %, (b) 50 %.**

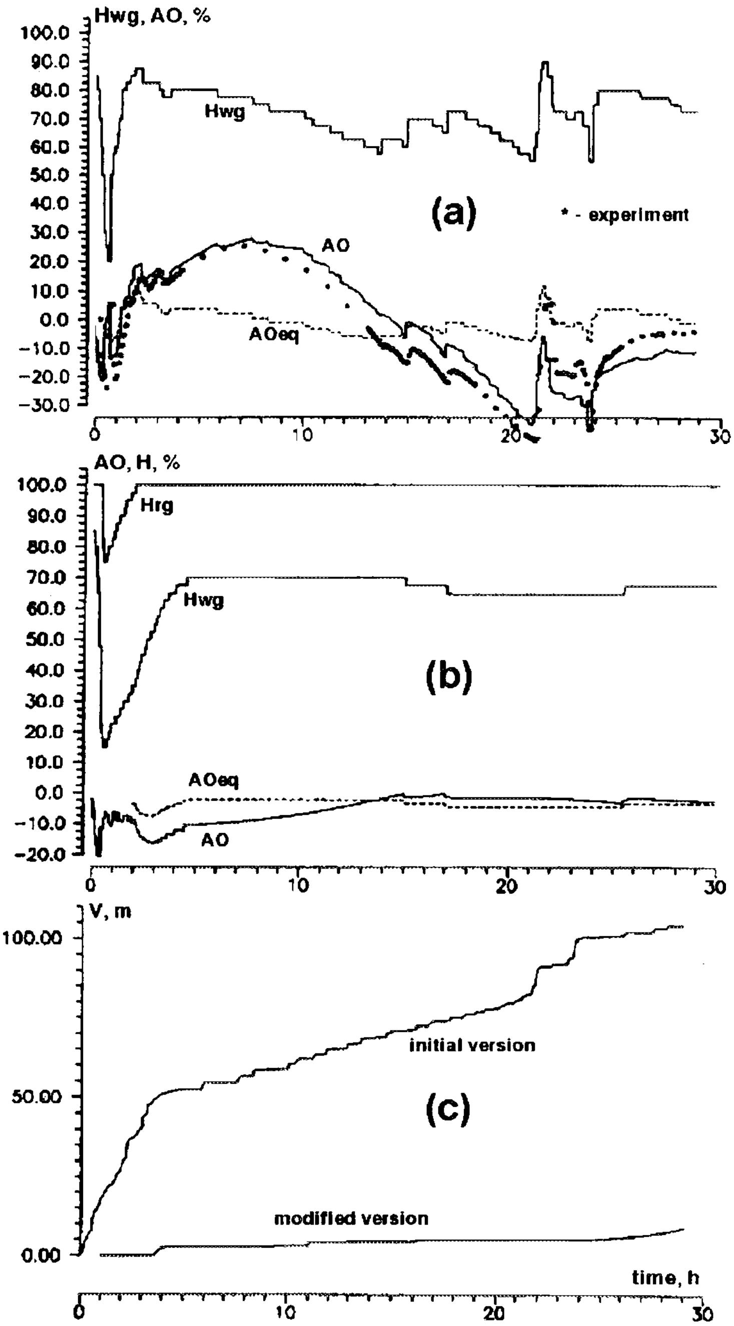

Figure 8 **Power drop from 100% to 50%:**
(a) initial control version;
(b) modified control version;
(c) volume waste in the boron control system.

To substantiate the above assumptions, PERMAK calculations were made for different layers along the core height. The results of the calculations are presented in Figs. 9 - 11.

With allowance for the above assumptions the linear power rating of the fuel in power cycling is calculated as follows:

- the fuel fragments (the fuel element sections) for the calculation of the linear power rating are chosen on the basis of the results of the BIPR simulation; in particular, a fuel element adjacent to the absorber guide tube is taken to determine the maximum rate of change in the linear power rating;
- for a given instant of the campaign, PERMAK calculation of the power density distribution throughout the fuel elements is performed for a single layer (at the half-height of the core) at different positions (in a given layer) of the CRs participating in the reactor control;
- the time variation of the linear power rating q_l is calculated by the above formula with the use of the K_k values determined by the BIPR calculations for the core cells containing the chosen fuel fragments; here the value of K_k (from the PERMAK calculation) is chosen (discretely varied) depending on the presence or absence of the CRs in the core layer enclosing the chosen fuel fragment;
- the analysis of the time variation of q_l yields the value of the rate of change in the linear power rating.

The changes in q_l in reducing the reactor power from 100% to 50% with a rate of 1%/*min* are plotted in Fig. 12. The maximum rate of change in the linear power rating is attained in the fuel assemblies where the WG control rods move to be 24 *W/(cm.min)* at a linear power rating change of 120 *W/cm*. In this case the average linear power rating changes with a rate of 1.6 *W/(cm.min)* by 83 *W/cm*. Thus, the allowance for the micro nonuniformity of the power density distribution in power cycling results in an excess of the local rate and magnitude of change in the linear thermal load on the fuel over the respective values averaged over the core by a factor of 15 and 1.5, respectively.

Layer number from the core bottom	18	19
9 →	1.171/083	1.725/312
7 →	1.168/083	1.723/312
5 →	1.171/083	1.759/312
3 →	1.172/083	1.759/312

Layer number from the core bottom	14	15	16	17
9 →	1.037/006	1.084/055	1.267/322	1.694/312
7 →	1.032/003	1.067/016	1.257/312	1.718/312
5 →	1.036/003	1.071/055	1.263/312	1.732/312
3 →	1.031/295	1.067/016	1.267/312	1.757/312

Layer number from the core bottom	8	9	10	11	12	13
9 →	1.104/043	1.072/051	1.210/315	1.082/322	1.220/160	1.796/315
7 →	1.119/068	1.068/012	1.132/315	1.050/003	1.227/159	1.834/315
5 →	1.125/068	1.072/012	1.137/315	1.050/003	1.231/159	1.895/315
3 →	1.125/068	1.070/012	1.139/315	1.050/003	1.235/159	1.868/315

Layer number from the core bottom	1	2	3	4	5	6	7
9 →	1.082/043	1.149/285	1.094/279	1.105/055	1.312/322	1.148/329	1.463/322
7 →	1.036/179	1.186/285	1.112/279	1.072/008	1.040/059	1.145/309	1.487/322
5 →	1.039/179	1.200/285	1.118/285	1.074/008	1.040/088	1.147/309	1.511/322
3 →	1.039/179	1.207/285	1.120/285	1.076/008	1.038/088	1.149/309	1.511/322

Figure 9 Distribution of the relative released power of the fuel elements with the maximum power density in the absence of clasters (value of the maximum / position of the maximum)

Layer number from the core bottom	18	19
9 →	1.163	1.714
7 →	1.160	1.705
5 →	1.163	1.741
3 →	1.164	1.742

Layer number from the core bottom	14	15	16	17
9 →	1.034*	1.066*	1.251*	1.694
7 →	1.042*	1.060*	1.225*	1.701
5 →	1.045*	1.055*	1.231*	1.715
3 →	1.038*	1.059*	1.235*	1.740

Layer number from the core bottom	8	9	10	11	12	13
9 →	1.106	1.053*	1.118*	1.040*	1.206*	1.798
7 →	1.116	1.062*	1.044*	1.058*	1.215	1.821
5 →	1.122	1.065*	1.048*	1.058*	1.218	1.829
3 →	1.122	1.063*	1.047*	1.059*	1.222	1.855

Layer number from the core bottom	1	2	3	4	5	6	7
9 →	1.083	1.136	1.049*	0.781*	1.127*	1.141*	1.468
7 →	1.036	1.162	1.073*	0.737*	1.008*	1.120*	1.469
5 →	1.038	1.176	1.011*	0.737*	0.995*	1.122*	1.494
3 →	1.039	1.162	1.011*	0.738*	0.992*	1.123*	1.493

Figure 10 The distribution of the relative released power of the fuel elements (with the maximum power density in the absence of clasters) in case of a claster lowered in fuel assembly 4 (asterisks denote the values which are not maximum in the fuel assembly)

Layer number from the core bottom	18	19
9 →	-0.7	-0.6
7 →	-0.7	-1.0
5 →	-0.7	-1.0
3 →	-0.7	-1.0

Layer number from the core bottom	14	15	16	17
9 →	-0.3	-1.7	-1.3	0.0
7 →	1.0	-0.7	-2.5	-1.0
5 →	0.9	-1.5	-2.5	-1.0
3 →	0.7	-0.7	-2.5	-1.0

Layer number from the core bottom	8	9	10	11	12	13
9 →	0.2	-1.8	-7.6	-3.9	-1.1	0.1
7 →	-0.3	-0.6	-7.8	0.8	-1.0	-0.7
5 →	-0.3	-0.7	-7.8	0.8	-1.1	-0.9
3 →	-0.3	-0.7	-8.1	0.9	-1.1	-0.7

Layer number from the core bottom	1	2	3	4	5	6	7
9 →	0.1	-1.1	-4.1	-29.3	-14.1	-0.6	0.3
7 →	0.0	-2.0	-3.2	-31.3	-3.1	-2.1	1.2
5 →	0.1	-2.0	-9.6	-31.4	-4.3	-2.2	1.1
3 →	0.0	-3.7	-9.7	-31.4	-4.4	-2.3	-1.2

Figure 11 The distribution of the relative deformation of the released power in the fuel element with the maximum power density in lowering a claster in fuel assembly 4 (as a percentage of initial power level)

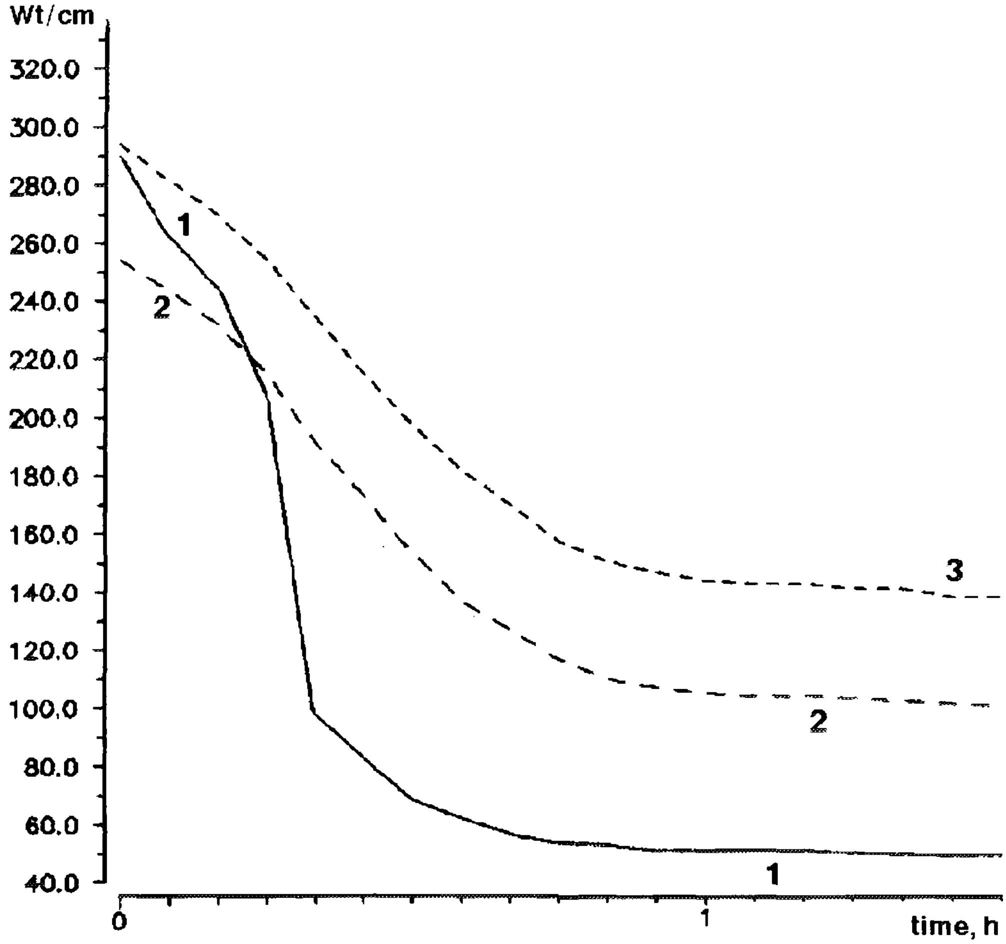

Figure 12 Variation of the linear power rate at the half-height of the core:
1 fuel assembly 4 (here the control group is moving);
2 fuel assembly 5;
3 fuel assembly 21.

Conclusion

The operating experience as well as the calculations and the experimental investigations show that in VVER-1000s under steady-state operating conditions at occasional power cycling it is essentially possible to ensure the control of the power density distribution with a high degree of reliability.

The work in progress now on the modernization of the means and methods of reactor control will allow the most complete realization of the given possibility. The operability of VVER-1000s under the conditions of power cycling is also under study.

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Improvement of Operation Efficiency for VVER-440 and VVER-1000 from TRIGON Fuel Assembly Design Features

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1 Introduction

Implementation of design and irradiation experience acquired from PWR/BWR fuel assemblies utilization has lead to TRIGON 440 and TRIGON 1000 features and will bring to VVER reactors operators the benefits of higher fuel cycle efficiency and confort of operation

Efficiency is measured mainly in number of assemblies to constitute a batch for a given cycle length. It may also represent the capability of higher reactor energy output. Confort of operation may be a translation of trouble free, heavy duty of the fuel during the cycle, in a word: reliability.

This second aspect includes the absorber assemblies as it is a core component important for the reactor safety and operation and as it should not add in anyway concerns to the day to day plant management.

2 Experience from PWR and BWR Assemblies in Reactor Operation

A brief survey of PWR and BWR experience follows.

- *Irradiation experience*
  Fuel assemblies with fuel rods containing pellets enriched up to 4.5% U5 have been irradiated in commercial nuclear power plants to an average burnup of 58000 *MWd/tU* (Figure 1) under normal operation conditions in order to explore the technological limits and evaluate the conservatism of the design. Feed back from on site and hot cell examination has confirmed the design potentiality.
  It is recalled that for example in a standard 900*MWe* plant average burnup of fuel enriched at 3.70% U5 with 1 year cycle and 1/4 core management is of the order of 44000 *MWd/tU* with a flexibility needed by a strategy of stretch out capability at every cycle.
- *Fuel cycle management experience*
  The trend is towards hybrid fuel management or IN-OUT loading strategy. The benefit obtained is twofold: economy and reduced pressure Vessel fluence or plant life extension. In view of this objective there is a limit without use of burnable poison mainly due to thermal hydraulic capability of the fuel assembly and the reactivity management. This limit is overcome by use burnable poison.
- *Use of an integrated burnable absorber*
  The gadolinium oxyde has been chosen mixed with Uranium dioxyde (natural or enriched).
  It is the only technology available in case of the VVER 440.

For the case of VVER 1000 pyrex rods may be used but they represent extra components which disposal is a difficulty, furthermore it take the place of an absorber assembly and reduces the core management flexibility.

- *Design, materials and technology*
  - Use of zircaloy 4 material generalized to fuel rod cladding and all structural parts situated along the active length.
  - Use of solid UO_2 ceramic pellets with controlled porosity and density, remarquable for their as built very low moisture content .
  - Use of bimetallic spacer grids.
  - Dismantable top nozzle.
  - Feasibility of incorporating a debris filter.
- *Moderating ratio*
  Increased moderating ratio towards PWR 17×17 standard has been reached by reducing fuel rod diameter, adding water rods and slight increase in fuel pitch for a better uranium utilization (Figure 2).
- *TRIGON 440 specific features*
  The channel type fuel assembly allows relative flexibility with respect to fuel bundle arrangement and internal pressure drop providing total fuel assembly pressure drop is identical.

3 TRIGON Design Features

TRIGON 440

TRIGON 440 (see Fig. 3) fuel assembly consists of an autonomous bundle comprising a Zircaloy structural skeleton to position the fuel rods. The bundle is contained in an hexagonal channel attached to the bottom nozzle. The removable top nozzle is connected to the upper end of the channel.

TRIGON 1000

TRIGON 1000 (see Fig. 4) original structure is derived from the PWR Cluster Control Concept and facilitates the direct introduction of benefits presently available to PWR operators.

Main fuel assembly characteristics are presented in Figure 5.

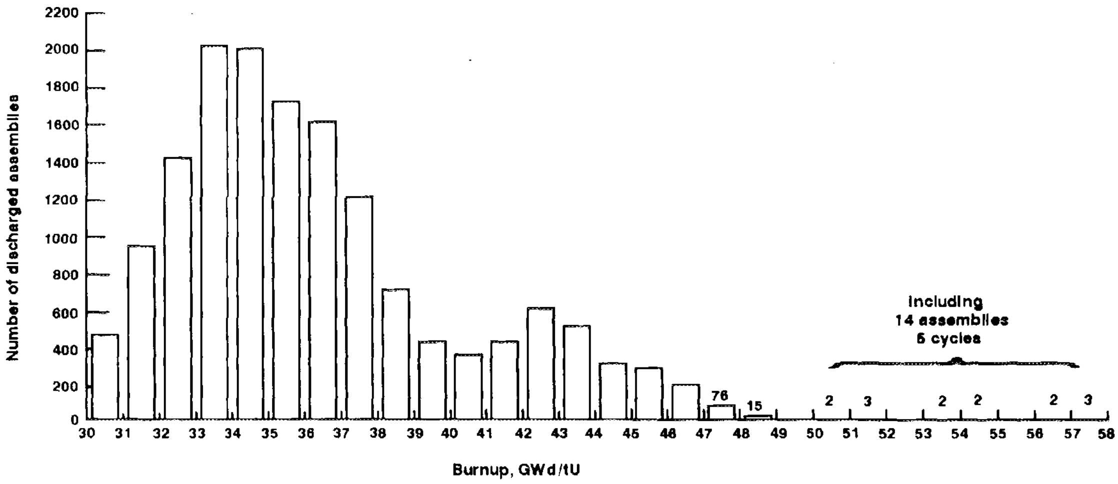

Figure 1 FRAGEMA 17×17 high burn-up experience as of June 30, 1994. Total number of discharged assemblies: 15254, including 3049 assemblies with 4 cycles

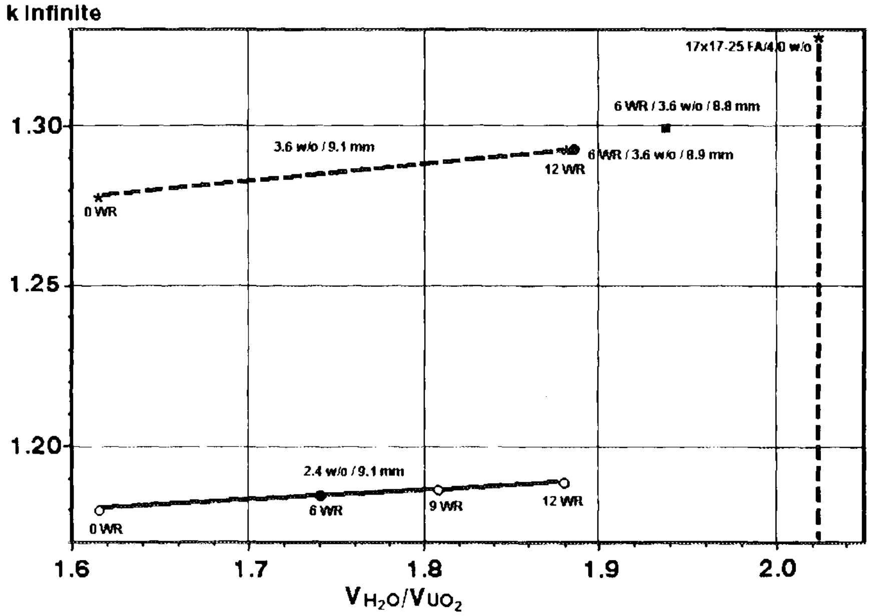

Figure 2 TRIGON 440 $K_{infinite}$ vs. water/fuel ratio for different fuel assembly designs

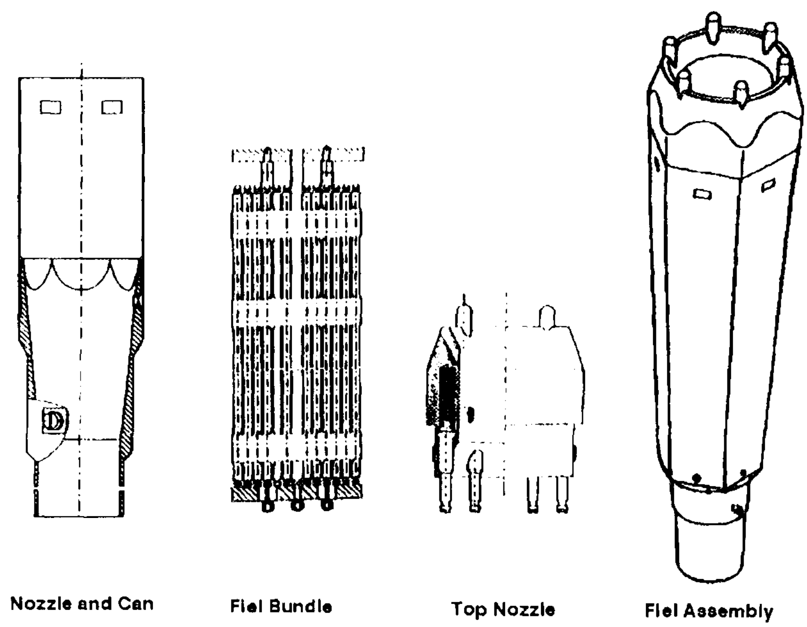

Figure 3 TRIGON 440 fuel product

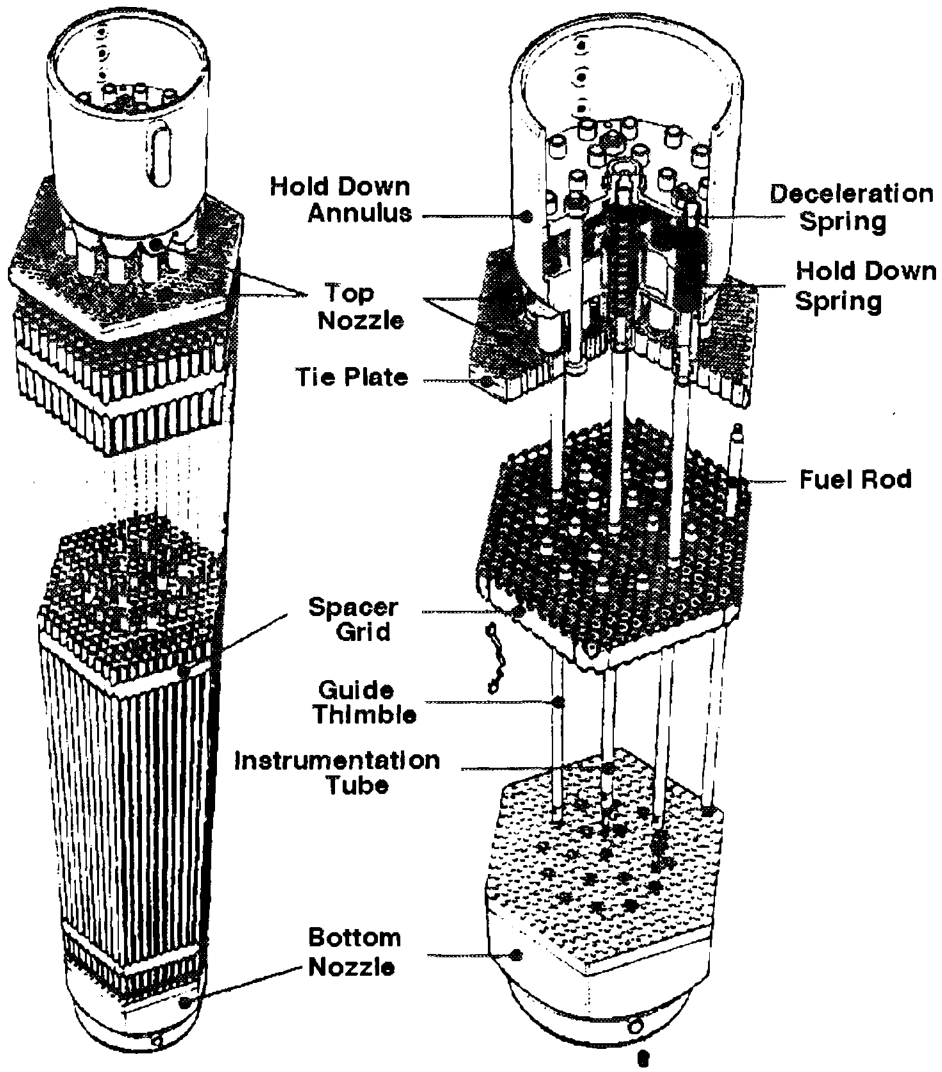

Figure 4 TRIGON 1000 fuel product

Fuel Assembly	TRIGON 440	TRIGON 100
Assembly geometry	Hexagonal	Hexagonal
Number of fuel rods	120	312
Rod pitch (trigonal array), *mm*	12.25	12.75
Thimble tubes	6	18
Instrumentation tubes	1	1
Overall assembly length, *mm*	3217	4570
Overall assembly width, *mm*	144.2	234
Active rod length, *mm*	2420	3780
Rod outside diameter, *mm*	8.8	8.9
Pellet length, *mm*	12.39	12.39
Pellet density, g/cm^3	10.41	10.41
Average linear heat generation rating, *W/cm*	130	176
Peak linear heat generation rating, *W/cm*	325	400
Clad material	Zry-4	Zry-4
Clad thickness, *mm*	0.55	0.6
Grid material (Stap. spring)	Zry-4/inconel	Zry-4/inconel
Average discharge burnup, *MWd/kgU*	>50	>50

Figure 5 TRIGON 440 and TRIGON 1000 characteristics

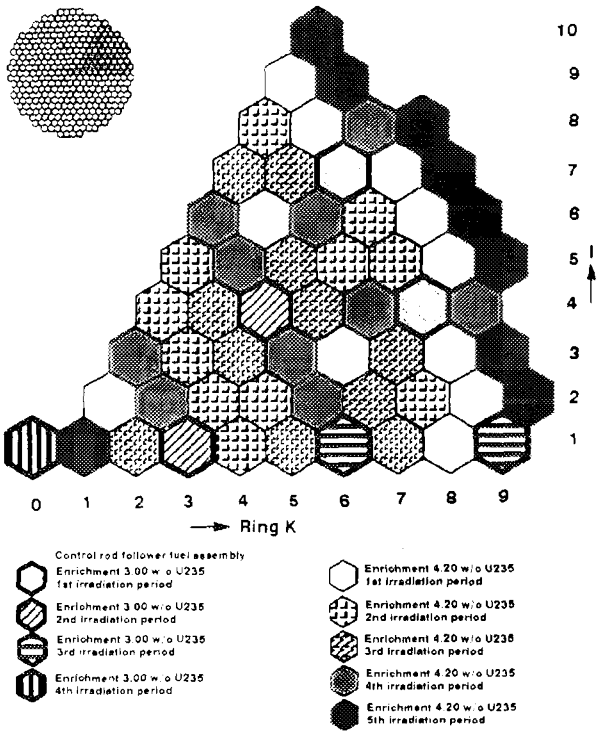

Figure 6 VVER-440 (core thermal power $1375 MW_{th}$) fuel loading pattern. Equilibrium cycle (average enrichment 4.02 w/o ^{235}U). 66 TRIGON reload fuel assemblies (4.02 w/o ^{235}U). 12 TRIGON reload fuel assemblies (3.00 w/o ^{235}U).

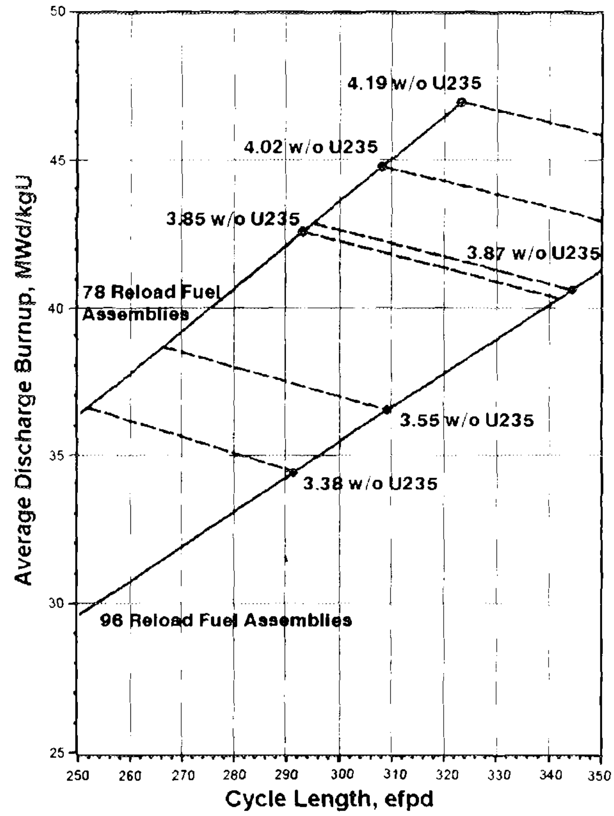

Figure 8 TRIGON 440 (core thermal power $1375 MW_{th}$). Equilibrium cycle. Average discharge burnup versus cycle length for In - Out fuel management. 6 additional waterflooded rods pers fuel assembly - outer diameter of fuel rod cladding 9.1 mm.

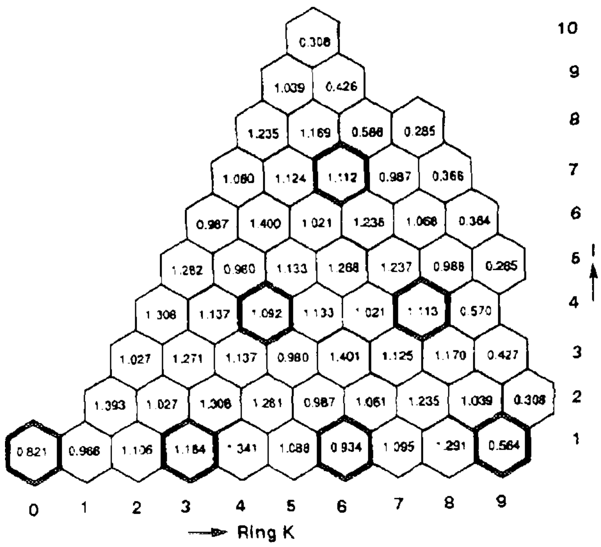

Figure 7 VVER-440 (core thermal power $1375 MW_{th}$), equilibrium cycle (average enrichment 4.02 w/o ^{235}U). Fuel assembly averaged values of the relative radial power density distribution. Beginning of cycle (6 efpd), hot full power, Xe-equilibrium. 66 TRIGON reload fuel assemblies (4.02 w/o ^{235}U). 12 TRIGON reload fuel assemblies (3.00 w/o ^{235}U). Outer diameter of FR cladding: 9.1 mm.

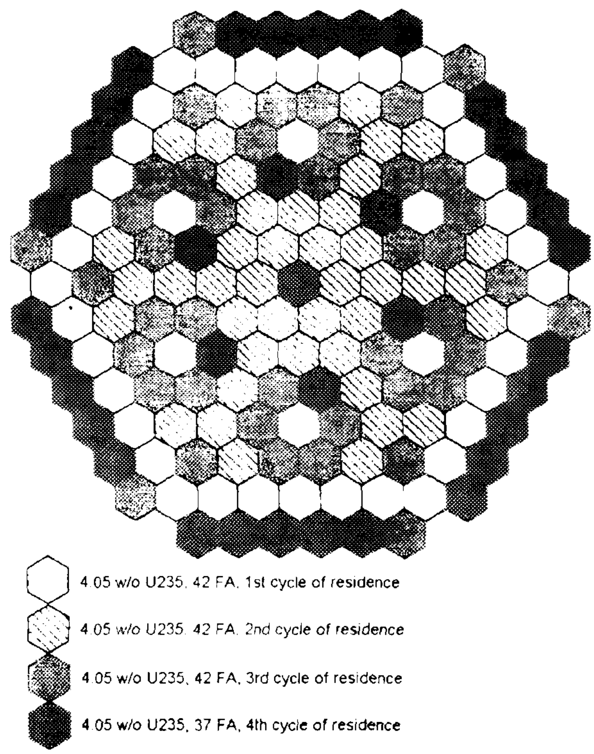

Figure 9 VVER-1000 equilibrium core design for In-Out fuel management: 4-year fuel cycle fuel loading pattern.

4 Example of Fuel Management in VVER-440

The core contains 312 independent fuel assemblies (IFA) and 37 control fuel assemblies (CFA).

The goal is to reach for example a 1/5 core reload strategy for a given cycle length with the appropriate enrichment.

A typical reload batch at equilibrium cycle will consist of:

$\left.\begin{array}{l}\text{66 IFA}\\ \text{12 CFA}\end{array}\right\}$ e.g. a total of 78 fuel assemblies

Figure 6 presents at equilibrium cycle the loading pattern obtained with an IN-OUT strategy without burnable poison.

Figure 7 presents the relative radial power distribution at the beginning of the 5th cycle.

Figure 8 represents the discharge burnup versus the cycle length for IN-OUT fuel management. Two types of specific situations may be commented :

1. From the limit which is related to the spent fuel pit design criterion with respect to critically an average enrichment of 4.02% U5 has been chosen that leads to an average burnup of 44900 *MWd/tU* and a cycle length of 308 EFPD.
2. With a typical 1 year cycle corresponding to 295 EFPD we obtain :
   - for an average reload enrichment of 3.87% U5;
   - an average batch burnup of 43000 *MWd/tU.*

The result shown on the picture indicate that the most depleted originally high enriched assembly will be essentially at the core periphery during their fifth cycle, on the contrary during their first cycle the originally high enriched fuel assemblies will be surrounded by 5th cycle fuel assemblies that will shield the pressure Vessel from high energy radiation.

It must be remarked that control fuel assemblies with a 3.0% U5 will only reside 4 cycles in the core.

One obvious consideration may be made with respect to radial power distribution: 1st cycle high enriched fuel assemblies see the highest radial relative power and 5th cycle high enriched fuel assemblies located at the core periphery see the lowest relative radial power.

Remark: the use of integrated burnable poison will allow to reach an even more IN-OUT pattern and optimize again fissile material utilization: saving on enrichment or increasing cycle length.

5 Example of Fuel Management in VVER-1000

A typical 1/4 core loading pattern strategy with a 1 year cycle length and an enrichment of 4.05 % U5 for the batch in presented in Figure 9.

The loading strategy is IN-OUT without the use of integrated burnable poison.

Comments: It can be seen that originally high enriched fuel assemblies are essentially located at the periphery during their 4th cycle therefore shielding the pressure vessel from reactive fresh 1st cycle fuel assemblies.

Relative power density (Figure 10) map indicates peak power in the fresh 1st cycle assemblies at the beginning of the cycle where they are introduced.

At the end of the same cycle (Figure 11) the peaks level up somewhat but stay high and the batch average burnup is around 49000 *MWd/tU.*

Remark: the use of integrated burnable poison will allow to have a more IN-OUT loading pattern. This will apply to first cycle high enriched fuel assemblies situated in the 6 core's corners in direct view from the pressure vessel and also to part of the second outer row fuel assemblies. The net result independently from pressure fluence reduction will be a lower neutron leakage and an improved fissile material utilization.

6 Increase in Plant Comfort of Operation

Improvement in Control Assembly technology for VVER 440 and 1000 is next described.

6.1 Control Assembly for VVER-440 (Figures 12 and 13)

The control assembly which consists in 2 parts : the absorber and the fuel, presents a potential weak point as its weight is borne by the fuel assembly hanging at the end of the drive shaft connection which supports the sum of the weights (static and dynamic).

The first improvement consists in a self braking absorber assembly which holds directly to the shaft body when in operation due to differential thermal expansion of constitutive materials. The second improvement is relevant to the prevention of the control assembly ejection due to drive mechanism casing failure. The improvement allows the control assembly to drop down by decoupling from the shaft therefore avoiding the consequence of such an accident.

6.2 Control Assembly for VVER-1000 (Figure 14)

Feed back from experience indicates some dual problems between original fuel assembly and the rod cluster control assembly (RCCA) such as free fall troubled by bending and friction in the guide thimbles after as little as 2 irradiation cycles.

First consideration: Fuel assembly mechanical design features

The design of TRIGON 1000 assembly includes total adequacy and mastering of the structure behaviour in areas such as elongation of the assembly, accomodation for fuel rod growth, guide thimble geometry and resistance to hydribing and irradiation effect, dimensionning of the assembly hold down spring pack and dimensioning of the spring designed to absorb the dynamic weight of the RCCA.

Second consideration: Rod cluster control assembly design features

"HARMONI" type RCCA have been adapted to VVER design with 18 absorber rods including such features as:

- Ion nitrided AISI 316L cladding for wear free long time warranted duty;
- Ag Ln Cd designed tip for resistance to long time in core station;
- spring loaded B_4C pellet column for high neutronic reactor scram efficiency with helium atmosphere and reduced internal temperature and benefits for creep, growth, swelling and interaction with the cladding.

7 Conclusion

Reference materials, design features as well as calibrated means of prediction of behaviour under irradiation obtained from light water reactor core operations allow TRIGON 440 and TRIGON 1000 as well as their control assembly matching counterparts to be used to increase efficiency of VVER reactor by improving fuel cycle cost from the reduced need of natural uranium to be enriched to the smaller number of fuel assemblies to fabricate and to be stored or reprocessed. Improved control assemblies bring themselves comfort to the plant operator with instrinsic progress in safety with respect to accidental situation and trouble free behaviour long time utilization in the reactor.

This represents the adaptation from PWR/BWR operation feed back to the design of VVER fuel and core components and it is probably one of the most efficient way to make aware VVER fuel operators that a supplier's competition is an advantageous way to be presented as soon as available: up to date technology, methods, tools, services for better operation and improved safety.

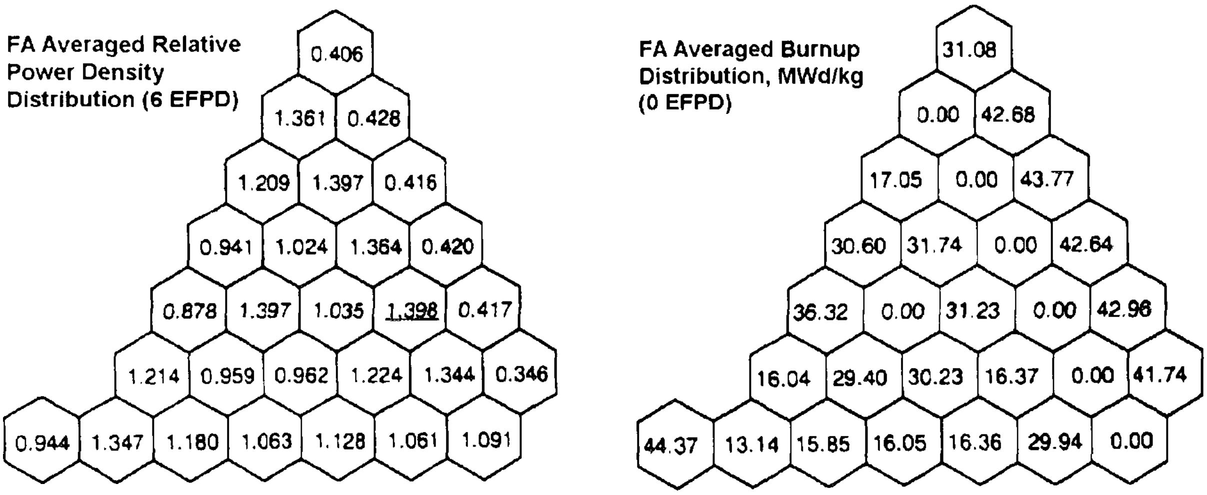

Figure 10 VVER-1000 equilibrium core for In-Out fuel management (4-year fuel cycle): FA averaged power density and burnup distributions at beginning of cycle.

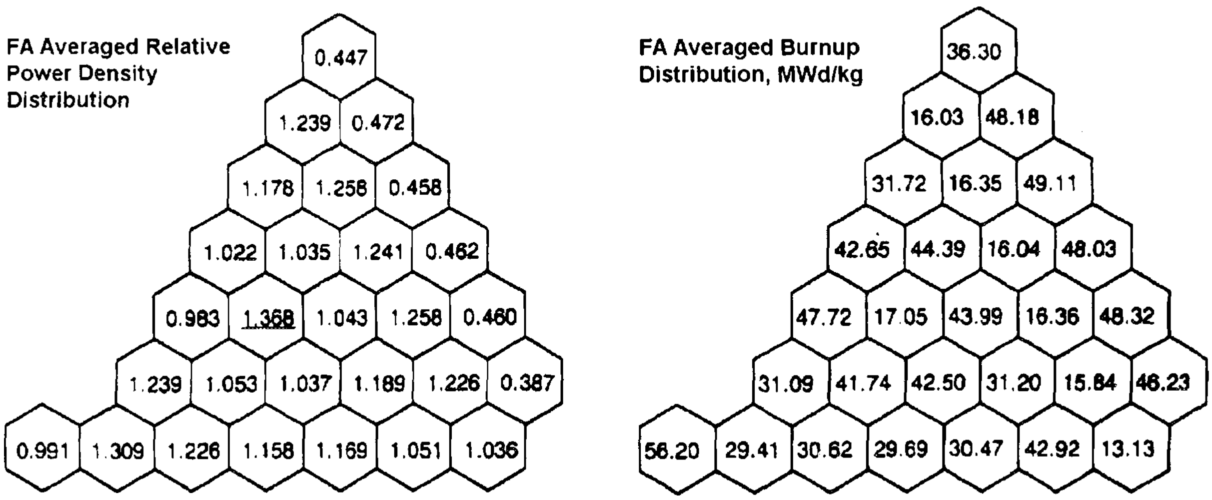

Figure 11 VVER-1000 equilibrium core for In-Out fuel management (4-year fuel cycle): FA averaged power density and burnup distributions at end of cycle (309 efpd).

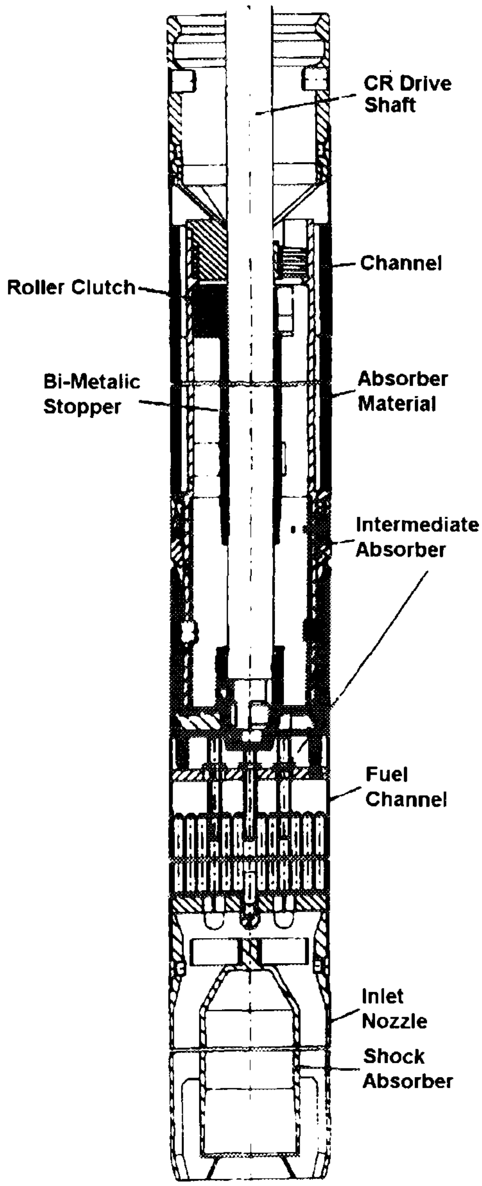

Figure 12 TRIGON 440 control assembly

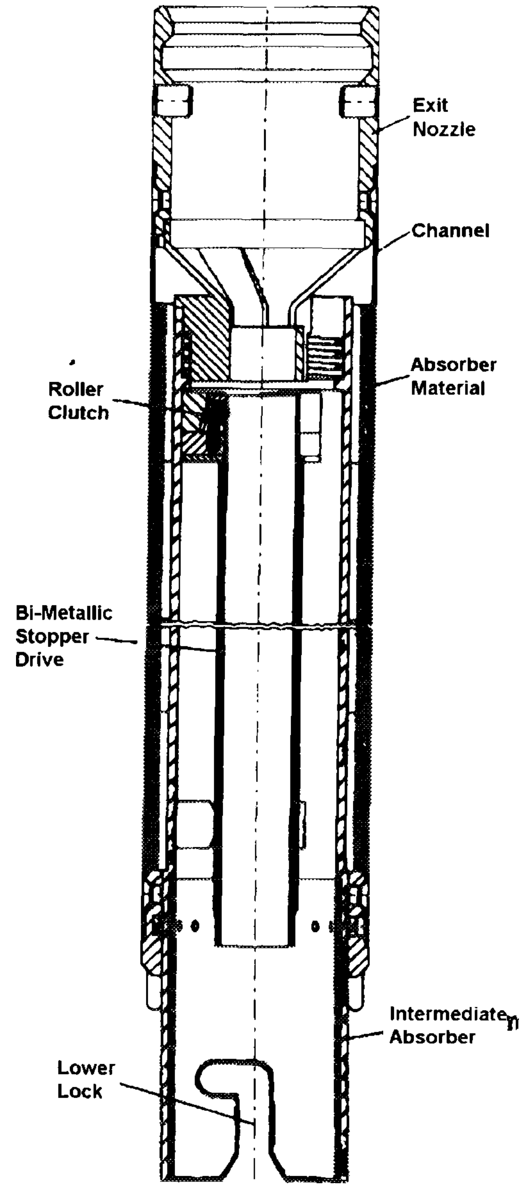

Figure 13 TRIGON 440 absorber assembly

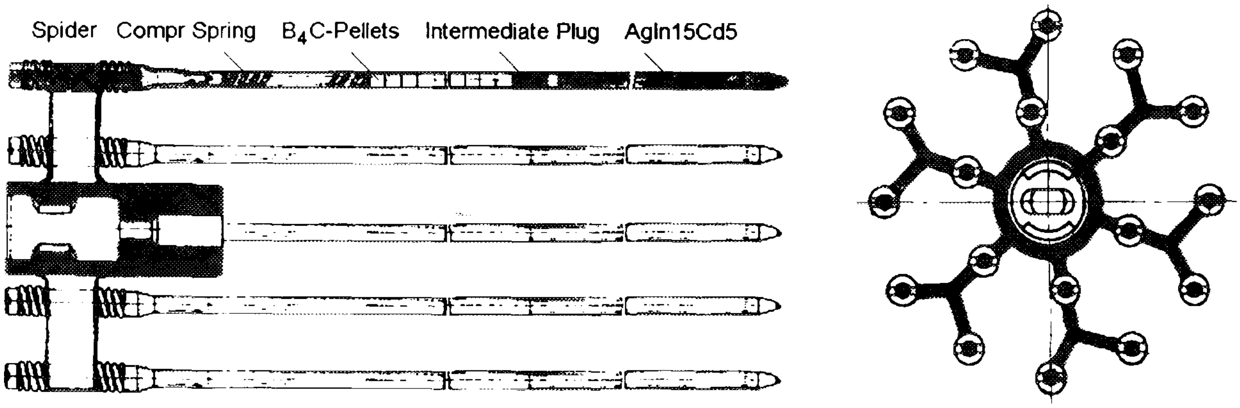

Figure 14 TRIGON 1000 control assembly

The International VVER Fuels Market

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1 The State of the Western Fuel Fabrication Market

Never before in the history of the Western nuclear fuel fabrication industry has there been a period of such intense competition among vendors. And this is occurring in what is truly an international market for nuclear fuel supply.

In the early years of nuclear power, utilities typically purchased fuel from their reactor vendors who were often located in their own country. Today, however, vendors in every nation are actively pursuing (and winning) contracts in other countries and for the reactors of other manufacturers. Even in Japan, which has always relied on domestic fabrication supply, consideration is being given to purchasing fuel from foreign vendors.

Because the international growth of nuclear power has lagged well behind the rates anticipated a decade or so ago, existing fabrication capacity in the West far exceeds the current and anticipated demand. In fact, current demand is not much more than half of the capacity available to supply it.

This has led to the aggressive competition noted earlier, especially in the USA where prices have fallen substantially, even as new and improved fuel designs are being brought to market.

2 The View to the East

Facing a heavily oversupplied market with no prospects for growth in the foreseeable future, and with the dramatic political changes in Eastern Europe during the past several years, Western vendors have turned eastward in an attempt to fill their fabrication plants. And the potential market they have seen there has brought broad smiles to the faces of marketing managers, production supervisors, financial executives and the stockholders of the Western vendors.

Focusing solely on pressurized water reactors, and excluding the plants in the Russian Federation, there is a possible long-term market of up to 44 units in Bulgaria, the Czech Republic, Finland, Hungary, Slovakia and Ukraine. These include:

- 20 VVER-440 reactors in operation, with an additional 4 units under construction
- 12 VVER-1000 units operating and 8 under construction.

As these markets have opened, Western vendors have aggressively pursued them, some independently, others in joint ventures. Even with the prospect that some of the older VVER units may be shut down, and even if some of the units currently under construction are not completed, this is certainly a market which could profitably support multiple suppliers.

3 The Benefits of Competition

Competition in the VVER fuel market promises significant benefits for the operators of these reactors. This will be especially true during the next several years when the marketplace will determine the identity and number of Western vendors that will participate in the market with the current Russian VVER fuel supplier and the market share of each one.

Among the principal benefits of this competition are:

- Lower cost - the most obvious benefit - the current U.S. market is the best example of the result of many vendors vying for a limited amount of business.
- More favorable contract terms and improved vendor cooperation with the customer in such areas as response to problems, sharing of information, etc.
- Accelerated technological development as vendors attempt to make their products more attractive through design improvements which enhance safety, improve fuel performance, expand operational flexibility and reduce fuel cycle costs.

There is also an indirect benefit to the utility's country in that most of the Western vendors have been seeking local subcontractors to provide fuel components and testing and analytical services in association with the supply of VVER fuel. Should any Western vendor obtain a sufficient number of contracts, there is a strong possibility that a VVER fuel fabrication plant might be built somewhere in Eastern Europe.

4 Active and Prospective Participants in the VVER Fuel Market

The current VVER fuel suppliers are TVEL and Westinghouse Electric Corporation (W). Other Western PWR fuel vendors who have bid but not yet secured fuel supply contracts for VVER plants, include Asea Brown Boveri Atom (ABBA), British Nuclear Fuels (BNFL), and European VVER Fuels GmbH (EVF). Their relationship to the international PWR fuels industry is shown on Fig. 1 and discussed below.

TVEL

All of the VVER fuel to date has been supplied by TVEL, an organization within the Russian

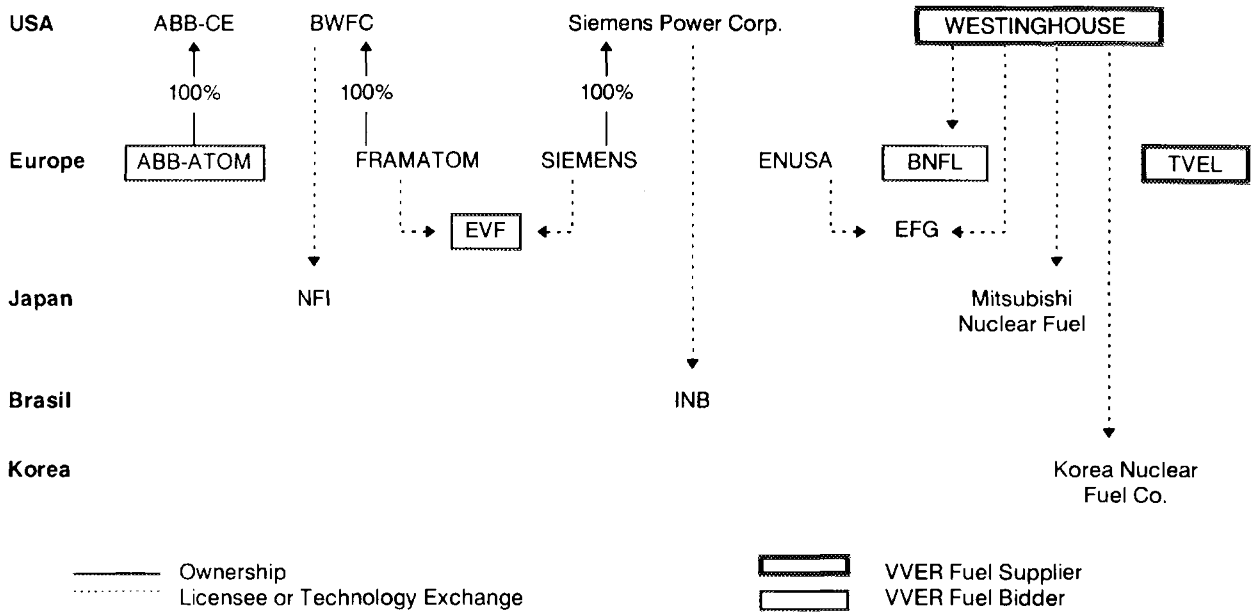


Figure 1 The PWR fuel fabricators

Ministry of Atomic Energy (Minatom), through its marketing and sales organization TENEX. TVEL coordinates the laboratories, design organizations, and fabrication plants to produce fuel for its customers. Within Russia, its customers are also a part of Minatom, and in other countries, the customers are independent utilities resembling Western utilities. The main contributors to the Russian fuels industry are the extensive facilities of the All-Russian Scientific and Research Institute of Inorganic Materials, Moscow; the Kurchatov Institute, Moscow; and Gydropress in Podolsk for design and development work. The fabrication plants and their products are:

- ⇨ Machinostroitelniy Zavod, Elektrostal
  - UO_2 pellets from UF_6
  - VVER 440 fuel assemblies
- ⇨ Chemical Concentrates Plant, Novosibirsk
  - VVER 1000 fuel assemblies
- ⇨ Ulba Metallurgical Plant, Ust-Kamenogorsk (Kazakhstan)
  - UO_2 pellets from UF_6
- ⇨ Chepetsky Mechanical Plant, Glazov
  - Zirconium alloy tubing, sheet, and rod from ore

The size and the capacity of these facilities exceed those of any Western vendor. All of these organizations are currently part of Minatom, but consideration is being given to restructuring this organization. TVEL and its facilities are expected to continue to be the dominant VVER fuel supplier for at least the near future.

ABB Atom

ABBA, a privately owned stock company, was primarily a supplier of BWR fuels until it purchased Combustion Engineering (CE), a U.S. firm that designs and builds PWRs and supplies PWR fuel. CE's fuel market is principally for reloads to its own NSSS in the U.S. With the help of CE's expertise in PWRs, ABBA established a PWR reload business in Europe and successfully sold fuel for W reactors in Belgium, Sweden and Switzerland, Framatome reactors in France and Siemens reactors in Germany.

The ABBA fuel design, development and production facilities are in Våsterås, Sweden. The CE design and metal component fabrication facilities are in Windsor, Connecticut and fabrication facilities in Hematite, Missouri - both in the USA.

BNFL

BNFL is a government owned corporation that provides all of the fuel to the UK's gas cooled Magnox and AGCR reactors. They have extensive experience in the production of these fuel assemblies as well as the production of UO_2 pellets from UF_6 for LWRs. Their entry into the PWR fuel fabrication field is relatively recent. Earlier this year, under a W license, they fabricated the first core of fuel for Sizewell B, the first PWR in the UK. They are a member of the European Fuels Group (EFG) which is a consortium of W and two of its licensees, ENUSA in Spain and BNFL, to supply PWR fuel to the West European market.

BNFL's fuel fabrication plant has the capability of providing complete assemblies with metal components provided by W. The plant is located in Springfields, England.

EVF

EVF GmbH, a German corporation located in Offenbach, was created in 1993 specifically for the supply of VVER fuel and is a partnership of 50% Siemens and 50% FRAMATOME and COGEMA of France. EVF represents the two largest PWR fuel vendors in Western Europe. Their facilities are extensive as listed in Table 1 and include

Table 1 Uranium fuel fabrication facilities - PWR/BWR fuel

Region	Owner/Operator	Plant Name&Location	Capacity, *t U / Year*
United States	B&W Fuel Co.	Lynchburg, VA	400
	ABB-CE[1]	Hematite, MO Windsor, CT	300
	Siemens Power Corp.[1]	Richland, WA	700 (to 1,200)
	General Electric[2]	Wilmington, NC	1,000
	Westinghouse	Columbia, SC	1,700
			4,100 (4,600)
Europe	ABB Atom[1]	Vasteras, Sweden	400
	ANF (Siemens)[1]	Lingen, Germany	400
	ENUSA[1]	Juzbado, Spain	200
	FBFC (Framatome)	Dessel, Belgium Romans, France Pierrelatte, France	400 800 250
	Siemens[1]	Hanau, Germany Karlstein, Germany	800 400
			3,250
	VVER Fuel: TVEL	Elektrostal	700
		Novosibirsk (assemblies only)	1,000
		Ust-Kamenogorsk (pellets only)	(2,650)
			1,700
Far East	Japan Nuclear Fuels[2]	Yokasuka City, Japan	750
	Korea Nuclear Fuel Co.	Taejeon, S.Korea	200
	Mitsubishi Nucl. Fuel	Tokai Mura, Japan	440
	Nuclear Fuels Ind.[1]	Kumatori, Japan Tokai Mura, Japan	265 200
			1,855

[1] BWR and PWR Fuel
[2] BWR Fuel Only

zirconium semi-finished product fabrication in France at CEZUS, zirconium alloy tube production in France at Zircotube and in Germany at Nuklear-rohr. FRAMATOME fuel design, development and fabrication facilities are distributed throughout France; and Siemens facilities are located in both Germany and the USA.

In the non-VVER fuel market FRAMATOME and Siemens still compete with each other.

WESTINGHOUSE

W is the largest PWR fuel supplier in the USA and, in addition, has agreements in Europe with EFG, and in Japan and Korea as shown in Fig. 1. The design and development facilities are located in the Pittsburgh, Pennsylvania area and its fuel plant is in Columbia, South Carolina. W contracts include reloads for CE and Babcock and Wilcox NSSS.

In addition, W owns Western Zirconium a producer of semi-finished zirconium alloy products in Ogden, Utah and a zirconium alloy tubing plant in Blairsville, Pennsylvania. In addition to its customers in the U.S., W has supplied fuel assemblies to reactors in Europe and Taiwan. Extensive component supply flows from the USA to its licensees.

Status of New Suppliers in the VVER Market

VVER-1000

The first request for competitive VVER fuel supply bids was issued in 1991 by Ceské Energetické Závody (CEZ) for the first cores and reload fuel for Temelín 1 and 2, a twin unit of 1000*MWe* VVERs under construction in the Czech Republic. W won the contract in 1993 in a competition that included ABBA, FRAMATOME, Siemens and TENEX. The independent technical evaluation of the bids was performed by the Stoller Corporation.

An extensive design and development program is currently in progress to complete and qualify the final design in 1995. The major objectives of the program are to produce a design that meets or exceeds initial performance goals, to assure that the fuel is compatible with the Russian design plant, and to license the core to Czech and U.S. regulatory standards. Some of the advanced features are described in the next section.

The program is carried out in W's facilities in the USA with the exception of cooperative work with Skoda, primarily in the areas of hydraulic testing and plant compatibility. Production of the fuel will be in the USA and the completed assemblies will be shipped directly from Columbia, to Temelín.

VVER-440

The first invitation by VVER 440 owners for competitive fuel supply bids was issued in 1992 by CEZ and Slovenský Energetický Podnik (SEP). It requested reload fuel for CEZ's four Dukovany Units and two of SEP's Bohunice Units, as well as first cores and reloads for SEP's four Mohovce Units. The bidders in this case included ABBA, BNFL, EVF, W and TVEL. The bid evaluation process narrowed the list to EVF, W and TVEL and the current negotiations are expected to identify the winner(s) in 1995. The initial bid evaluation was a joint CEZ-SEP effort, but each utility is negotiating its final contract individually. The Stoller Corporation assisted the utilities in the preparation of the bid specification, the technical, economic and commercial evaluation of the bids and is currently assisting in the contract negotiations.

5 Principal Differences Between Western and VVER Fuels

Physical Features and Related Design Considerations

The most obvious physical difference between Western and VVER Fuels is the square vs. hexagonal lattice geometry. To appropriately evaluate the nuclear behavior of hexagonal lattices, Western vendors have had to modify their existing nuclear codes, and subsequently qualify the calculational capability of the modifications. Existing thermal-hydraulic (e.g., subchannel analyses), mechanical design (e.g., fuel performance, seismic analysis), and safety/LOCA analyses methods are generally directly applicable to hexagonal assemblies with little or no modification. Perhaps the one unique modeling development required by Western vendors to appropriately analyze the spectrum of Condition III and IV events is related to the horizontal steam generator in VVER plants vs. the vertical generators in Western PWRs.

Regarding assembly mechanical design, the design concept of the VVER spacer grids is different from their Western counterparts. The VVER grids act primarily as springs without tight restraint of the fuel rods; the grids are closely spaced to control vibration. The Western grids have a more rigid structure and rod restraint with springs and hard stops and, as a result, fewer grids are needed. If the Western grid concept is to be introduced into VVER fuel, detailed testing and modeling are required to assure the adequate hydraulic, thermal, and mechanical performance of the assembly. There are also significant differences in the designs of the top and bottom nozzles.

Of necessity, all Western fuel assembly mechanical designs must be compatible with the core internals, fuel handling equipment, spent fuel storage, and shipping containers. If reload fuel is being supplied, it must also be compatible with the residual Russian fuel.

Considering thermal and hydraulic parameters, VVER-1000 cores are quite similar to those of Western PWRs, falling somewhere in between Western 3 and 4 loop plants. The VVER-1000 assembly has a somewhat lower water-to-fuel ratio then its Western counterparts which has important nuclear related implications. In neither the VVER-1000 or Western PWRs are assembly shrouds employed; additionally, control rod geometric designs are similar.

On the other hand, the VVER-440s have fuel assembly and core features quite different from either the VVER-1000 or Western PWRs. These include, but are not limited to, shrouded assemblies (which eliminates cross-flow modeling in thermal-hydraulic analyses), and control assemblies which have fuel assembly followers. These raise a potential pellet-clad interaction issue associated with the movement of the fueled portion of the control assembly into the core.

There are also basic material differences between VVER and Western PWRs. All the VVER reactors operate with Zr-1% Nb cladding in a KOH and NH_3 water chemistry, while all Western reactors operate with Zircaloy 4 cladding in a LiOH water chemistry. Both, of course, use boric acid for reactivity control. Since Zircaloy 4 can operate satisfactorily in the more corrosive LiOH atmosphere, its introduction into VVERs should pose no problem, but confirmation is desirable.

The greater hydrogen pickup potential of Zircaloy 4 compared to Zr-1%Nb also needs to be monitored, as would be the greater oxygen sensitivity of Zr-1%Nb if it were to be introduced in Western plants.

The LOCA performance of Zircaloy 4 in VVERs should be satisfactory, based on Western safety testing.

Previously, we alluded to the necessity for mechanical compatibility. For transitional cores (mixed cores containing VVER and Western fuel) there are also nuclear and thermal-hydraulic compatibility issues. The most important of these is pressure drop compatibility to prevent flow starvation of any assemblies.

Operational Considerations

VVERs are currently operated on annual fuel cycles with discharge burnups generally in the low 30*GWd/tU* range. In contrast, current Western PWRs operate annual, eighteen month and two-year fuel cycles with significantly higher discharge burnup levels (mid-40 to low 50*GWd/tU*) leading to longer in-core residence time for fuel assemblies. Annual cycles are prevalent in Western European PWRs, and eighteen month cycles dominate in U.S. PWRs. In the West, the selections of cycle length

Table 2 **Principal advanced VVER fuel features offered or under development**

FEATURE	MAJOR ADVANTAGES					UNDER DEVELOPMENT BY	
	Reduced Fuel Cycle Costs	Reduced O&M Costs	Pressure Vessel Protection	Improved Fuel Reliability	Improved Operatiing Flexibility	W for Temelin	TVEL
Zirconium Alloy Grids	o	o				✓	✓
Zirconium Alloy Guide Tubes	o	o					✓
Advanced Zirconium Alloys	o	o					
Burnable Absorbers:							
IFBA	o					✓	
Gd_2O_3	o						✓
Axial Blankets	o					✓	
Optimized Fuel Rod Diameter	o					✓	
Water Rods	o						
Extended Discharge Burnups	o					✓	✓
Advanced In-Core Fuel Management	o		o			✓	✓
Debris Resistance: Filter, Long End Plug				o		✓	
Removable Top Nozzle	o			o		✓	✓
High Performance Spacer Grids	o				o	✓	
Longer Lived Control Rods (VVER-1000)		o				✓	

and discharge burnup are based upon cost/benefit analyses (including fuel costs, O&M costs), operating and system factors (especially cycle length influence on plant availability), and regulatory concerns (including number of interactions with regulators and need for a mid-cycle maintenance outage). In the West, a modest experience base exists for fuel assemblies irradiated in the 45 - 55*GWd/tU* range.

6 Advanced Features Offered by Vendors

The recent competition in the VVER fuels market has been the incentive for the development of advanced design features for the improved reliability, economy and operational flexibility of the fuel. Major advanced design features and their vendors are summarized in Table 2.

The two currently active suppliers, TVEL and W, are developing advanced features that will be, or can be, implemented in the near future. Those under development by W for TemelHn are indicated by a single asterisk on Table 2, and those under development by TVEL by a double asterisk. Given sufficient time and development funds, it is Stoller's belief that both TVEL and all Western vendors can provide the entire range of features shown on Table 2. Comments on Table 2 follow.

Zircaloy 4 grids and guide tubes are part of the W Temelin fuel design and are also offered by other Western vendors. TVEL has Zr-1%Nb grids under irradiation in LTAs and offers these as an option.

All of the vendors offer designs that have the nuclear and mechanical capability to achieve higher burnups. Advanced fuel management cores, low leakage fuel management for example, are also offered. Low leakage fuel management and perhaps longer and higher burnup fuel cycles increase the demand for burnable absorbers. For such applications, fuel suppliers are moving away from so-called discrete burnable absorbers which occupy guide thimbles, to integral burnable absorbers which are mixed with or are on the surface of the UO_2 fuel. Typical among these are gadolinia mixed with the UO_2, and ZrB_2 deposited upon the cylindrical fuel pellet surfaces. The former is used by all vendors except W, the latter is being provided by W to Temelín, and is in commercial use in Western fuel. Advantages of the integral burnable absorbers include better power distribution and moderator temperature coefficient control, lower end-of-cycle residual reactivity penalties and elimination of separate handling and disposal.

Although debris has not been a problem in VVERs, debris resistant features are available. For example, long lower end plugs in fuel rods will be delivered to Temelín by W.

Removable top nozzles (RTN) to permit replacement of failed fuel rods, rather than replace the entire assembly, will be part of the design delivered by W to Temelín. TVEL has designed and built several LTAs with this feature and intends to irradiate them. Versions of RTNs for VVER fuel have also been designed by the other vendors. The application of RTNs necessitates repair stands for the spent fuel pool which are offered by the vendors as well. W is in the process of designing the first one for Temelín.

The current VVER 1000 control rods are life limited by B_4C swelling, irradiation assisted stress corrosion cracking (IASCC) of the steel clad, and B_4C washout. The W Temelín design, and those offered by other vendors, will use AgInCd alloy in their tips, the point of highest exposure, to extend the control rod life.

7 Entering the Competitive Marketplace

Most VVER reactor operators have not yet been involved in a competitive fuel procurement. For those contemplating such a procurement, we offer a few suggestions.

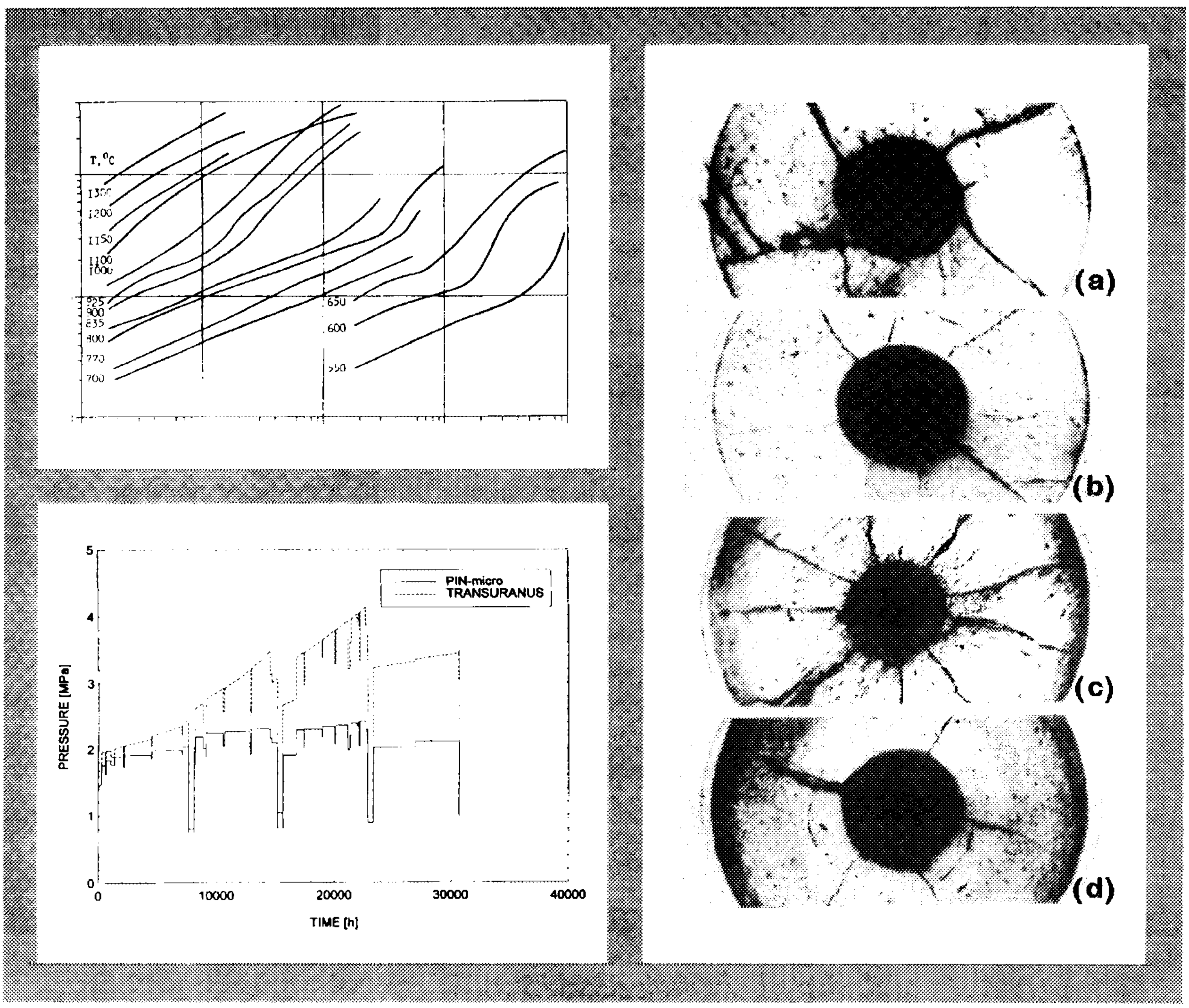
- ⇨ Become familiar with the various vendors and their products and services. Invite the vendors to visit your facilities to make presentations and hold discussions on how they can satisfy your requirements.
- ⇨ Involve all areas of your company in the process, including engineering, operations, purchasing, economics and finance.
- ⇨ Prepare well thought out, detailed bid specifications defining your technical, economic and commercial requirements.
- ⇨ Perform comprehensive evaluations of the technical, economic and commercial portions of the proposals with emphasis on development of VVER fuel designs and analytical methods and licensing the fuel and methods in your country.
- ⇨ At the conclusion of the evaluation, and to ensure that competition is maintained, develop a short list of no more than two or three vendors and negotiate the final contracts with each before making the award.

The political and market changes of the past several years have opened a new era in fuel procurement for VVER reactor operators. It is an era filled with challenges for the utilities and there will undoubtedly be some bumps in the road.

We are convinced, however, that the rewards of participating in the new VVER fuel market will far outweight the effort involved.

Part 2

VVER Fuel Behaviour Modelling and Experimental Support



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APPLIED RESEARCH IN NUCLEAR ENGINEERING

REACTOR PHYSICS

The Reactor Physics Div. of INRNE has compiled an International Data and Computer Code Library with about 85 software products for VVER core analysis:

- calculation of cross sections and resonance integrals;
- spectrum calculations, generation of group constants libraries;
- burnup and fuel reloading calculations;
- point kinetics, spatial kinetics and related calculations of neutronic and thermal dynamic characteristics, modelling of time-dependent accidents;
- evaluated VVER operational data files (preparation, processing and management);

The INRNE staff have developed 3-D neutron diffusion codes for steady-state calculations and core simulation of VVER-440 reactors in Kozloduy and have generated the few-group diffusion parameters and boundary conditions, which after verification in comparison with experimental data have been applied for operational in-core fuel management calculations in Kozloduy NPP.

INRNE have the biggest group of specialists in applied reactor physics in Bulgaria with the total experience of about 120 man-years, performing the operational support of Kozloduy NPP since its start and maintaining intensive international collaboration.

REACTOR SYSTEM THERMAL HYDRAULICS AND FUEL PERFORMANCE ANALYSIS

The activities of INRNE staff in this field are centered on development of models and computer code assessment and application for VVER reactor thermal hydraulics and fuel behaviour analysis under steady-state and transient conditions, including coupled core thermal hydraulics and fuel rod thermal mechanics analysis. The software tools include well-known computer codes RELAP, COBRA, SSYST, PIN and others, which have been verified for VVER calculations.

The activities for operational support of Kozloduy NPP include core and fuel safety margins determination for specific Kozloduy refuellings - such analyses are being regularly performed in INRNE, especially since 1989.

A total of appr. 100 man-years of experience have been gained in INRNE in the fields of reactor thermal hydraulics and fuel thermal mechanics in both normal-operation and accident analyses.

SAFETY AND RELIABILITY ASSESSMENT

INRNE pioneered the use of Probabilistic Safety Analysis methods in Bulgaria, including use of PSA techniques for structural system reliability analysis and for optimisation of NPP system design and maintenance.

Risk assessments are being performed for consequences of NPP accidents as well as for research reactor and radwaste storage facilities.

A total of approximately 50 man-years of experience have been gained in INRNE which is directly connected with different levels of probabilistic safety assessment and system analysis projects.

VVER Fuel Behaviour Modelling and Experimental Support

Panel Discussion Report

Panel Chairmen: Yu. Bibilashvili [1]***, K. Lassmann*** [2]

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1 Fuel Behaviour Models and Codes: General

Reliable prediction of fuel behaviour is a basic requirement in order to perform safety based calculations, for design purposes and fuel performance predictions. Therefore, computer codes are needed.

The ultimate goal of modelling is a description of fuel behaviour in both normal and abnormal conditions. From this knowledge, operating rules can be derived to prevent fuel failures (the clad is the first safety barrier) and the release of fission products to the environment, also, in extreme cases, to prevent escalation of fuel and core damage and the consequential hazard.

The safety limits on fuel mainly involve the control of a number of key phenomena many of which are interrelated:

- **Fuel temperature** to avoid reaching high temperatures in overpower and coolant faults to prevent fission products release, melting and fuel disruption.
- **Fission gas release** from the fuel pellets to prevent excessive release resulting in build-up of rod internal pressure, resulting in turn in clad lift-off, ultimately leading to failure and escape of radiologically hazardous species to the reactor coolant.
- **Clad temperature** to avoid dry-out with the possibility of degrading clad mechanical properties and exacerbating corrosion and hydrogen pick-up.
- **Pellet-Clad Interaction (PCI)** to avoid fuel failures during power ramp (only applied in few countries).

In addition, it is fundamental to the future of nuclear power that reactors can be run economically to compete with other forms of power generation. As a consequence, the development of the understanding of fuel performance and the embodiment of that knowledge in codes allow for more realistic predictions of performance. This in turn leads to a reduction in operating margins, thus improving operational economics.

Beside the development of codes for fuel operated in steady state, transient and accident conditions, it was found that it is important and necessary to develop codes and models describing the behaviour of the spent fuel while in storage, including VVER fuel after discharge, covering the temperature range encountered (< 350°C). Data are particularly missing for temperatures ~200°C. The effect of water chemistry and storage in air or in a non oxidizing atmosphere should be further studied and modelled.

2 Computer Codes Available for VVER Fuel Modelling

TRANSURANUS: This code has been developed by CEC and verified against a large fuel data base. Although not verified against VVER data, some interest in its use was shown by several Eastern Countries.

PIN-Micro: This code is based on GAPCON-Thermal, and several versions have been developed in Eastern Countries. This code has been verified against a limited set of data in some countries and against a more extensive data base in some others. The code is not fully proved, however, post-test results on FUMEX carried out by Bulgaria showed a great improvement of results.

START-3: Developed in Russia, this code has been validated extensively and has been used very successfully after feedback modifications from the FUMEX programme. Examples of verification were presented on MR experiments, FUMEX and NPP fuel. The verification work should be shown in more detail to reach a clearer idea of the code's capability.

ENIGMA: Used in Finland, it has been verified and qualified against a large fuel data base, VVER fuel experiments (SOFIT) and PIE results of VVER 440 fuel.

FRAPCON: Is used in Turkey on VVER fuel. Although well handled this code is not benchmarked against VVER data. Some of the temperatures obtained must be considered as very conservative since no pellet relocation model was used.

RAPTA: (LOCA, RIA, Severe accidents) has been developed in Russia. The experimental basis, although interesting and promising, needs further clarification.

3 Verification of Computer Codes

The Russian codes PIN-micro, START-3, PIN-mod1, RAPTA, Russian MATPRO - are physically based and verified on MR, MIR and other experimental results simulating LOCA and severe accidents. From the Russian side it was stated that Russia has already given a full set of fuel high temperature properties to European colleagues and that Russia intends to publish externally more experimental data for code verification. The Russian MATPRO could be given on an exchange basis.

Concerning the high burnup, Russia will perform re-irradiation of refabricated and full scale fuel rods from power reactors.

IAEA is working on establishing a data base: from Halden, Canada, Finland, Riso experiments. The database will contain only checked data acquired by well instrumented experiments. Some relevant Russian experimental data will be included as well.

It is recommended that the following aspects should be dealt with in future meetings:

- Description of concepts and models included in the codes.
- Report on verification of these codes with emphasis on temperatures at high burnups and fission gas release.

4 Computer Codes: Further Development

It was generally agreed during the meeting that:

1. Models and codes should be physically based and well verified against well defined experiments addressing main "physical effects". Theoretical efforts are needed to investigate the fuel behaviour at high burnup (rim effect, stress, thermal conductivity, etc.).
2. There are generally two kinds of codes available: those developed by fuel vendors and oriented to a specific fuel; and codes based on "physical models". It was agreed that "mechanistic codes" were preferred to address the large range of fuels and situations encountered.

5 Computer Codes: Licensing

It was agreed that modelers must be involved in the development of criteria and safety regulations.

Codes should be licensed. Nowadays this licensing procedure is the result of an agreement between safety authorities and utilities and/or code developers.

It was suggested that the Agency should establish in 1995 a set of basic criteria for licensing of both fuel and codes. It was suggested that it could be done within the framework of the safety convention.

6 New Experiments

In this meeting, interesting new results of measurements of fission gas release at higher burnup on VVER 440 and VVER-1000 were presented. The threshold in burnup for increased fission gas release is situated around 45 *MWd/kg*. Logically, the release is faster for VVER 1000 (warmer fuel). More results are needed to accurately define this increased rate and predict fuel behaviour.

Experiments for local high burnup effects: PCMI, rim effect, fuel thermal conductivity degradation should be further addressed. Halden experiments on VVER fuel will fulfil these needs for fuel dimensional changes (densification, swelling) and verification of FGR at low linear rates. It was recommended that Halden experiments are carried out up to high burnup. Halden experiments do not replace experiments on refabricated high burnup fuel rods.

Temperature increase at high burnup is a question of concern for all types of light water reactor fuel, it can increase fission gas release dramatically. The increase of the fuel temperature at high burnup has been observed on an irradiation in the MR reactor; however this phenomenon related to the RIM effect and the thermalconductivity degradation cannot be fully studied in MTR reactors (different neutron spectrum). To get a better knowledge and possibly find countermeasures it is recommended that high burnup irradiations in MTR reactors and more detailed examinations of VVER fuel at high burnup are carried out. Acquisition of such data is of major importance for normal operation and safety analysis.

Up to now water side corrosion is not a problem in VVER reactors.

Main Concepts and Objectives of Fuel Performance Modelling and Code Development

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1 Introduction

Since the commissioning of the first commercial nuclear power station at Shippingport, 35 years of development have been spent improving the reliability, operating flexibility, safety and economics of fuel elements. Reliable performance requires that the fuel behaves in accordance with stringent nuclear, thermo-hydraulic and mechanical design specifications. The very low current failure rates of approximately 0 to 0.005% prove that, from fabrication to discharge of fuel elements, the complex problems including the material behaviour during irradiation are well understood.

Besides reliability strong economical arguments justify further research and development: the annual fabrication costs of fuel assemblies for all reactors world-wide represent approximately 10 - 20 billion dollars per year.

The present status of Light Water Reactor (LWR) fuel element performance was presented at an international topical meeting ealier this year [1], a summary is given in [2]. Key issues are advanced zirconium alloys together with improved fuel element designs with the goal to further improve load following and to enhance the burnup. In addition, many specific questions such as the behaviour of MOX fuel, Gadolinium fuel, or the managing of failed fuel rods were discussed.

The detailed understanding of complex fuel rod behaviour could only be achieved by large irradiation programs such as the Halden, Studsvik, Riso and many other projects and by detailed modelling. In most cases only macroscopic quantities (such as temperature, fission gas release etc.) are measured and these are the result of the interplay between many complex micro/macro processes in the fuel (diffusion, cracking, bubble formation etc.). Therefore only theoretical models which take into account these complex processes can be used to adequately treat the tremendous amount of experience and experimental evidence available. We cannot review or even mention the many publications on fuel rod modelling and must refer to an international conference organized by IAEA [3] which gives the present status.

Theoretical fuel rod modelling has made a major contribution to the successful evolution of fuel rod design. Whereas up to the late seventies reactor operation was supported only by rather simple calculations, the situation has changed drastically. Safety requirements, considerations towards minimising the nuclear waste and economics require detailed analysis of fuel performance under normal, off-normal and accident conditions. In fact, there is a clear trend in increasing the number and size of fuel rod modelling groups in research, by manufacturers and licensing authorities. Thus it seems to be worthwhile to review the basic concepts of fuel rod modelling and discuss their limitations.

2 What Has to Be Modelled?

The macroscopic behaviour of a fuel rod is to a large extent determined by microscopic processes. Theoretical models should therefore include the relevant local phenomena but should of course also be able to predict macroscopic consequences such as radial and axial deformations of the cladding, cladding corrosion, cladding failure, pressure build-up by fission gas release etc. Figs.1 and 2 give two typical examples for macroscopic results taken from References [4] and [5]. The large scatter is due to different loadings and designs. It is expected from modelling that these differences are explained.

The following three figures show examples of fuel microstructures imaged by Transmission Electron Microscopy for LWR fuel rods which have been subjected to different irradiation conditions, and demonstrate the dependence of the macroscopic fuel behaviour on the local UO_2 microstructure.

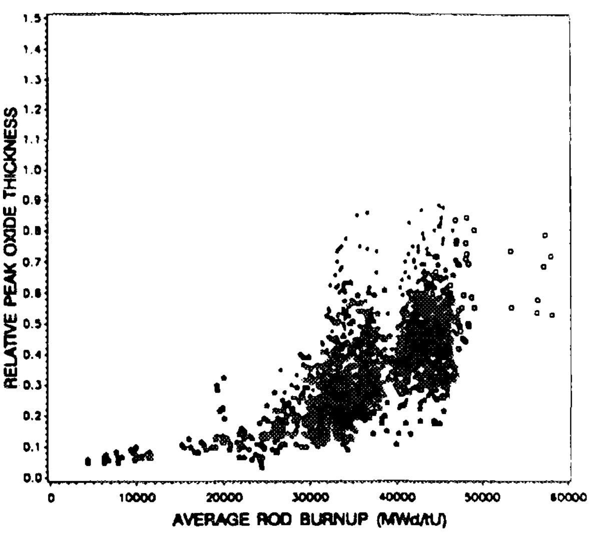

Figure 1 Relative peak oxide thickness as a function of average rod burnup according to Ref. [3]

Fig. 3 is an example of the fuel structure at the rim of a high burn up LWR fuel irradiated under steady state conditions to an average burn up of 75 *GWd/tM*, but with a local burn up at the fuel rim of about 200 *GWd/tM*. The original pre-irradiation fuel structure of large UO_2 grains with a typical grain size of 10 μm is seen to have restructured into very much smaller subgrains, typically 0.2 μm in diameter, separated by low angle grain boundaries - an example of one of the microstructural changes associated with the so-called "Rim Effect". The subgrains are free of dislocations and fission product precipitation. This transformation into a fine scale subgrain structure plays an important role in the local fission product release from the fuel.

Under steady state irradiation conditions at relatively low burn up most of the fission products remain in solid solution within the fuel matrix, or are contained in very small bubbles (< 10 *nm* diameter) or precipitates which can be redissolved by interaction with fission spikes. However, if the rating is increased above the steady state level higher fuel temperatures can lead to extensive redistribution of the fission products which can migrate rapidly to form bubbles and precipitates. Examples of such effects are demonstrated in Figs. 4 and 5 which are micrographs taken on LWR fuel samples which had been transient tested to a 25% increase in linear power for 24 *hrs.*, following steady state irradiation to a burn-up of approximately 4.5% FIMA.

In Fig. 4 large fission gas bubbles with diameters of 200 *nm* are found to have nucleated and grown on an extensive dislocation network, which serves as a rapid diffusion path for the fission gas in the fuel matrix. The formation of such bubbles leads to local swelling of the fuel (in this case about 4%) which can lead to very significant macroscopic effects when integrated over the fuel cross section. In addition the formation of fission gas porosity within the fuel matrix influences the local fuel conductivity.

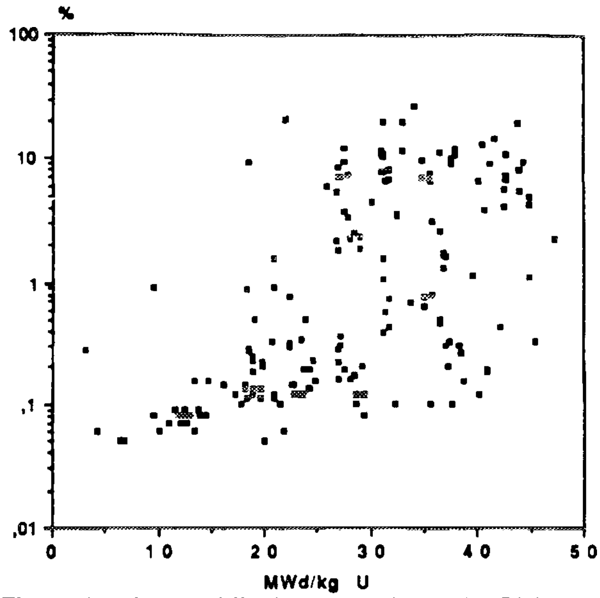

Figure 2 Integral fission gas release for BWR rods according to Ref. [4]

Figure 3 Transmission Electron Micrograph of the rim of a high burn up LWR fuel sample, showing the restructuring into a fine scale subgrain structure, free of dislocations or fission product precipitation

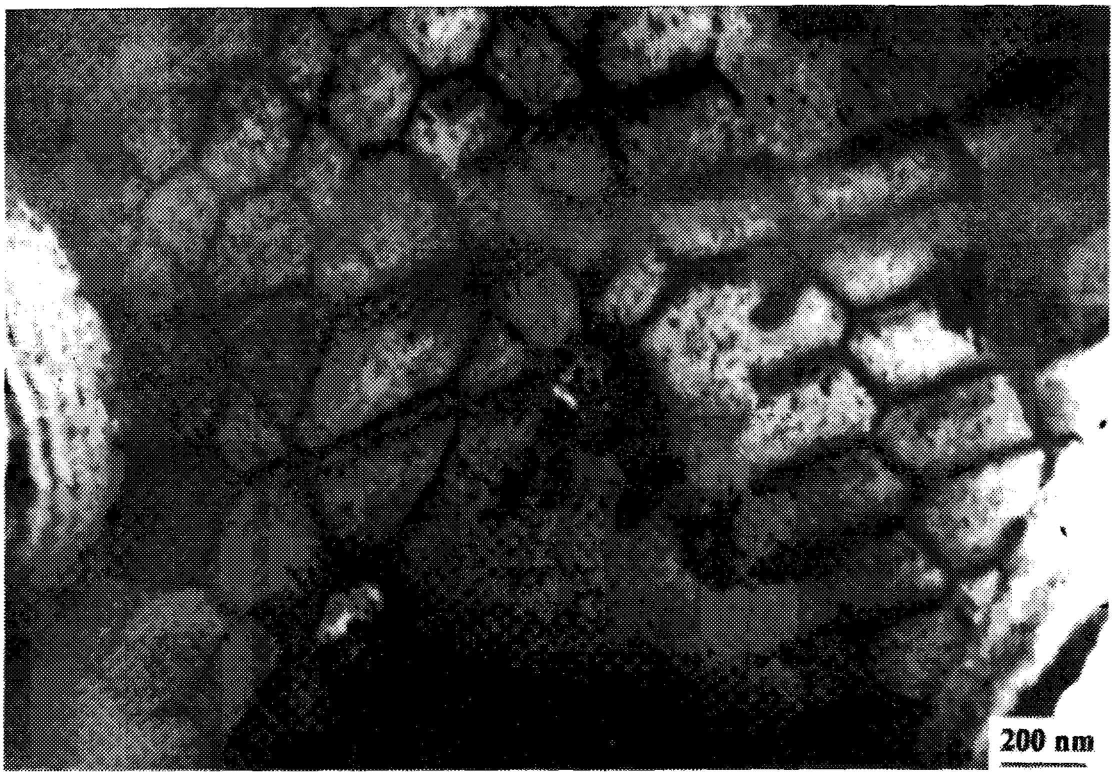

Figure 4 **Transmission Electron Micrograph of a transient tested LWR fuel sample showing the growth of large fission gas bubbles on a dislocation network within the UO_2 matrix**

A similar effect is demonstrated for the solid fission products in Fig. 5, taken on an LWR fuel sample with a similar transient history. A very large precipitate of the metallic fission products (Pd, Mo, Tc, Ru, Rh - a so-called 5-Metal particle identified by Energy Dispersive X-ray Analysis in the electron microscope) has grown within the fuel matrix, associated with extensive finer precipitation along the dislocation lines. The formation of such solid state precipitates also leads to fuel swelling, and, in addition, has important consequences for the distribution of released fission products following a reactor accident.

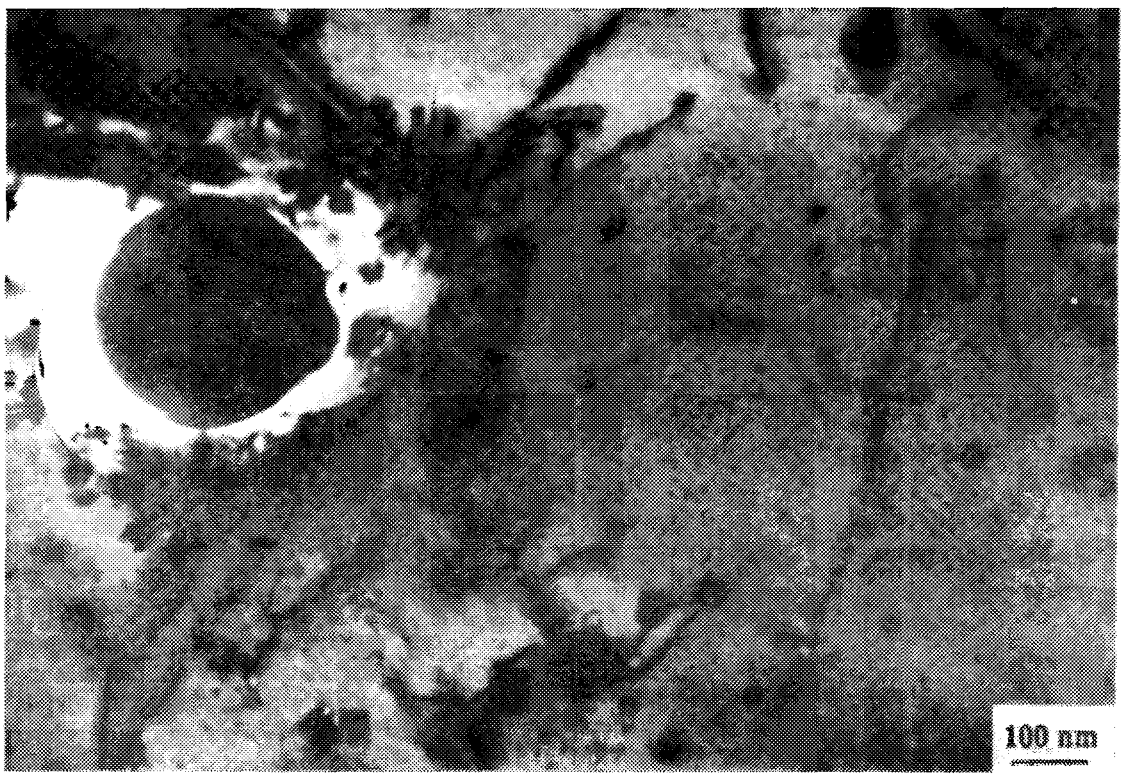

Figure 5 **Transmission Electron Micrograph of a transient tested LWR fuel sample showing the growth of a very large precipitate of the metallic fission products within the fuel matrix**

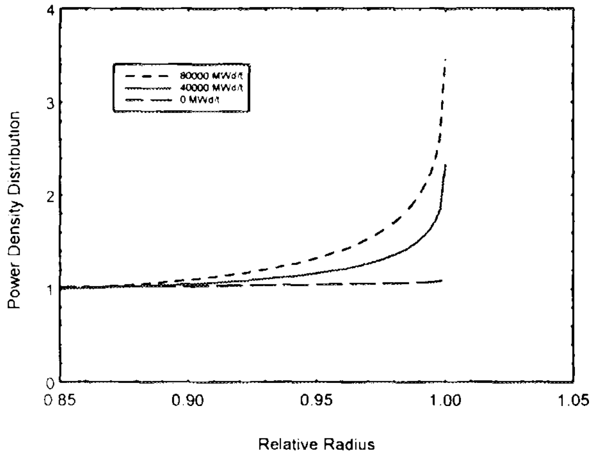

Figure 6 Radial power density distribution as a function of the relative fuel radius

These examples indicate just a few of the microscopic processes occurring at a local level within the fuel matrix which can have a determining influence on the macroscopic behaviour of a fuel pin.

3 Basic Equations

As already indicated, the fuel rod behaviour is determined by thermal, mechanical and physical processes such as densification, swelling, gas release, irradiation damage etc. Most of the physical processes depend exponentially on temperature and are highly non-linear functions of stress. Any fuel rod model must therefore include the solution of the heat conduction equation and the principal mechanical equations, i.e. equilibrium and compatibility, together with constitutive equations. These equations form the theoretical structure into which physical models must be incorporated.

3.1 Basic Equations of Heat Transfer

The calculation of temperatures in a fuel rod is one of the primary goals of fuel element modelling. The accuracy of these calculations influences such strongly temperature-dependent physical phenomena as fission gas diffusion and release, restructuring creep, thermal expansion, etc. In the following the uncertainties of the thermal analysis are outlined.

The basic principle of energy conservation must be applied and the thermal analysis in a fuel rod is governed by the heat conduction equation

$$c\,\rho\,\frac{\partial T}{\partial t} = \nabla \cdot \lambda \nabla T + q''' \qquad (1)$$

where: c = specific heat, ρ = density,
T = temperature, λ = thermal conductivity,
q''' = power density.

As can be seen, the local temperature T depends on local material properties (thermal conductivity λ, specific heat c, density ρ) and the local power density q'''. In some models the radial distribution of the power density q''' is neglected. However, Fig. 6 clearly demonstrates that at high burnup this simplifying assumption is no longer valid. The temperature T also depends on time dependent boundary conditions and heat transfer coefficients (coolant-to-fuel rod and gap conductance).

Due to the numerous nonlinearities involved, only numerical techniques are possible which are of crucial importance because they determine the numerical reliability (stability) and to a large extent the total computer cost. Standard techniques such as finite difference or finite element are used and need not be discussed here.

It is pointless to ask what has the largest influence on the temperature distribution in a fuel rod for all real or postulated conditions. Nevertheless, under steady-state normal operation the gap conductance and the thermal conductivity λ dominate. However, this does not imply that the specific heat c and density ρ are of minor relevance. Since in most cases the thermal conductivity is only indirectly obtained from measurements of diffusivity, these two properties are of equal importance.

3.1.1 Thermal conductivity of LWR fuel

The thermal conductivity of LWR fuel has been extensively investigated, both theoretically and experimentally. A pioneering work on UO_2 is that of H.E.Schmidt [6] and the theoretical understanding is summarized by Hyland [7]. Three contributions are identified: conduction through lattice vibrations (phononic term), conduction through free electrons and a small contribution due to radiation. It is generally accepted that at lower temperatures the phononic term (2) dominates.

$$\lambda_{phonon} = \frac{1}{a + bT} \qquad (2)$$

The introduction of solid fission products and the formation of fission gas bubbles should decrease the thermal conductivity λ. Since point defects (oxygen interstitials or vacancies) and fission products act as phonon scatterers an extended phononic term of the form

$$\lambda_{phonon} = \frac{1}{a + a_x x + a_{bu} bu + bT} \qquad (2a)$$

seems to be justified, where $a_x x$ accounts for the effect of hypo- or hyperstoichiometry and $a_{bu} bu$ for the burn-up effect. The variable x is the stoichiometric deviation $|O/M - 2|$.

Today we have clear evidence for the degradation of the thermal conductivity λ with burn-up from three sources:

1. Fuel centre line temperature measurements performed at Risoe on refabricated fuel rods with different designs and burn-up levels [8].
2. Fuel centre line temperature measurements performed at Halden on specifically designed fuel rods with a small pellet-to-clad gap, negligible fission gas release and fuel restructuring proved the degradation of λ with burn-up. A reduction of 6 - 8% per 10 *Mwd/kg* UO_2 was found by Kolstad and Vitanza [9].

3. Measurements performed on SIMFUEL (SIMulated high burn-up nuclear FUEL) with an equivalent burn-up of 3 and 8 atom% provided data of the intrinsic conductivity (i.e. without gas bubbles, cracks etc.) and proved a degradation of the thermal conductivity λ with burn-up. The reduction was approximately linear with burn-up, Lucuta et al. [10].

Direct centre line temperature measurements on irradiated fuel rods give only an integral information over the entire fuel rod, whereas the measurements performed on SIMFUEL give local values.

All findings are in agreement with what is expected from theory. It can be concluded from the general Eq. (2a) that the degradation of the thermal conductivity λ with burn-up is more pronounced at lower temperatures. Kolstad and Vitanza obtain the factor a_{bu} > > 0.014 - 0.016 *mK/W* per atom percent burnup, Lucuta et al. obtain almost the same factor although their correlation is more complicated.

Apart from the temperature, the thermal conductivity of UO_2 also depends on the porosity *P*, the oxygen-to-metal ratio O/M (stoichiometry) and the burn-up bu. These relations are known for nearly 20 years [11].

Porosity in a ceramic material invariably decreases the thermal conductivity. Geometry (morphology, size, shape, orientation) and physical properties (emissivity of the solid, properties of the gas trapped inside the pores) are of importance. Theoretically, all of these effects are in principle understood. The problem, however, is the determination of the detailed relations. At the beginning of irradiation a given fabrication porosity (pores with a given size distribution and morphology) exists, which decreases during irradiation. Simultaneously, depending on irradiation conditions, new classes of porosity evolve, mainly at grain boundaries but also in the interior of the grains. In addition, macroscopic and microscopic cracking of the fuel takes place that provides further internal volumes. Consequently, porosity correction factors will always remain a source of uncertainties.

Hypo- and hyperstoichiometric fuel, i.e. fuel with a nonzero value of x has a lower thermal conductivity than stoichiometric fuel since introducing point defects (vacancies or interstitials) increases phonon scattering. As in the case of the porosity correction, the difficulty arises from how to obtain the necessary details of the local stoichiometry. The redistribution of oxygen in nonstoichiometric fuels is not completly understood. Chemical processes certainly complicate the picture.

Recently, a new dependence of λ was found which results from an obviously structural change of the UO_2 fuel near the surface at high burn-up, in the so-called "rim zone". Thus, the burn-up effect consists of two contributions: a general degradation in the thermal conductivity of the fuel and the development of the "rim zone" which may act as a thermal barrier.

At the pellet surface of LWR fuel the local burn-up is enhanced due to the build-up of plutonium. Above a local threshold burn-up of 68000 *MWd/tU* microstructural changes (loss of optically definable grain structure, i.e. low angle subgrains on a very fine scale, change of porosity) and a lower dislocation density, lower density of intragranular fission gas bubbles and precipitates was observed [12]. Because of the microstructural changes and the high quantities of fission gas present in this zone, there is a potential for athermal fission gas release. The thermal conductivity of the rim zone is unknown. An increased porosity and a lower concentration of fission gas in the matrix (i.e. a higher concentration of fission gas on the grain boundaries) implies a lower thermal conductivity than would be expected from the burn-up effects discussed above. Since the heat flux density is highest at the pellet surface such a rim effect would further increase fuel temperatures. A first attempt to quantify this effect was undertaken by C. Bagger, M.Mogensen and C.T.Walker. [13]. The authors used direct centre line temperature measurements, temperature "markers" (grain growth and xenon diffusion data) to construct a radial temperature profile. From known boundary conditions they concluded that the thermal conductivity of the rim zone is considerable lower (factor 5 to 10) than in the rest of the fuel, thus the rim zone acts as thermal barrier. This finding, however, needs further clarification.

3.1.2 Heat transfer coefficient between fuel and cladding

The heat transfer coefficient between fuel and cladding (gap conductance) is an important contributor defining the temperature of the fuel. At high burn-up where the gap is closed, the influence is not as pronounced as at the beginning of irradiation. Nevertheless, the gap still acts as a thermal barrier. Gap conductance models depend on temperature, emissivity, gas composition, gas and contact pressure and on surface morphology (roughness, waviness) which may change significantly at high burn-up. Consequently, the surface morphology and composition of LWR fuel and cladding must be known which is especially at high burn-up not the case. The gap conductance at high burn-up needs investigation.

In most fuel rod performance codes, the classical gap conductance model of Ross and Stoute [14] is applied (the contact term is omitted for the sake of simplicity):

$$h_{gas} = \lambda_{gas} / \left(s + a_1 \, \Delta R + l_{ex} \right) \tag{3}$$

where s is the gap width, ΔR is the sum of surface roughnesses, a_1 a fitting parameter and l_{ex} is the sum of gas extrapolation lengths. An extension of this model is the URGAP model [15] which takes the increase of surface area by the roughness into account:

$$\frac{1}{h_{gap}} = \frac{1}{h_{f,g}} + \frac{1}{h_{cl,g}} + \frac{1}{h_g} \tag{4}$$

where:

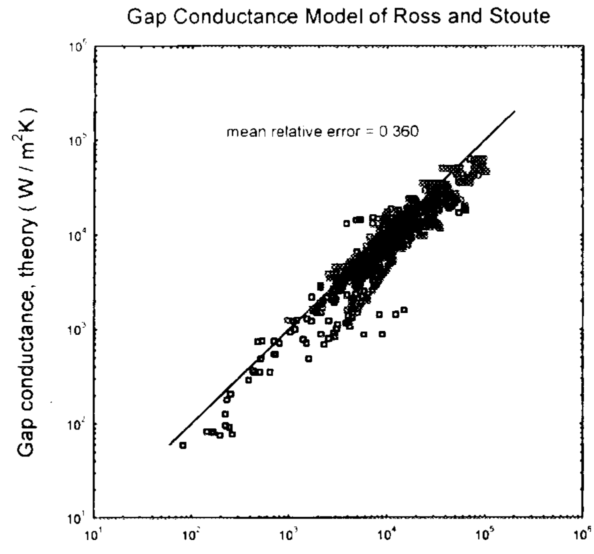

Figure 7 Comparison of the gap conductance model of Ross and Stoute [14] with experimental data

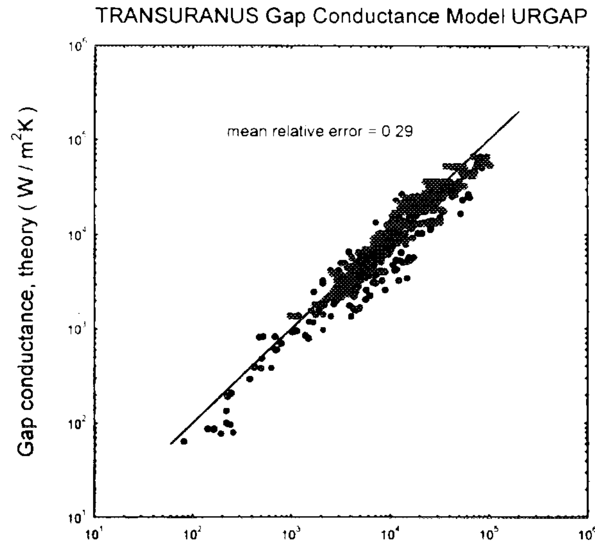

Figure 8 Comparison of the URGAP gap conductance model [15] with experimental data

$$h_{i,g} = \frac{\lambda_{gas}\left\{\beta_1\,\beta_2\,\Delta R_i\,(1-\frac{\lambda_{gas}}{\lambda_i})+1\right\}}{\beta_2\,\Delta R_i}; \quad i = \mathrm{f,cl}$$

$$h_g = \frac{\lambda_{gas}}{s + l_{ex,f} + l_{ex,cl}}$$

and β_1 and β_2 are fitting parameters.

The gap conductance has been extensively investigated by out-of pile and in-pile experiments. The data base of the Institute for Transuranium Elements consists of approximately 1000 well characterized data covering different material pairings, gas compositions, gas pressures, gap widths, contact pressures and surface roughnesses. Therefore, the error of this important submodel can be investigated since all important input parameters are known. The goal of the following investigation is to find out to what extend theoretical details can be justified by the data. It should be noted that the data bases of other submodels are less extensive and that the uncertainty of gap conductance must be considered as low compared with other submodels.

Both, the standard Ross-Stoute model and the URGAP model were investigated. Different options for the thermal conductivity of a gas mixture, the correlation of Lindsay and Bromley [16] and that of Tondon and Saxena [17] were analysed together with different correlations for the gas extrapolation length. A very simple correlation and a more complicated one were analysed which take a standard correlation for accomodation coefficients into account. The results are shown in Table 1 and Figs. 7 and 8. The conclusion is obvious: a good agreement can be obtained with a relatively simple correlation. Details such as the influence of gas accomodation coefficients cannot be inferred from the data simply because the spread is too large. The general conclusion is that beside details of gap conductance model the average relative error is in the range of 30%.

Similar to the discussion of the influence of porosity and stoichiometry on the thermal conductivity of the fuel the problem lies in applying the right data for gap conductance models. At the beginning of the irradiation where the gap is open, the gap size is determined by highly uncertain relocation of the pellet fragments. At high burn-up the integral effect of all irradiation processes, especially gas release (influence on gas composition) and swelling (influence on contact pressure) contribute to an uncertainty of the gap conductance which is certainly larger than the 30% uncertainty of well characterized gap conductance data.

3.1.3 Discussion of the uncertainties of the thermal analysis

The basic equations and most of the dominating processes are well understood. An exception are the properties of the new structure formed at the edge of the fuel ("rim zone") at high burnup which are unknown. For some correlations or models it is possible to identify individual errors. It is interesting to note that these individual errors are not often mentioned and it is suggested that these errors should be more addressed. They would form the basis for probabilistc analyses. The more important problem arises when these correlations or models are applied. Most of the important details of the model vary with burnup and it turns out that some of this information is simply lacking during the irradiation. Examples are the pore distribution and morphology or simple gap size, gas composition and pressure. One can imagine that it will be possible to further reduce uncertainties of the thermal

Table 1 Mean relative error of gap conductance of the two gap conductance models and the data base

Mean relative error of gap conductance	Ross and Stoute model	Urgap model
1. Thermal conductivity of gas mixture according to Lindsay and Bromley [16]; accomodation coefficients are taken into account	0.360	0.306
2. Thermal conductivity of gas mixture according to Tondon and Saxena [17]; accomodation coefficients are taken into account	0.312	0.290
3. Thermal conductivity of gas mixture according to Lindsay and Bromley; accomodation coefficients are **not** taken into account	0.360	0.311
4. Thermal conductivity of gas mixture according to Tondon and Saxena; accomodation coefficients are **not** taken into account	0.316	0.293

analysis, however, it will never be possible to completly eliminate them.

3.2 Mechanical Equations

The structural analysis of a fuel rod is governed by three basic mechanical principles: equilibrium, compatibility and constitutive equations. The equilibrium and compatibility equations are the standard equations of structural analysis. The constitutive equations, also called stress-strain relations or as material equations, are well known in their simplest form as Hooke's law. A further specialisation of the generalised Hooke's law is the case in which the material is assumed to be linear, isotropic and elastic. The theory of an elastic-plasticity solid is already a very complex and broad subject, however, the fuel behaviour is much more complex due to local swelling, densification and cracking. The fuel must be considered as anisotropic, with different behaviour under tension and compression. Plastic deformations may be caused within short time scales by high stresses and by lower stresses within large time scales. These stresses cause cracking of fuel pellets (macroscopic cracks) but may also cause cracking on a microscopic scale (microcracks). The volume change by densification and swelling is extremely complex since structural changes such as shrinkage of pores, the formation of new bubble population, grain growth and grain decoration and specific irradiation induced damage are involved. How can this complex material behaviour be encapsuled in a constitutive equation?

The presently adopted approach follows the classical approach one by adding strain increments $d\varepsilon_i$ from different mechanisms to obtain the total strain increment $d\varepsilon_{tot}$, for instance:

$$d\varepsilon_{tot} = d\varepsilon_{elastic} + d\varepsilon_{thermal} + d\varepsilon_{plastic} + d\varepsilon_{creep} + \\ + d\varepsilon_{swelling} + d\varepsilon_{densification} + d\varepsilon_{crack-rel} + \cdots \quad (5)$$

where "crack-rel" means a strain increment due to cracking and relocation. Traditionally, plastic and creep strains are distinguished and it is assumed that the volume is kept constant during short time plastic or long time creep deformation.

It is obvious that this concept of material behaviour has some limitations since all contributions are considered as independent. In fact, since all of these proccesses act simultaneously, one of the general difficulties in measuring the behaviour of the fuel during irradiation is to single out specific mechanisms. Clearly, the constitutive equation of fuel needs to be re-evaluated, but it is unknown to the authors whether research is being done in this area.

In the context of this evaluation of uncertainties it is sufficient to briefly discuss the individual contributions. Thermal strains are well known. Swelling may be split into swelling due to solid fission products (volume change ≈ 0.7% per atom percent burnup) and swelling due to gaseous fission products. The latter is usually small (volume change ≈ 0.1 - 0.3 % per atom percent burnup) but may be significant at temperatures above 1400°C. Thermal and irradiation induced densification is in the range of a few percent depending how stable the fuel is.

Theoretically, the most difficult problem is how to treat the cracked fuel. As can be seen from Eq. (5), crack and relocation may be treated as a fictitious volume change, however, this is only one possible concept. A three-dimensional sketch of a .deformed and cracked fuel pellet is shown in Fig. 9. Pellet cracking and relocation can be separated into two mechanisms:

(a) Mechanism 1: The elastic strain prior to cracking is redistributed, i.e. the pellet volume increases and the stress level in the pellet is reduced. This stress reduction depends strongly on the number of cracks and the crack pattern. The following number gives an order of magnitude: 4 to 6 cracks reduce the stress level by a factor of 5 - 10. Fig. 10 shows that not only the stress is reduced but changes also its distribution.

(b) Mechanism 2: depending on the gap size, a relocation, i.e. a gross movement of fuel fragments occurs.

The first mechanism can be modelled whereas the second one by its nature can be treated only empirically. These problems are discussed in greater detail in [18]. It is to be noted that in most situations the second mechanism is by far more important so that details of the first mechanism are masked by the inevitable uncertainties of the relocation process itself.

A closer analysis of the individual contributions reveals that the total volume change, i.e. the sum of all individual volume changes is uncertain which directly affects gap closure and pellet cladding mechanical interaction. Consequently, due to cracking, relocation and all other effects, the resulting local stresses in the fuel are rather uncertain and great care must therefore be taken if highly stress dependent models are involved.

4 Basic Concepts

In the following chapter three more general concepts are briefly outlined which have been discussed controversially since the beginning of modelling the fuel rod behaviour.

4.1 One-Dimensional Models versus 2-D or 3-D Models

In most of the fuel rod models the theoretical and computational effort is reduced by reducing the geometric dimensions. Relatively simple solutions can be obtained, if the geometric problem is confined to one-dimensional, plane and axisymmetric idealisation (1-D models). A 2-D description (r-θ or r-z coordinates) which is usually based on finite elements techniques significantly increases the computational effort. Even 3-D models are under development [19], however, they are restricted to the analysis of very specific problems. It would be ideal if 3-D analyses would be used to produce adjustment factors for 2-D or even 1-D models [20]. However, the authors are not aware that a systematic attempt has ever been made.

One-Dimentional and 2-D models may have very different characteristic behaviour. From the assumptions made in 1-D models it is clear that deformations of fuel and cladding must differ in both approaches. Fig. 11 shows a schematic comparison of the mechanical interaction between fuel and cladding in a 1-D and a 2-D description. It is evident that local stresses, especially the stresses in the vicinity of the contact area vary significantly. Closer analyses show that local stresses may differ by a factor of 2 to 4. One of the most important limitations of 1-D model is therefore that local mechanical problems cannot be predicted by this approach.

One-dimensional models are well developed and widely used. However, more local details are needed for further optimisation of fuel rods. Therefore a clear trend can be observed: 1-D analyses are augmented by 2-D ones, both approaches being complimentary.

4.2 Steady-State versus Transient Modelling

In the past a "steady-state modelling" was discussed in contrast to a so-called "transient" modelling. Both terms were never clearly defined and different people understood different things by this terminology. This steady-state modelling related to the analysis of normal irradiation behaviour with emphasis on the thermal behaviour. The author suspects that the wording "steady-state modelling" originated from the solution of the simple steady-state heat conduction equation. However, it turned out that the definition of a transient during which specific transient models are to be switched on (and after this transient to be switched off again) is impossible. This means that the development of time-dependent models which also take gradients of loadings or other parameters into account is indispensable in all areas.

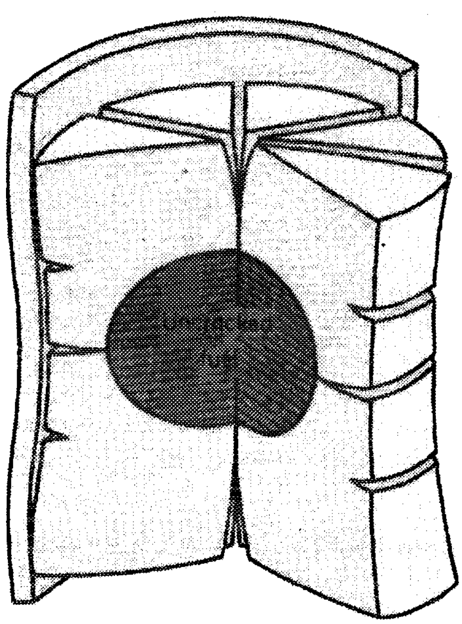

Figure 9 **Three-dimensional sketch of a deformed and cracked fuel pellet (from Ref. [18])**

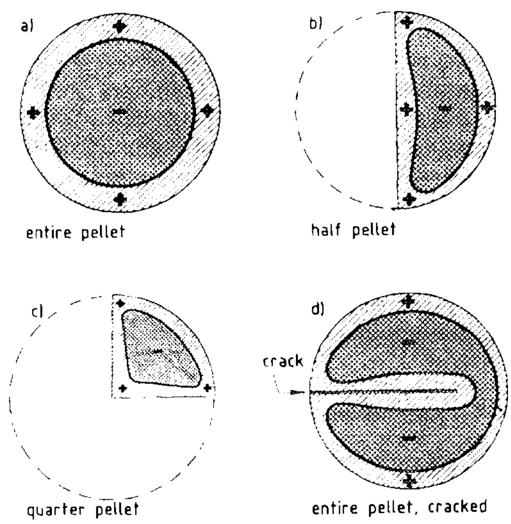

Figure 10 Thermoelastic, hydrostatic stress field in different pellet geometries:
+ tensile stress,
- compressive hydrostatic stress
(results of Mezzi, quoted in Ref. [18])

It is not sufficient to develop a model in an isolated manner, just to describe a particular phenomenon. The model itself should be consistent, reacting in a physically correct manner to the variations in all of the relevant parameters. Such models are certainly more difficult to develop but their usage is straight forward. It should not be forgotten that it was extremely difficult to use some of the simple steady-state models. For instance, years ago, frequently used fission gas release models were of the type:

$$f_i = a_i, \quad \text{if } T_1 \leq T < T_2 \qquad (6)$$

where f_i is the fractional fission gas release in the temperature range i between temperatures T_1 and T_2. Eq. (6) can be applied without problems if the fuel temperatures remain constant. For the more realistic case of varying temperatures, Eq. (6) cannot be applied without a specific convention of what amount of gas is to be released in a given time step. Since this convention is nothing else but a pure substitution for a time-dependent, more detailed model, a generally valid convention cannot exist and therefore such simplified models are more or less useless.

4.3 Empirical versus Mechanistic Models

There is no doubt that physical models need to be "correct". But it will be shown below that for nearly all models or processes the most important input parameter such as temperature and local stress exhibit significant uncertainties. Consequently, the complexity of models must be seen in the light of other approximations made. Sometimes it is claimed that detailed models or purely mechanistic models are based upon "first principles". However, it is difficult if not impossible to describe materials behaviour without some free parameters. Fission gas release and swelling is an area in which such attempts were made. Despite highly sophisticated models, the predictions are in almost all cases too uncertain to justify a fuel rod performance code in which all mechanistic models available are included. The relevance of mechanistic models is a more fundamental one since only such detailed models can determine whether the basic processes are understood and what mechanisms are relevant. Fuel rod performance codes used in research may therefore have included complex models whereas standard fuel rod performance codes used preferentially for predictions may include somewhat simplified models which are consistent with the other uncertainties. These type of models are sometimes called "models of adequate complexity".

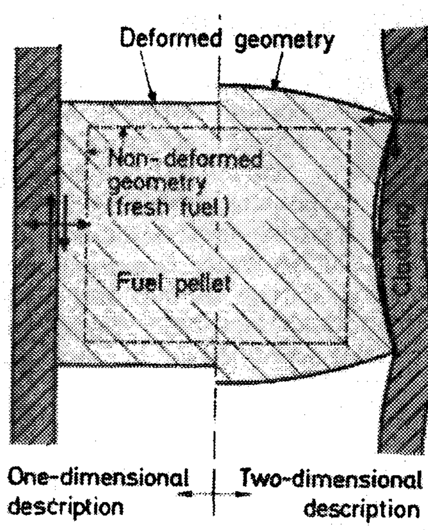

Figure 11 Schematic comparison of the mechanical interactions between fuel and cladding in a one-dimensional (left diagram) and a two-dimensional description (right diagram)

5 Conclusions

Any fuel rod model must include the solution of the heat conduction equation and the principal mechanical equations, i.e. equilibrium and compatibility, together with constitutive equations. These equations form the theoretical structure into which physical models are incorporated. We have identified basic limitations from the different assumptions made for the solution of the governing equations. It is evident that the thermal and the mechanical analysis are strongly coupled and therefore errors are propagated. The individual correlations and processes are also not error free and in only some cases can individual errors be estimated. There is for almost all processes a good understanding of the dominating parameters. However, the unsolved problem is how local quantities such as pores and grain structure, stresses etc. evolve during the irradiation. Thus, the main source of uncertainty lies in wrong input data for local processes.

The problem of determining fission gas release and swelling correctly even after more than 20 years of many specific experiments must be attributed to this category of uncertainties: the more detailed a fission gas release model is, the more local and hence uncertain information is used. For instance, all detailed fission gas release models include a specific gas interlinkage model which determines whether gas can be kept on grain boundaries. However, in order to analyse the interlinkage network, local stresses are needed which exhibit the highest uncertainty. This explains why fission gas release and swelling is so sensitive to alterations in the number of cracks or of the crack structure.

As a final example the centre line predictions of two codes for the FUMEX blind exercise [21] are given in Fig. 12. The selected temperatures cover a wide range of burnup and ratings. Besides the trend to over predict temperatures fair agreement is obtained. This figure is representative of the pres-

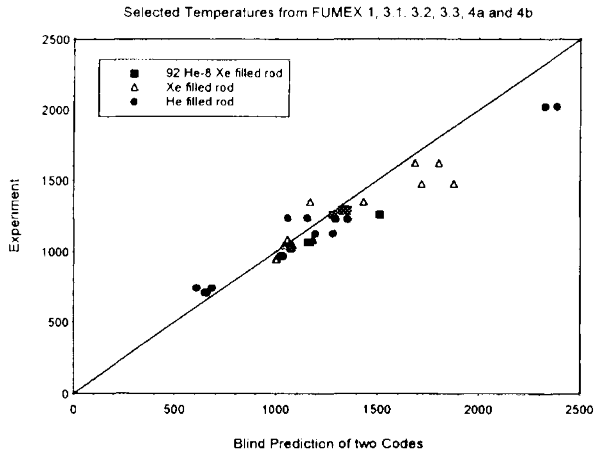

Figure 12 Comparison between measured temperatures and those predicted by two different codes taken from the blind FUMEX code comparison exercise [21]

ent state-of-the-art of the thermal analysis. It should be noted that in order to obtain such an agreement all other models, the mechanical model, physical models (swelling, identification, and fission gas release etc.) and correlations (thermal conductivity, creep etc.) must be correct. It will be very difficult to improve the predictability of fuel performance codes significantly. However, it must also be stated that since the D-COM exercise in 1984 [22], fuel rod performance codes have been improved considerably and must be now considered as mature tools for further optimisation of fuel rods.

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TRANSURANUS: A Fuel Rod Analysis Code Ready for Use

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1 Introduction

The basic concepts of fuel rod performance codes are discussed in a contribution to this conference [2]. In the following paper the TRANSURANUS code [3], a so-called 11/2-dimensional code is presented. The TRANSURANUS code was developed at the Institute for Transuranium elements and is now in use in various institutions through Europe. It is based on a clearly defined mechanical/mathematical framework into which the detailed models describing the physical behaviour can easily be incorporated. The code is well structured and therefore easy to understand. In this paper emphasis is put on the description of this mechanical/mathematical framework rather than the physical models or available options. The latest developments concerning high burn-up models are outlined.

2 General Concept

The mechanical/mathematical concept of the TRANSURANUS code consists of a superposition of a one-dimensional radial and axial description (the so-called quasi two-dimensional or 11/2-D models).

The fuel rod is divided into axial slices and at a given time t the rod is analysed slice per slice. After all slices have been analysed, the slices need to be coupled together which means that quantities such as the inner pin pressure or the axial friction forces between fuel and cladding are determined (axial coupling).

The structure of the TRANSURANUS code reflects the structure of the theoretical model:

Level 1 is the main driver of the code in which the time integration is organised. The new time $t_{n+1} = t_n + \Delta t$ is determined, where the time step Δt is the minimum of many different time step criteria given by stability criteria or criteria to control the accuracy. This time step control refined over many years of code development is a very important feature of the code.

Level 2 controls the axial loop over all slices, the axial coupling and its convergence. A special technique is applied to minimize computer memory.

Level 3 controls the analysis of a slice, i.e. at this code level the thermal and mechanical analysis is performed for which all physical models are needed. A clear distinction is made between explicit and implicit models. For implicit (or mixed explicit-implicit) model special procedures for obtaining convergence are necessary.

One of the main advantages of this clear structure is that the user can easily incorporate new models. The user is more or less only concerned with the decision whether a model is explicit or implicit and he needs only a limited knowledge how to interface this model with the TRANSURANUS variables.

3 Basic Equations

In the following sections the basic equations are briefly outlined in order to make the underlying theoretical concepts better understandable.

3.1 Thermal Analysis

Thermal analysis of the whole fuel rod is obtained by a superposition of one-dimensional radial and axial energy conservation equations [4]. Here, only simplified equations are given. The energy equation (heat conduction equation) for fuel, cladding and structure is

$$c\,\rho\,\frac{\partial \vartheta}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(\lambda r \frac{\partial \vartheta}{\partial r}\right) + q''' \qquad (1)$$

whereas the energy equation for the coolant is given by

$$c\,\rho\,\frac{\partial \vartheta}{\partial t} + c\,\rho\,w\frac{\partial \vartheta}{\partial z} = \frac{q''_{cl,c}\,2\pi r_{cl,o}}{A} + q''' \qquad (2)$$

where: c = specific heat, ρ=density, ϑ = temperature, r = radius, λ= thermal conductivity, q'' = heat flux density, q''' = power density, w = coolant velocity, A = area, cl,o = cladding-outer, cl,c = cladding-coolant and z = axial coordinate.

The main features of the thermal analysis are summarized as follows:

1. The solution method includes the well-known Finite Difference Method and the Finite Element Method as special cases. The standard usage is an optimum combination of both which makes the solution extremely accurate.
2. The methods includes explicit, implicit or Crank-Nicholson integration procedures. Standard usage for transient conditions is the Crank-Nicholson scheme.
3. Phase changes (melting, boiling) are considered.
4. An important aspect of the thermal analysis is the heat transfer between fuel and cladding. The TRANSURANUS code includes a detailed model for this gap conductance which is fully described in Reference [5]. The URGAP model is available on request.

3.2 Mechanical Analysis

The mechanical analysis consists of the calculation of stresses, strains and the corresponding deformations. Dynamic forces are in general not treated and the solution is therefore obtained by applying the *principal conditions* of *equilibrium* and *compatibility* together with *constitutive relations*.

In the following a brief overview is given which should allow the reader to understand the basic mechanical concepts underlying the TRANSURANUS code. More details are given in References [3] and [6].

In order to derive an adequate solution, the following assumptions are made:

1. The geometric problem is confined to one-dimensional, plane and axisymmetric idealization, i.e. the axial deformation is constant across the radius (modified plane strain condition).
2. The elastic constants are isotropic and constant within a cylindrical ring, a so-called coarse zone.
3. The constitutive equations are given by

   $$\varepsilon^{tot} = \varepsilon^{el} + \varepsilon^{ex} \tag{3}$$

   where tot = total, el = elastic, ex = sum of all nonelastic strains,

   $$\varepsilon = \begin{Bmatrix} \varepsilon_r \\ \varepsilon_t \\ \varepsilon_a \end{Bmatrix}; \; r = \text{radial}, \; t = \text{tangential}, \; a = \text{axial}.$$

   One important theoretical concept of the TRANSURANUS code is that all volume changes due to different processes such as densification and swelling, cracking, etc. are expressed via strains.

The assumptions, together with the compatibility equations

$$\varepsilon_r = \frac{du}{dR}, \; \varepsilon_t = \frac{u}{R}; \; \varepsilon_a = \text{constant} = C_3 \tag{4}$$

and the equation of equilibrium

$$\frac{d\sigma_r}{dR} = \frac{\sigma_t - \sigma_r}{R} \tag{5}$$

lead to the classical semianalytic solution of the problem. The radial deformation is obtained by

$$\begin{aligned} u(R) = \frac{1-2\nu}{2(1-\nu)} & \left\{ \frac{1}{R} \int_{R_i}^{R} R\,(\varepsilon_r^{ex} + \varepsilon_t^{ex})\, dR \right\} + \\ & + R \int_{R_i}^{R} \frac{\varepsilon_r^{ex} - \varepsilon_t^{ex}}{R}\, dR + \frac{\nu}{1-\nu} \frac{1}{R} \int_{R_i}^{R} R\,\varepsilon_{tot}^{ex}\, dR + \\ & + C_1 R + \frac{C_2}{R} \end{aligned} \tag{6}$$

where $R=r+u$, R = radius of the deformed geometry and r = radius of the reference geometry.

Similar expressions result for the stresses and strains. Eq. (6) shows that equilibrium is always taken at the deformed geometry. In addition, geometrical second order effects are taken into account approximately (not shown here). The constants C_1, C_2 and C_3 are determined from the well-known boundary conditions.

The integrals included in the solutions (e.g. Eq. 6) must be evaluated numerically. This is the reason why the solution is called a *semianalytic solution*. From the theoretical assumptions it follows that fuel and cladding can be divided into an arbitrary number of rings (coarse zones) which are further subdivided in fine zones in order to allow for the numerical integration. Since any discretization can be chosen this is called a variable multizone concept.

Creep of fuel and cladding shows a highly non-linear behaviour with regard to stress where stress exponents of 5 and more must be anticipated. Therefore the calculation of creep is not an easy task and the corresponding numerical procedures must be designed very carefully. In the TRANSURANUS code both principal techniques, explicit and implicit treatments, are used. The treatment of plasticity and cracking is given in Reference [3].

In the case of a whole fuel rod axial friction forces between fuel and cladding are generated which are calculated by a one-dimensional axial model called *URFRIC* (TRANSURANUS FRICtion model [7]).

4 Flexibility of the TRANSURANUS Code

Up to now no distinction was made between the different types of fuel rods and in fact this was not necessary as can be seen from the basic equations. The general concept of the model is that the basic equations apply to all type of fuel rods and reactor conditions. However, specific models are needed for specific problems. The complete set of models and options available is fully described in Table 1 of Reference [3].

The basic equations of material conservation are applied to plutonium redistribution (model of Bober et al. [8]), to pore migration and redistribution of oxygen (OXIRED model [9]). Different models are optional for densification, gas release and swelling, relocation, for predicting the fuel structure, cladding failure, the radial power density, the formation and closure of the cenral void (FBR), the waterside corrosion etc.

In order to be as flexible as possible the relevant material properties such as the elastic constants, the thermal conductivity, the specific heat, the density etc. are formulated in specific subroutines which allow for the incorporation of different correlations. At present up to 30 different correlations for fuel and cladding are possible and the TRANSURANUS code includes material data for all reactor types. Possible fuel materials are oxide, mixed oxide, carbide and nitride, the cladding materials are Zircaloy, steel and niobium and the coolant may be water, sodium, sodium-potassium and helium.

Similar to the formulation of the basic material properties, the general boundary conditions were formulated as flexible as possible and different geometries ranging from an analysis of the fuel alone to the analysis of Fuel, Cladding, Coolant and Structure.

The TRANSURANUS code is written in standard FORTRAN 77 which does not allow for variable dimensioning. However, the radial and axial discretization is very flexible through the usage of pseudo-variable dimensioning.

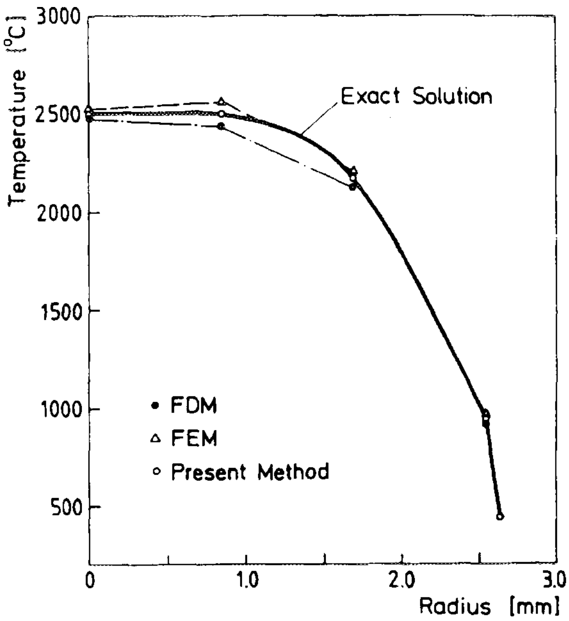

Figure 1 **Radial temperature distribution in a fuel rod (test case). Compared are the analytic, exact solution with different options of the TRANSURANUS code (FDM = finite difference method, FEM = finite element method, Present solution = optimum of accuracy)**

5 Verification of the Basic Equations and Numerical Aspects

The TRANSURANUS code has been verified extensively by three steps:

1. Verification of the numerical techniques by comparison with analytic solutions or by comparison with other techniques.
2. Verification of specific models with experiment.
3. Verification of the TRANSURANUS code by comparison with irradiations which is an ongoing activity.

In the following examples of the three verification steps are given.

5.1 Verification of the Numerical Techniques

During the development of the TRANSURANUS code all possibilities were used to test the basic equations regarding accuracy, computer time consumption and stability. In the following a few examples shall be given in order demonstrate the capabilities of the TRANSURANUS code.

Thermal analysis:

The numerical techniques of the thermal analysis have been carefully designed to be fast reliable and accurate: examples are shown in Figure 1. Special emphasis was put on the problem of obtaining convergence and the total amount of numerical effort [10]. This is the prerequisite for the analysis of complicated power histories in which the thermal analysis has to be done up to several thousand times for a single analysis.

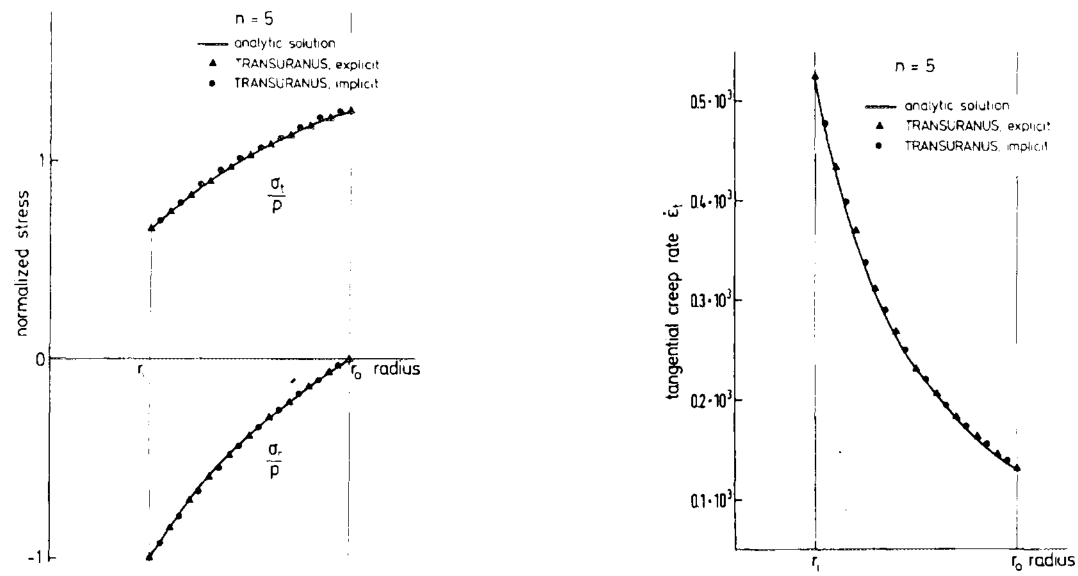

Figure 2 **Radial distributions of normalized stresses and tangential creep rate for a Norton exponent of n = 5. Compared are the analytical solutions with the explicit and the implicit numerical solution**

Mechanical analysis:

The mechanical analysis was also tested extensively. In particular, the highly nonlinear creep was analysed using explicit and implicit techniques. The agreement between both techniques and the analytic solution was confirmed for a thick-walled cylinder. Figure 2 may serve as an example. It is worthwile to note that the explicit technique cannot be applied for standard fuel conditions because the high rates in the centre of the pin, which determine the stability criterion, would lead to extremely small time steps. In fact, the difference in computer time between an explicit and an implicit technique is up to a factor of 5000 for a standard irradiation history.

Mass Transfer:

Plutonium redistribution and pore migration are solved by Finite Difference techniques where the balance equation for each zone is written in conservative form. Fully implicit formulations are used and the resulting systems of equations are penta- and bidiagonal which can be solved effectively. The numerical schemes of radial plutonium redistribution and pore migration can be

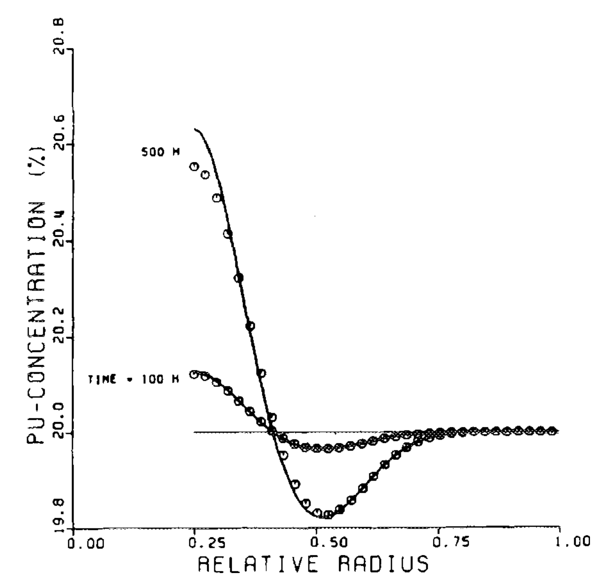

Figure 3 Comparison between the analytic, asymptotic solution of Clement (solid line) and the numerical solution (open circles). Inner and outer fuel radii are 0.75 and 3 mm; centre temperature 2550K, fuel surface temperature 1100 K, initial plutonium concentration 0.2.

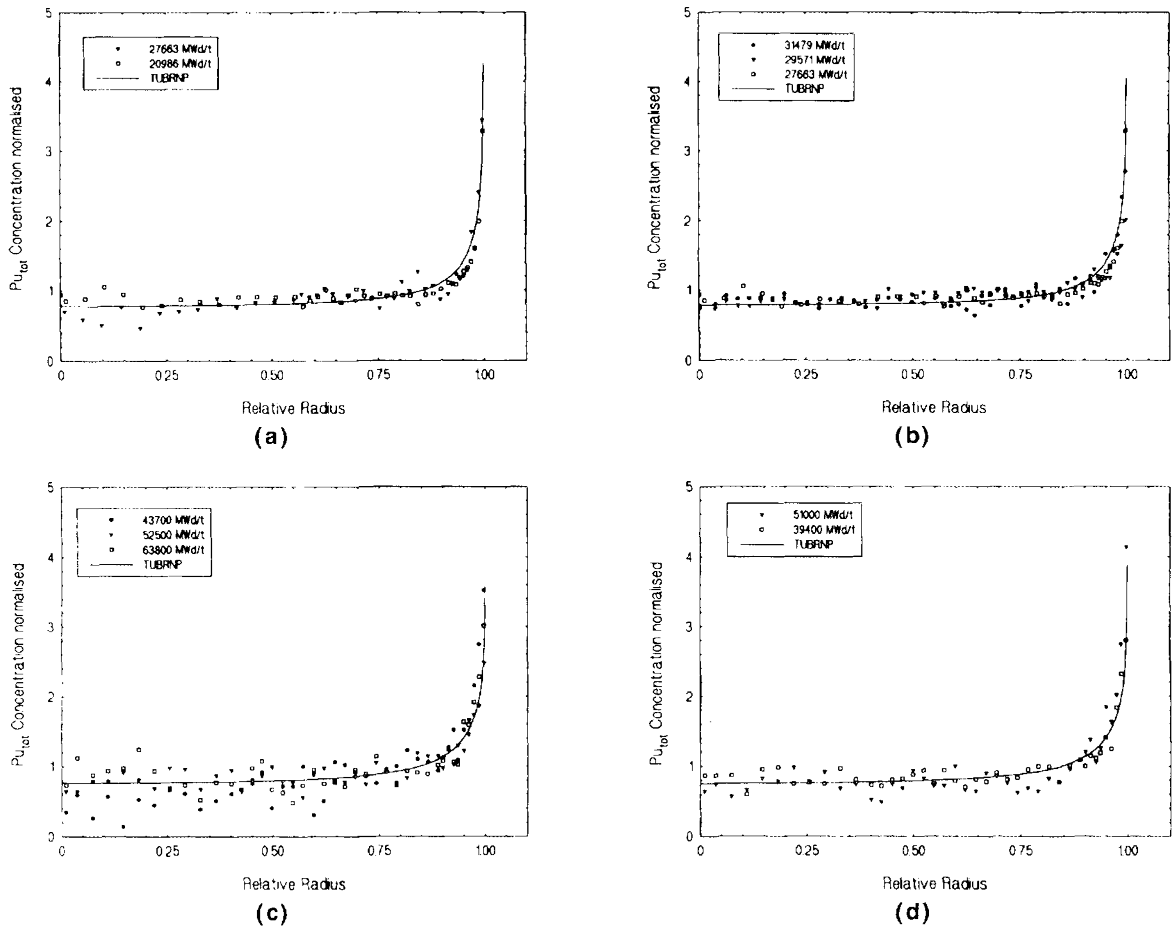

Figure 4 Comparison of the radial total Pu concentration measured with that calculated by the *TUBRNP* model: (a), (b) STRO fuels with enrichment $^{235}U=2.9\%$, calculations with TUBRNP for 25000 and 29000 MWd/t; (c), (d) EPRI fuels with enrichments $^{235}U=5.75\%$ and 8.25%, calculations with TUBRNP for 55000 and 45000 MWd/t.

compared with asymptotic solutions given by Clement [11] which are valid as long as the change from the initial conditions is relatively small. Comparisons between numerical and the analytic solutions of plutonium redistribution are shown in Fig. 3 for two different time steps: the agreement is excellent. The slight deviation between the solutions at the inner boundary at 500 *h* is due to the fact that the asymptotic solution is already of limited use under these conditions.

5.2 Verification of Specific Models

The TRANSURANUS code includes many physical models which cannot be discussed here. One of the recently developed model is the TRANSURANUS burn-up model *TUBRNP* [12]. This model calculates the local concentration of several isotopes by the following burn-up equations:

$$\frac{dN_{235}(r)}{dbu} = -\sigma_{a,235} N_{235}(r) A \tag{7a}$$

$$\frac{dN_{238}(r)}{dbu} = -\sigma_{a,238} \overline{N}_{238} f_1(r) A \tag{7b}$$

$$\frac{dN_{239}(r)}{dbu} = -\sigma_{a,239} N_{239}(r) A + \sigma_{c,238} \overline{N}_{238} f_1(r) A \tag{7c}$$

$$\frac{dN_{240}(r)}{dbu} = -\sigma_{a,240} \overline{N}_{240} f_2(r) A + \sigma_{c,239} N_{239}(r) A \tag{7d}$$

$$\frac{dN_{241}(r)}{dbu} = -\sigma_{a,241} N_{241}(r) A + \sigma_{c,240} \overline{N}_{240} f_2(r) A \tag{7e}$$

$$\frac{dN_{242}(r)}{dbu} = -\sigma_{a,242} N_{242}(r) A + \sigma_{c,241} N_{241}(r) A \tag{7f}$$

The local concentrations of ^{238}U and ^{240}Pu, $N_{238}(r)$ and $N_{240}(r)$, are written as $\overline{N}_{238} f_1(r)$ and $\overline{N}_{240} f_2(r)$ where $f_i(r)$, $i=1,2$ are radial shape functions with normalisation factors defined by,

$$\frac{\int_{r_{in}}^{r_{out}} f_i(r) r dr}{r_{out}^2 - r_{in}^2} = 1, \quad i=1,2 \tag{8}$$

where r_{in} and r_{out} are the inner and outer fuel radii. The shape functions take into account the resonance absorption in ^{238}U and ^{240}Pu that lead to the formation of ^{239}Pu and ^{241}Pu, respectively. These distribution functions can be interpreted as the combination of a constant production of $^{239,241}Pu$ from thermal neutron capture plus a highly non-linear term for production due to resonance absorption: an example is given in Figure 4.

The *TUBRNP* model has been used to analyse the radial distribution of Pu in MOX fuels [13], an example is shown in Figure 5. It has also be used to determine the thickness of the so-called "rim-zone" [14], see Figure 6.

5.3 Verification by Comparison with Experiments

LWR conditions have been analysed by experiments from the following projects: Halden, Riso, Studsvik, Tribulation and others, FBR conditions were mainly tested within the CABRI project. Here, only an example from the FUMEX exercise organised by IAEA [15] shall be given. Blind predictions were made for 10 complicated irradiations performed in the Halden reactor. Figure 7 shows the comparison between predicted and measured temperatures for different burnup levels, gas compositions etc. The agreement is good although the tendency to overpredict high temperatures needs further clarification.

6 Conclusions

The following conclusions can be drawn:

1. The TRANSURANUS code is a quasi two-dimensional (11/2-D) code designed for the treatment of a whole fuel rod for any type of reactor and any situation. The fuel rods found in the majority of test or power reactors can be analysed for very different situations, as given for instance in an experiment, under normal, off-normal and under accident conditions. The time scale of the problems to be treated may range from milliseconds to years.
2. The TRANSURANUS code consists of a clearly defined mechanical/mathematical framework into which physical models can easily be incorporated. This framework has been extensively tested and the programming very clearly reflects this structure. The code is well structured and therefore easy to understand.
3. The code has a comprehensive material data bank for oxide, mixed oxide, carbide and nitride fuels, Zircaloy and steel claddings and different coolants. The subroutines which include the material properties are of identical structure and the incorporation of new data is straightforward.
4. The code can be employed in different versions, as a deterministic and a statistical code. It is evident that the relevance of statistical analysis is increasing: this is an area for future growth.
5. For the user the following practical aspects are of relevance:
   - The implementation on any computer is simple since the TRANSURANUS code is written in standard FORTRAN 77.

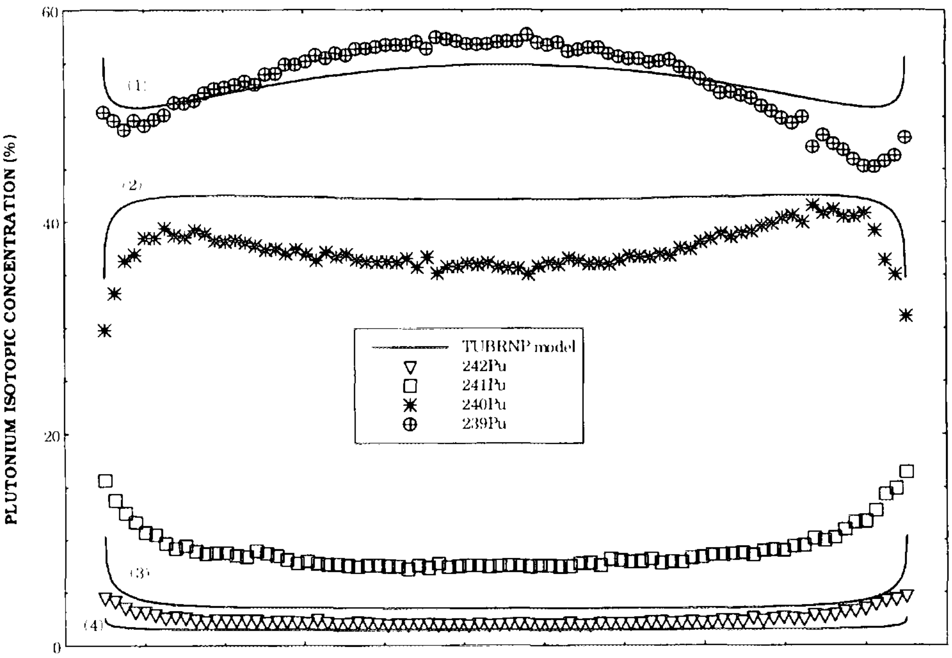

Figure 5 The isotopic plutonium composition plotted as a function of fuel radius as determined by SIMS (PWR) and predicted by *TUBRNP*. The curves labelled 1 to 4 correspond to the predictions for isotopes 239 to 242.

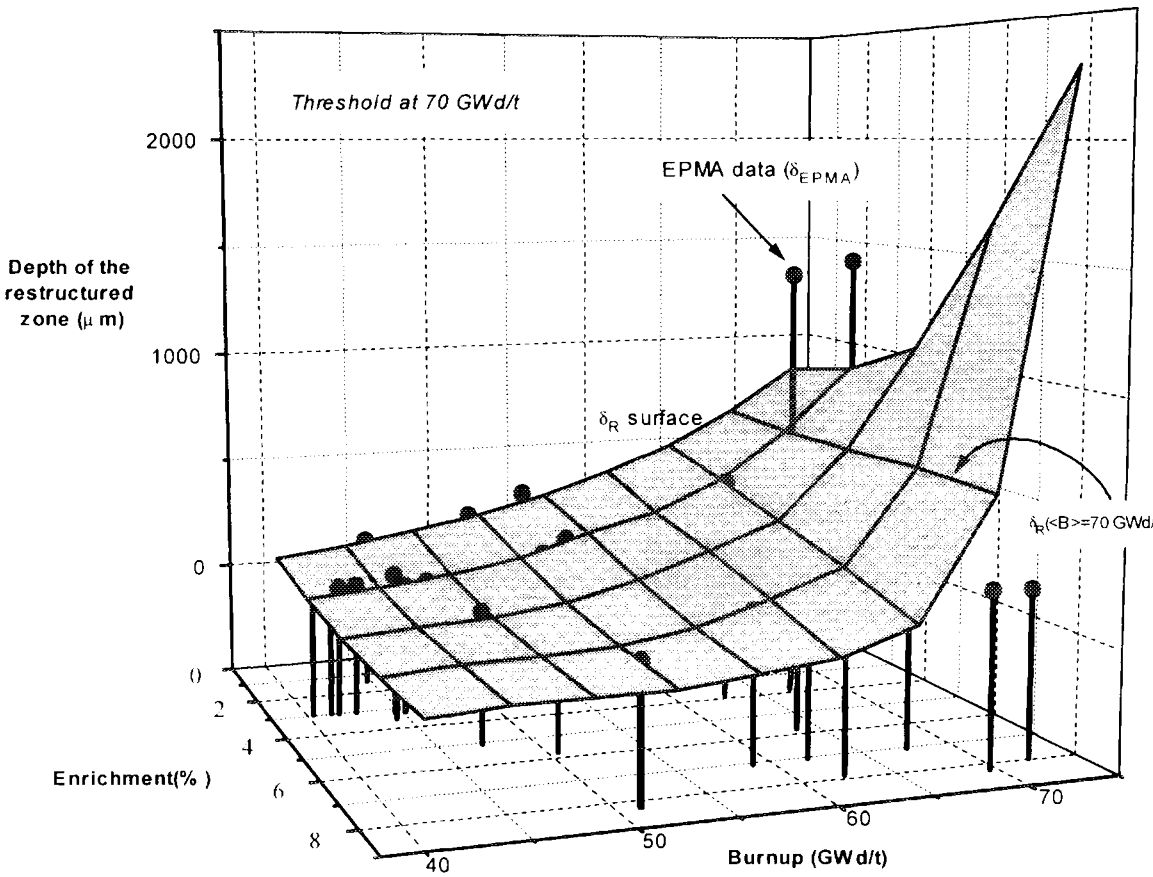

Figure 6 The depth of the restructured zone as a function of enrichment and average burnup. The surface is the prediction (δ_R) of the *TUBRNP* model with a threshold value of τ_b =70 GWd/t. The points are the values of the depth of this zone derived from the Xe profiles (δ_{EPMA}).

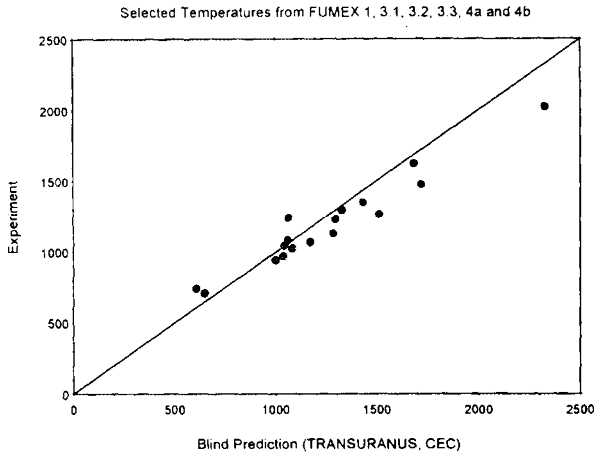

Figure 7 Comparison between blind temperature predictions employing the TRANSURANUS code (CEC version) with measured data of the FUMEX exercise [15]

- The TRANSURANUS code system consists of 2 preprocessor programms (MAKROH and AXORDER) which largely helps in setting up new data cases. A new Windows-based interactive interface is under development. One postprocessor, the very detailed plot program URPLOT, enables the user to plot all important quantities as function of the radius, the axial coordinate or the time. The postprocessor URSTAT may be used to evaluate statistical analyses.
- The TRANSURANUS code exhibits short running times. For example a simple LWR analysis of a whole fuel rod takes about 20 *s* CPU time on a modern workstation. A data case with a very complicated power history and detailed discretization may take approximately 2 minutes. It is important to note that all techniques were chosen in such a way that the computer costs depend more or less linearly on the discretization.

The TRANSURANUS code is now in use in various institutions through Europe and is now available to all interested parties.

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Overview of Fuel Testing Capabilities at the OECD Halden Reactor Project

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1 Introduction

The OECD Halden Reactor Project was established in 1958 as a joint undertaking of the OECD Nuclear Energy Agency through an agreement between nuclear centers of OECD countries sponsoring an experimental research program with the Halden Boiling Water Reactor (HBWR), which is owned and operated by the Institutt for Energiteknikk, Norway. Following the first agreement, eleven more have been entered into, the present one ending in December 1996. Some joint program and bilateral test plans reach well beyond that date already now and provide the starting point for further agreements. The reactor, which is continuously renewed and technically upgraded, has a life-time to beyond 2105, making it a reliable facility for execution of the most long-ranging test plans.

The early research programs were aimed at demonstrating the operability of HBWR type reactors and included extensive physics and dynamic studies. Then the emphasis gradually shifted to performance investigations on light water reactor fuels and materials and the development of computerized supervision and control systems. The main areas of activity defined in the present three years program from 1994-1996 are:

- high burnup fuel performance, safety and reliability;
- degradation of in-core materials and water chemistry effects;
- man-machine system research.

From the beginning, fuel performance and reliability investigations were supported by the development and perfection of in-core rod instruments. The measurement capabilities are expanded through development of experimental rig and loop systems where reactor fuel and material can be tested under light water reactor conditions, including prototypic PWR and BWR water chemistries.

It is believed that experimental studies and modeling of VVER fuel behavior can benefit from experience with western LWR fuel. In keeping with the theme of the seminar, the paper therefore gives an overview of testing capabilities and applications mainly aimed at exploring mechanisms of fuel behavior, also as related to high burnup. Examples of fuel performance are taken from data provided by the Halden Project for the IAEA Co-ordinated Research Program FUMEX (Fuel Modeling at EXtended burnup, [1]).

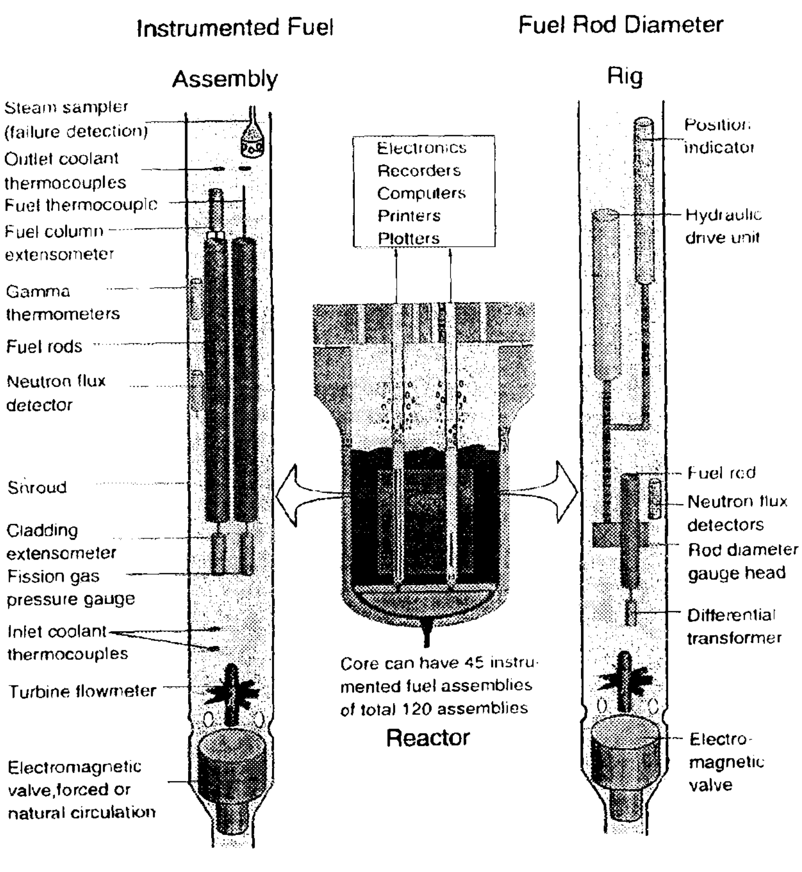

Figure 1 **Principal layout of HBWR with instumented fuel assemblies. HBWR: natural circulation boiling heavy water reactor with max. power of 25 MW; The ccolant water temper. is 240°C, corresponding to operating pressure of 33.6 bar.**

2 Instrumented Fuel Assemblies and Irradiation Rigs

An instrumented fuel assembly represents a test train in which one set of fuel rods can be tested, while irradiation rigs designate test hardware in which fuel rods are exchangeable. A number of heavily instrumented rigs to suit different test objectives have been developed. Some of them are illustrated in Fig. 1 and described below. Many other types of rig not further explained here have been utilized over the years. They are associated with testing of fuel from special reactors, e.g. advanced gas-cooled reactor, advanced thermal reactor. Also rigs for studies of LWR core component materials may be included in this category. In general, the Halden reactor provides versatile means to simulate conditions prevailing in planned and existing commercial nuclear power plants.

Base irradiation rig

As the name implies, this rig is used mainly for burnup accumulation prior to specific testing in other rigs. A control of rod power is provided through flow channel instrumentation (turbine, coolant thermocouples) and neutron flux measurements.

Gap meter rig

This rig allows the assessment of the fuel-to-cladding gap by squeezing the cladding onto the fuel, measuring force and deflection. Hot fuel expansion, relocation and swelling have been determined in this way in previous experimental periods.

Diameter measurement rig

In this rig, a diameter gauge can be moved along the fuel rod during operation. The change of cladding diameter in response to various modes of operation and burnup can be obtained with a micrometer precision.

Ramp rig

Ramps and overpower tests are effected in various ways: by moving rods from low to high flux positions during operation, using (movable) neutron absorption shields, depressuring coils with helium-3, or combinations of these. Also daily load following and frequency control modes of operation can be executed with these designs and have in fact been investigated.

Gas flow rig

This type of rig allows the exchange of fuel rod fill gas during operation. The gap thermal resistance and its influence on fuel temperature can thus be determined. It is also possible to analyze swept out fission products for assessment of fuel structure changes and fission gas release [2]. Further possibilities are indicated in section 4.4.

Instrumented fuel assembly

Designs differ largely with specific testing purposes and requirements. Rods can be heavily instrumented with thermocouples, pressure transducers and elongation detectors and stacked in several clusters. An example of a possible combination is shown in Fig. 2, which schematically depicts the arrangement and instrumentation of test rods for comparative testing of VVER fuel. The start of this experiment is foreseen for 1995 and the results will be available for the joint program of the Halden Project. The data are expected to provide information on fission gas release, thermal and mechanical behavior.

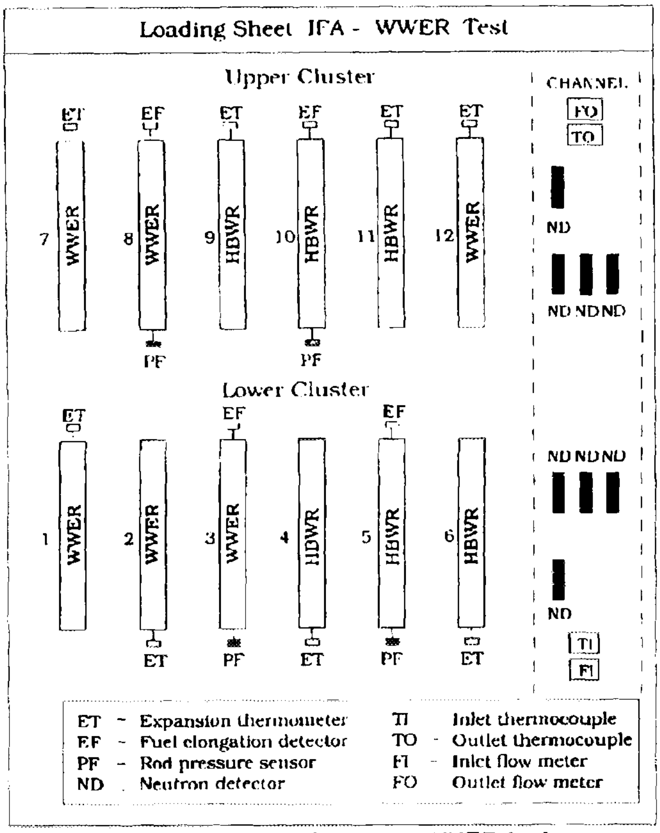

Figure 2 Layout of comparative test of VVER fuel. The instrumentation will provide data on thermal performance, fission gas release and fuel dimensional behaviour (swelling, densification)

3 Instrumentation Capabilities

While PIE ascertains the state existing at the end of irradiation, in-core instrumentation combined with expedient experimental techniques and test designs can give information on when and how phenomena occurred for the entire in-core service. Trends developing over several years, slow changes occurring on a scale of days or weeks and transients from seconds to some hours can be captured by the same instrumentation, altogether making up the total picture of fuel behavior.

The Halden Project has more than thirty years of experience in performing in-core measurements and a variety of sensors have been developed for this purpose. Most of them are designed and produced in-house, while a few - such as thermocouples and self-powered neutron detectors - are acquired commercially. They are typically applied in fuel performance investigations to obtain information on:

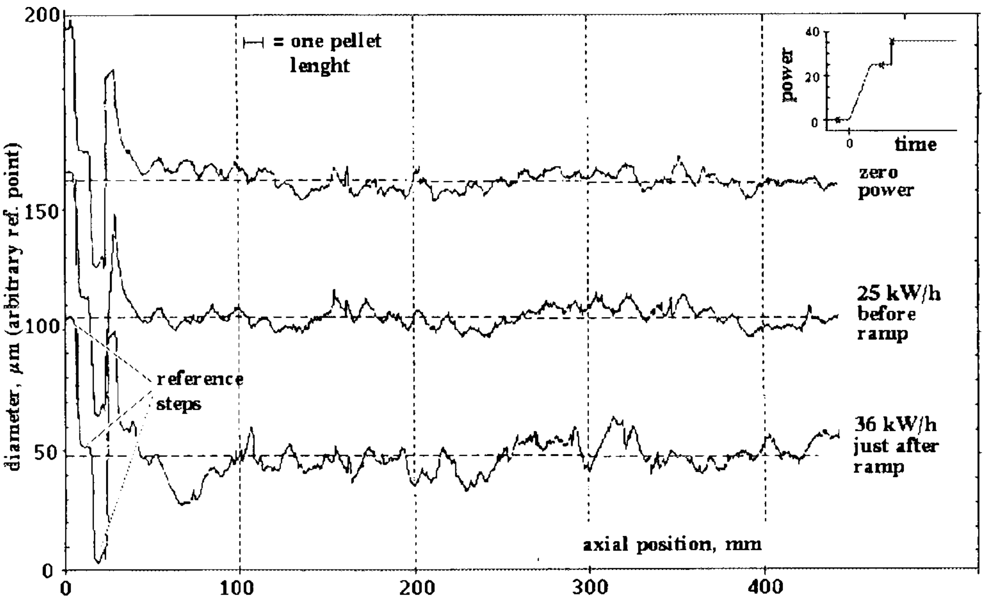

Figure 3 Diameter traces obtained during ramp testing (FUMEX-5). Cladding diameter changes can be measured with high precision relative to calibration steps in the end plug (left side of figure)

- fuel centre temperature and thermal property changes as function of burnup;
- fission gas release as function of power, operational mode and burnup;
- fuel swelling as affected by solid and gaseous fission products;
- pellet-cladding interaction manifested by axial and diametral deformations.

The following instruments are utilized for this purpose:

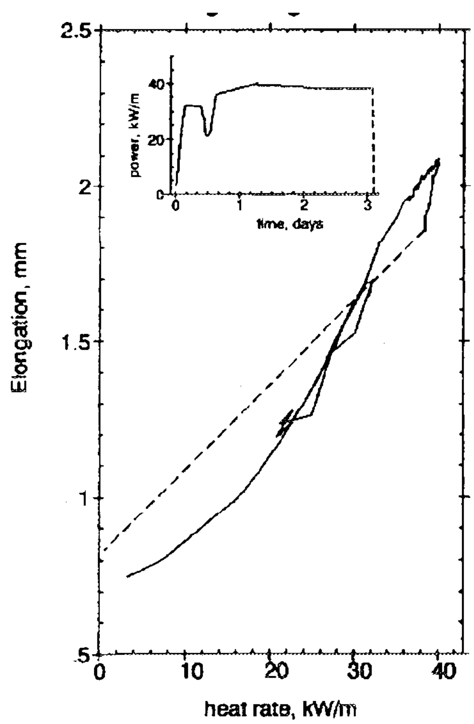

Figure 4 Cladding elongation measured during power ramp and following holding period. The data indicate relaxation at power and permanent length increase

- fuel thermocouples or expansion thermometers, which measure the fuel center temperature;
- bellows pressure transducers, which provide data on fission gas release by measuring the rod inner pressure;
- fuel stack elongation detectors, with which densification and swelling behavior can be assessed;
- cladding diameter gauge, with which radial deformation can be determined as function of power, holding time after ramping and burnup. Typical diameter traces are shown in Fig. 3 (FUMEX 5);
- cladding elongation detectors, which provide data on the onset and amount of pellet - cladding interaction, permanent deformation as well as relaxation capabilities of fuel and cladding as function of power and burnup (e.g. Fig. 4).

These instruments (with the exception of the fuel thermocouple and the diameter gauge) operate by means of a magnetic core inserted in a differential transformer (LVDT). Thus no cable penetration into the fuel rod is required and the fuel rods can be unloaded/reloaded without losing the measurement capabilities.

4 Rod Design and Irradiation Techniques

Investigations of fuel performance parameters, especially at high burnup, have to deal with a number of experimental problems, i.e. the time required for burnup accumulation, the demand of instrumentation to function reliably and the need for a separation of an increasing number of phenomena. The Halden Project has developed and applied techniques which make it possible to obtain reliable data for all relevant burnups, from beginning-of-life to ultra high exposure reaching 100 *MWd/kg* UO_2.

4.1 Design Features for Simulation of Burnup Effects

Increasing burnup in general incurs increasing uncertainties in data interpretation, e.g. fuel temperature changes can be effected by a combination of causes like fission gas release, changes in gap size (densification swelling, clad creep-down), and conductivity degradation. Test design with controlled and known influential parameters therefore facilitate the assessment of separate effects.

Fill Gas Composition

Fission gas release (an effect with considerable spread in model prediction) can be simulated by known additions of xenon to the fill gas. Fuel modeling codes must be able to predict the resulting change of the temperature-power relationship in a satisfactory manner. An example of the difference in fuel temperature caused by 8% and 23% xenon compared to pure helium fill gas is shown in Fig. 5 (FUMEX-4).

Small initial gap

Xenon fill gas in combination with a small as-fabricated gap (50 - 100 *μm*) simulates a high burnup situation (large amount of released fission gas, closed gap due to fuel swelling and clad creep-down) which is quite important for safety assessments. Gap conductance models cam be verified with measured data from tests with such a design without the uncertainties caused by high burnup. Examples of this can be found in the FUMEX-3 data set [1].

4.2 Fast Burnup Accumulation

Burnup representative of extended fuel cycle discharge values can be accumulated in an accelerated manner (about 20 *MWd/kg* UO_2 per year) by using highly enriched fuel and special geometries.

Small diameter rods

Thin rods (ca. 6 *mm* pellet diameter) are operated at a high fission rate per unit volume. Linear heat ratings and temperature profiles typical of LWR fuel can thus be obtained. Several such experiments are presently under irradiation in the HBWR. One has reached about 80 *MWd/kg* UO_2; its purpose is the investigation of fuel conductivity degradation and fission gas release at high burnup. Some results from the earlier stage of irradiation (up to 50 *MWd/kg* UO_2) were presented in [3].

Isothermal disks

This technique consist of an alternate series of fuel and molybdenum disks. The latter functions as an effective sink for the heat from the fuel disks. With a suitable gap between cladding and molybdenum disk and a proper choice of fill gas, the disk temperature can be selected independent of the specific power. In this way, it is possible to operate at very high fission density (fast burnup accumulation) while keeping the fuel temperature at low or moderate levels. The technique provides near isothermal conditions in the fuel disks which are very suitable as PIE specimens.

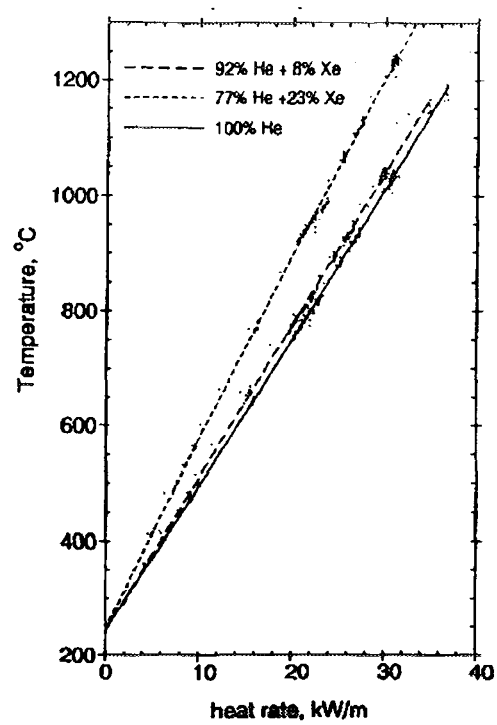

Figure 5 Fuel temperatures obtained with different fill gaz compositions, simulating the effect of fission gas release

4.3 Re-Instrumentation of Irradiated Segments

Re-instrumentation of fuel segments previously irradiated to high burnup has been used for several experiments. The sensors comprise pressure transducers, fuel thermocouples and cladding elongation detectors. Re-instrumentation with the latter is especially simple since it only requires the external fitting of a magnetic core to the segment, while the other two involve more elaborate techniques:

Pressure transducers

An instrumental head containing a bellows pressure transducer and a drill is welded onto the end plug of an irradiated rod. The reaction of the bellows to pressure changes in the fuel rod is picked up during further irradiation with an LVDT (linear variable differential transformer). Continuous measurements of the internal gas pressure give precise information about the fission gas release as a function of time and burnup. This is shown in Fig. 6 (FUMEX-6) where the onset of fission gas release is clearly detected.

Fuel thermocouples

Re-instrumentation of irradiated fuel rods with fuel thermocouples is part of the current Halden Project program. The technique, based on Riso experience, is fully implemented in the hot cell if Institutt for Energiteknikk at Kjeller, Norway and is applied to joint program and bilateral tests.

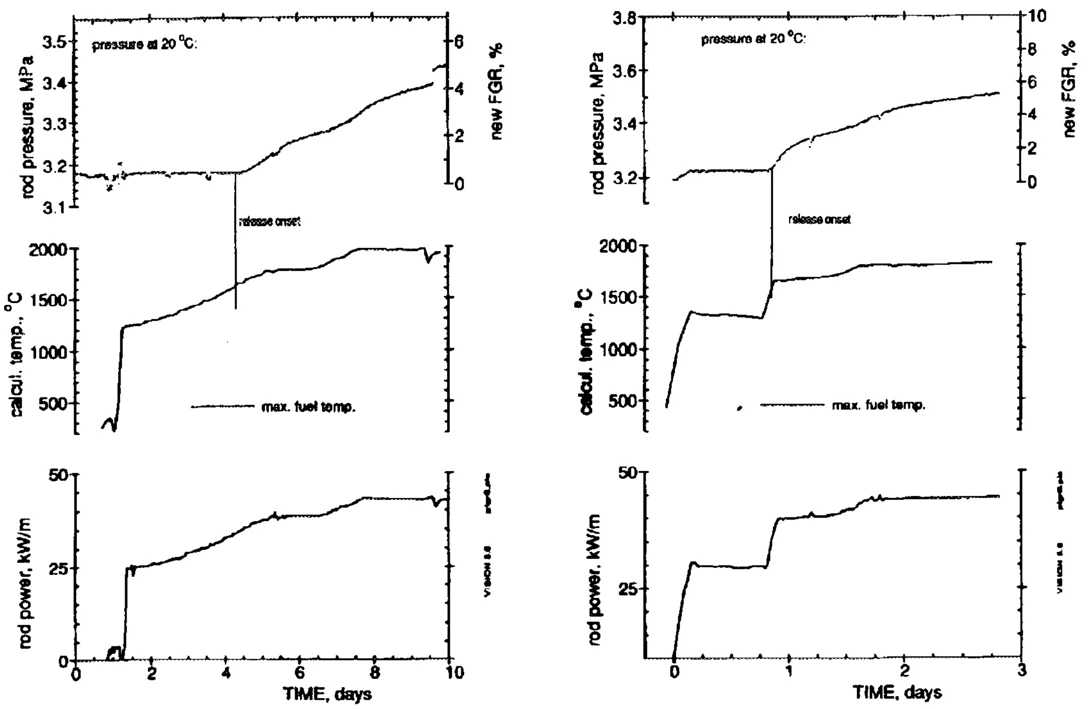

Figure 6 Examples of pressure response (fission gas release) to slow and fast power ramps (FUMEX-6). The rods were re-instrumented with pressure transducers after base irradiation. The onset of fission gas release is clearly detected and occurs at about the same power in both cases.

4.4 Experiments to Study Separate Effects

Most phenomena occurring in fuel rods during irradiation are interrelated, in particular via the influence of temperature. The dependencies tend to become more complex with burnup, while uncertainties increase. The possibility to study some effects separately or in relative isolation is therefore of great advantage for fuel performance modeling. Of particular relevance for thermal properties and fission gas release investigations are the Halden Project's gas flow rigs. In these rigs, the gas contained in the fuel rods can be changed and the pressure can be varied. Volatile fission products can be swept out, cold trapped and analyzed. With this type of rig, a number of parameters can be addressed:

Gap size

The hydraulic diameter or gap can be determined from the resistance that the fuel stack offers against the gas flow driven by a given differential pressure between the ends of a rod. Gradual gap closure with increasing burnup has been verified by this method.

Effect of pressure on temperature

The pressure and gas-type dependent temperature jump distance is an important parameter in fuel temperature calculation models. Variation of the fill gas pressure at constant power has an influence on the fuel centre temperature measured simultaneously. This effect (decreasing temperature with increasing pressure) is more pronounced for helium than for argon or xenon and increases with burnup - probably due to structural changes which create a growing of surfaces where the effect can take place.

Gas composition in the fuel-cladding gap

By changing the gas in a rod, the separate effects of several parameters can be investigated. For fission gas release, the influence of temperature can be assessed while keeping the power density constant, thus separating thermal and athermal release. It is also possible to determine the influences of fuel vs. gap heat conductance since mostly the latter is changed when replacing the gas filling, e.g. helium with argon.

Sweeping out and analysis of fission products

Volatile fission products swept out with the purging cover gas can be cold-trapped and analyzed using on-line gamma-ray spectrometry. The method allows the determination of release-to-birth ratios which are controlled by thermal and athermal diffusion processes. By analyzing the release rates of several short-lived isotopes it is possible to assess the surface-to-volume ratio which is a sensitive indicator of grain boundary porosity interlinkage and resintering [2].

4.5 Noise Analysis and Utilization of Transient Data

The technique of noise analysis is being used, in particular in conjunction with fuel thermal data, to monitor long-term changes which have an influence on fuel temperatures. These are essentially gap closure, fission gas release and conductivity changes which influence the time constant of the fuel. Also transient temperature data, obtained with a recording system automatically activated during scrams, can be used to infer response (time constant) changes with burnup. These techniques have the advantage of being independent of the exact power and temperature. They provide supplementary information to steady state analyses and usually corroborate the results obtained therefrom. An example of scram data evaluation is shown in Fig. 7 (FUMEX-4). The time constant changed at 17 *Mwd/kg* UO_2 in response to fission gas release caused by operation at higher powers and temperatures than before.

5 Summary

Fuel performance and reliability investigations at the OECD Halden Reactor Project are supported by a variety of irradiation rigs, suitable irradiation techniques and a range of instrumentation. They have been perfected and applied over the years to assess separate effects as well as integral fuel behavior. Rod instrumentation (fuel thermocouples, bellows pressure transducer, cladding and fuel elongation detector) can also be attached to segments pre-irradiated in commercial LWRs, especially for high burnup behavior and fuel failure studies.

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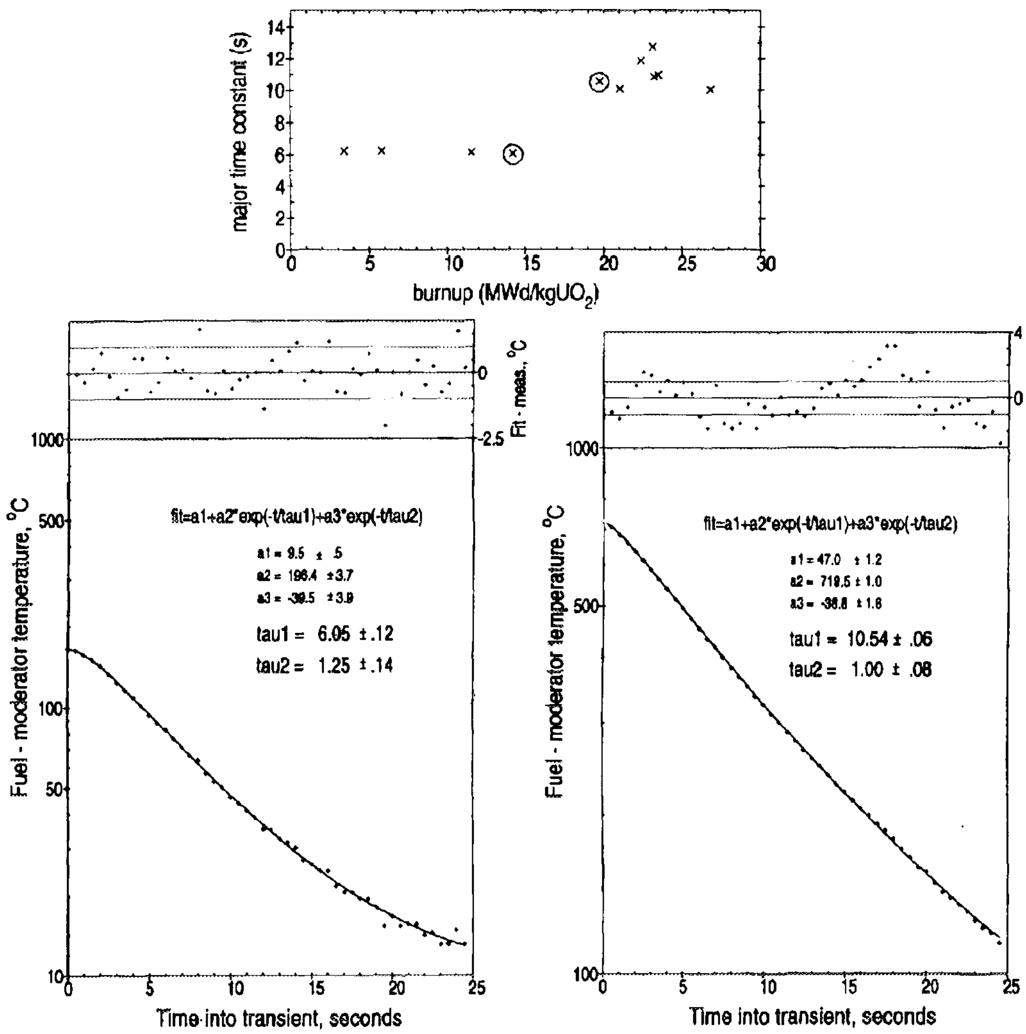

Figure 7 **Transient fuel temperature response after reactor scram. The fuel time constant is a measure of stored heat dissipation capability. Fission gas release occured during operation following the scram at 14.5 MWd/kg, and the time constant increased.**

BG9600365

Assesment of VVER Fuel Condition in Design Basis Accident

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This paper gives a review of design and experimental studies of fuel behaviour in design basis accidents relevant to a fuel cladding temperature rise.

The VVER fuel conditions is assessed based on calculations by the RAPTA code developed at ARSRIIM. The first version of the code appeared in late 70-ies [1]. Nowadays the fifth version of the code, specifically RAPTA-5, has been developed. Compared to the preceding version [2] the code comprises new models: non-axisymmetrical local ballooning of cladding, cladding straining within a fuel bundle taking account of mutual effects. The models of cladding straining and rupture, cladding material oxidation and oth. have been backfitted with due account for new experimental data.

To prepare a new version of the code the newly developed and backfitted models were tested using experimental data. For instance, the serviceability of the calculation of the process of non-isothermal high temperature oxidation of Zr-1%Nb alloy was determined by comparing with the relevant experimental data [3]. As a whole the codes verified by comparing with the results of experiments with fuel bundles as well as with design data obtained using other codes.

RAPTA-5 code performs interrelated thermo-mechanical calculations of a fuel conditions taking account of high temperature interactions. The results of code calculations are:

- temperature field of a fuel rod including assessments of power density due to chemical interactions;
- stress-strained condition of cladding including determination of location, time and strain of rupture;
- assessment of the extent of chemical interactions (cladding material oxidation, cladding-fuel interaction etc.);
- assessment of hydrogen release;
- assessment of cross section blockage.

The input data for the RAPTA code are results of calculations of neutron physics and thermal hydraulics parameters of the core: power density of fuel, coolant condition (composition, pressure, flow rate), etc. The calculations of this kind are executed by the chief designer of a reactor unit, i.e., OKB Hydropress.

The main result of the calculations of fuel rod condition in design basis accidents is the check-up of the ability to meet the criteria for the emergency core cooling system acceptance:

- not to exceed the temperature of UO_2 melting down at any point in the core;
- not to exceed the fuel cladding temperature of 1200°C;
- local depth of fuel cladding oxidation must not exceed 18% of the initial wall thickness;
- a fraction of reacted Zr must not be higher than 1% of its mass in fuel claddings. There are also requirements placed on the maximum blockage of a core;
- the feasibility of discharging a core and its components after a design basis accident. For accidents relevant to a quick reactivity increase the specific threshold energy of fuel rupture must not be exceeded at any moment of life.

Comparison between the above criteria in terms of fuel for VVER type reactors with those for PWR's demonstrates the full coincidence in the set of the criteria. In many cases also quantitative parameters agree.

The process describing models designed for the RAPTA-5 code are based on dependences and constants derived experimentally using commercially produced materials, fuel rod simulators, fuel bundles. The major processes are

- cladding material straining and rupture;
- high temperature interaction of a cladding material with steam, a spacer grid material (stainless steel) and UO_2 (in case of a contact);
- UO_2 - steam interaction have been studied in a wide range of temperatures and rates of temperature - force loading.

The unique method (continuous weighting of a specimen) used for studying the oxidation of a cladding material [4, 5] revealed same specific features. For instance, upon rather long-term holding the "weight gain - time" curves show a transition. The metallographic analysis confirmed the relationship between accelerated oxidation and oxide film separation. The ranges of this effect in terms of temperature - time have been revealed.

Another specific feature of oxidation is relevant to the process of oxidation during cladding material straining [6]. Based on the generated experimental data to be used for design calculations, a set of conservative dependences was recommended for the description of oxidation kinetics. For the temperature range of 900 - 1200°C and the interaction time up to 900 *sec* the following dependence is a conservative one:

$$\Delta m/A = 920 \cdot \exp(-10410 / T)\ t^{1/2} \qquad (1)$$

where: $\Delta m/A$ - is the specific weight gain, *mg/cm²*, T is temperature in K, and t - time, *sec*.

Eq. (1) is an analogue of Baker-Just dependence recommended for PWR cores with fuel cladding in Zry-4. It is to be noted that Eq. (1) and other relations that enter into the set of conservative dependences for VVER cores give an assessment of oxidation with a margin also in case of fuel cladding straining.

The strain property of a cladding material were studied both in initial tension and in experiments with cladding and fuel rod simulators. The investigations performed with iodine containing environment in a cladding (the concentration of 1 mg/cm^3) show that the effect of iodine takes place up to 700°C. The availability of iodine lowers the rupture strain of cladding.

Cladding-spacer grid material interaction was studied using diffusion pairs in vacuum as well as cladding and spacer grid simulators in steam. It is shown that steam available in a system has a significant effect on the minimum temperature of eutectics formation [7].

UO_2-steam interaction depends significantly on oxygen potential and steam composition. At the standard composition of steam and temperatures typical of design basis accident UO_2 does not essentially react with steam.

In the problem relevant to properties of fuel and structural materials a special position is taken by the mechanical properties of high temperature oxidized fuel claddings, since this issue is closely related to the margins in the acceptance criterion in terms of cladding material oxidation. Recently papers have appeared [8, 9] that question the validity of the criterion. Tests were conducted for compression of oxidized cladding section in the direction normal to the axis of the specimen symmetry. Similar results were published by us earlier [10]. However, these results cannot be related to the needed experimental validation of the criterion, since, first, the experiments were performed at room temperature. Second, this does not comply with actual scenarios of design basis accidents, and as has been mentioned in [11] the chosen type of cladding loading does not comply with the loads experienced by fuel rods in accidents and subsequent fuel handling, shipping included.

The experiments carried out in Russia using oxidized specimens (cladding sections, cylindrical microspecimens, rings) demonstrated a significant increase of ductility with temperature.

Most dangerous for fuel rods during accidents are modes of core flooding with cold water. Fig. 1 shows the results of testing for thermal resistance of fuel simulators. The vertical line is a boundary of an allowable region of oxidation, namely 1200°C; the sloping line is a boundary of 18% oxidation of a cladding material. The experimental data shown by Fig. 1 pertain to cladding testing without axial limitation of relocation. The recent similar data on axially limited relocation indicate with in this case too there is the needed margin for oxidation. All these results were obtained for indirectly heated fuel claddings which is basic since at significant oxidation (more then 1%) experiments with directly heated claddings (by passing directly electrical current through a specimen) give large uncertainties in terms of temperatures.

Taking account of the commercial introduction of new Zr alloys as cladding materials in the IAEA framework recommendations are to be issued in terms of methodology support of determining the fuel acceptance criteria.

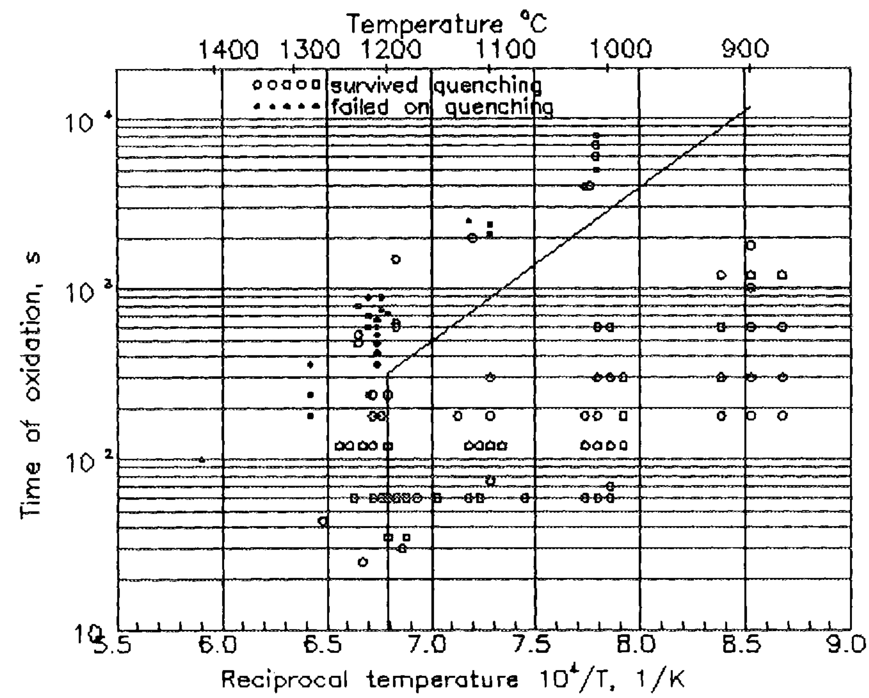

Figure 1 Testing of dummy fuels by thermal shock (cladding 9.1×7.74 mm, Zr1%Nb)

Conclusions

1. The fuel condition in design basis accidents is assessed on the basis of the verified code RAPTA-5.
2. The code manes use set of high temperature physico-chemical properties of the fuel components as determined for commercially produced materials, fuel rod simulators and fuel rod bundles.
3. The VVER fuel criteria available in Russia for design basis accidents do not generally differ from the similar criteria adopted for PWR's.

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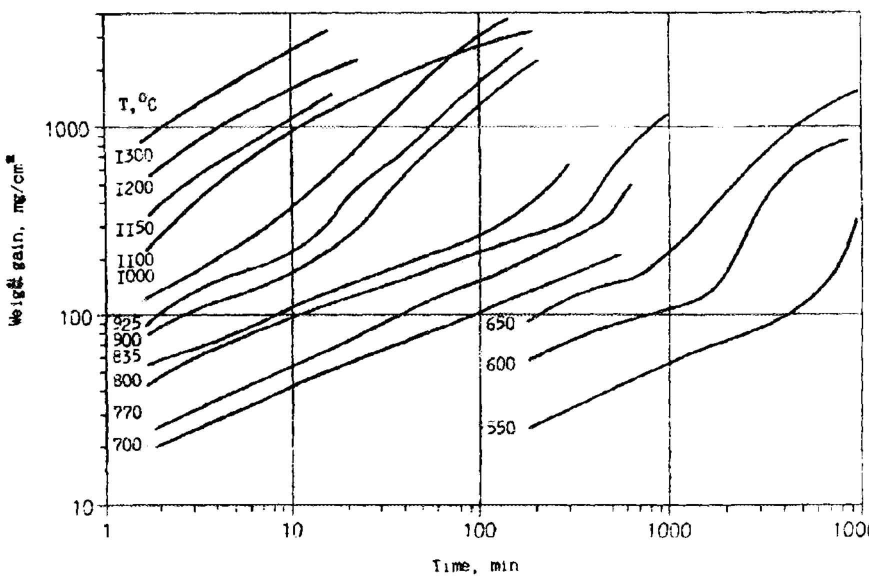
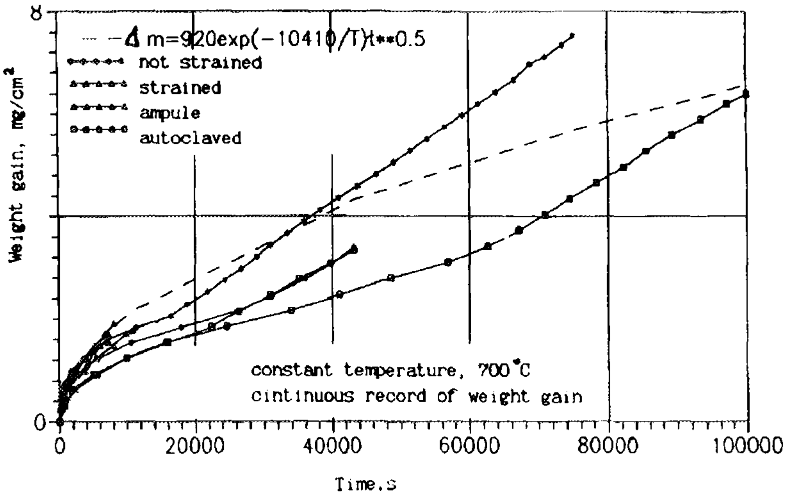
Note by the Publisher

Additional figures were presented by the authors at the Seminar, that were not mentioned in the text. As we consider them important for understanding of the discussed phenomena, these pictures are next included without specific order or numbering. Their captions, however, correspond to certain paragraphs in the text above.

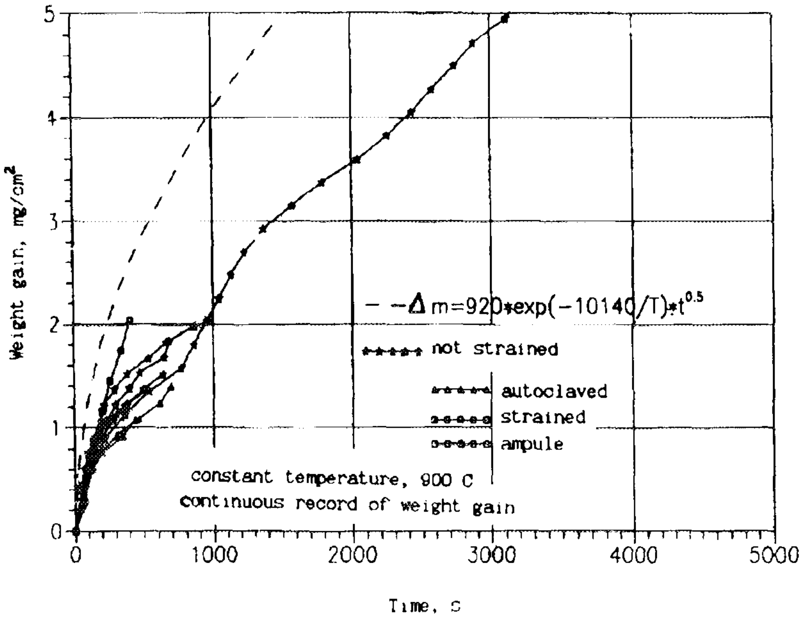


Weight gain of VVER-type claddings as oxidized in steam at athmospheric pressure

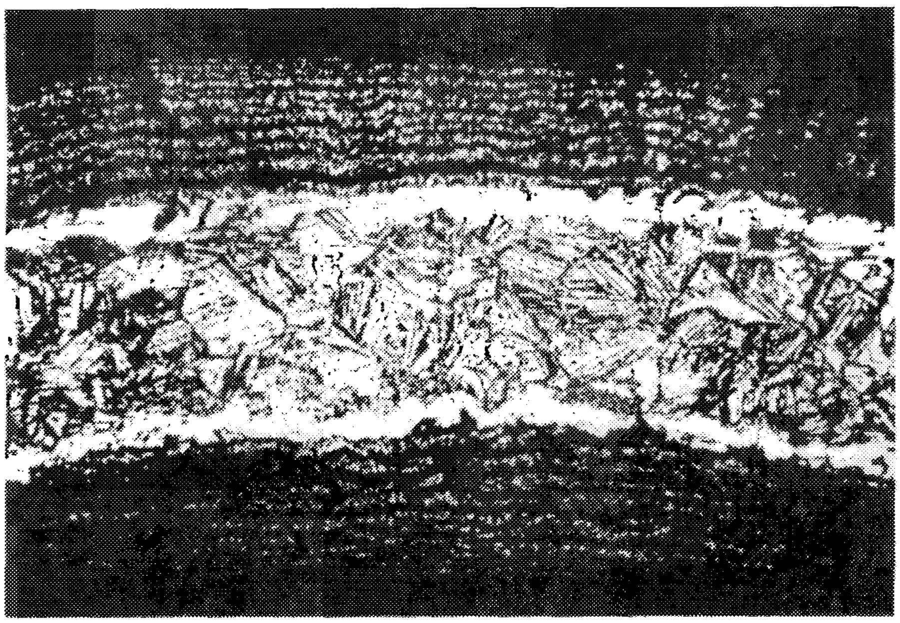
Weight gain of tubular samples 9.15×7.72 as oxidized in steam



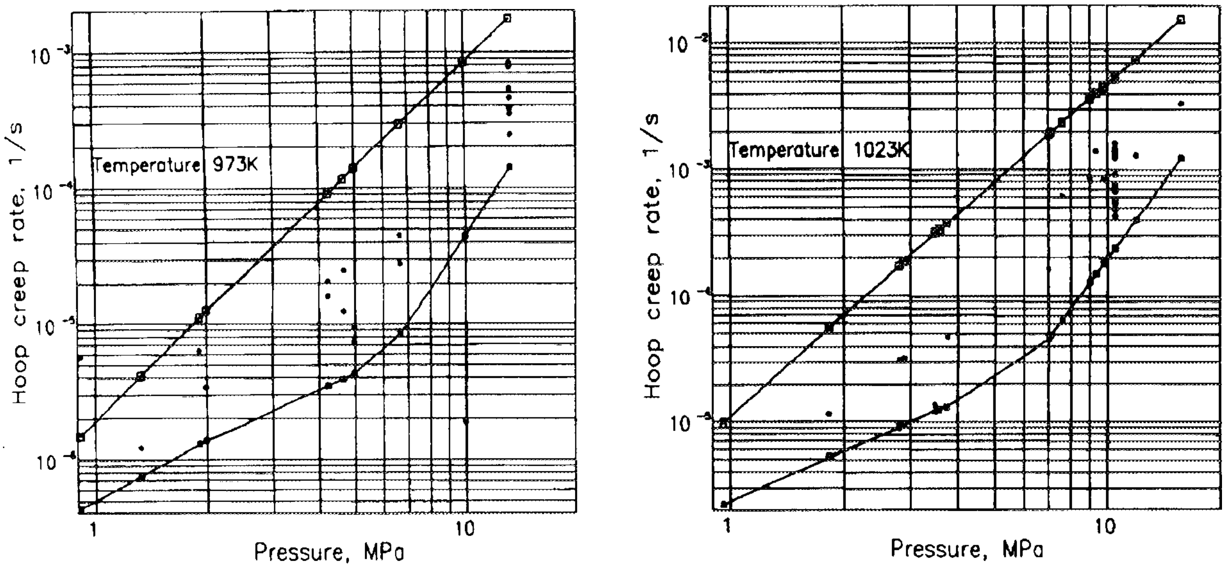
Weight gain of tubular samples 9.15×7.72 as oxidized in steam



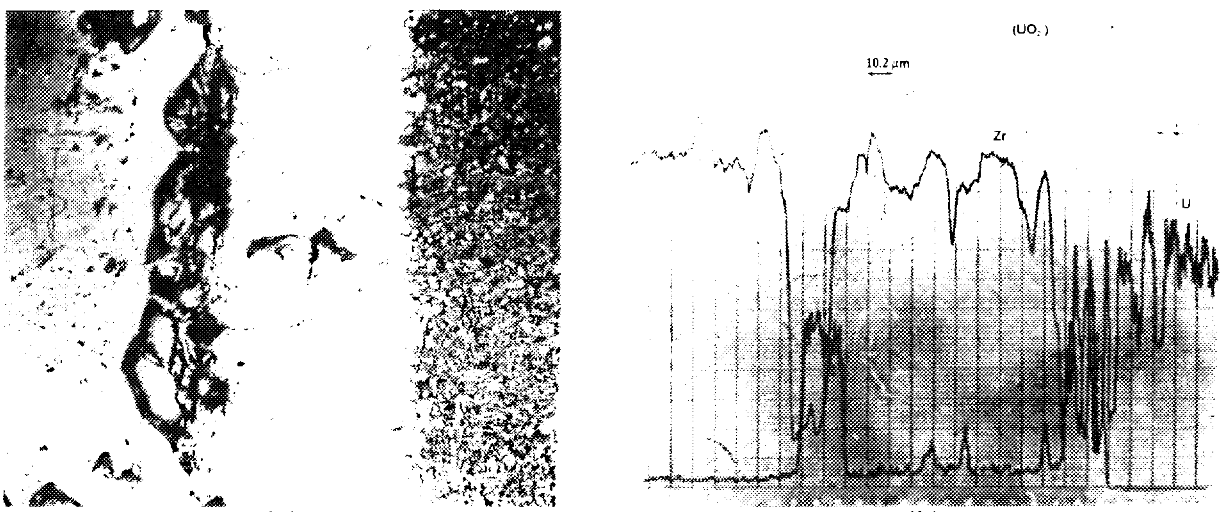
Microstructure of steam oxidized specimen



T=1000 °C, τ=8040 s

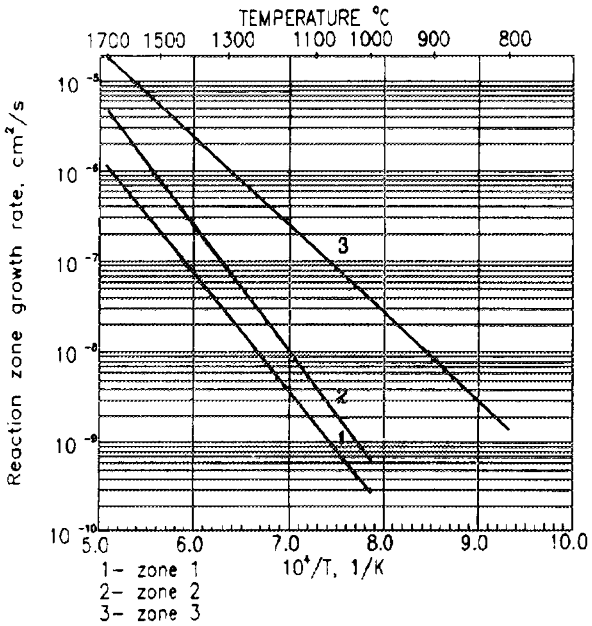


Dependence of creep rate on internal pressure (hot, undeformed clad):
••• experimental data; □□□ calculated (1983); ▪▪▪ calculated (1986).

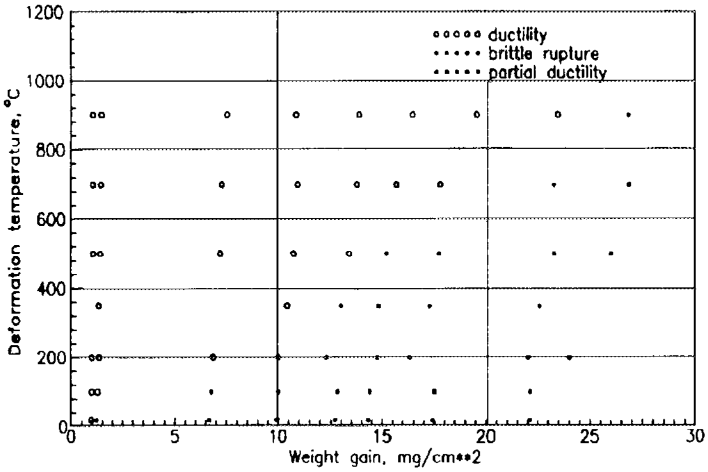


(a) (b)

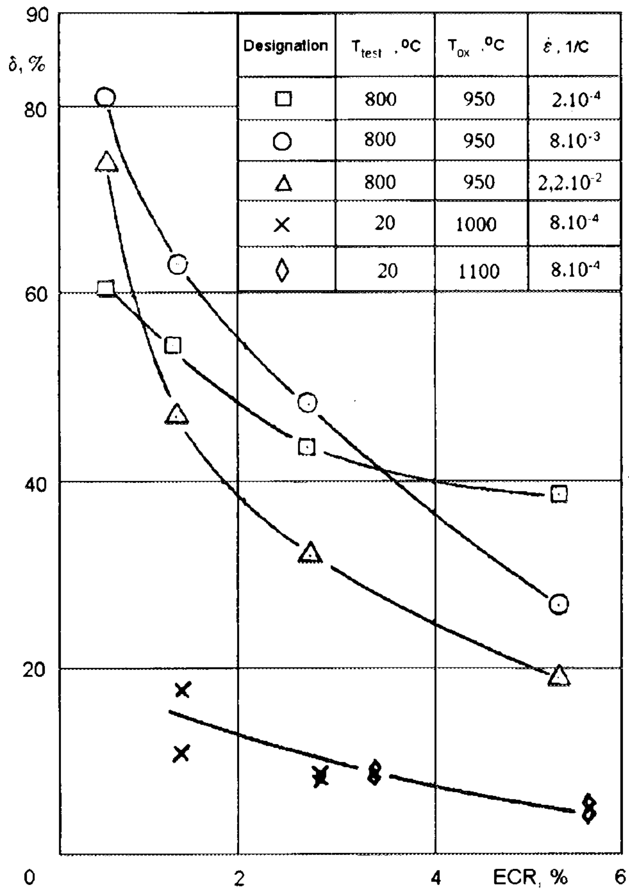
Reaction zone of Zr1%Nb / UO_2 couple (1500°C, 15 min, 300×) - (a);
Zr and U contents (b).



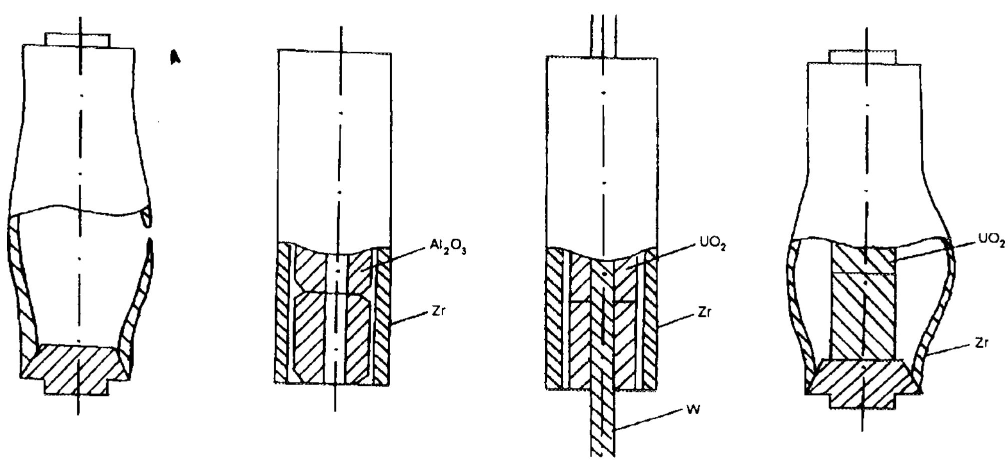
Reaction zone growth rates of Zr1%nb/UO_2 couple



Ductility of steam-reacted Zr1%Nb claddings. Ring compression results.



Per cent elongation in active tension of cylindrical microspecimens



Schematics of dummy fuel rods (see also Fig. 1 in the text).

BG9600366

Influence of Fuel-Cladding System Deviations from the Model of Continuous Cylinders on the Parameters of VVER Fuel Element Working Ability

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In the programs of fuel rod computation, which are used to validate its working ability during the normal operation, fuel and cladding are usually presented in the form of coaxial cylinders, which can change their sizes, mechanical and thermal-physical properties. But some deviations from this scheme are typical to the real fuel element: axial asymmetry of the fuel-cladding system (due to the oval form of the cladding, cracking and other types of the fuel pellet damage, axial asymmetry of the volumetric heat release), gaps between the pellets (and heat release peaking in the fuel near the gag), chamfers in the pellets.

As a result of these deviations actual fuel rod parameters of working ability (temperature, stresses, thermal fluxes relieved from the cladding, geometry changes) in some locations can greatly vary from the ones calculated due to continuous coaxial cylinders (CCC) model. The influence of these deviations is extremely importane, because they are a part (or are to be a part) of mechanical excess coefficient, which is very important while calculating the fuel rod.

Lets review the influence of these factors using specific examples.

The calculations were carruid out by the developed by the author two-dimensional codes (x-y and r-z), based on the Finite Elements Method (FEM) for calculation of temperature fields, stressess, deformations in the elements of fuel rods.

It was considered that:

- the properties of fuel, cladding and gas media inside the cladding are typical for VVER fuel rods;
- coolant temperature at the reviewed location is 600°C;
- cladding-coolant heat removal coefficient is constant and equal to $1.5{\cdot}10^4$ *W/m²*.

For more clear results, the influence of the relative power density reduction along the fuel radius was not considered.

The initial stage of operation is reviewed. It is considered that there is no fuel cracking, creep deformation is not taken into account.

Let's consider that the sum of the parameter of roughness of fuel and cladding

$B=C\,(R_{a1}+R_{a2})$

and temperature jump length is equal to 8 *μm*.

1 Axial Asymmetry

Let s review the VVER fuel rod:

$R_{co}=4.6$ *mm*, $R_{ci}=3.9$ *mm*,

$R_{po}=3.8$ *mm*, $R_{pi}=1.15$ *mm*

at the initial temperature 300 *K*. Here and later: R is radius; c and p indexes mean that it either belong to "cladding" or to "pellet", respectively; o and i indexes denote outer or inner surface.

Let's assume that the volumetric density of heat release $q_v=1.109$ *W/m³* (at 300 K), that means that linear thermal load $q_l=412$ *W/cm*.

1.1 Excentricity of Fuel and Cladding

Excenticity of fuel and cladding is $e=69$ *μm* (after thermal expansion fuel touches the cladding at the angle of $\varphi=0$ - see Fig. 1).

Table 1 lists the calculated values of temperature T and shifts d.

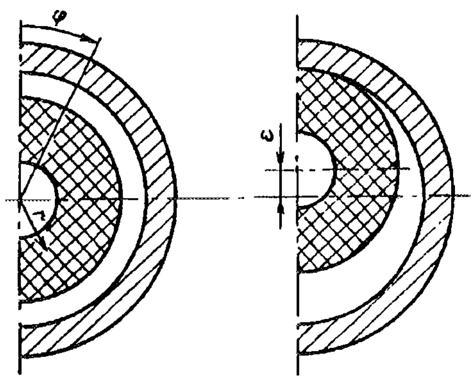

Figure 1 Fuel-cladding excentricity

Table 1 Fuel-cladding excentricity

Parameter	$e=0$	$e=69\,\mu m$		
	for all φ	$\varphi=0$	$\varphi=\pi/2$	$\varphi=\pi$
T_l, K	1545		1514	
T_{pi}, K	1961	1729	1923	2099
T_{po}, K	1073	768	1065	1322
T_{ci}, K	692.4	708	690	681
T_{co}, K	640.6	647.6	639.7	635.6
d_{po}, μm	53	39	51	63
d_{ci}, μm	8.2	8.4	8.0	7.9
d_{co}, μm	9.5	9.8	9.4	9.3

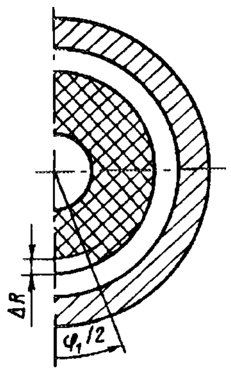

Figure 2
Fuel pellet spall

Table 2 **Spall of fuel pellet** ($\varphi_1=\pi/14$; r=0.19 μm; T_f =1554 K).

Temperature	φ				
	0	5π/7	25π/28	27π/28	π
T_{pi}, K	1962	1971	1976	1977	1978
T_{po}, K	1072	1073	1079	1042	- (gas)
T_{ci}, K	692.4	693.1	692.2	684.3	676.1
T_{co}, K	640.6	640.9	640.3	637.0	634.8

Table 3 **Ovality of cladding** w = 62.5 μm (T_f =1497 K)

Temperature	φ = 0	φ = π/2
T_{pi}, K	1962	1853
T_{po}, K	1261	781
T_{ci}, K	674	721
T_{co}, K	633	652
d_{po}, μm	54.1	46.5
d_{ci}, μm	8.75	7.4

1.2 Spalls of Fuel Pellets

Lets discuss the case $\varphi_1=\pi/14$, when Δr=0.19 *mm* (q_1 = 410.5 *W/cm*) - see Fig. 2. The calculation results are listed in Table 2.

1.3 Ovality of the Cladding

Lets specify the ovality in the form

$R_{ocio,ij}=R_{cio,ij}+W\cos2\varphi$

If W=61.5 μm (after thermal expansion fuel touches the cladding at $\varphi=\pi/2$). The results of calculations are listed in Table 3.

1.4 Axial Asymmetry of the Volumetric Density of Heat Release

Lets assume that $q_v = q_{vo}\,(1 + 0.1\cos\varphi)$.

In this case T_f=1544 K; T_{pi} is equal to 2023 K, when φ=0 and it is equal to 1895 *K*, when $\varphi=\pi$; T_{po} is equal to 1088 K, when φ=0 and to 1056 K, when $\varphi=\pi$.

We can make a conclusion [2], that axial asymmetry of the fuel-claddin system leads to slight reduction of the fuel average temperature T_f, when the average gap is preserved (cases 1.1, 1.3, 1.4). It is all due to the fact that redistribution of thermal flux takes place through the gap - the majority of heat is transferred through the part of the gap having higher thermal conductivity, than in the case with axial symmetry. It leads to reduction of maximum temperature T_{max} in the pellets without central hole (when q_1=412 *W/cm*: T_{max}=2244 K, T_f=1609 K when e=0; T_{max}=2225 K, T_f=1575 K when e=64 *cm*), the maximum fuel temperature increases in the pellets with central hole. The value of this increase depends on the linear load, fuel-cladding gap and radius of the central hole in the fuel [4].

In case the average gap becomes larger (1.2) the avrerage and maximum temperatures of the fuel become higher.

In all the cases with axial symmetry heat fluxes relieved from the cladding are redistributed (the intensity of the flux can be increased in some local part by 20%).

1.5 Influence of Axial Symmetry on the Pellet-Cladding Interaction (PCI) [6]

It is well known that the main mechanisms responsible for violation cladding tightness is PCI, which mainly occurs at the increase of linear load. The effect of interaction can be strengthened due to radial cracks within the pellets, which are the concentrators of tensile stresses, deformations and agressive environment.

Let's review the influence of volumetric heat release density axial asymmetry and radial crack in the fuel of the stressed-deformed state of the cladding. It was assumed that pellets and cladding are tightly keyed to each other and that they are deformed together without any sliding in the conditions of flat deformation of the cylinder with free ends. It was considered that at q_1:

- outer pressure (16 *MPa*) - from the part of the coolant and the inner one (5 *MPa*) - from the part of the gas inside the cladding influence the fuel-cladding system;
- there is no tension in the fuel;
- there is no fuel-cladding contact pressure;
- fuel has either no cracks or one radial crack, the shores of which are in contact with each other (at zero pressure). The depth of the crack is 0.58 *mm*;
- the fuel burnup is 30 *MWd/kg* (this was used to calculate heat release and power density along the fuel radius, as well as fuel-cladding gap conductivity).

In p. 1.5 we assumed that there took place abrupt increase of the linear load up to q_{12}. When reviewing the case for volumetric power density axial asymmetry, it was considered that

$$q_1 - q_v = \text{const}(\varphi), \text{ if } q_{12} : q_{va}\ (1 + 0.1\cos\varphi).$$

Lets review the example when the power is increased from $q_1 = 200$ *W/cm* ($T_{\max} = 914$ K, $T_f = 802$ K) to 300 *W/cm* ($T_f = 929$ K).

Table 4 Calculated temperatures and circular stresses at the inner and outer surfaces of the cladding:

1. Fuel-cladding axial symmetric system;
2. In the fuel there exists the crack of an angle coordinate $\varphi = \pi$, axial symmetric volumetric heat release;
3. No crack in the fuel, axial asymmetric volumetric heat release.

$q_1 = 200$ *W/cm*, $q_{12} = 300$ *W/cm*					
Case	1	2		3	
	for all φ	$\varphi=0$	$\varphi=\pi$	$\varphi=0$	$\varphi=\pi$
T_f, K	928.5				
T_{pi}, K T_{co}, K	1117 609.5	1117 609.5		1147 611.5	1087 607.5
S_i, *MPa* S_o, *MPa*	114 106	182 116	114 106	138 132	91 80

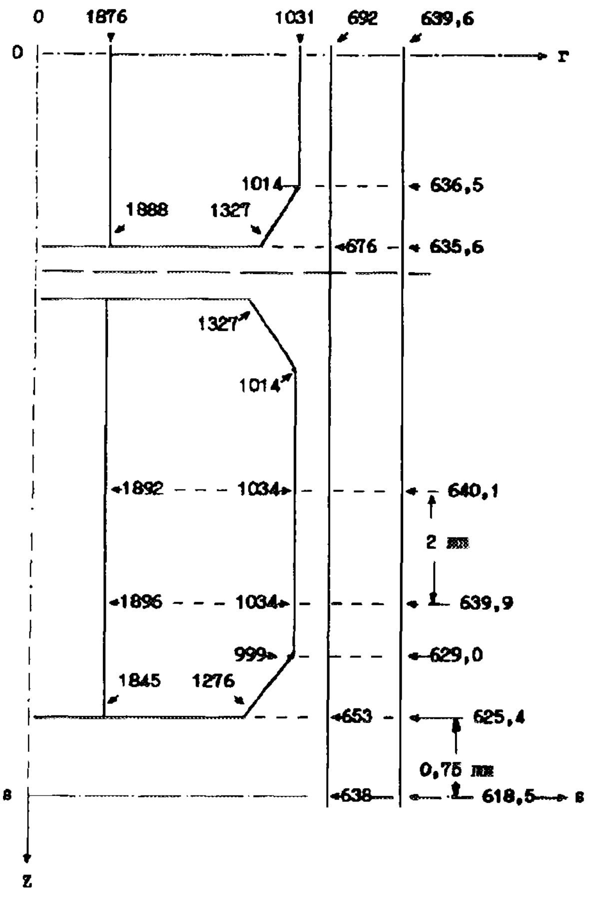

Figure 3 Calculated temperature of the fuel rod points, where the gap between two pellets is 1.5 mm. Symmetry axis is O-z; **symmetry surfaces are** 0-r **and** s-s.

The averaged (in respect to the intermediate environment) results of calculations are listed in Table 4.

We would like to note that here we do not discuss the shock tensile influence of fuel on the cladding, which is possible at cracking of the non-damaged at q_f fuel.

2 Deviations Within the Height of the Fuel Rod

It is evident that presence of spalls and chamfers on the pellets leads to the increase of the average gap of the fuel-cladding system, and thus to the increase of the maximum and average fuel temperature (for some 20 K when q_f =412 *W/cm*) and to skight change of heat fluxes relieved from the cladding [3, 5].

Lets discuss the influence of the gap between the pellets on the temperature field in the fuel rod. It is known that in the gap area there takes place local increase of the heat release density into the fuel of this fuel rod and into the neighboring ones. This is caused by moderation of the neutron spectrum due to the increse of the moderator fraction and decrase of the neutron absorption. The typical size of the area with increased heat release in the fuel of the fuel rod with the gap 1.5 *mm* is some 5 - 10 *mm* (calculated values used are presented by Dr. G. Bogachev): near the gap heat release density goes up by 6% at the distance of 8 *mm* - by 0.1%.

Fig. 3 presents the calculated temperature of the fuel rod points, when the gap between two pellets is 1.5 mm and sticking of other pellets after thermal expansion. It was considered that the pellets have chamfers, typical for VVER fuel rod·pellets, R_{co}=4.6 *mm*, R_{ci}=3.88 *mm*, R_{po}=3.78 *mm*, R_{pi}=1.15 *mm*, length of the pellets is 10 *mm*, q_v=1·10^9 *W/m*^{3}; q_v and q_l decrease due to the fuel thermal expansion (in r and z direction) was considered. It is seen that presence of the gap causes the temperature splash and thermal flow from the cladding to the splash of heat release in the neighbouring fuel rods as well (when the gap is 1.5 *mm*) - by 0.4 - 1% in local parts (1 - 2 *cm*) of the neighbouring fuel rods. In this case temperature goes up by 0.4 - 1%.

We would like to point out that the calculation results show the slight heat eschange between the neighbouring pellets.

It is shown that consideration of these deviations, as a rule, leads to the increase of the maximum fuel temperature (in the VVER pellets, characteristic by large central hole), tempertaure of the cladding, thermal flux, relieved by the coolant from the cladding and stresses in the cladding.

To conclude it all, it is important to note, that it is necessary to consider these factors for both validation of the fuel element working ability and interpretation of the experimental results.

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Application of Modified Version of SPPS-1 - HEXAB-2DB Computer Code Package for Operational Analyses of Fuel Behaviour in VVER-440 Reactors at Kozloduy NPP

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1 Introduction

SPPS-1 [1, 2] code has been used at Kozloduy NPP since 1986 for reactor physics calculations. Numerous comparisons with operational data have proved it as an adequate workable tool for basic VVER-440 neutron-physics characteristic calculations. PCs of higher order implementation (32 bits etc.) and the availability of 32 bits compiler for FORTRAN have allowed improvement and development of code capabilities. A modified version of SPPS-1-HEXAB-2DB code system has been developed and implemented at Kozloduy NPP for the purposes of operational analysis and power peaking factors and reactor core critical parameters predictions at VVER-440s.

In the present report are presented the results of calculations done by the use of the new SPPS-1 code version and their corresponding neutron physics characteristics, resulted from experiments at Kozloduy NPP VVER-440s. The results of fuel rod power distribution calculated by HEXAB-2DB are presented too.

2 Method for Operation Simulation of Reactor Core with 349 Assemblies (Unit 4) and with 313 Fuel Assemblies and 36 Dummy Fuel Assemblies (Unit1).

A new type of calculation was used during the simulation of reactor operation regime, so was performed a combination of fuel cycle calculation with Xe-135 transient, taking into consideration fuel burn-up increase. Beside for fuel burn-up and Pm-149 and Sm-149 concentrations distributions, the new algorithm calculates the space distributions of I-135 and Xe-135 concentrations, which also are recorded at the fuel burn-up file. In this way it is possible to perform the reactor operation simulation during the initial period after start-up or shut-down and to be predicted the reactor core critical parameters during transients.

The results of a Unit 4, cycle 11 restart simulation are presented at Table 1, covering a period since the 11th fuel cycle beginning till the 80th hour,

Table 1 Simulation of the core operation at the Unit 4 start-up at the beginning of Cycle 11

Time t_0 after start-up [*h*]	Operational parametres				Calculated $C^{crit}_{H_3BO_3}$, [*g/kg*] by simulation	$\Delta C_{H_3BO_3}$, *g/kg*
	N_T, %	H_{VI}, *cm*	t °C	$C_{H_3BO_3}$, *g/kg*		
0	HZPC	190	256	9.16	9.56	-0.40
0	HZPC	54	255	8.54	8.60	-0.06
0	HZPC	99	259	8.68	8.84	-0.16
0	28	132	262	8.42	8.76	-0.34
2	30	143	263	8.42	8.77	-0.35
6	43	195	265	8.42	8.69	-0.27
15	50	195	266	7.82	8.00	-0.18
17	50	204	266	7.75	7.95	-0.20
23	50	187	266	7.75	7.63	0.12
26	50	200	267	7.57	7.45	0.12
29	50	187	267	7.45	7.34	0.11
41	50	199	267	7.13	7.28	-0.15
49	53	173	268	7.07	7.08	-0.01
57	53	177	268	6.95	7.07	-0.12
65	53	185	268	6.94	7.06	-0.12
66	55	189	268	6.94	7.08	-0.14
67	57	192	269	6.94	7.11	-0.17
72	55	188	269	6.88	7.10	-0.22
80	55	192	269	6.93	7.10	-0.17

HZPC - Hot Zero Power Critical condition when a steady state level has been established at thermal power operation N_t = 55% nominal power. The time to is considered since the restart moment and the measured core parameters are as follows:

N_t thermal power;

H_{VI} critical height of the operating VI-th group of control assemblies;

t reactor core mean temperature;

$C_{H_3BO_3}$ measured critical boric acid concentration of the coolant.

The last two columns include the critical boric acid concentration $(C_{H_3BO_3})^{crit}$ calculated for the operational parameters, and calculation's accuracy $\Delta C_{H_3BO_3} = C_{H_3BO_3} - (C_{H_3BO_3})^{crit}$.

The similar data of a Unit 4, cycle 12 restart simulation are shown at Table 2.

The very good accuracy of simulation is obvious for the power gaining transient processes. In cases of biggest errors, the calculation provides a conservative estimation of the expected boric acid critical concentration.

Table 2 Simulation of the core operation at the Unit 4 start-up at the beginning of Cycle 12

Time t_0 after start-up [*h*]	Operational parametres				Calculated	
	N_T, %	H_{VI}, *cm*	t °C	$C_{H_3BO_3}$, *g/kg*	$C^{crit}_{H_3BO_3}$, [*g/kg*] by simulation	$\Delta C_{H_3BO_3}$, *g/kg*
0	HZPC	198	256	9.30	9.77	-0.47
14	33	198	263	8.54	8.67	-0.13
16	35	210	264	8.54	8.65	-0.11
24	50	193	268	7.90	8.04	-0.14
27	50	193	268	7.81	8.00	-0.19
50	38	196	263	7.44	7.78	-0.34
65	40	187	264	7.37	7.76	-0.39
99	40	178	265	7.37	7.59	-0.22
113	35	173	264	7.37	7.58	-0.21
114	35	173	264	7.31	7.63	-0.32
137	37	177	265	7.31	7.63	-0.32
185	40	199	266	7.31	7.59	-0.28
189	44	200	266	7.31	7.57	-0.26
204	54	165	268	7.25	7.37	-0.12
210	55	189	267	7.25	7.37	-0.12
216	55	192	268	7.12	7.31	-0.19
236	55	192	268	7.06	7.21	-0.15
254	55	192	268	7.0	7.19	-0.19
260	55	170	272	7.0	7.18	-0.18
270	72	193	275	6.94	6.93	0.01
281	75	200	275	6.94	6.83	0.11
291	75	203	275	6.57	6.71	-0.14

Table 3 Simulation of the core operation at and after the Unit 1 scram accounting Xe-135 transient

Time t_0 after start-up [*h*]	Operational parametres				Calculated	
	N_T, %	H_{VI}, *cm*	t °C	$C_{H_3BO_3}$, *g/kg*	$C^{crit}_{H_3BO_3}$, [*g/kg*] by simulation	$\Delta C_{H_3BO_3}$, *g/kg*
-10.0	55	182	267	5.20	5.16	0.04
0.0	55	182	268	5.20	5.14	0.06
0.0	0	-	255	-	-	-
8.0	HZPC	125	256	5.02	4.81	0.21
13.5	HZPC	125	256	5.02	5.07	-0.05
13.5	6	102	262	5.02	4.76	0.26
17.7	25	90	262	5.02	4.96	0.06
19.7	55	106	270	5.02	4.93	0.09
24.7	55	144	269	5.45	5.45	0.00
30.0	55	168	268	5.27	5.52	-0.25
48.0	55	202	268	5.39	5.26	0.13
56.0	55	175	268	5.14	5.05	0.09
60.0	55	187	268	5.14	5.03	0.11
72.0	55	192	268	5.08	5.09	-0.01
86.0	55	189	268	5.14	5.16	-0.02

The results of Unit 1's reactor operation simulation is presented at Table 3, covering the time interval since the moment t_0=0.0, about 102 full power days of 17th fuel cycle. Following a steady operation at 55% nominal thermal power (N_t=55%) a sharp scram type-1 took place at the moment 0.0 *h*. The values $\Delta C_{H_3BO_3}$ have confirmed again the high calculation accuracy of SPPS-1 new version.

A file recorded the calculated fuel burn-up and the Pm-149, Sm-149, I-135 and Xe-135 distributions at the moment of 24.7 *h* after the scram type-1 took place (at the same time a map of the measured assemblies' outlet temperatures had been done). This file was used further as an input to compare the calculated assemblywise relative power distribution factors with the corresponding measured factors. The comparison data resulted from the use of the new SPPS-1 version for transients simulation are presented in Table 4.

The results, presented at Table 4 show that the new SPPS-1 version for simulation of reactor core transients describes in adequate way the reactor core status.

Experience of the calculations done over Kozloduy NPP VVER 440s shows, that the new algorithm complies to a great extend with the algorithms of on-line simulators, as the difference is that the data on reactor operational regime are set by the user, while in the case of on-line simulators, data are provided directly from the special instrumentation data supply code subsystem.

Table 4 Error in SPPS-1 K_q calculation of ^{135}Xe transient (Unit 1, Cycle 17)

24.7 *h* after the scram	Comparisons
K_{eff}	1.00036
ΔK_{eff}[%] - criticality calculation error	0.036
$K_v^{max,calc}$ (n, m)	1.718 (292, 6)
Maximal values of K_q $K_q^{ave,meas}/K_q^{calc}$ (n)	1.294 /1.260 (277)
K_q^{calc} / $K_q^{ave,meas}$ (n)	1.260 /1.294 (277)
Maximal positive error	9.5
No. of assembly (type)	194 (2A)
Maximal negative error	- 13.2
No. of assembly (type)	139 (3A)
σ - mean square error	4.1
Mean error 1A (84)	0.4
Mean error 2A (93)	0.1
Mean error 3A (77)	- 1.2
Error peripherial assemblies	3.0

$\Delta = (K_q(n)^{calc} - K_q(n)^{ave,meas})\ 100\%$;
$K_v^{max,calc}$(n,m) - calculated maximal core volume power peaking factor in assembly number n, in the m-th axial segment, at order from the bottom to the top of the core, in the calculation each assembly containing 20 segments

3 New SPPS-1 Code Algorithm for Estimation of Assemblywise Power Peaking Factors Distribution in Reactor Core

The old SPPS-1 code algorithm for comparison of the calculated and the experimentally gained assemblywise power peaking factors distributions presumes negligible deviations from the 60° rotational symmetry and is reasonable only to be used for evaluation of calculated results accuracy. The new algorithm developed, which is used as well for reactor cores without 60° rotational symmetry is aimed to be used as well for analysis of the measured assemblywise power peaking factors. The new algorithm averages again thermo-couple indications of the assemblies, which are located at symmetrical positions. In contrast to the old algorithm, which averages the temperature values at assembly outlets, the new algorithm averages the differences between the measured t^{meas} and the calculated t^{calc} temperature values for each assemblies group at symmetrical positions. Following a rejection of unacceptably deviated data, the resulted mean deviation is added to the calculated temperatures per each assembly of the group at symmetrical positions and the result is considered as a restored or averaged temperatures distribution at assembly outlets t^{ave}. The restored temperature values of assembly outlets are used for calculation of their relative power $K_q^{ave,meas}$.

The new algorithm pays a special attention to the experimentally gained power peaking distribution factors K_q^{meas}, which have been calculated on the basis of individual temperatures at assembly outlets t^{meas}. When K_q^{meas} is estimated, the algorithm uses only the adjusted by the requirement for thermal balance parameters - reactor core inlet temperature, reactor power or reactor core coolant flow.

The analysis of thermal couples individual indications, could be used to identify the cause for value deviations from the predicted calculated values. A number of options for this purpose have been implemented in the new algorithm.

The average deviations between the calculated temperatures and the measured ones, their mean square error over the whole reactor core had been calculated as well as per each commutation device. In order to follow-up the automatic control assemblies status, there was implemented a calculation of the deviations between the measured temperatures and the calculated ones, over the first and the second surrounding annuli at each automatic control assembly.

The important case from safe operations point of view is the case of indication for a higher temperature at assembly outlet than the predicted temperature. The causes for this could be established to a great extend, if temperature and power values are analysed over the assemblies of the first and the second surrounding annuli.

Table 5 **Main results in SPPS - 1 calculation, compared with the measured assemblywise relative power distributions $K_q(n)$ for different moments T [FPD] of the Cycle 17 of Unit 2**

T, FPD	15.30	16.29	18.25	23.13
Parameters of the core condititons	$H_{VI}=181cm$ $T_{in}=263.7°C$ $N_T = 98\%$ $C_B=0.942g/kg$	$H_{VI}=181cm$ $T_{in}=263.5°C$ $N_T = 98\%$ $C_B= 0.942g/kg$	$H_{VI}=194cm$ $T_{in}=263.5°C$ $N_T = 98\%$ $C_B=0.932g/kg$	$H_{VI}=193cm$ $T_{in}=264.1°C$ $N_T =95\%$ $C_B=0.986g/kg$
Keff	1.00471	1.00435	1.00550	0.99987
ΔKeff -calculation error [%]	0.471	0.435	0.550	-0.013
Max. K_q				
$K_q^{ave,meas}/K_q^{calc}$ (n)	1.253 / 1.244 (219)	1.254 / 1.243 (219)	1.251 /1.250 (201)	1.257 /1.249 (201)
$K_q^{calc}/K_q^{ave,meas}$ (n)	1.257 /1.243 (226)	1.257 / 1.239 (226)	1.254 /1.241 (226)	1.255 /1.234 (226)
Maximum positive error	14.5	13.7	14.1	8.4
No of assembly (type)	174 (2A)	174 (2A)	174 (2A)	230 (1B)
Maximum negative error	- 7.6	- 6.7	- 5.6	- 6.2
No of assembly (type)	29 (1A)	29 (1A)	29 (1A)	29 (1A)
σ mean-square error	3.8	3.6	3.6	3.4
mean error 1A (96)	- 1.6	- 1.3	- 1.8	- 1.0
mean error 2A (84)	2.8	2.9	2.8	2.0
mean error 3A (84)	- 1.7	- 1.8	- 1.3	- 1.4
error peripherial assemblies	- 3.6	- 3.2	- 2.8	- 2.8

Error $K_q(n) = (K_q(n)^{calc} - K_q(n)^{ave,meas})100$ %; n - location number in 360° sector

Table 6 **Main results in SPPS - 1 calculation, compared with the measured assemblywise relative power distributions Kq(n) for different moments T [FPD] of the Cycle 17 of Unit 2**

T, FPD	24.08	42.13	43.10	48.00
Parameters of the core condititons	$H_{VI}=197cm$ $T_{in}=264.5°C$ $N_T =95\%$ $C_B=0.986g/kg$	$H_{VI}=208cm$ $T_{in}=263.8°C$ $N_T =97\%$ $C_B=0.909g/kg$	$H_{VI}=197cm$ $T_{in}=263.7°C$ $N_T =98\%$ $C_B=0.898g/kg$	$H_{VI}=197cm$ $T_{in}=263.5°C$ $N_T =98\%$ $C_B=0.888g/kg$
Keff	0.99982	0.99996	0.99947	0.99862
ΔKeff - calculation error [%]	- 0.018	- 0.004	- 0.053	- 0.138
Max. K_q				
$K_q^{ave,meas}/K_q^{calc}$ (n)	1.256 / 1.248 (201)	1.243 / 1.246 (226)	1.244 /1.247 (57)	1.236 / 1.246 (226)
$K_q^{calc}/K_q^{ave,meas}$ (n)	1.254 /1.240 (226)	1.246 / 1.243 (226)	1.247 /1.243 (226)	1.246 / 1.231 (57)
Maximum positive error	7.7	13.4	19.1	19.9
No of assembly (type)	174 (2A)	174 (2A)	174 (2A)	174 (2A)
Maximum negative error	- 6.8	- 6.3	- 6.1	- 6.9
No of assembly (type)	29 (1A)	29 (1A)	29 (1A)	29 (1A)
σ mean-square error	3.2	3.6	3.9	4.1
mean error 1A (96)	- 0.9	- 0.5	- 1.0	- 1.1
mean error 2A (84)	1.9	1.4	2.0	2.7
mean error 3A (84)	- 1.2	- 1.3	- 1.4	- 1.9
error peripherial assemblies	- 2.3	- 2.2	- 2.2	- 2.9

Error $K_q(n) = (K_q(n)^{calc} - K_q(n)^{ave,meas})100$ %; n - location number in 360° sector

Table 7 Unit 2, Cycle 17

N01 N06/kind X - Y	T [fpd] NT [%]	15.30 98	16.29 98	18.25 98	23.13 95	24.08 95	42.126 97	43.1 98	48.0 98
281 49 / 1A 05 - 52	δ1 [%] δ2 [%] δ3 [°C] δ4 [°C]	6.0 1.6 1.6 1.3	6.4 1.8 1.7 1.3	9.0 1.3 2.4 2.1	2.2 - 6.6 0.6 2.3	-0.1 -7.1 0.0 1.8	11.6 1.5 3.1 2.7	10.6 2.0 2.9 2.3	11.7 1.0 3.2 2.9
218 30 / 2A 09 - 54	δ1 [%] δ2 [%] δ3 [°C] δ4 [°C]	7.1 -2.6 1.9 2.6	7.5 -2.7 2.0 2.7	7.2 -3.1 1.9 2.8	6.0 0.0 1.6 1.6	7.7 -1.4 2.0 2.4	9.8 -0.6 2.6 2.8	9.6 -0.8 2.6 2.8	11.4 -2.4 3.1 3.7
168 49 / 1A 11 - 28	δ1 [%] δ2 [%] δ3 [°C] δ4 [°C]	8.8 1.6 2.4 2.0	10.0 1.8 2.7 2.2	8.4 1.3 2.3 1.9	10.7 -6.6 2.8 4.6	9.2 -7.1 2.4 4.3	11.2 1.5 3.0 2.6	11.8 2.0 3.2 2.6	9.9 1.0 2.7 2.4
43 25 / 1A 19 - 54	δ1 [%] δ2 [%] δ3 [°C] δ4 [°C]	-13.5 -0.5 -3.8 -3.4	-10.7 -2.0 -3.0 -2.3	-14.7 0.4 -4.2 -4.3	-23.4 0.5 -6.5 -6.6	-18.8 1.1 -5.2 -5.5	-11.2 -2.4 -3.1 -2.3	-10.6 -2.4 -3.0 -2.1	-10.7 -2.6 -3.0 -2.1

The code provides an option to be printed information on the ten highest measured coefficients of relative power, which is presented at Table 8 and Table 12. The abbreviations **N1** and **ERS1** concern the number of temperature controlled assemblies from the first surrounding annulus and respectively the average difference between the measured and their calculated relative power values in percents, while **N2** and **ERS2** are the same characteristics, concerning the second surrounding annulus of the pointed assembly. In case when the values **ERS1**, **ERS2** of the first and the second surrounding of an assembly are negative, it could be concluded that the high value of its measured coefficient of relative power K_q^{meas}, could not be due to an increased neutron flux.

At Tables 5, 6 and 7 are presented the new possibilities of SPPS-1 code for comparison of the calculated results with temperature control data as well as for analysis of thermal couple indications, for selected assemblies during the 17th fuel cycle of Unit 2 (313 loaded fuel assemblies together with 36 dummy fuel assemblies at the periphery).

At Tables 5 and 6 are presented for comparison purposes values of maximal experimental averaged data over symmetric groups of thermo-couples, relative power peaking factors $K_q^{ave,\,meas}$, the corresponding calculated K_q^{calc} and the average calculation errors for the different type of assemblies.

At Table 7 are followed the differences δ_i (i=1, 2, 3, 4) of individually measured values from the corresponding averaged values as well as from the calculated ones. Data are presented for several assemblies indicated by their calculation numbers **N01, N06** (within the 60° sector of 53 assemblies), which are of the types: **1A** (for the first year in the reactor core, enrichment 3.6%), **2A** (for the second year in the reactor core, enrichment 3.6%), **3A** (third year in the reactor core, enrichment 3.6%) and the corresponding **X-Y** coordinates.

The symbols used have the following meanings:

δ_1 difference between the measured individual relative power factor K_q^{meas} and the corresponding calculated one K_q^{calc}:

$$\delta_1 = (K_q^{meas} - K_q^{calc})\ 100\%;$$

δ_2 difference between the measured averaged relative power factor over a group of symmetrical thermo-couples $K_q^{ave,\,meas}$ and the corresponding calculated one K_q^{calc}:

$$\delta_2 = (K_q^{ave,\,meas} - K_q^{calc})\ 100\%;$$

δ_3 difference between the measured temperature at assembly outlet t^{meas} and the corresponding calculated one t^{calc}: $\delta_2 = t^{meas} - t^{calc}$, [°C];

δ_4 difference between the measured temperature at assembly outlet t^{meas} and the corresponding averaged value t^{ave} over the group symmetrical thermo-couples: $\delta_4 = t^{meas} - t^{ave}$, [°C].

The δ_2 characteristic is considered as calculation accuracy for the corresponding group of symmetric assemblies.

It is obvious from Table 7, in cases the thermo-couples at assembly N01 = 168 outlets indicate temperature values, which are not only higher than the calculated results, but also are higher than the measured values which are averaged over groups of symmetrical thermo-couples $K_q^{ave,\,meas}$. The additional analysis of data from Table 8 leads to the conclusion that the measured increased temperature values at the above discussed assembly are not of neutron-physics origin. The similar data for assembly N01=43 with no doubt indicate that the thermo-couples, been located in these positions provide strongly reduced values, without any physics causes. All δ_2 values are within the limits of calculation accuracy, as it could be seen from their comparison to the maximal errors, presented in Tables 5 and 6.

Table 8 Unit 2 , Cycle 17

N01	N06	X-Y	t^{meas}	$t^{ave,meas}$	t^{calc}	Kq^{meas}	$Kq^{ave,meas}$	Kq^{calc}	N1	ERS1	N2	ERS2
15.3 FPD												
168	49	11-28	302	300.0	299.6	1.293	1.221	1.205	6	0.4	6	1.2
16.29 FPD												
168	49	11-28	302.5	300.3	299.8	1.305	1.223	1.205	6	-0.8	6	0.6
18.25 FPD												
168	49	11-28	302.3	300.4	300.0	1.290	1.219	1.206	6	-0.2	6	0.4

Table 9 Main results in SPPS - 1 calculation, compared with the measured assemblywise relative power distributions Kq(n) for different moments [FPD] of the Cycle 17 of Unit 1

T, FPD	9.53	11.43	30.15	42.41	58.56
Parameters of the core condititons	$H_{VI}=203cm$ $T_{in}=261.8°C$ N_T = 95 % $C_B=1.140 g/kg$	$H_{VI}=195cm$ $T_{in}=261.3°C$ N_T = 100% $C_B= 1.100 g/kg$	$H_{VI}=181cm$ $T_{in}=261.4°C$ N_T = 96% $C_B=0.990 g/kg$	$H_{VI}=183cm$ $T_{in}=262.5°C$ N_T = 95% $C_B=0.997 g/kg$	$H_{VI}=191cm$ $T_{in}=263.3°C$ N_T = 95% $C_B=0.921 g/kg$
Keff	1.0046	1.0052	1.0061	1.0017	1.0023
ΔKeff -calc. error [%]	0.46	0.52	0.61	0.17	0.23
Max. K_q					
$K_q^{ave,meas}/K_q^{calc}$ (n)	1.277 / 1.245 (189)	1.273 / 1.241 (292)	1.271 /1.240 (292)	1.274 / 1.239 (292)	1.275 / 1.235 (292)
$K_q^{calc}/K_q^{ave,meas}$ (n)	1.250 /1.255 (207)	1.251 / 1.233 (207)	1.250 /1.254 (207)	1.247 / 1.246 (207)	1.240 / 1.236 (207)
Max. positive error	8.9	10.2	7.5	6.8	10.5
No of assembly (type)	194 (2A)	120 (2A)	194 (2A)	194 (2A)	120 (2A)
Max. negative error	- 6.5	- 5.5	- 4.2	- 6.7	-4.8
No of assembly (type)	139 (3A)	310 (1A)	310 (1A)	139 (3A)	293 (1A)
σ mean-square error	3.0	2.9	2.6	2.8	3.2
mean error 1A (84)	- 1.0	- 1.6	- 1.8	- 0.9	-1.6
mean error 2A (93)	1.5	2.2	1.6	1.6	1.5
mean error 3A (77)	- 1.0	- 1.1	- 0.3	-1.1	-0.6
error peripheral assemblies	0.6	- 2.3	- 0.6	0.1	1.6

Table 10 Main results in SPPS - 1 calculation, compared with the measured assemblywise relative power distributions Kq(n) for other different moments [FPD] of the Cycle 17 of Unit 1

T, FPD	60.46	71.44	76.19	88.00	94.32
Parameters of the core condititons	$H_{VI}=188cm$ $T_{in}=262.3°C$ N_T = 95% $C_B=0.921 g/kg$	$H_{VI}=195cm$ $T_{in}=262.1°C$ N_T =95% $C_B=0.935 g/kg$	$H_{VI}=185cm$ $T_{in}=263.1°C$ N_T =95% $C_B=0.844 g/kg$	$H_{VI}=182cm$ $T_{in}=262°C$ N_T =95% $C_B=0.801 g/kg$	$H_{VI}=179cm$ $T_{in}=259.9°C$ N_T =55% $C_B=0.909 g/kg$
Keff	1.0015	0.9968	1.0019	1.0006	1.00096
ΔKeff - calc. error [%]	0.15	- 0.32	0.19	0.06	0.096
Max. K_q					
$K_q^{ave,meas}/K_q^{calc}$ (n)	1.274 /1.236 (292)	1.270 / 1.234 (292)	1.266 /1.232 (292)	1.276 / 1.232 (292)	1.262 / 1.230 (189)
$K_q^{calc}/K_q^{ave,meas}$ (n)	1.241 /1.225 (207)	1.237 / 1.242 (207)	1.238 /1.237 (207)	1.237 / 1.245 (207)	1.248 / 1.255 (277)
Max. positive error	10.0	9.8	9.6	9.5	9.2
No of assembly (type)	194 (2A)	34 (2B)	34 (2B)	120 (2A)	120 (2A)
Max. negative error	- 8.4	- 8.8	- 5.0	- 4.7	- 12.2
No of assembly (type)	293 (1A)	139 (3A)	293 (1A)	74 (1A)	139 (3A)
σ mean-square error	3.4	3.4	2.9	3.2	3.5
mean error 1A (84)	- 2.1	- 1.4	- 1.7	- 1.6	- 1.1
mean error 2A (93)	2.1	0.6	0.5	0.8	2.0
mean error 3A (77)	- 0.8	- 0.4	- 0.1	- 0.2	- 1.9
error peripheral assemblies	0.5	2.1	1.8	2.5	- 0.6

Analogous results for actual 17-th cycle of Unit 1 (313 loaded fuel assemblies together with 36 dummy-fuel assemblies at the periphery) are shown at refuelling pattern and to follow-up Tables 9, 10, 11 and 12.

The new algorithm provides an option for checking the correctness of the the reactor core symmetry. When compared with the experimental data, the calculation ones describe the refuelling particularities and could be used for analysis of the differences between the measured and the predicted temperature values.

4 Other New Options in the Modified SPPS-1 - HEXAB-2DB Code

Beside the above presented options, a new algorithm is developed and implemented to SPPS-1, for the purpose of making choice of refuelling in a dialogue mode. The required information is provided in a fast and appropriate form for choosing of refuelling patterns. The new options, which have been implemented allows: each selected refuelling pattern to be studied at different control group assembly heights; to perform an automatic connection with HEXAB-2DB code, for the purpose of fuel rod power peaking factors calculation and to calculate the main safety related neutron-physics characteristics during the fuel cycle. In this way, a large amount of refuelling patterns could be selected and checked in order to make decisions on an optimal refuelling pattern. Fuel cycle characteristics can be evaluated too.

The enhanced scope of analyses of fuel rods, requires information for power peaking factors along the entire length of the concerned assemblies. This reason imposed development and implementation of an option in HEXAB-2DB for calculation of reactor core layers, at the level of the automatic control assembly absorbers.

The calculated fuel rod power peaking factors of the assemblies K_{kk} and the maximal linear power density **Gl** for different moments of the design cycle 12, Unit 4, are presented on Figures 1 - 4. It is obvious that the values of **Gl** do not exceed the limit value of 325 *W/cm*.

The computer codes SPPS-1-HEXAB-2DB are used also for preparation of neutron-physics input data for the fuel thermo-mechanic analyses (by means of thermo-mechanical computer code PIN-MICRO [5]). A set of neutron-physics data for 1 fuel rod of reactor core Unit 4, available for PIN-MICRO are presented at Tables 13 and 14.

Finally it could be pointed, that the experience from the use of the modified SPPS-1-HEXAB-2DB code system confirms the provision of improved availability of operational analysis, prediction of Kozloduy NPP VVER-440s safe operations and fuel behaviour estimation.

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Table 11 Unit 1, Cycle 17

N01 N06/kind X - Y	T [FPD] NT [%]	11.43 100	30.15 96	42.41 95	58.56 95	71.44 95	76.19 95	88.0 95	94.32 55
146 49 / 1A 13 - 58	δ1 [%] δ2 [%] δ3 [°C] δ4 [°C]	14.2 1.2 4.0 3.6	14.2 3.3 3.8 2.9	17.4 - 2.3 4.6 5.2	10.8 4.3 2.9 1.7	13.6 3.6 3.6 2.6	13.6 5.0 3.6 2.3	9.5 3.1 2.5 1.7	18.1 2.3 3.0 2.6
13 43 / 1A 21 - 44	δ1 [%] δ2 [%] δ3 [°C] δ4 [°C]	12.2 -2.9 3.4 4.2	14.1 -2.5 3.8 4.4	20.6 0.0 1.6 1.6	21.7 1.4 5.7 5.4	14.7 3.5 3.9 3.0	12.9 3.2 3.4 2.5	15.0 4.2 4.0 2.9	20.2 2.8 3.4 2.9
26 46 / 1A 20 - 47	δ1 [%] δ2 [%] δ3 [°C] δ4 [°C]	4.2 3.2 1.2 0.2	3.4 3.0 0.9 0.0	10.7 -6.6 2.8 4.6	6.8 3.6 1.8 0.9	2.9 3.6 0.8 -0.3	2.8 2.9 0.7 -0.1	4.8 3.9 1.3 0.2	3.0 0.6 0.5 0.4
27 49 / 1A 20 - 49	δ1 [%] δ2 [%] δ3 [°C] δ4 [°C]	7.2 1.2 2.0 1.7	7.0 3.3 1.9 1.0	35.8 6.2 9.5 7.8	13.4 4.4 3.5 2.4	9.6 3.6 2.5 1.5	8.8 4.5 2.3 1.1	10.1 3.2 2.7 1.8	12.8 2.3 2.1 1.7
37 30 / 2A 19 - 42	δ1 [%] δ2 [%] δ3 [°C] δ4 [°C]	1.6 -1.7 0.4 0.9	2.4 0.5 0.6 0.5	-23.4 0.5 -6.5 -6.6	12.3 -0.9 3.2 3.4	9.1 0.5 2.4 2.3	6.2 -0.1 1.6 1.7	11.4 0.3 3.0 2.9	23.9 -2.3 3.9 4.3

Table 12 Unit 1, Cycle 17

N01	N06	X-Y	t^{meas}	$t^{ave,meas}$	t^{calc}	K_q^{meas}	$K_q^{ave,meas}$	K_q^{calc}	N1	ERS1	N2	ERS2
58.56 FPD												
37	30	19-42	304.2	300.8	301.0	1.359	1.227	1.236	4	-0.4	4	11.6
146	49	13-58	301.7	300.0	298.8	1.263	1.198	1.155	2	-0.5	5	2.3
76.19 FPD												
37	30	19-42	302.7	301.0	301.1	1.297	1.234	1.235	4	-3.9	5	7.1
146	49	13-58	302.5	300.2	298.9	1.289	1.203	1.153	2	2.4	5	1.5
88.0 FPD												
37	30	19-42	302.6	299.7	299.6	1.348	1.237	1.234	4	-2.9	5	8.0
146	49	13-58	300.0	298.3	297.5	1.249	1.185	1.154	2	-1.8	5	-1.7

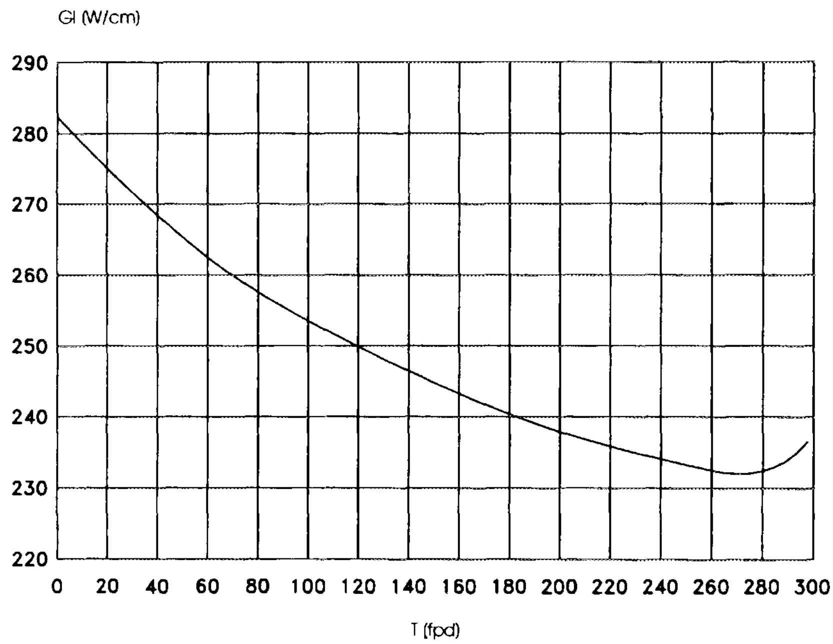

Figure 1 Maximal linear power density Gl [W/cm] evolution, calculated by SPPS-1 - HEXAB-2DB for the design cycle 12, Unit 4.

Table 13 Neutron-physics data of the fuel rod for thermo-mechanical analysis.

Cycle	N	T days	T full power days	Nt,% thermal power	Coolant temperature increment Δt [°C]	t, °C average coolant temperature	t_{in}, °C inlet coolant temperature	Average fuel rod linear power density Gl, [*kW/m*]
	1	0.0	0.0	75	20.5	269	259	13.8
	2	3.6	2.7	100	26.9	275	262	18.0
	3	6.6	5.7	55	15.1	268	260	10.0
9	4	8.96	7.0	100	26.8	276	263	17.8
	5	148.96	146.8	100	26.7	277	264	17.4
	6	291.66	289.5	100	26.8	276	263	17.1
	7	297.13	294.97	55	15.1	268	260	9.7
		the reactor is stopped for refuelling - 45 days						
	8	342.13	0.0	70	19.2	268	258	12.4
	9	346.13	2.8	100	26.8	276	263	17.5
	10	360.83	17.5	0	0	260	260	0.0
		the reactor is stopped for 7 days						
	11	367.83	17.5	70	19.1	270	260	12.0
	12	369.51	18.7	100	26.7	277	264	17.3
	13	388.44	37.6	60	16.4	270	262	10.7
	14	390.78	39.0	100	26.7	277	264	17.3
	15	530.64	178.7	0	0	260	260	0.0
		the reactor is stopped for 7 days						
	16	537.64	178.7	85	22.9	274	263	14.1
10	17	539.29	180.1	100	26.8	276	263	16.5
	18	604.91	245.3	60	16.2	274	266	10.1
	19	606.24	246.1	100	26.8	276	263	16.3
	20	618.06	257.9	58	15.9	268	260	9.7
	21	621.0	259.6	100	26.7	277	264	16.3
	22	632.6	271.2	0	0	260	260	0.0
		the reactor is stopped for 6 days						
	23	638.6	271.2	70	18.9	274	265	11.5
	24	640.45	272.5	100	26.7	277	264	16.3
	25	651.45	283.5	55	15.1	268	260	9.2
	26	695.83	309.32	45	12.4	268	262	7.5
		the reactor is stopped for refuelling - 62 days						
	27	757.83	0.0	55	15.1	269	261	8.6
	28	764.56	3.7	86	23.2	274	262	13.7
	29	766.53	5.4	100	26.7	278	265	16.0
	30	775.53	14.4	55	15.0	270	263	8.9
	31	781.53	17.7	100	26.6	278	265	16.0
	32	885.83	122.0	55	15.0	270	263	8.4
	33	888.20	123.3	100	26.6	278	265	15.3
	34	925.40	160.5	55	15.1	269	261	8.2
	35	967.03	183.4	0	0	260	260	0.0
		the reactor is stopped for 2.5 days						
	36	969.53	183.4	54	14.8	268	261	8.2
	37	971.33	184.37	0	0	260	260	0.0
		the reactor is stopped for 102 days						
	38	1073.33	184.37	55	15.0	270	263	8.3
11	39	1084.11	190.3	0	0	260	260	0.0
		the reactor is stopped for 1 day						
	40	1085.11	190.3	55	15.0	270	263	8.3
	41	1093.84	195.1	75	20.4	272	262	11.4
	42	1098.33	198.2	100	17.7	278	265	15.1
	43	1138.18	238.0	56	15.3	270	262	8.2
	44	1139.97	239.0	100	26.6	278	265	15.0
	45	1163.97	263.0	60	16.4	270	262	8.7
	46	1165.64	264.0	100	26.6	278	265	15.1
	47	1194.24	292.6	55	15.1	268	260	8.0
	48	1211.95	301.8	0	0	260	260	0.0
		the reactor is stopped for 1 day						
	49	1212.95	301.8	50	13.7	268	261	7.6
	50	1272.04	332.0	50	13.8	267	260	7.5
		the reactor is stopped for refuelling - 141 days						
	51	1413.04	0.0	100	26.7	278	264	8.1
12	52	1563.04	150.0	100	26.7	278	264	8.2
	53	1710.65	297.61	100	26.7	278	264	8.3

Table 14 **Kz - axial power distribution of the assembly containing the considered fuel rod (the assembly is divided in 10 axial segments)**

N	1	2	3	4	5	6	7	
	0.425	0.410	0.365	0.416	0.552	0.794	0.654	top
C	0.801	0.770	0.681	0.776	0.906	1.115	0.930	
y	1.053	1.017	0.894	1.018	1.072	1.150	0.990	
c	1.202	1.172	1.061	1.167	1.133	1.099	1.006	
l	1.278	1.260	1.208	1.252	1.152	1.049	1.041	
e	1.300	1.299	1.333	1.290	1.161	1.026	1.106	
	1.271	1.289	1.386	1.283	1.167	1.031	1.166	
9	1.173	1.209	1.330	1.209	1.151	1.046	1.196	
	0.958	1.002	1.110	1.009	1.040	0.998	1.133	
	0.539	0.572	0.632	0.580	0.667	0.693	0.778	bottom

N	8	9	10	11	12	13	14	15	16	17	
	0.446	0.425	1.0	0.541	0.463	0.464	0.475	1.0	0.734	0.698	top
C	0.783	0.748	1.0	0.918	0.795	0.780	0.805	1.0	1.080	1.037	
y	0.989	0.953	1.0	1.107	0.987	0.942	0.988	1.0	1.168	1.136	
c	1.137	1.108	1.0	1.193	1.117	1.050	1.108	1.0	1.153	1.138	
l	1.238	1.220	1.0	1.221	1.204	1.146	1.190	1.0	1.110	1.110	
e	1.285	1.285	1.0	1.213	1.251	1.236	1.236	1.0	1.071	1.084	
	1.277	1.296	1.0	1.176	1.256	1.287	1.246	1.0	1.044	1.068	
10	1.201	1.238	1.0	1.100	1.207	1.266	1.206	1.0	1.021	1.053	
	1.020	1.067	1.0	0.944	1.054	1.120	1.065	1.0	0.952	0.986	
	0.624	0.660	1.0	0.588	0.665	0.710	0.680	1.0	0.667	0.691	bottom

N	18	19	20	21	22	23	24	25	26	
	0.684	0.720	0.707	0.730	1.0	0.817	0.749	0.731	0.815	top
C	0.968	1.020	0.991	1.022	1.0	1.121	1.035	1.005	1.109	
y	1.031	1.092	1.045	1.089	1.0	1.162	1.094	1.047	1.138	
c	1.043	1.089	1.046	1.083	1.0	1.123	1.083	1.042	1.109	
l	1.066	1.072	1.061	1.067	1.0	1.072	1.062	1.054	1.075	
e	1.095	1.065	1.085	1.060	1.0	1.034	1.053	1.075	1.047	
	1.122	1.071	1.107	1.068	1.0	1.013	1.058	1.097	1.027	
10	1.134	1.079	1.118	1.078	1.0	1.003	1.069	1.109	1.014	
	1.082	1.038	1.069	1.041	1.0	0.957	1.035	1.066	0.964	
	0.776	0.754	0.771	0.761	1.0	0.698	0.763	0.776	0.702	bottom

N	27	28	29	30	31	32	33	34	
	0.614	0.579	0.560	0.551	0.579	0.655	0.670	0.647	top
C	0.979	0.927	0.898	0.882	0.913	0.959	0.972	0.899	
y	1.146	1.100	1.072	1.032	1.075	1.047	1.078	0.960	
c	1.198	1.168	1.148	1.135	1.140	1.098	1.099	0.994	
l	1.191	1.180	1.172	1.187	1.157	1.119	1.093	1.064	
e	1.158	1.168	1.171	1.197	1.155	1.120	1.087	1.123	
	1.112	1.140	1.156	1.179	1.143	1.114	1.089	1.160	
11	1.048	1.092	1.118	1.132	1.113	1.101	1.092	1.178	
	0.931	0.983	1.015	1.019	1.021	1.039	1.049	1.137	
	0.622	0.663	0.689	0.686	0.703	0.747	0.770	0.838	bottom

N	35	36	37	38	39	40	41	42	
	1.0	0.814	1.0	0.811	1.0	0.714	0.765	0.711	top
C	1.0	1.125	1.0	1.118	1.0	0.991	1.054	0.981	
y	1.0	1.181	1.0	1.173	1.0	1.046	1.121	1.059	
c	1.0	1.142	1.0	1.136	1.0	1.076	1.104	1.066	
l	1.0	1.081	1.0	1.078	1.0	1.086	1.070	1.057	
e	1.0	1.031	1.0	1.032	1.0	1.086	1.045	1.057	
	1.0	1.000	1.0	1.003	1.0	1.086	1.036	1.072	
11	1.0	0.984	1.0	0.989	1.0	1.086	1.038	1.094	
	1.0	0.942	1.0	0.950	1.0	1.048	1.009	1.080	
	1.0	0.701	1.0	0.709	1.0	0.781	0.759	0.823	bottom

Table 14 (continued)

N	43	44	45	46	47	48	49	50	
	0.708	0.755	0.726	0.866	0.753	1.0	0.824	0.937	top
C	0.928	1.006	0.915	1.138	0.947	1.0	1.054	1.174	
y	0.957	1.065	0.900	1.132	0.956	1.0	1.061	1.172	
c	0.972	1.059	0.897	1.063	0.962	1.0	1.062	1.105	
l	1.033	1.043	0.945	1.007	1.019	1.0	1.056	1.035	
e	1.090	1.040	1.052	0.981	1.073	1.0	1.045	0.984	
	1.131	1.052	1.151	0.984	1.113	1.0	1.038	0.957	
11	1.161	1.077	1.222	1.009	1.146	1.0	1.041	0.953	
	1.145	1.071	1.232	1.017	1.141	1.0	1.023	0.941	
	0.875	0.832	0.959	0.802	0.890	1.0	0.796	0.741	bottom

N	51	52	53	
	0.510	0.663	0.891	top
C	0.841	0.971	1.144	
y	1.031	1.075	1.139	
c	1.149	1.107	1.078	
l	1.220	1.117	1.028	
e	1.250	1.121	1.003	
	1.233	1.121	1.001	
12	1.155	1.101	1.008	
	0.984	1.014	0.975	
	0.629	0.711	0.733	bottom

4A/59 N of years in the core, fuel type (enrichment)/ N in 60°-sector (59 assemblies)
1.632 Kkk - maximal relative power factor of the fuel rods in the assembly
108.0 Gl [W/cm] - maximal linear power density in the assembly

1A/57 1A/58
1.329 1.571
274.4 213.1

1A/53 1A/54 3A/55 4A/56
1.114 1.213 1.445 1.679
281.6 277.2 174.3 91.7

3A/48 3A/49 1A/50 1A/51 4A/52
1.094 1.094 1.166 1.395 1.672
195.4 199.1 276.6 265.8 114.8

2A/42 1A/43 3A/44 1A/45 1A/46 4A/47
1.113 1.106 1.099 1.125 1.363 1.674
236.0 275.3 198.4 282.5 277.9 118.9

2A/35 3A/36 2A/37 3A/38 1A/39 1A/40 4A/41
1.108 1.098 1.137 1.132 1.124 1.390 1.674
228.5 198.1 228.3 204.9 282.1 265.4 91.9

3A/28 2A/29 3A/30 2A/31 3A/32 1A/33 3A/34
1.088 1.109 1.100 1.136 1.101 1.165 1.443
201.0 240.2 194.9 227.8 198.0 276.3 174.3

2A/20 3A/21 2A/22 3A/23 1A/24 3A/25 1A/26 1A/27
1.109 1.089 1.109 1.098 1.104 1.095 1.212 1.570
223.2 202.0 239.9 196.6 274.2 199.0 277.0 212.9

2A/11 2A/12 3A/13 2A/14 2A/15 3A/16 1A/17 1A/18 4A/19
1.091 1.110 1.087 1.108 1.113 1.093 1.114 1.328 1.631
219.2 223.2 200.7 227.0 235.6 195.1 281.4 274.2 108.0

3B/01 2A/02 2A/03 3A/04 2A/05 2A/06 2B/07 3A/08 1A/09 3B/10
1.045 1.098 1.100 1.077 1.114 1.102 1.078 1.121 1.164 1.421
159.4 218.5 219.9 189.1 235.8 239.5 191.0 207.5 282.6 150.6

Figure 2 Assemblywise distribution of the power peaking factors of the fuel rods in the assembly Kkk and of maximal linear power density Gl [W/cm] at the beginning of Cycle 12, Unit 4.

4A/59									
1.564									
100.8									
1A/57	1A/58								
1.302	1.509								
236.0	188.5								
1A/53	1A/54	3A/55	4A/56						
1.121	1.217	1.372	1.609						
243.5	241.5	154.3	86.4						
3A/48	3A/49	1A/50	1A/51	4A/52					
1.068	1.068	1.170	1.364	1.586					
174.8	175.2	240.2	230.4	105.2					
2A/42	1A/43	3A/44	1A/45	1A/46	4A/47				
1.105	1.114	1.071	1.135	1.336	1.588				
215.4	244.3	175.3	244.2	238.8	109.1				
2A/35	3A/36	2A/37	3A/38	1A/39	1A/40	4A/41			
1.091	1.077	1.107	1.090	1.133	1.360	1.604			
210.4	181.6	203.9	179.3	243.6	230.1	86.6			
3A/28	2A/29	3A/30	2A/31	3A/32	1A/33	3A/34			
1.076	1.100	1.072	1.106	1.073	1.169	1.370			
190.0	222.3	178.6	203.6	175.0	239.9	154.2			
2A/20	3A/21	2A/22	3A/23	1A/24	3A/25	1A/26	1A/27		
1.096	1.072	1.099	1.078	1.112	1.069	1.215	1.508		
211.4	190.1	221.8	180.7	243.5	175.3	241.1	188.4		
2A/11	2A/12	3A/13	2A/14	2A/15	3A/16	1A/17	1A/18	4A/19	
1.081	1.095	1.076	1.090	1.105	1.068	1.121	1.302	1.563	
209.3	211.2	190.0	209.1	215.1	174.6	243.3	236.0	100.7	
3B/01	2A/02	2A/03	3A/04	2A/05	2A/06	2B/07	3A/08	1A/09	3B/10
1.046	1.090	1.087	1.064	1.105	1.100	1.056	1.081	1.171	1.352
159.7	210.2	209.6	280.4	219.9	221.3	181.3	181.0	244.6	135.7

Figure 3 **Assemblywise distribution of the power peaking factors of the fuel rods in the assembly Kkk and of maximal linear power density Gl [W/cm] at 150 FPD of Cycle 12, Unit 4.**

4A/59									
1.511									
102.2									
1A/57	1A/58								
1.287	1.448								
218.9	176.2								
1A/53	1A/54	3A/55	4A/56						
1.128	1.219	1.327	1.550						
229.0	225.8	149.8	89.6						
3A/48	3A/49	1A/50	1A/51	4A/52					
1.075	1.077	1.181	1.340	1.526					
177.0	174.4	225.5	214.2	106.8					
2A/42	1A/43	3A/44	1A/45	1A/46	4A/47				
1.102	1.119	1.071	1.147	1.312	1.529				
214.3	235.2	175.0	228.3	219.9	110.6				
2A/35	3A/36	2A/37	3A/38	1A/39	1A/40	4A/41			
1.083	1.071	1.097	1.067	1.145	1.336	1.546			
208.9	183.9	201.3	175.9	227.9	213.9	89.9			
3A/28	2A/29	3A/30	2A/31	3A/32	1A/33	3A/34			
1.068	1.096	1.071	1.097	1.070	1.180	1.325			
192.6	220.2	183.0	201.3	174.2	225.3	149.7			
2A/20	3A/21	2A/22	3A/23	1A/24	3A/25	1A/26	1A/27		
1.088	1.065	1.096	1.072	1.120	1.077	1.219	1.447		
212.4	193.7	220.4	184.1	235.4	174.4	225.8	176.1		
2A/11	2A/12	3A/13	2A/14	2A/15	3A/16	1A/17	1A/18	4A/19	
1.078	1.088	1.068	1.084	1.102	1.076	1.129	1.286	1.511	
212.8	212.5	192.8	210.0	214.5	177.2	229.2	218.8	102.2	
3B/01	2A/02	2A/03	3A/04	2A/05	2A/06	2B/07	3A/08	1A/09	3B/10
1.037	1.093	1.082	1.065	1.093	1.094	1.058	1.074	1.178	1.305
236.5	216.1	211.2	186.6	216.7	218.1	172.8	179.5	228.7	127.4

Figure 4 **Assemblywise distribution of the power peaking factors of the fuel rods in the assembly Kkk and of maximal linear power density Gl [W/cm] at the end of Cycle 12, Unit 4.**

Results of Calculation of VVER-440 Fuel Rods (Kolskaya-3 NPP) at High Burnup

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The results of calculations of thermal physical characteristics of fuel rods of two fuel assemblies (#1 and #2), which were operated within 5 - 8 and 5 - 9 core fuel loadings of Unit 3 of the Kolskaya NPP, respectively, are presented in this paper. These fuel assemblies were unloaded off the reactor, being actually tight (as the cladding tightness control of the NPP showed), and sent for the post-irradiation testing.

1 Calculation Code

PIN-04 (PIN-micro) code [1] was previously used in the USSR to validate the working ability of VEER-440 fuel rods during normal operation regimes. Now for the same purposes we also use the code for fuel rod thermal physical calculations PIN-mod1 (PIN-mod2) [2, 3], which is designed for modeling of VVER reactor type fuel rods behavior in a quasisteady-state operation. In comparison with PIN-micro code, the following models have been changed in PIN-mod2 code: the model of fission gas release out of fuel, calculation of relocation, fuel-cladding gap thermal conductivity calculation, burnup influence on the fuel thermal conductivity has been taken into account, as well as the effect of fuel creep upon the increase of fuel diameter, the approach for setting the input data while calculating burnup has been changed, etc. The average burnup along the fuel cross-section is used as a parameter to consider the influence of burnup on the increase of the fission gas release out of fuel. We would like to note that in [3] it is shown that for VVER fuel rods when, for example, average burnup along the cross-section is 48 *MWd/kg*, burnup in the thin layer of the outer fuel surface can reach 96 *MWd/kg*. During calculation of the temperature field the values of the relative power density reduction along the fuel radius, depending upon the achieved burnup, were used [3]. It was considered that fast neutron flux along the fuel rod height is proportional to thermal load; proportion coefficient was calculated depending upon the linear thermal load and the achieved burnup [3].

In PIN-type codes while modeling behavior of fuel and cladding, their presentation in the form of integral coaxial cylinders (in respect to the height) values of geometric and structural parameters are used.

2 Input Data for Calculations

The results of neutron physical calculations for Kolskaya NPP and the reactor system operational history (power, coolant inlet temperature, coolant heating in the reviewed fuel assemblies) were used for developing power history of fuel rods.

Figure 1 presents the simplified power history of the 3rd Unit of the Kolskaya NPP reactor system within the time period since 24.09.1986 (beginning

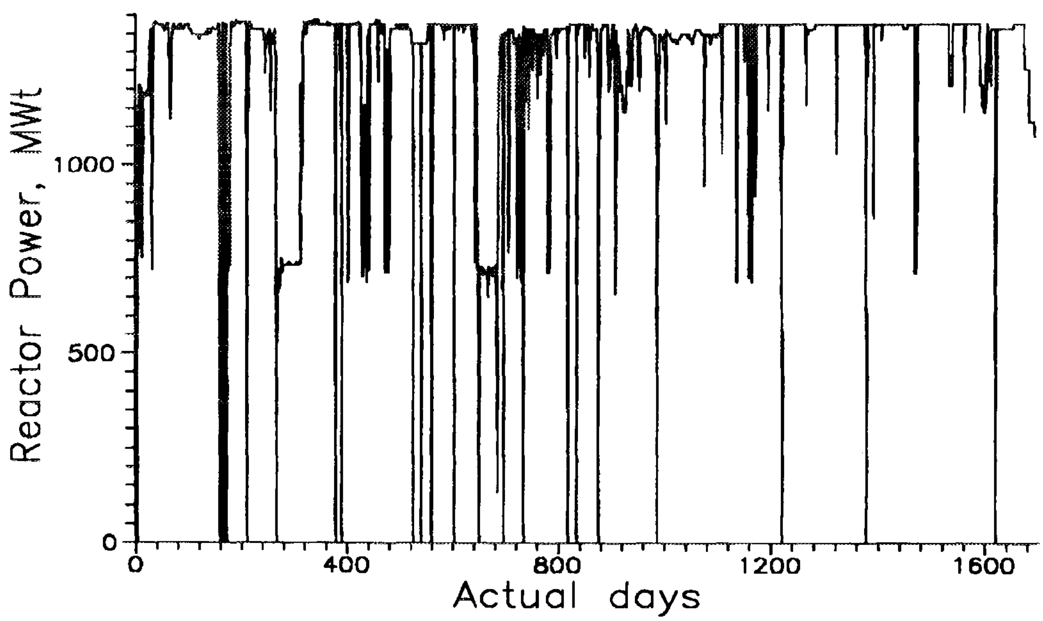

Figure 1 Kolskaya-3 NPP reactor system power history

of the 5th fuel cycle) till 12.10.1991 (end of the fuel cycle). Periods between the planned annual fuel reloading were excluded in Fig. 1. Power history of fuel rods was presented in actual (and not in effective) days of operation. In this case planned annual reloadings were excluded. For the simplification of presentations of figures, it was considered that the coolant temperature is 200°C, when the reactor is at zero power. Short-term distortions of the power density along the height of the fuel assembly, which van take place if the power of the unit is changed, are not presented in the power history of fuel rods.

Several power histories of fuel rods were developed for each fuel assembly. In all cases the relative profile of power density along the height of the fuel rod was obtained out of the results of neutron-physical calculations for fuel assembly.

Case 1.

While developing power history of the "average within the fuel assembly" fuel rod, the average value of the linear thermal load for any of the presented moments of time was obtained as the quotient when fuel assembly power at a given moment of time was divided by 124 (number of fuel rods within the fuel assembly) and by 243 *cm* (the length of fuel stack).

Cases 2 - 4 relate to one specific fuel rod, which has maximum deep (for its fuel assembly) burnup. The fuel rod power history was developed in two ways: with and without consideration of the excess coefficients.

Case 2.

When the fuel rod power history was developed without consideration of excess coefficients, the average value of the linear thermal load $q_{lav}^{(2)}$ was obtained as a quotient when fuel rod power at this particular time moment was divided by 243 *cm.*

Case 3.

While developing fuel rod power history with the consideration of excess coefficient in respect to burnup, average value of linear thermal load:

$$q_{lav}^{(3)} = 1.04\, q_{lav}^{(2)}$$

Case 4.

While developing fuel rod power history with the consideration of excess coefficient in respect to linear load, average value of linear thermal load:

$$q_{lav}^{(4)} = k\, q_{lav}^{(2)}$$

where k is a coefficient considering technological deviations during fuel fabrication and errors of neutron-physical calculations; it was considered that $k=1.06$. In this case it was considered that k influence caused only the increase of linear load 1.06 times; k influence was not considered while calculating burnup and fission products storage in the fuel.

After thus developing fuel rod power history (in effective operation days), consideration of the real power history of the reactor system was introduced.

In the calculations it was considered that:

- the initial fuel stack length is 242 *cm*;
- fuel pellets are flat ended;
- fuel enrichment for U-235 is 4.4%;
- fuel grain average size is 7 μm;
- fuel mass for case 1 is 1095 *g* for the fuel rod out of fuel assembly 1 and for the fuel rod out of assembly 2 it is 110 *g*; for cases 2 - 4 fuel mass is 1091 *g* for the fuel rod out of fuel assembly 1 and for the fuel rod out of fuel assembly 2 it is 1098 *g*;
- outer diameter of the cladding is 9.15 *mm*;
- filling gas - helium (98%) when the filling pressure P_{fil} .

As the parameters of fuel pellets and cladding have the spread in values, than three calculations were carried out for the variants:

MAX : maximum effective gap:
inner cladding diameter $D_{ci}=7.8$ *mm*;
pellet outer diameter $D_{po}=7.54$ *mm*;
pellet linear diameter $d=1.2$ *mm*;
fuel initial density $p=10.43$ g/cm^3 ;
filling gas pressure $P_{fill}=0.5$ *MPa*;
volume of the gas collector $v_g=3.7$ cm^3.

AVER: average effective gap:
$D_{ci}=7.76$ *mm*; $D_{po}=7.565$ *mm*;
$d=1.6$ *mm*; $p=10.58$ g/cm^3;
$P_{fill}=0.6$ *MPa*; $v_g=4.2$ cm^3.

MIN : minimum effective gap:
$D_{ci}=7.72$ *mm*; $D_{po}=7.59$ *mm*;
d=2.0 *mm*; $p=10.73$ g/cm^3;
$P_{fill}=0.7$ *MPa*; $v_g=4.7$ cm^3.

It is important to note that MAX or MIN case can not be realized in reality within the height of the fuel rod (in this case the requirements of the regulatory documents relating to the mass and height of the fuel stack will not be satisfied), but it is possible that one pellet with the parameter values described as MAX variant gets into the very part of the cladding, the value of the inner diameter of which corresponds to the MAX value. Due to small heat leaks between the pellets in the axial direction [2] maximum temperature in this pellet will be close to the MAX variant. That is why as a real case we are to review the results of calculations for the AVER case, basing on the results of calculations of the fuel rod local characteristics (maximum temperature, maximum fuel/cladding interaction) for MIN and MAX cases as the limiting possible ones.

It was considered that the limiting value of the fuel resintering is restricted by the density 10.68 g/cm^3, at the same time change of fuel diameter can not exceed 0.4%.

Figures 2 and 3 present dependencies of the average linear thermal load of the fuel rod out of fuel assemblies 1 and 2, respectively, vs. time for Case 2.

Figures 4 and 5 present calculated maximum fuel temperature of the fuel rod out of fuel assemblies 1 and 2, respectively, vs. time for Cases 2 AVER.

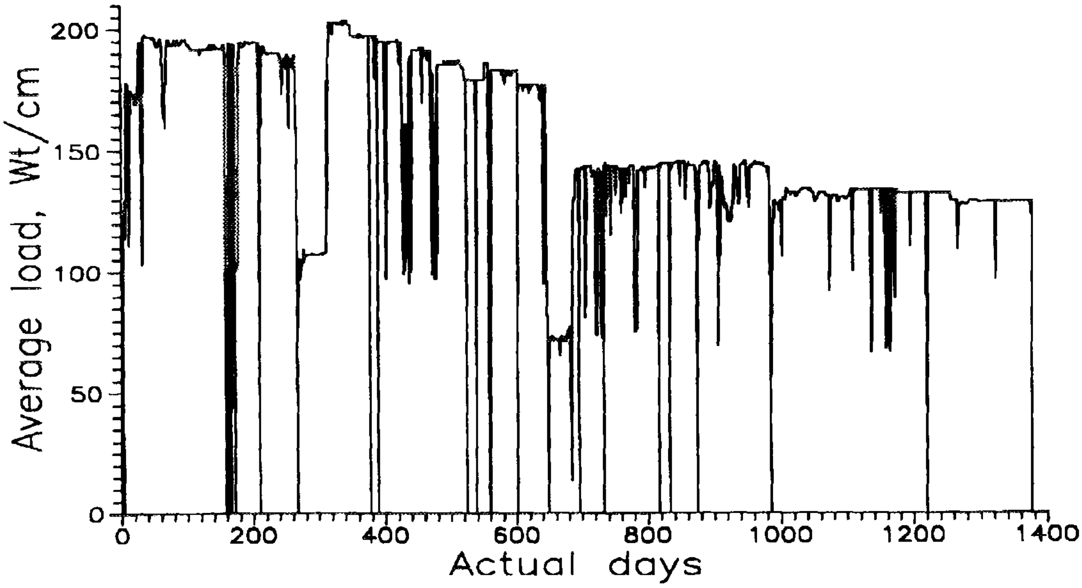

Figure 2 Dependence of the fuel rod average linear thermal load (assembly # 1) vs. time for case 2.

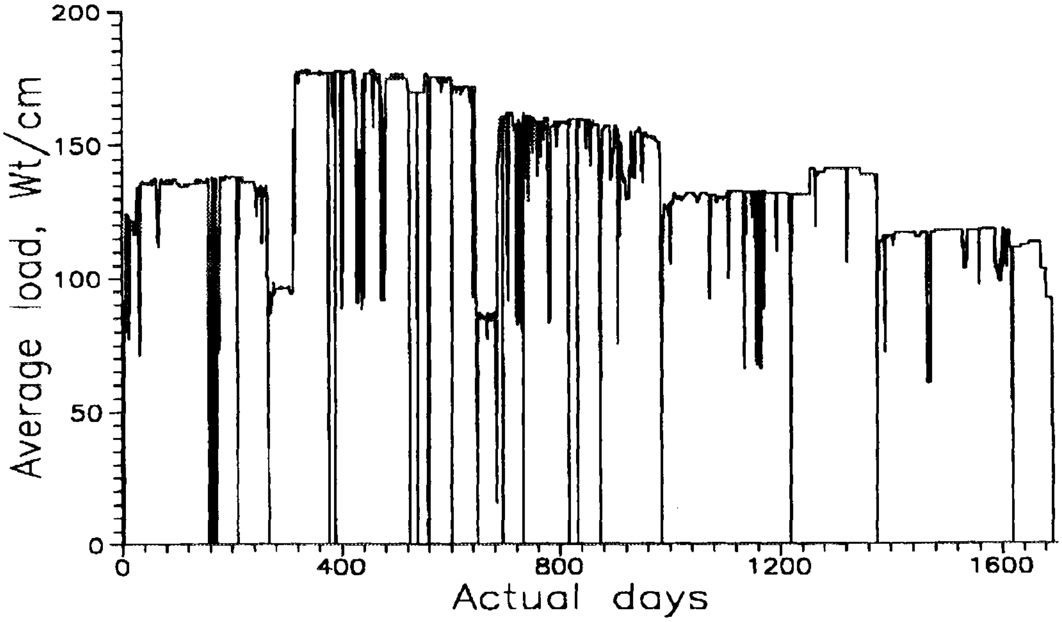

Figure 3 Dependence of the fuel rod average linear thermal load (assembly # 2) vs. time for case 2.

Figures 6 and 7 present calculated fuel temperature of the fuel rod out of fuel assemblies 1 and 2, respectively, vs. time for Cases 4 AVER.

Figures 8 and 9 present calculated values of the gas media pressure inside the cladding of the fuel rod out of assemblies 1 and 2, respectively, vs. time for Cases 2 AVER.

The calculation results are listed in Tables 1 - 2.

3 Conclusions

The results of thermal-physical calculations of fuel rods of fuel assemblies, which have achieved deep burnup during 4-year (> 46 *MWd/kg*) and 5-year (> 48 *MWd/kg*) fuel cycles of the 3rd Unit of Kolskaya NPP are presented in the paper.

For the calculations the average fuel rod in the fuel assembly and the fuel rod with the maximum burnup were selected.

The preliminary comparison of the calculation results with the results of post-irradiation examinations [5] (fission gas release from 0.7 to 1.3% for the fuel rods of fuel assembly 1; form 1.5 to 3.7% for the fuel rods from assembly 2; pressure inside the cladding at the end of campaign at normal conditions is from 0.87 to 1.13 *MPa* for the fuel rods of fuel assembly 1 and from 0.95 to 1.4 *MPa* for fuel rods of fuel assembly 2; decrease of the cladding radius is up to 35 μm for the fuel rods of fuel assembly 1 and up to 30 μm for the fuel rods of fuel assembly 2, etc.) showed very satisfactory agreement. In the future the improvement of the model for calculation fission gas release and creep of the cladding is planned on the basis of the results of post-irradiation examination.

The results show that the fuel rod completely preserves its working ability; fuel temperature does not exceed 1300°C; fission gas release does not

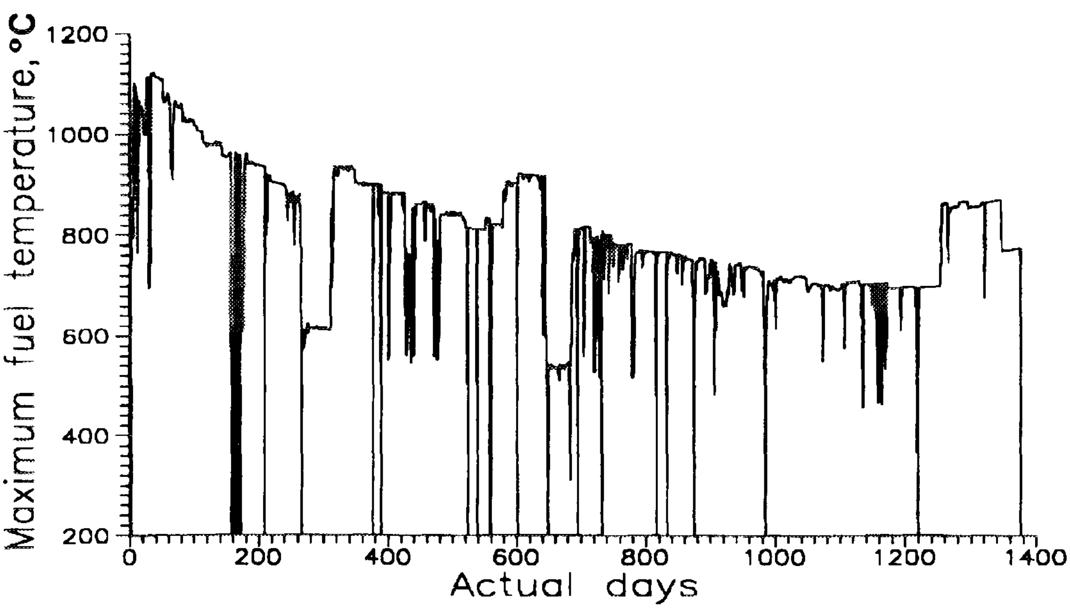

Figure 4 Calculated maximum fuel temperature (assembly # 1) vs. time for case 2 - aver.

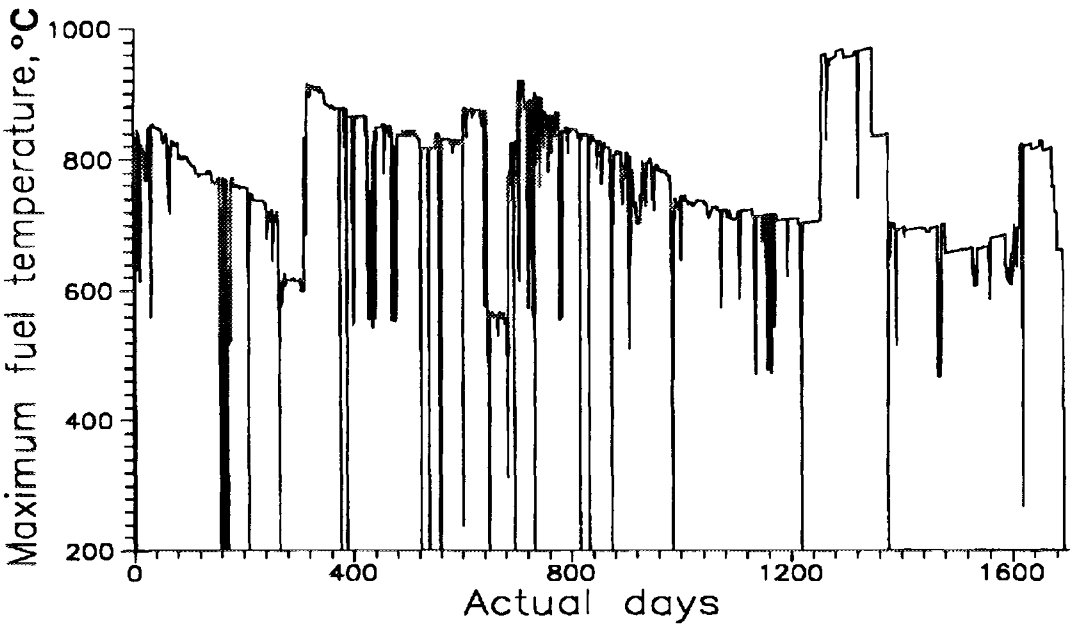

Figure 5 Calculated maximum fuel temperature (assembly # 2) vs. time for case 2 - aver.

Table 1 Calculated values for assembly # 1

	Case 1	Case 2			Case 3	Case 4
variant	aver	max	aver	min	aver	aver
Burn	46.7	52.2	52.2	52.5	54.3	52.2
B_{max}	51.8	57.9	57.8	58.1	60.1	57.8
T_{max}, °C	954	1265	1121	932	1154	1171
T_t, °C	545	649	571	509	583	586
F, %	0.56	3.13	1.17	0.73	1.96	1.53
P_{max}, *MPa*	2.02	3.02	2.63	2.70	3.19	2.87
P_c, *MPa*	0.83	1.11	1.00	1.06	1.17	1.07
dr_f, μm	35		48		53	
dr_c, μm	-27		-32		-34	

Table 2 Calculated values for assembly #2.

	Case 1	Case 2		Case 3	Case 4
variant	aver	max	aver	aver	aver
Burnup	49.6	57.8	57.7	60.0	57.7
B_{max}	55.8	64.9	64.8	67.4	64.8
T_{max}, °C	917	1209	972	1028	1036
T_t, °C	511	631	549	563	565
F, %	0.63	7.88	2.99	5.56	4.00
P_{max}, *MPa*	2.07	5.55	3.60	5.12	4.17
P_c, *MPa*	0.87	2.03	1.41	1.97	1.60
dr_f, μm	44		64	70	
dr_c, μm	-24		-30	-31	

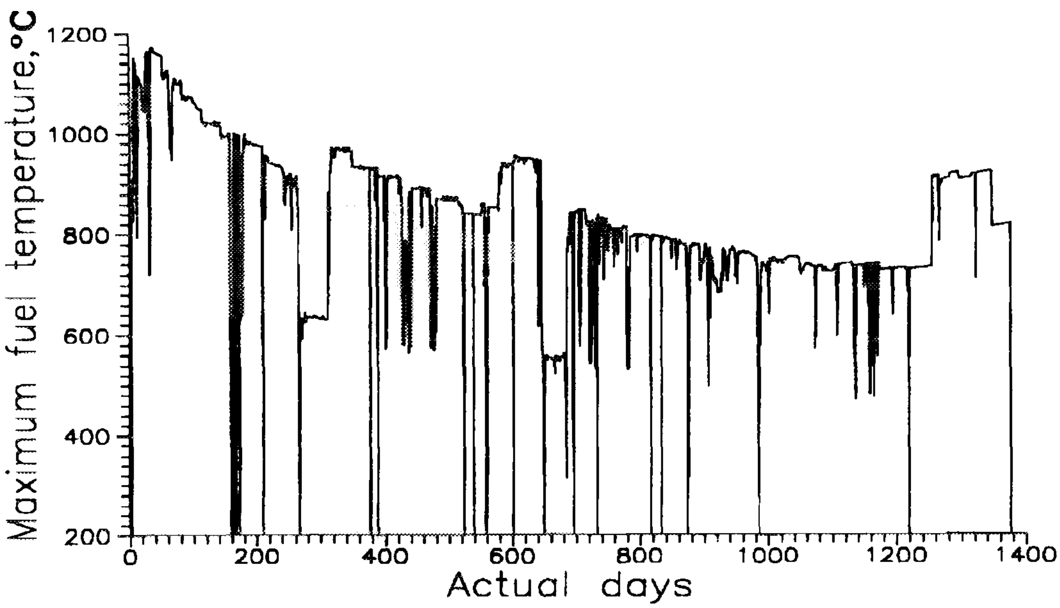

Figure 6 Calculated maximum fuel temperature (assembly # 1) vs. time for case 4 - aver.

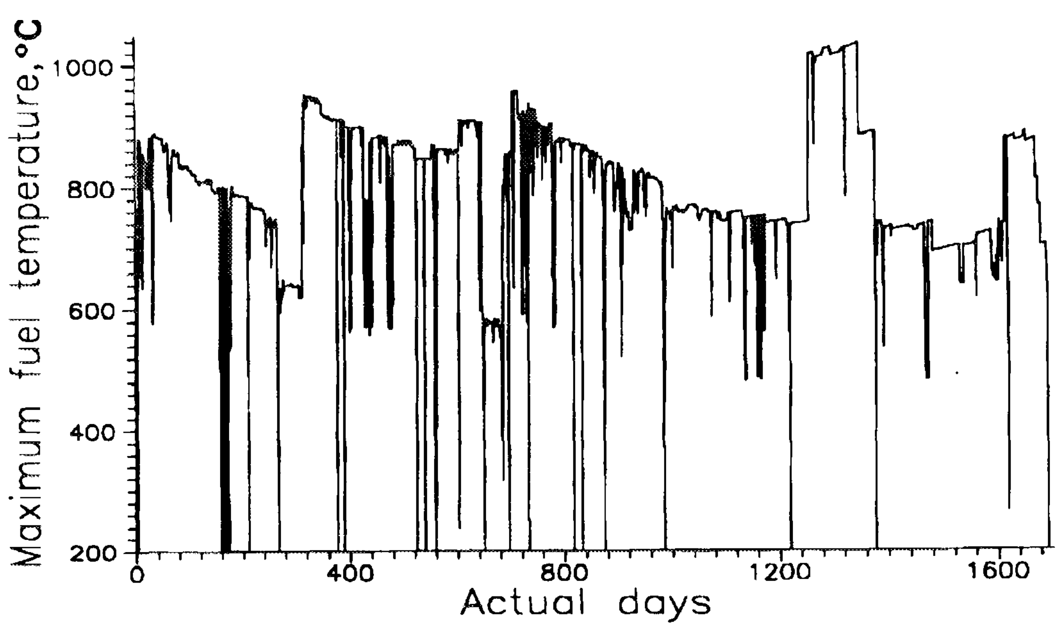

Figure 7 Calculated maximum fuel temperature (assembly # 2) vs. time for case 4 - aver.

exceed 4%; maximum gas pressure inside the cladding does not exceed 4 *MPa*; gas pressure inside the cladding at the end of campaign does not exceed 2 *MPa*.

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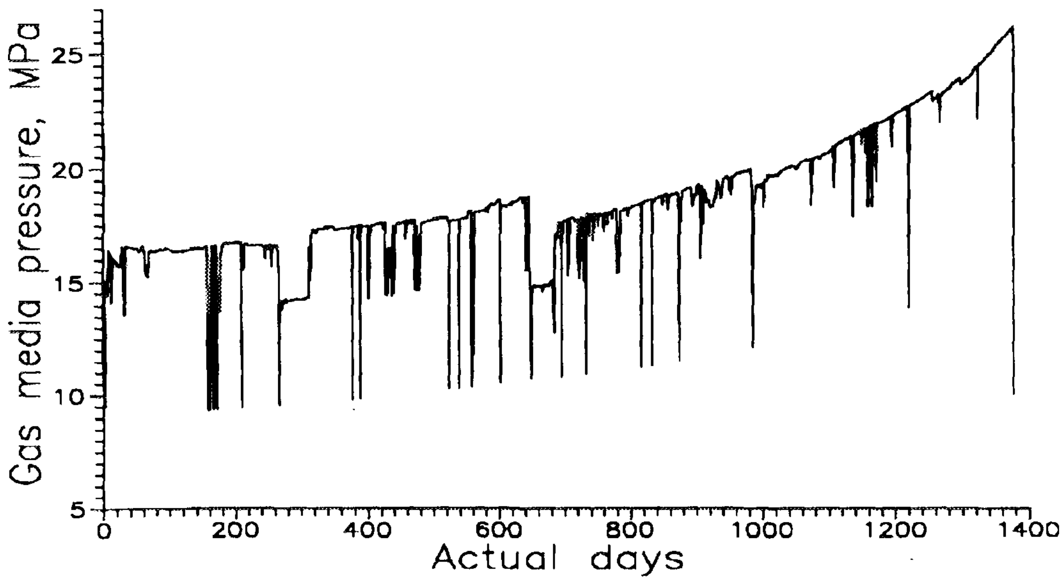

Figure 8 Calculated fuel rod gas pressure (assembly # 1) vs. time for case 2 - aver.

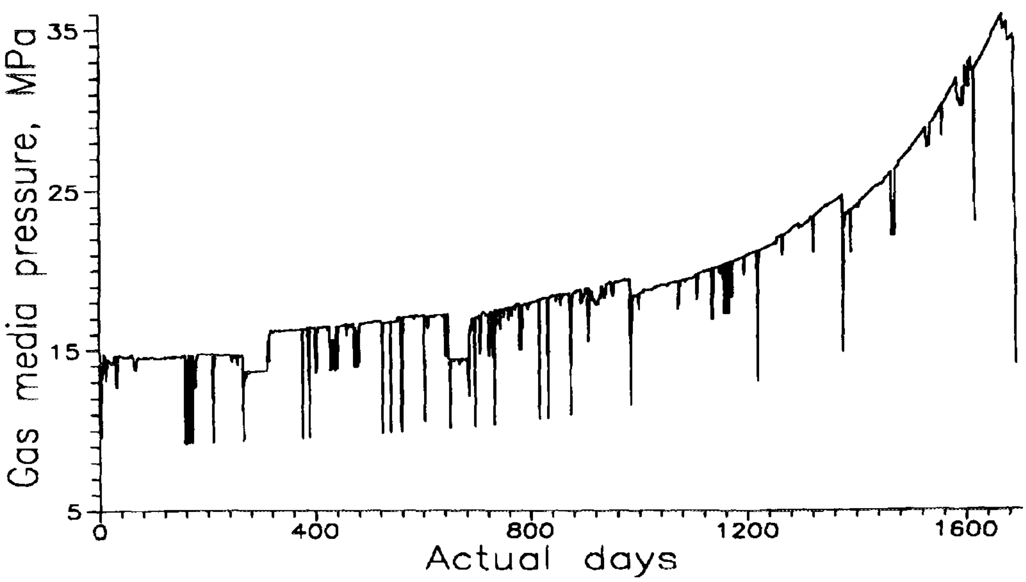

Figure 9 Calculated fuel rod gas pressure (assembly # 2) vs. time for case 2 - aver.

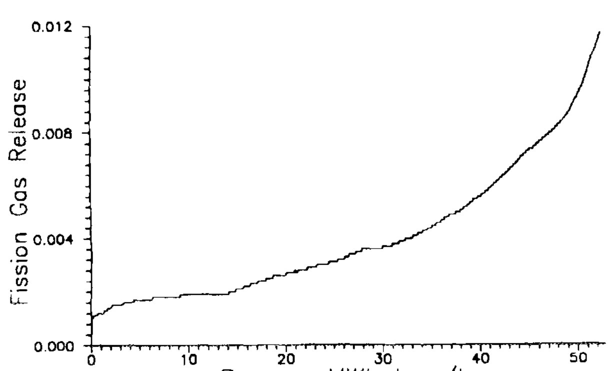

Figure 10 Fission gas release (assembly # 1) vs. burnup for case 2 - aver.

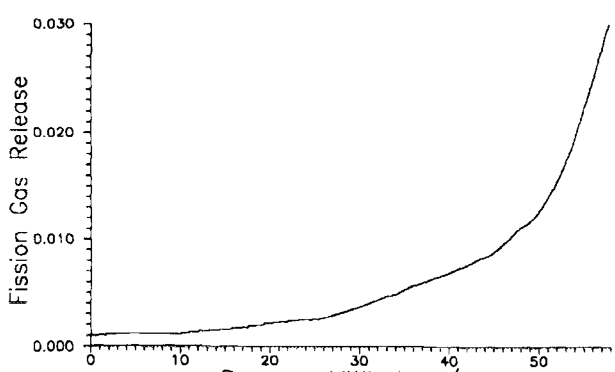

Figure 11 Fission gas release (assembly # 2) vs. burnup for case 2 - aver.

Analysis of VVER-440 Fuel Performance Under Normal Operating Conditions

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1 Introduction

VVER-440 type reactors are similar to western LWR's in many aspects. The only difference is the hexagonal fuel assembly configuration. While dealing with fuel rods, there is no significant difference, effecting thermal parameters. Fundamentally, the phenomena governing nuclear fuel behavior are expected to be the same for LWR and VVER-440 reactors. Although FRAPCON-2 is specifically used to simulate LWR fuel behavior, it also can be used for VVER-440 reactors.

FRAPCON-2 computer code has been developed for the calculation of thermal and mechanical response of typical water cooled reactor fuel rod under steady-state and long term transient operations. The code calculates the heat transfer between fuel rod components, elastic and plastic deformation characteristics, fission gas release and cladding oxidation.

2 Methodology

FRAPCON-2 computer code is used in this study to analyze the fuel behavior under typical VVER-440 reactor operating conditions. The code is capable of utilizing different models for mechanical analysis and gas release calculations. Heat transfer calculations are accomplished by a collocation technique by the method of weighted residuals. Mechanical analysis calculations are performed by finite element techniques. Temperature and burnup element properties are evaluated using MATPRO package.

Material properties of Zr-1%Nb which is used as cladding in VVER-440 reactors are not provided in this package. However, Zr-1%Nb alloy has comparable mechanical properties with annealed Zircaloy-4 [1, 2]. Furthermore, oxidation characteristics of both alloys are not very different. Hence, Zircaloy-4 is used as a substitute for Zr-1%Nb.

Mac-Donald-Weisman gas release model is extensively used in the calculations. This model determines gas release in two steps: escape of gas atoms form the fuel matrix and release of gas atoms trapped at grain boundaries and dislocations. Probability of trapped particle release per unit time and the fraction of the released gas atoms reaching to the surface are evaluated as functions of local temperature and density.

Two different mechanical analysis models are used in present calculations. Both models consider small displacement of the fuel and cladding. The first model FRACAS-I considers rigid pellet and neglects stress induced deformation in the pellet. Thermal expansion, swelling and densification are the only sources for pellet deformation. Fuel is assumed to be isotropic. Creep deformation of the fuel is neglected in this model. On the other hand, the second model FRACAS-II considers deformable fuel pellets. Therefore, stress induced deformations are also included to the model.

It was assumed that the reactor was operated for total 920 days that is the total length of three consecutive cycles. The burnup achieved as a result of this operation is 42,000 *MWd/tU*. Characteristics of the standard fuel rod used throughout the calculations are given in Table 1. The power and burnup history is shown in Figure 1. The reactor is shut down at the end of each cycle.

The fuel element is divided into 7 axial and 11 radial nodes in fuel region. Axial power profile is assumed to have a cosine shape. Axial-power profile peak-to-average ratio is taken to be 1.4.

Table 1 Parameters of VVER-440 used in analysis

Parameter	Value
Core	
Core length	2.5 *m*
Core diameter	2.88 *m*
Average burnup	28600 *MWd/t*
Maximum burnup	42000 *MWd/t*
Lattice	Triangular
Fuel rods	
Rod pitch	122 *mm*
Outer diameter	9.1 *mm*
Cladding thickness	0.65 *mm*
Active length	242 *cm*
Average linear power	127 *W/cm*
Cladding material	Zr-1%Nb
Fill gas	Helium
Fuel pellets	
Outer diameter	7.55 *mm*
Inner diameter	1.4 - 1.6 *mm*
Density	10.4 g/cm^3
Material	UO_2
U-235 enrichment	2.4 + 3.6%
Primary circuit	
Coolant pressure	12.4 *MPa*
Coolant temp. at inlet	268°C
Coolant temp. at outlet	301°C
Coolant flow rate	42000 m^3/h

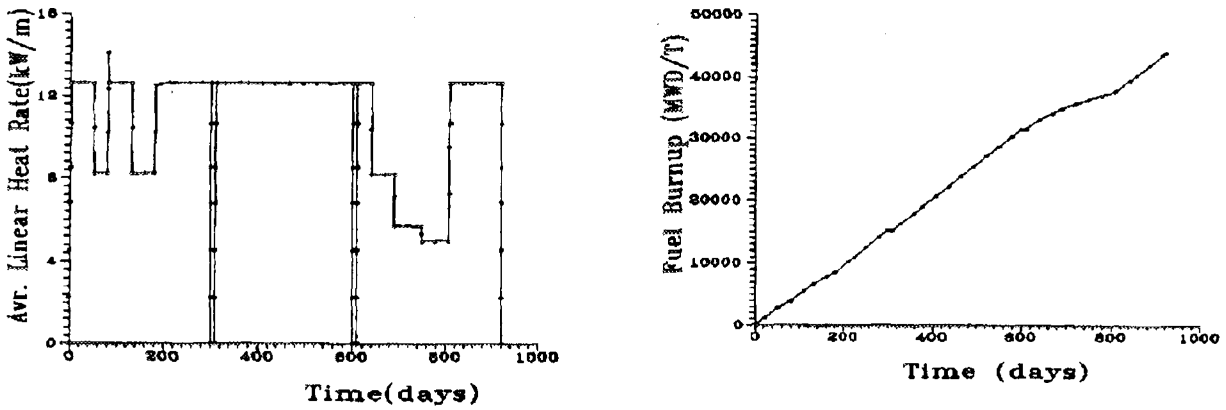

Figure 1 Power history and corresponding burnup

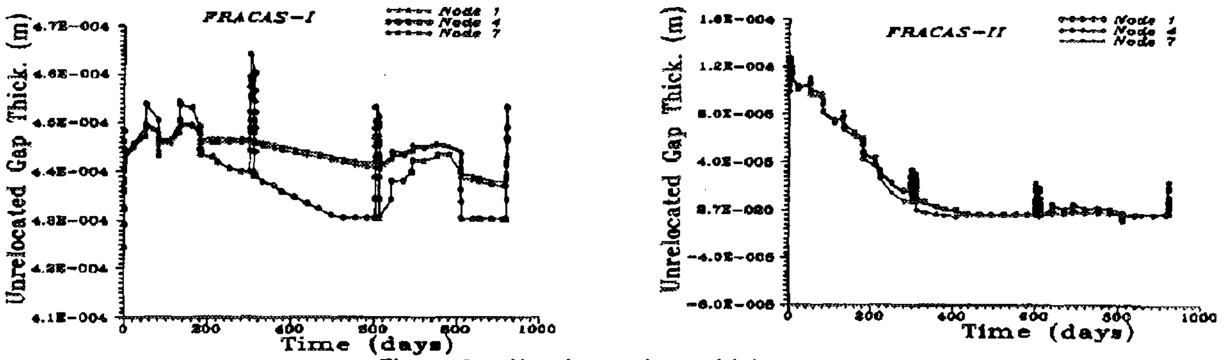

Figure 2 Unrelocated gap thickness

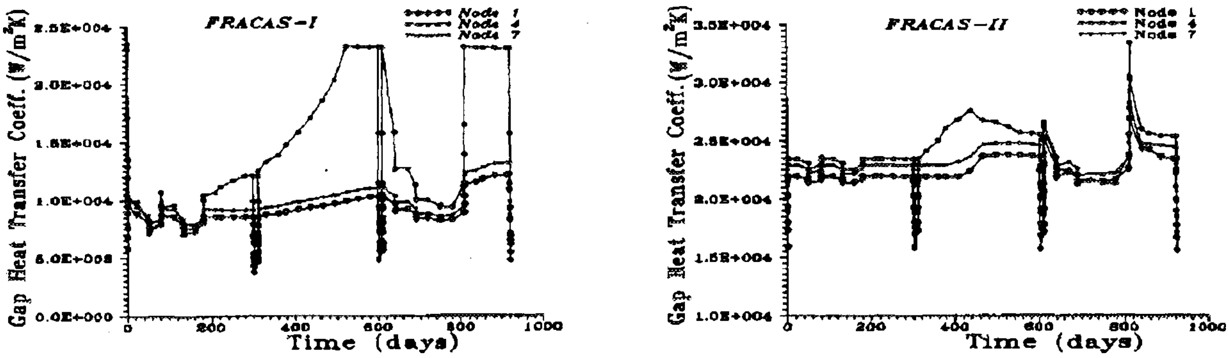

Figure 3 Gap heat transfer coefficient

3 Conclusions

Results of the fuel rod behavior analysis are represented for the three axial nodes: bottom node (Node 1), central node (Node 4) and top node (Node 7). The variation of several fuel rod parameters are investigated through the prescribed power history. Results obtained by FRACAS-I and FRACAS-II model are used to make a comparison.

The most significant effect of the selected mechanical models are observed for the unrelocated gap thickness (Fig. 2). FRACAS-I results in an average gap thickness of 0.45 *mm*. This is greater than the initial gap thickness. This might be due to underestimation of radial expansion originated from axial stresses. In this case densification overcomes the other factors and the central node shows the minimum gap thickness which is in the order of 0.43 *mm*. On the other hand, FRACAS-II produces more reasonable results. Steadily decreasing gap size is observed during the initial operation. Then gap is almost closed after about 200 days of operation in all axial locations. Naturally, this observation affects the remaining fuel rod parameters.

The second parameter which is directly related to the gap size is the heat transfer coefficient of the gap (Fig. 3). Heat transfer coefficient given by FRACAS-I is a direct function of the gap size. The maximum heat transfer coefficient is at the central node and it is about 24000 W/m^2K. Other nodes have significantly lower heat transfer coefficients. FRACAS-II results in almost closed gap regime, heat transfer coefficients are relatively high and the maximum heat transfer coefficient could go up to 33000 W/m^2K at the central node. All other nodes have relatively close heat transfer coefficients.

Gap size observations may be complemented by fuel and clad axial elongation calculations (see Figs. 4, 5). Fuel is rather free for expansion and the maximum elongation is calculated to be about 3% for the central node in FRACAS-I model. However, FRACAS-II results in a maximum of 2% axial elongation at the same location. This is due to the effect

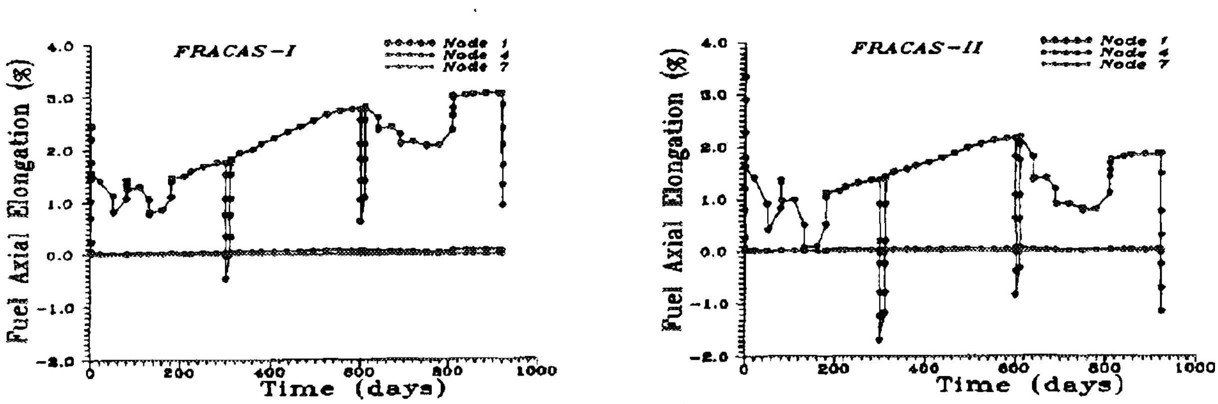

Figure 4 Fuel axial elongation

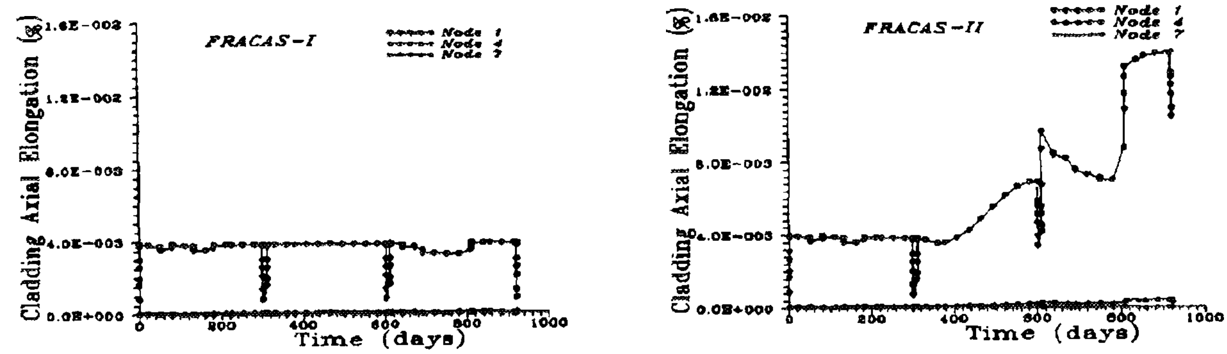

Figure 5 Cladding axial elongation

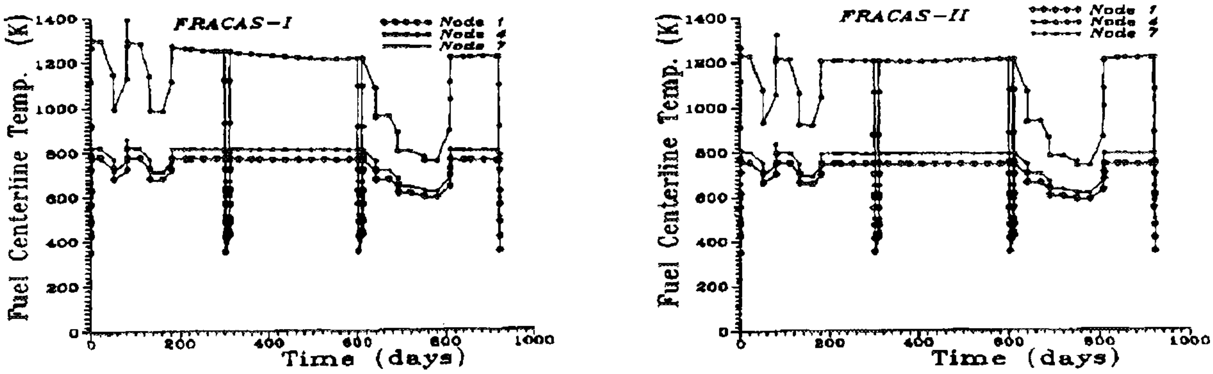

Figure 6 Fuel centerline temperature

of axial stress produced in the fuel. The result of this observation is greater radial expansion and smaller gap size for the FRACAS-II model. Both models produce negligible axial elongation at the top and bottom nodes.

Fuel centerline temperature is another parameter to indicate fuel performance (Fig. 6). Centerline temperature in both mechanical model selections closely follow the prescribed power history. Since gap heat transfer coeffic, centerline temperature prediction is slightly higher in this case. On the average, fuel centerline temperature is 1250 K, whereas temperatures at the top and bottom nodes are about 800 K. In all cases, temperatures are considerably below the melting temperature of fuel material and does not cause any safety problem. The results of low fuel temperatures is the observation of relatively low fission gas release (Fig. 7). Only 1% of all constituents of filling gas belongs to fission products. The maximum gas pressure observed during the operation is about 7 *MPa* (Fig. 8).

Another important parameter is the zirconium oxide thickness at cladding surface (Fig. 9). This result is independent of the mechanical model used in calculations. Zirconium oxide thickness at the central node is about 8 microns at the end of the operation. Since cladding surface temperature increase only up to 600 K, it does not promote excessive oxidation. The thickness of oxide layer is very low compared to an acceptable safety limit of 17% of the total cladding thickness. Stored energy in cladding is below limits for safe operation (Fig. 10).

Results show that calculations with deformable pellet approximation with FRACAS-II model could provide better information for the behavior of a typical fuelrod. Calculations indicate that fuel rod failure is not observed during the operation. All fuel rod parameters investigated in this study are found to be within the safety limits. However, these calculations should be extended for transient conditions such as a loss of coolant accident in order to make better assessment for reactor safety.

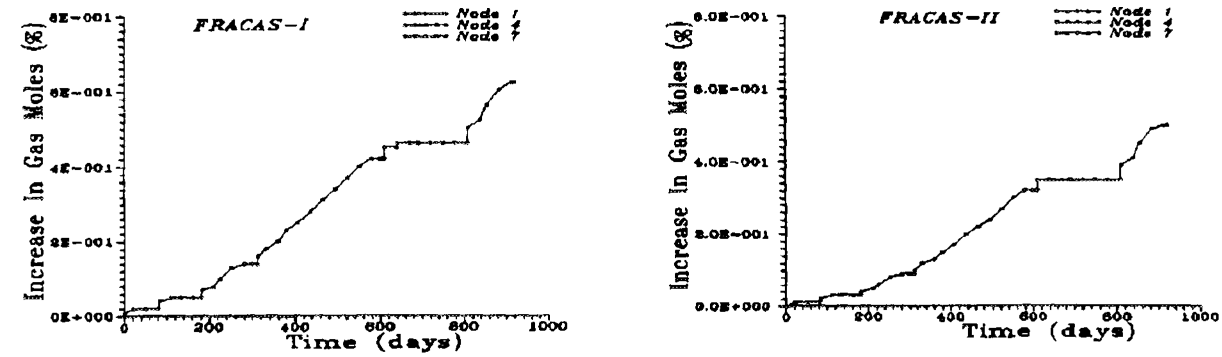

Figure 7 Increase in gas moles

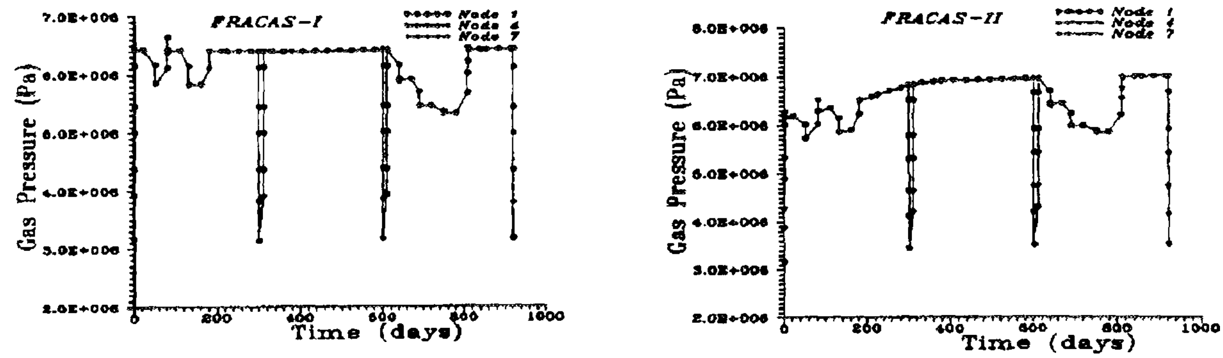

Figure 8 Gap gas pressure

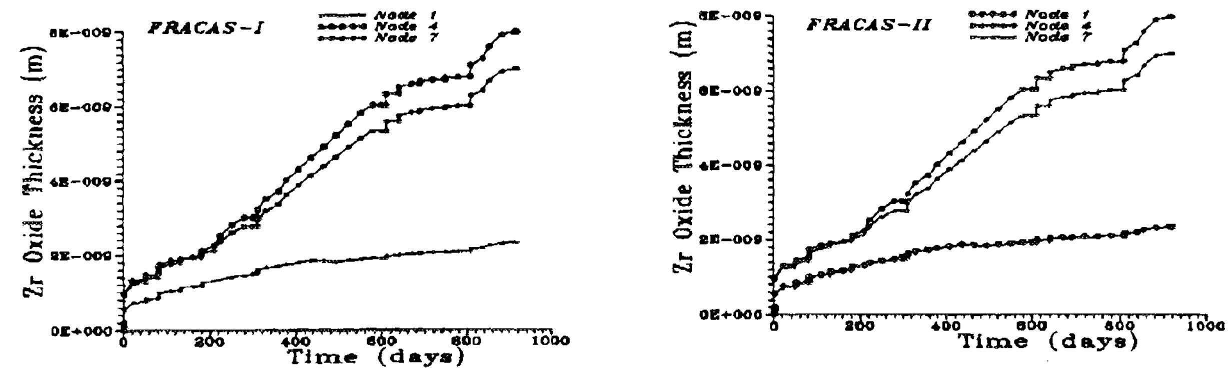

Figure 9 Zr oxide thickness

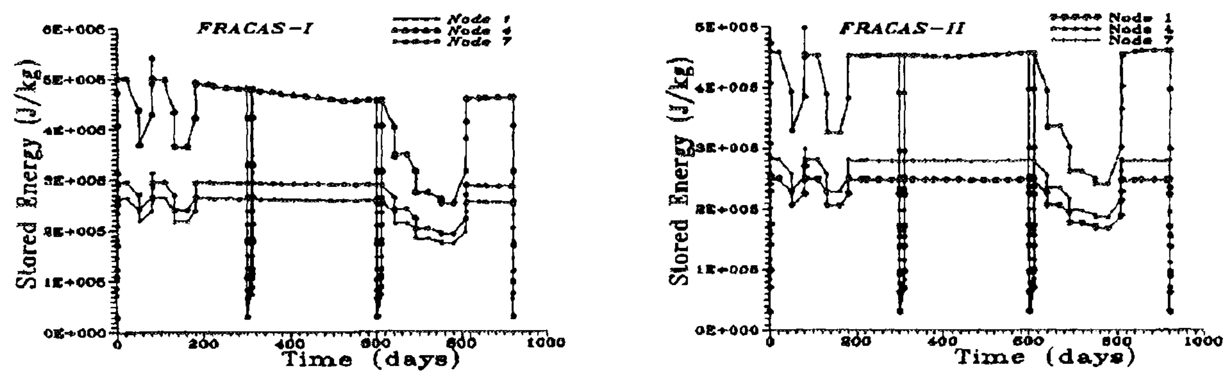

Figure 10 Stored energy in cladding

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Experimental Support of VVER-440 Fuel Reliability and Serviceability at High Burnup

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Nuclear power plants with the VVER-440 reactor type are successfully operated in Russia and abroad, demonstrating the high level of reliability. By now the total operation time of the VVER-440 units covers more than 300 reactor years. The average fuel burnup achieves 36 *MWd/kg* at the average enrichment 3.6 *mass%* in U-235 and three-year fuel campaign.

However, the improvement of NPP economic indices along with the high level of safety and reliability of their operation is important for the successful development of atomic power. Therefore, fuel burnup increasing to a great extent defining the fuel cycle price is a problem of large practical and scientific importance.

In 1990-1991 the operation of fuel assemblies was successfully completed at the third unit of Kola NPP during four-year and five-year fuel cycles up to the average burnup 46 and 48 *MWd/kg*, respectively.

To estimate experimentally the fuel assembly states and to define their serviceability allowance a pair of them was carried to the RIAR hot laboratory and studied by the non-destructive and destructive examination methods. Results of the examination are given in this paper.

1 Fuel Assembly Characteristics and Operation History

The fuel assemblies examined were manufactured in May 1986, with following characteristics:

- Zr-1%Nb alloy was used as a material of fuel element claddings and end plugs. The sintered pellets of uranium dioxide enriched in ^{235}U up to 4.4% served as a fuel. The wrapper tube of a cross flats dimensions 144 *mm* was made of Zr-2.5Nb. Head, tail and spacing grids are made of Cr18Ni10Ti steel.
- the value of helium pressure was 9.120 *mm* and inner diameter - 7.755 *mm*. The pellet outer diameter in an average of 7.560 *mm* and the inner channel diameter - 1.65 *mm*. An average outer diameter of the pellets was 7.560*mm*, internal channel diameter - 1.65 *mm*. The fuel height diameter varied from 10.0 to 12.4 *mm*.

Fuel assembly No.1 was operated during 5 - 8 core fuel loadings and fuel assembly No.2 - during 5 - 9 core fuel loadings of the Kola NPP - unit 3, i.e. during four and five years, respectively.

The inlet reactor coolant temperature and its heating during different operation periods did not exceed 256°C and 30°C, respectively. The primary circuit pressure was 12.5 *MPa*. The average fuel burnup was 46.20 and 48.18 *MWd/kg* for fuel assembly No. 1 and No. 2, respectively (calculations were performed by BIPR-7 program).

It has been experimentally found that the maximum fuel burnup of fuel assemblies No.1 and No.2 achieved was 58.3 and 64.0 *MWd/kg*, respectively.

The maximum linear energy release of fuel assembly No.1 and No.2 did not exceed 319 and 264 *W/cm* and decreased up to 110 and 88 *W/cm* at the end of operation.

From the results of leak testing the fuel assemblies did not include unsealed fuel elements.

2 Post-Reactor Investigation Results

A number of parameters determining the fuel serviceability was estimated in the process of post-reactor investigations. Among these parameters are as follows:

- fuel assembly and element dimensional stability;
- fuel element claddings corrosion and hydriding;
- fission gas release and inner fuel element pressure;
- fuel macro- and microstructure.

2.1 Fuel Assembly Overall Dimensions Change

The hexagonal wrappers of both fuel assemblies did not change their dimensions significantly after operation.

The maximum value of the average cross flat dimension of the wrappers for fuel assembly No.1 and 2 was 144.8 and 144.9 *mm*, respectively, at the initial dimensions $144.2^{+14}_{0.5}$ *mm*. The greatest value of axes defection vector of fuel assembly No.1 was 0.7 *mm* at the mark 1000*mm* from the wrapper top. On the average, the length of wrapper ribs increased by 3.5 *mm* (0.13%) and 3.9 *mm* (0.14%) for fuel assemblies No.1 and 2 respectively.

2.2 Fuel Element Length Change

The length change depending on burnup of 35 fuel elements in fuel assembly No.1 and 2 is presented in Fig. 1. One can see that no statistically significant relation between elongation and burnup is available in 70 fuel elements placed on both fuel assemblies diagonals.

The average values of fuel element elongation in fuel assembly No.1 and 2 are in good agreement with the relation obtained previously for the VVER-440 fuel and given in Fig. 2 [1]. The fuel elements are seen to elongate proportionally to the fuel burnup at the rate 0.01% per 1 *MWd/kg*. The average fuel element elongation over a range from 13 to 50 *MWd/kg* is well described by the linear function of the burnup:

$\Delta L = 0.013B - 0.135$

where: ΔL - average fuel rod elongation, %; B - fuel burnup, *MWd/kg*.

2.3 Fuel Element Diameter Change

The VVER-440 fuel element diameters are decreased as a result of cladding creep under coolant pressure 12.5 *MPa*. The diameter variation along the fuel element length is presented in Fig. 3. The average diameter decrease in the central part of fuel element compared to that of gas volume region comes to 0.07 *mm* (0.75%) for fuel assembly No.1 and 0.06 *mm* (0.65%) for fuel assembly No. 2.

Along with the smooth diameter decreasing of fuel elements in both fuel assemblies the local diameter increasing of cladding (effect of bamboo stick) was seen with the pitch multiple to the fuel pellets length (10 - 12 *mm*). The amplitude of periodical diameter change was 10 - 30 μm on these areas. The average diameter change of one fuel element of fuel assembly No.1 on the area 1300 - 1500 *mm* is presented as an example in Fig. 4. The fact that the coordinates of the cladding deformation maximums are in agreement with those of the pellets ends was verified by metallographical examinations.

Change of the VVER-440 fuel element outer diameter depending on burnup is presented in Fig. 5 [1]. Prior to burnup 35 *MWd/kg* a decrease of fuel element diameter was observed. At high burnup the diameter practically remained constant. At high burnup a "hot" clearance disappeared between fuel and cladding. Then the cladding diameter was determined by the behavior of swelling fuel with the local diameter increasing at the contact sites with the pellet ends [2].

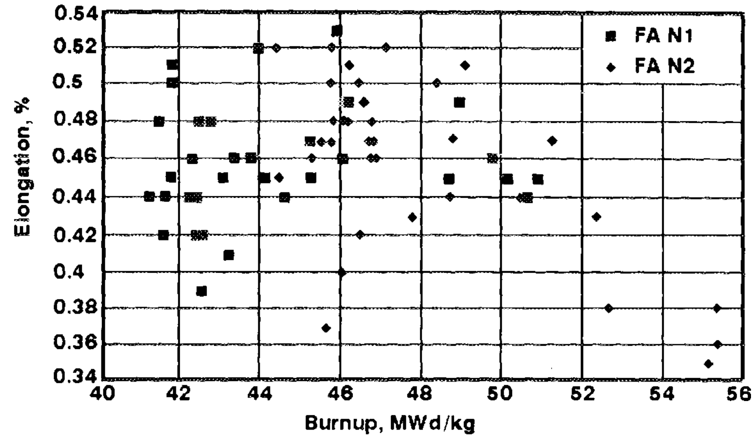


Figure 1 Dependence of fuel rod elongation on fuel burnup

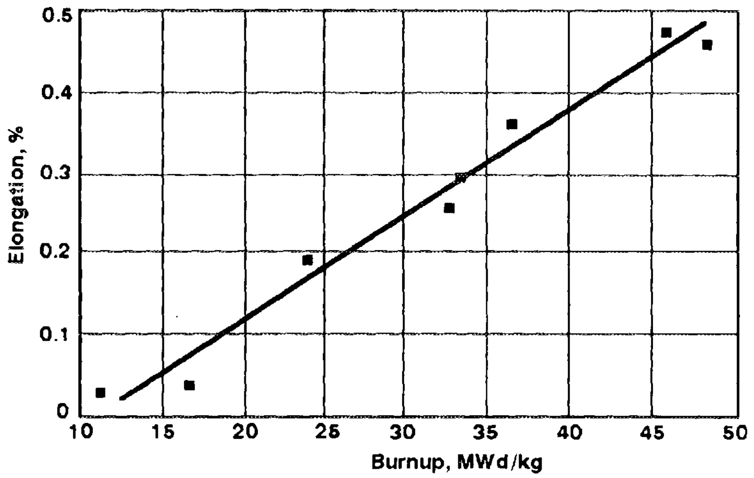


Figure 2 Dependence of fuel rod elongation on average fuel burnup

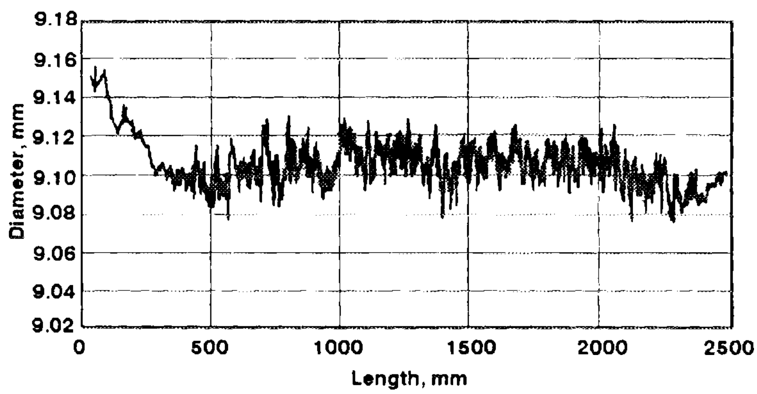


Figure 3 Distribution of fuel rod diameter size along length: fuel rod No. 7 from the fuel assembly No. 2.

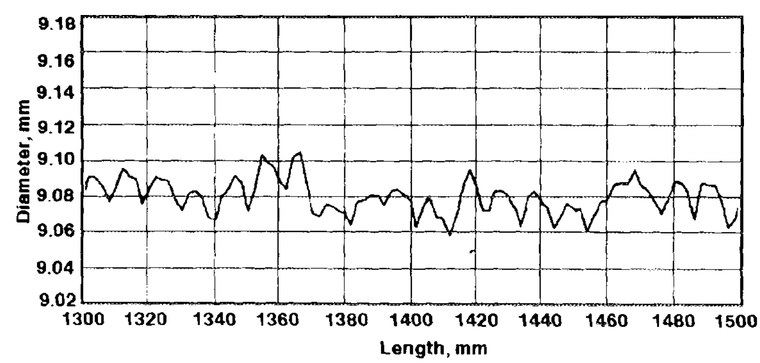


Figure 4 Distribution of fuel rod diameter size along length: fuel rod No. 92 from the fuel assembly No. 2.

The assessment of residual cladding strain by Birger method [3] allowed to calculate the tangential strain arisen at close contacts with fuel. The strain accounts for 38 - 50% from conditional yield strength at operating temperature 350°C.

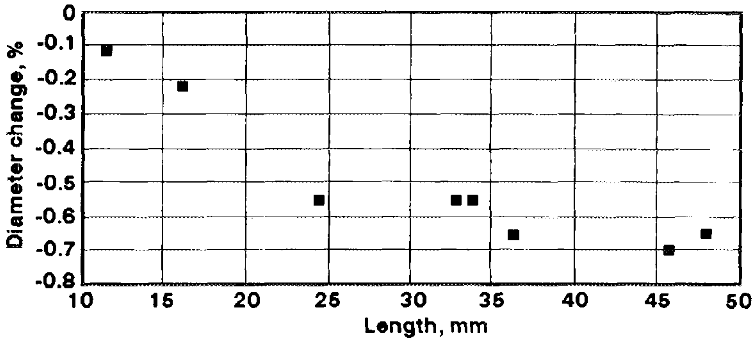


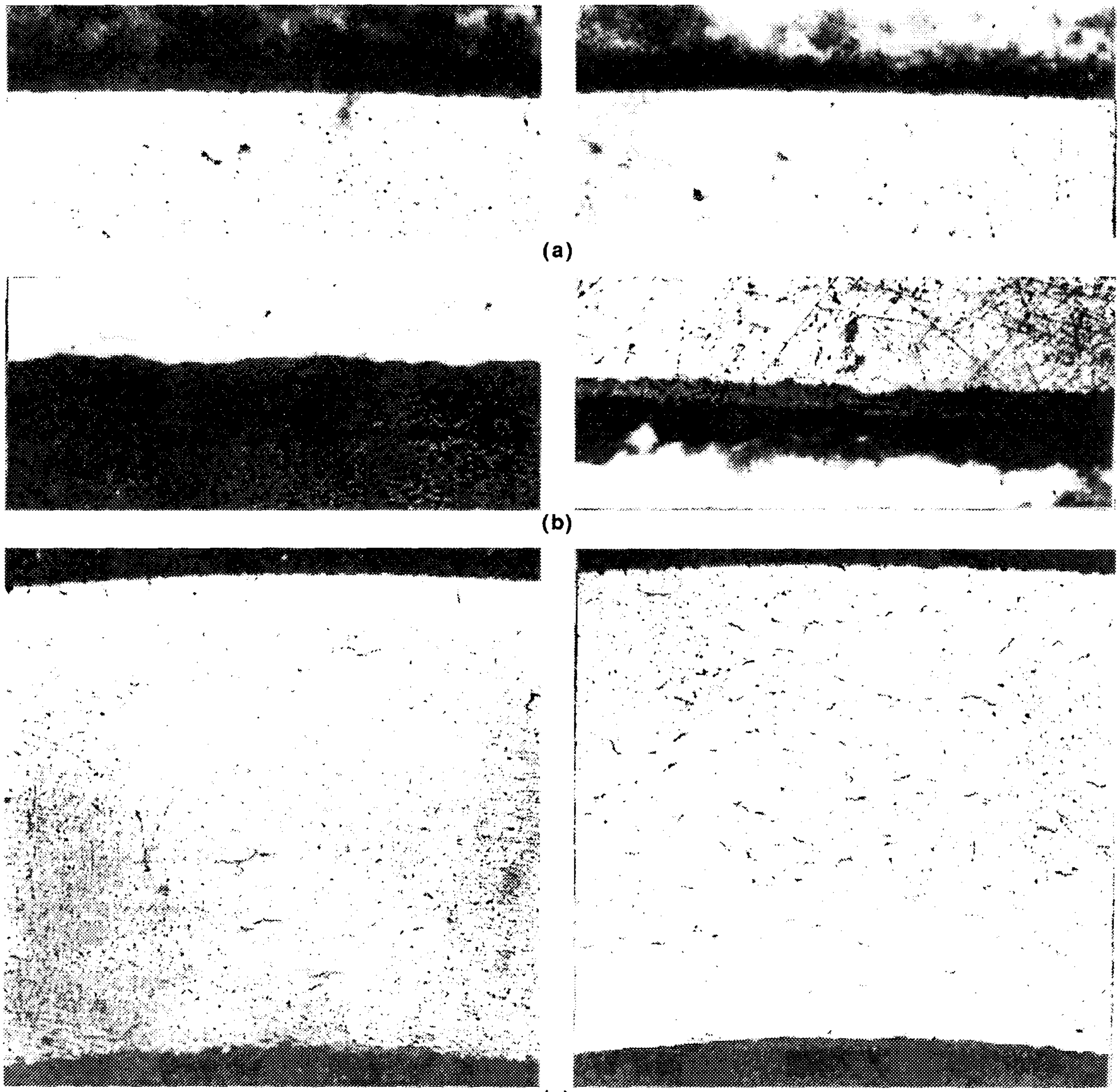
Figure 5 Dependence of fuel rod diameter change on burnup

2.4 Fuel Element Cladding Corrosion and Hydriding

Results of the investigation of fuel element claddings of fuel assembly No.1 and 2 verified the previously ascertained fact of high corrosion resistance of Zr-1%Nb alloy used as the VVER fuel element cladding material [1].

The thickness of uniform oxide film on the outside cladding surface was 3 - 7μm (Fig. 6a). The uniform film was closely bonded with the main material. On the inner surface its thickness (Fig. 6b) varied from 8 to 17 μm.

The size of random oriented hydride releases did not exceed 100 μm (Fig. 6c). Its cladding volume portion was negligible that verified a poor hydriding of the former. Hydrogen qualitative



(a)

(b)

(c)

Figure 6 Oxide film on the outer (a) and inner (b) surfaces (a) of irradiated fuel rods (×400) and characteristic structure (c) of irradiated claddings (×100).

analysis of cladding by spectral-isotopic method showed that the hydrogen content did not exceed $1.3 \cdot 10^{-2}$ *mass%*.

2.5 Fuel Element Cladding Mechanical Properties

The mechanical properties of cladding were determined with the annular specimens of 3 *mm* height. The cross tension test was performed with the deformation rate 1mm/min at temperature 20°C and 350°C.

Short-term mechanical properties of cladding material were practically the same for both fuel assemblies and did not depend on the specimen arrangement in fuel element height.

Yield strength at 350°C was no less than 330 *MPa*, while ultimate strength was no less than 360 *MPa* but at testing temperature 20°C they were 480 and 580 *MPa*, respectively.

The uniform relative elongation practically was the same at temperature 20°C and 350°C and changed over a range from 3.7 to 7.3%. The total relative elongation was less than 13% in all cases.

The results obtained are in good agreement with the previous conclusion on the stability of the short-term mechanical properties of Zr-1%Nb alloy cladding over a wide range of burnups [1].

2.6 Fission Gas Release from Fuel and Gas Pressure

The irradiated fuel elements gas pressure of fuel assembly No.1 and 2 at the romm temperature was within the range of 0.87 - 1.13 and 0.95 - 1.40 *MPa*, respectively (initial pressure 0.6 *MPa*).

The relative gas release from fuel assembly No.1 was 0.7% on the average and 1.5% for fuel assembly No.2. The relative gas release independent of fuel burnup was 0.5% for all previously examined VVER-440 fuel assemblies that the average fuel burnup was from 12 to 37 *MWd/kg* [1]. The gas release as a function of average fuel burnup value in the fuel elements of fuel assembly No.1 and 2 is presented in Fig. 7. One can see that within the range 43 - 60 *MWd/kg* the gas release linearly increases at the range 2.7% per 10 *MWd/kg*. Further experimental verification is required for this conclusion.

2.7 Fuel Macro- and Microstructure

The qualitative fuel structure along the whole length of the investigated fuel element was the same. The pellets cracked by 5 - 8 fragments, but preserved its initial configuration. The central hole diameter did not change under irradiation (Fig. 8).

UO_2 grain growing was not observed in the center of the fuel pellets. The average grain size ranged between 8 and 14 μm from one fuel element to another - that is in agreement with fuel grain size variation in the initial state (Fig. 9).

Results of the density measurement by hydrostatic weighting of 6 - 8 *g* fuel pellet fragments shows that irradiated pellet density decreased by $1.2 \pm 4.0\%$, compared to that of the initial state.

3 Conclusion

Results of post-reactor examination of two VVER-440 fuel assemblies spent at the Kola NPP third unit during 4 and 5 fuel cycles showed that the VVER-440 fuel elements and assemblies serviceability conserved to maximum fuel burnup 64 *MWd/kgU* under operation.

The examination showed that the mechanical fuel pellets-cladding interaction was observed at the average fuel burnup above 45 *MWd/kg* that occurred with increasing the local cladding diameter at the areas of pellets end arrangement (bamboo stick). Again the rate of fission gas release from fuel increased at burnup mire that 45 *MWd/kg*.

However no from this parameters limiting the fuel serviclife, i.e. fuel assembly and rods geometrical stability, cladding corrosion and hydriding and mechanical properties change, fuel swelling and state, fission gas products release achieve the critical value at the indicated fuel burnup.

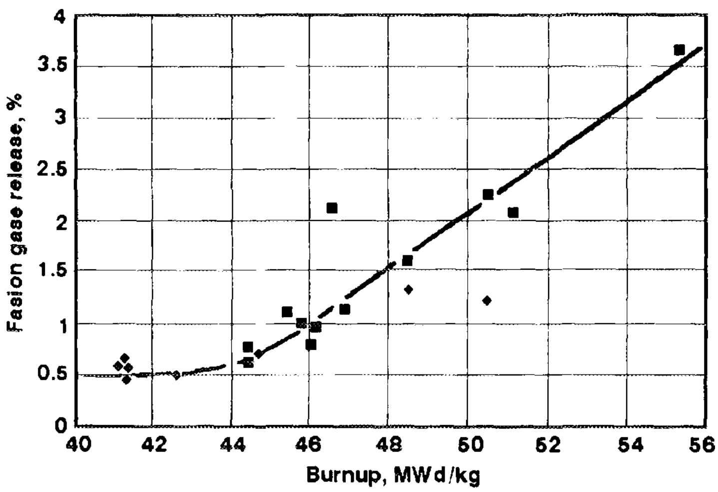


Figure 7 Dependence of fission gas release on burnup

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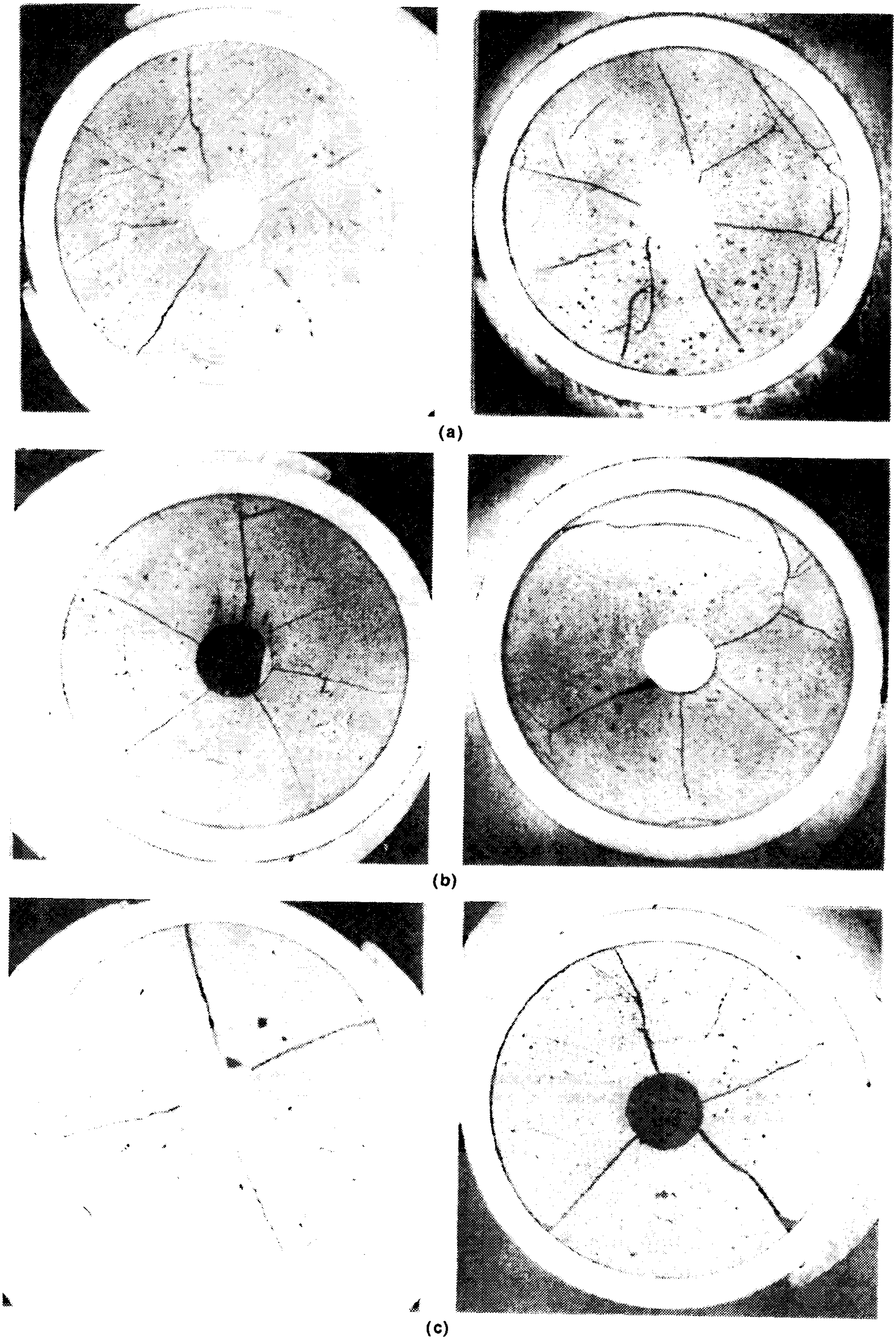

Figure 8 Macrostructure of fuel pellets at 425 (a), 1260 (b) and 2470 mm (c) from top of the fuel rod

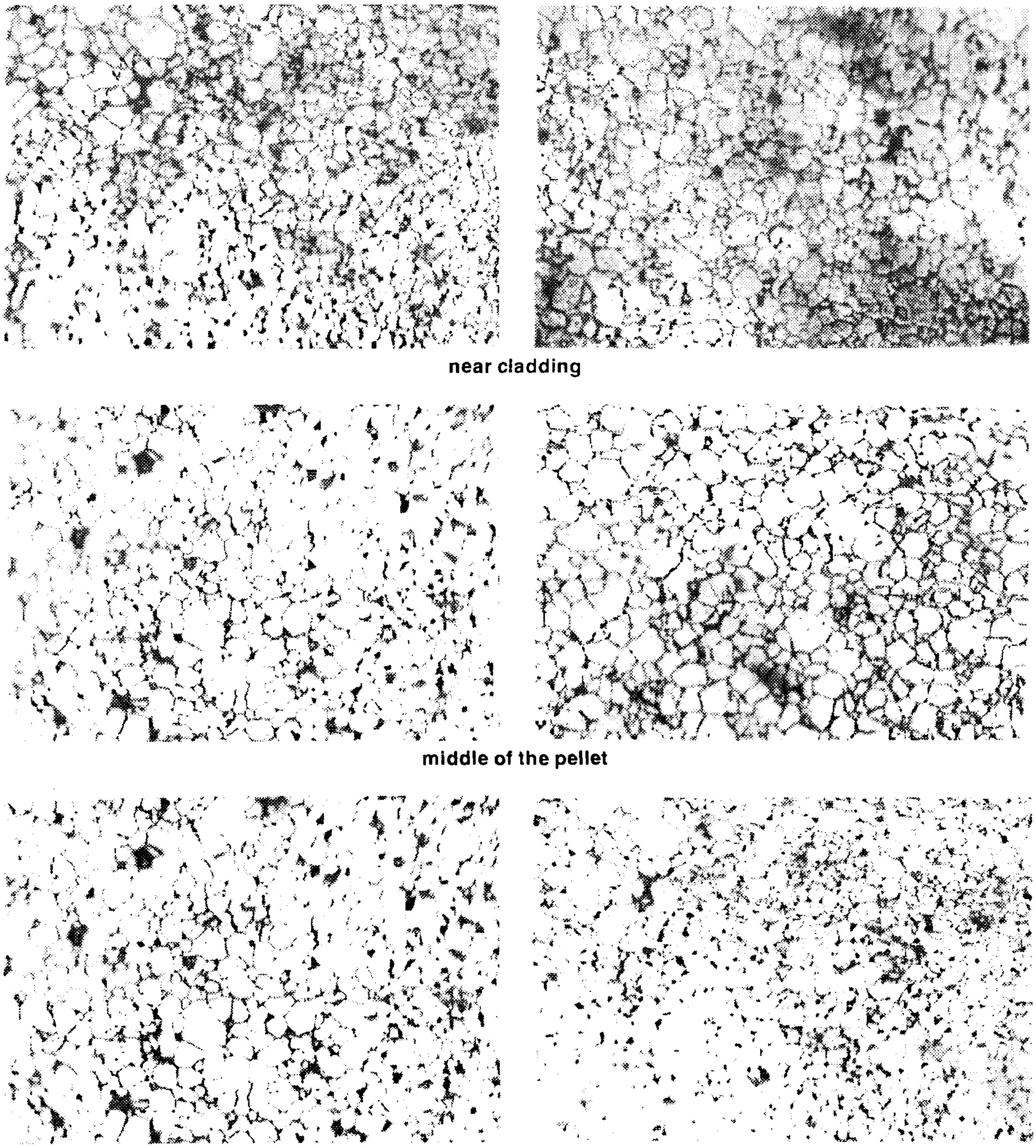

Figure 9 Microstructure of fuel pellets irradiated up to burnup of 54 MWd/kg (FA No. 1) and up to 62 MWd/kg (FA No. 2) under the local maximum linear power of 319 W/cm (FA No. 1) and 264 W/cm (FA No. 2).

BG9600371

VVER-440 Fuel Rod Performance Analysis with PIN-Micro and TRANSURANUS Codes

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1 Introduction

The reliable prediction of the fuel rod behaviour at normal operational, off-normal and accident conditions is rather important for the safe operation of VVER reactors and for high efficiency of the fuel utilisation.

Since two years activities have been initiated in the Institute for Nuclear Research and Nuclear Energy of the Bulgarian Academy of Sciences to perform thermomechanical calculations and analyses for chosen fuel assemblies (FAs) with different power histories of the Kozloduy NPP VVER-440 units, employing the PIN-micro code [1]. A very important experience has been gained through the participation of the Bulgarian group in the IAEA Coordinated Research Programme FUMEX [2]. This activity has been conducted in co-operation with highly experienced specialists from the Russian Kurchatov's Institute and has been kindly supported by experts from the European Institute for Transuranium Elements. In addition to the analyses made with the PIN-micro code, first analyses were performed utilising the TRANSURANUS VVER version which is under development.

2 The PIN-Micro Code

The PIN-micro code is a steady-state, quasi-two-dimensional code, developed on the base of the GAPCON-THERMAL-2 code [4]. It has been improved and verified against VVER fuel rods by introducing specific material property correlations [5].

The models and correlations included in the PIN-micro code describe the radial temperature distribution of fuel and cladding, the radial deformation of fuel pellets due to thermal expansion, relocation, densification and swelling, fuel grain growth and restructuring, the radial deformation of cladding due to thermal expansion, creep and irradiation growth, the axial elongation of fuel and cladding, the fission gas release (FGR), the gas composition and pressure, the gap size or pellet-to-cladding contact pressure.

The PIN-micro code has been verified by international experiments, as well as on Russian and Czech experiments, specific for VVER. The Russian experiments have been performed in the MR reactor in the Kurchatov's Institute [6]. The behaviour of VVER-440 and VVER-1000 FAs has been investigated by comparing measured and calculated results.

The complexity of the problems which could be solved by PIN-micro was restricted by:

- a simplified mechanistic solution;
- a steady-state solution;
- no cladding failure criteria;
- no model for pellet-cladding interaction (PCI);
- no specific models at extended burnup.

3 The TRANSURANUS Version Used

The original TRANSURANUS code version is described in References [7, 8]. The main features of the code are:

- VVER 1½-D code (superposition of 1-D radial and 1-D axial description), suitable for integral fuel rod thermal and mechanical analysis, taking complicated power histories into account.
- VVER steady-state and transient solutions, accounting for time and burnup dependent processes, such as fuel restructuring and grain growth, densification, creep, swelling, heat transfer coefficient degradation, FGR, etc.
- VVER empirical and semi-empirical modelling, based on experimental observation, as well as physically based description of the fuel rod behaviour.

The present status with emphasis on modelling high burn-up phenomena is given in a contribution to this conference [9]. Although the TRANSURANUS code has been extensively verified using irradiation data from very different sources, the standard version cannot be applied to a VVER fuel rod mainly because of the different cladding material. A first step towards the development of a specific TRANSURANUS-VVER version has been made by introducing VVER cladding data and a specific VVER correlation for the thermal conductivity of the fuel [3]. The most important TRANSURANUS-VVER correlations are the creep rate $\dot{\varepsilon}_{clad}^{creep}$ of the Zr+1%Nb cladding and the thermal conductivity λ_{fuel} for 95% dense fuel:

$$\dot{\varepsilon}_{clad}^{creep} = \left(1 + 7\, e^{-\frac{t}{200}}\right) 36.5\, \sigma_{eff}^{5.4}\, e^{-\frac{25600}{T}} + \tag{1}$$
$$+ 1.346 \cdot 10^{-15}\, \Phi\, e^{-\frac{8500}{T}} \sinh\left(3.703 \cdot 10^{-2} \sigma_{eff}\right)$$

$$\lambda_{fuel} = \frac{1}{3.77 + 0.0258T} + 1.1 \cdot 10^{-6}\, T + \tag{2}$$
$$+ 1.01 \cdot 10^{-13}\, T^3\, e^{7.2\,T}$$

Both correlations have been published by Strijov et al. in 1988. Evidently, eq. (2) is not burn-up dependent. In view of the well established burn-up dependency of the thermal conductivity of UO_2, the TRANSURANUS standard correlation has been used instead of eq. (2).

It is doubtful whether other TRANSURANUS standard models and correlations apply for VVER fuel rods. Consequently, all predictions should be seen as a first attempt rather then a meaningful prediction. Especially fission gas release should be taken with great caution since local fuel data is not available.

4 Calculated Results

4.1 Input Data

For this comparison the two highest loaded fuel rods of the FAs, which were irradiated in VVER-440 with different power histories, have been selected. The most important operational data, such as power histories, axial power profiles and the primary system parameters, have been taken from Reference [10]. For these two FAs, FA1 and FA2, a set of the most probable average values of all geometrical and technological parameters has been used (see Table 1). The power histories of FA1 and FA2 differ because of the different reloading schemes.

The geometrical representation of the considered fuel rods contains 10 axial segments (in both codes) and 20 or 13 radial segments (in PIN-micro and TRANSURANUS, respectively).

On the basis of the two VVER-440 real cases described, a comparison between PIN-micro and TRANSURANUS codes has been performed, using identical input data.

Table 1 Geometric and technological parameters for the selected variables

Central hole diameter	1.6 *mm*
Fuel outer diameter	7.565 *mm*
Cladding inner diameter	7.76 *mm*
Cladding outer diameter	9.15 *mm*
Diametral gap	0.195 *mm*
Gas inner pressure	0.6 *MPa*
Fuel density	10.6 g/cm^3

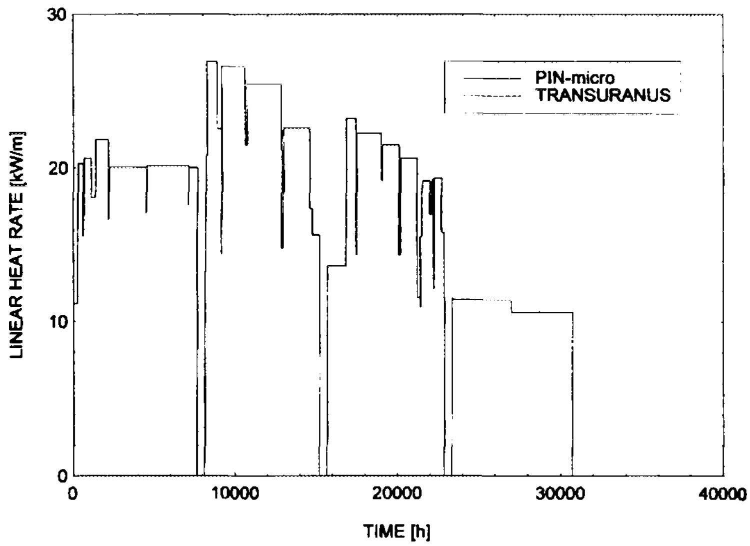

Figure 1 Local linear heat rate, FA1

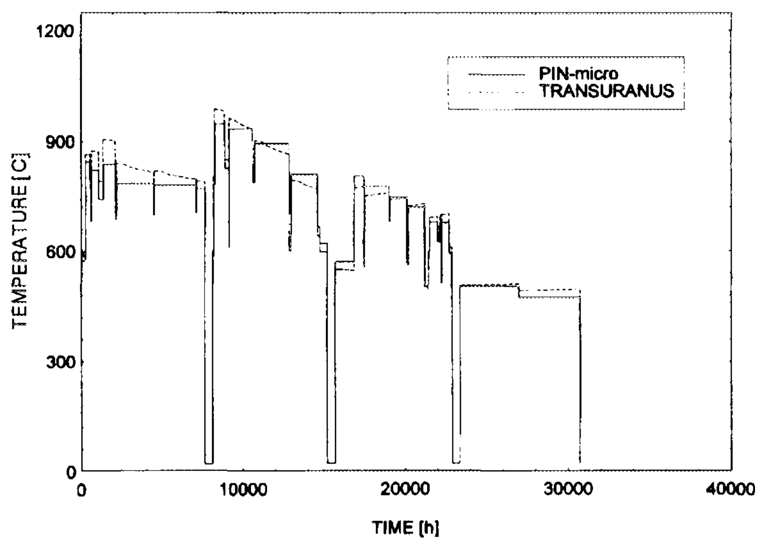

Figure 2 Fuel central temperature, FA1

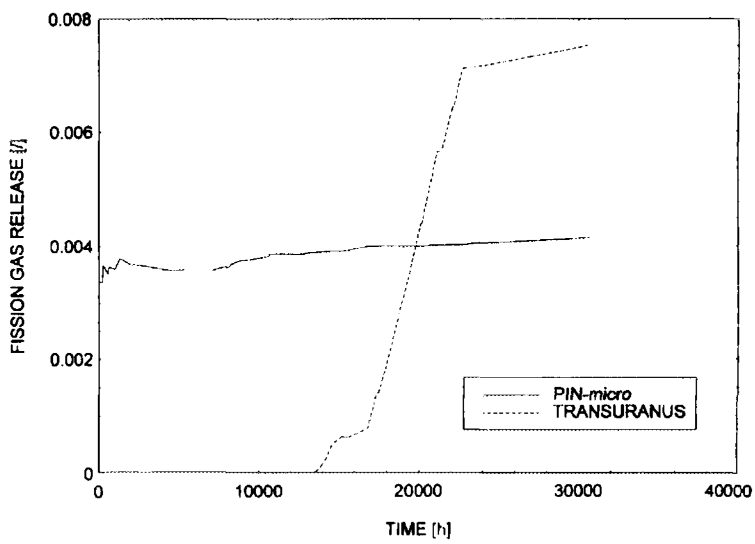

Figure 3 Fission gas release, FA1

4.2 Results and Discussion

The results of the calculations obtained by PIN-micro and TRANSURANUS for the fourth axial segment are compared in Figs. 1 - 5 for FA1 and in Figs. 6 - 10 for FA2.

The complicated power histories are shown in Fig. 1 and Fig. 6 as a function of time. In Fig. 2 - 5, and Figs. 7 - 10, fuel central temperature, FGR, inner gas pressure and gap size, calculated by both codes, are given.

Fig. 2 and Fig. 7 show a satisfactory overall agreement of the fuel centre temperature, except at low burnup, where TRANSURANUS predicts about 100°C higher temperatures. This difference may be due to different densification and relocation models.

The kinetics of fission gas release (FGR) differ mainly in the initialisation threshold of about 10000 hours given by TRANSURANUS for the gas release as well as in the achieved final level being higher of 50% to 80% (Figs. 3 and 8).

It is difficult to understand discrepancy of the inner gas pressure presented in Fig. 4 and Fig. 9. This discrepancy needs further investigation.

The gap size dependences presented in Fig. 5 and Fig. 10 indicate differences of fuel densification, relocation and swelling, especially at low and intermediate burnups, leading to different time of gap closure.

It should be very clearly and strongly emphasised, that all calculations performed by the TRANSURANUS code must be considered only as preliminary. The discrepancies between the TRANSURANUS and PIN-micro can be understood by difference of models and material properties.

The TRANSURANUS code has not yet been verified against VVER fuel rod experimental data and it needs a thoroughly verification. Nevertheless, the results obtained are physically quite reasonable and demonstrate the ability of the code to analyse the VVER fuel rod behaviour. The cases investigated do not yet need the use of the advanced capabilities of TRANSURANUS, compared with PIN-micro, to treat the fuel rod behaviour at fast transients and accidents, which may lead to PCI, cladding ballooning or rod failure.

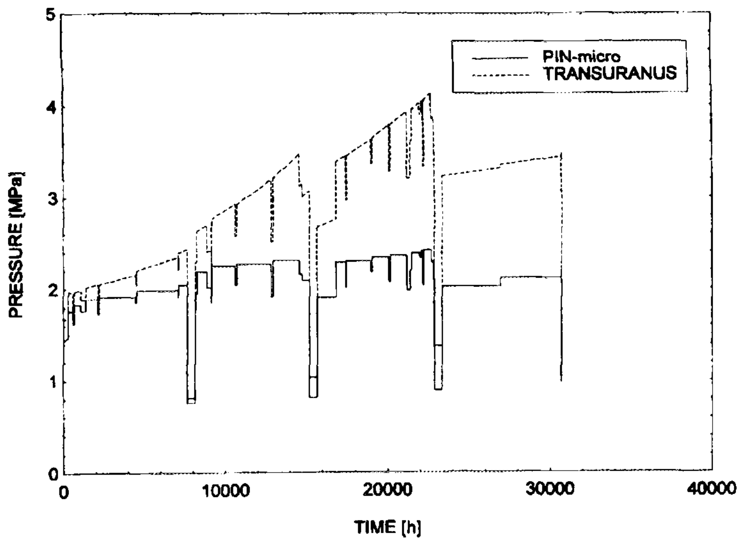

Figure 4 Inner gas pressure, FA1

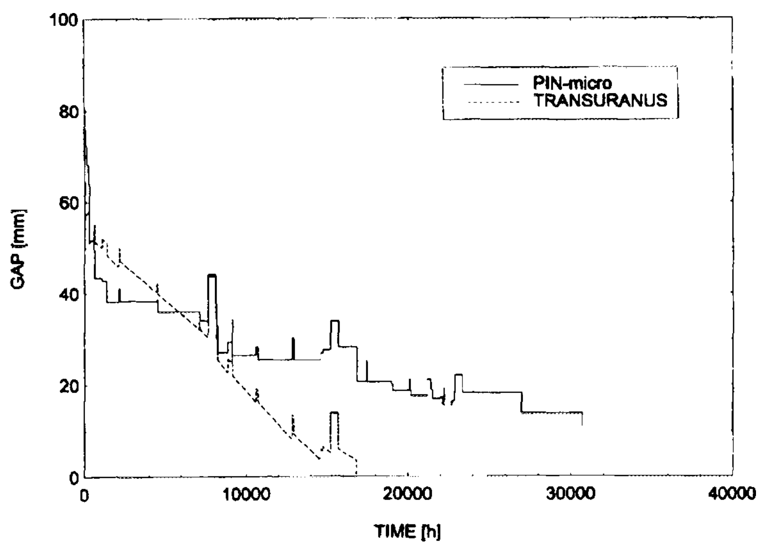

Figure 5 Gap size, FA1

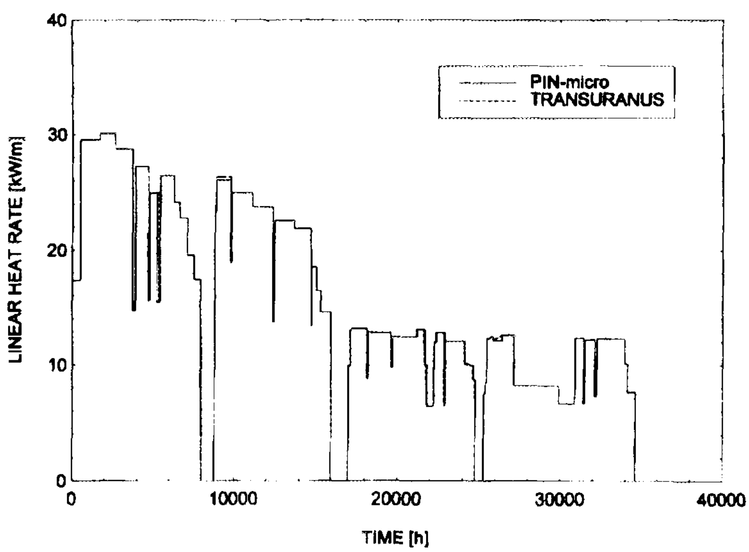

Figure 6 Local linear heat rate, FA2

5 Conclusions

The results presented allow to draw the following conclusions:

1. The PIN-micro code, as an accepted reference code for VVER fuel rod steady-state performance analysis, predicts adequately the thermal and mechanical behaviour of the two considered VVER fuel rods. Moreover, PIN-micro is restricted to be applied to transients and at extended burnups. In this very preliminary comparison between the PIN-micro and TRANSURANUS codes, the results obtained by PIN-micro have to be considered as reference ones.
2. This first and preliminary comparison of the calculated results obtained by PIN-micro and the TRANSURANUS codes shows reasonable agreement and the discrepancies can be explained by the lack of thoroughly VVER oriented verification of TRANSURANUS.
3. It might be expected that the advanced TRANSURANUS code could be successfully applied for VVER fuel rod thermal and mechanical analysis after incorporation of all necessary VVER specific properties and models for the Zr+1%Nb cladding, for the fuel and for the fuel rod as a whole and after validation against VVER experimental and operational data.

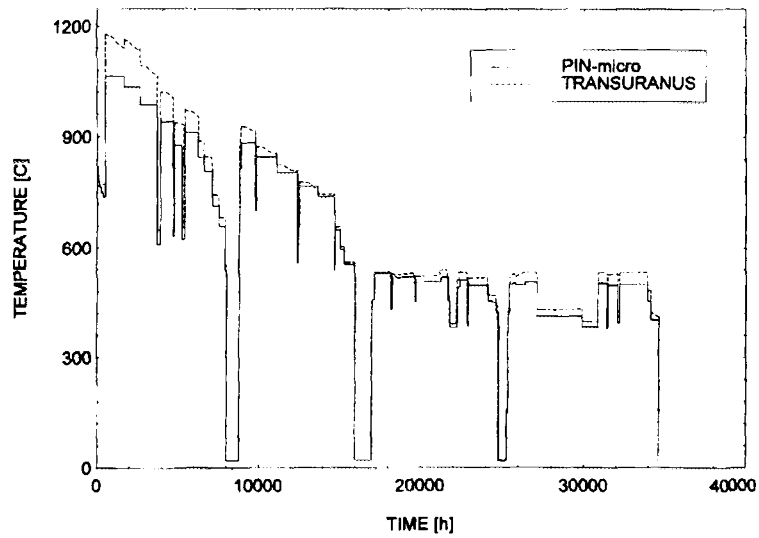

Figure 7 Fuel central temperature, FA2

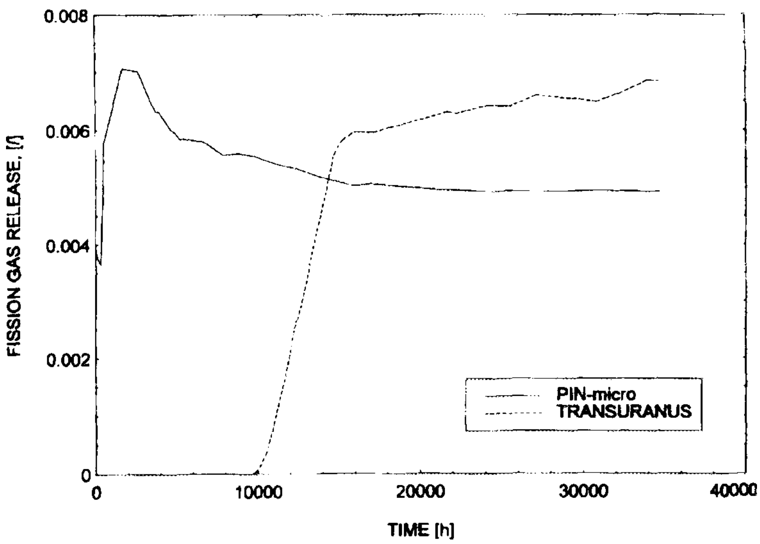

Figure 8 Fission gas release, FA2

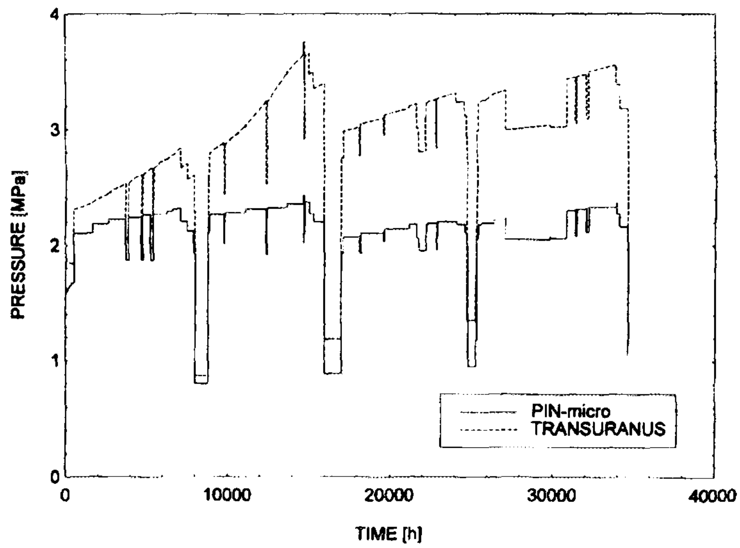

Figure 9 Inner gas pressure, FA2

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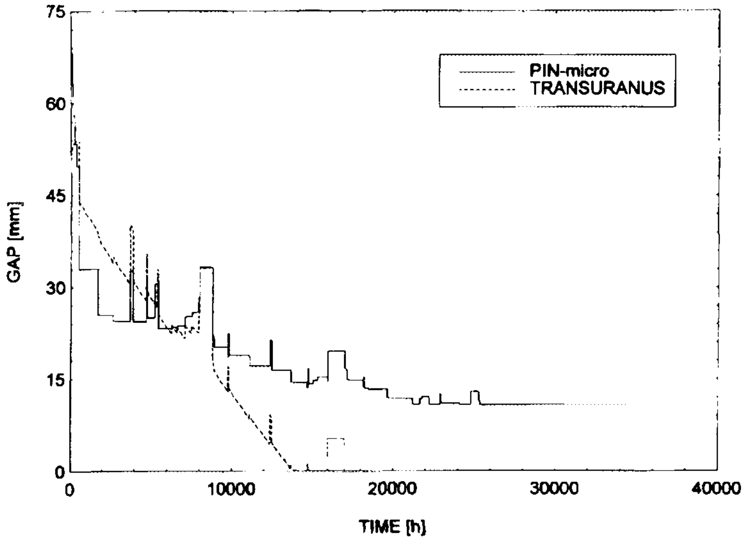

Figure 10 Gap size, FA2

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FUMEX Cases 1, 2, and 3 Calculated Pre-Test and Post-Test Results

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1 Introduction

The aim of the Bulgarian participation in the calculations of the complex FUMEX experiments is to obtain first knowledge and experience in the fuel rod behaviour modelling and prediction, and to improve the models in the only available at the present time computer code PIN-micro against fuel rod operational regimes at extended burnup. Such experience is expected to contribute directly to the better understanding of the fuel rod performance and to its more efficient and safe operation in the Bulgarian nuclear power plants. Four VVER-440 units and two VVER-1000 units are operating in the Kozloduy NPP in Bulgaria, but up to 1993 the fuel performance has never been analysed.

The calculations have been performed by a group of beginners in the field of the fuel behaviour modelling, working in two Bulgarian institutions. The work is supported by the valuable collaboration with experienced Russian specialists from the Kurchatov Research Center.

2 Experiments FUMEX 1, 2 and 3

Three of the given six FUMEX experiments were calculated, namely FUMEX 1, 2 and 3, where no PCMI is expected. The main characteristics of the FUMEX 1, 2 and 3 fuel rods are given in Table 1.

The FUMEX 1 experimental rod has relatively low power, temperatures, FGR and sufficiently large gas volume and filling gas pressure (He). The achieved final burnup has an intermediate value of ~ 34 *Mwd/kg* UO_2. Following the prescribed procedure, the fractional density of the densified fuel is calculated to be 0.9613.

The FUMEX 2 experimental rod is designed with a small diameter and it is nearly twice shorter to achieve rapid accumulation of burnup. Its linear power is nearly twice higher, compared with the FUMEX 1. The filling gas pressure is similarly high enough to oppose the significant FGR expected. The achieved final burnup is rather high, over 50 *MWd/kg* UO_2. The given fractional density of the densified fuel is 0.965. The necessity to predict the gas pressure within the frames of the given plot format (it must not exceed 30 *bars*) imposed a modification of the FGR model.

The FUMEX 3 consists of 3 short rods named **a**, **b** and **c**, each one with varying gap size, initial grain size and filling gas composition. The average linear power is relatively high and roughly constant until the final start-up, where it ramps to an about 50 - 80% higher value. The filling gas pressure is low (1 bar) compared with the first two FUMEX cases. The rods **b** and **c** are initially filled with pure Xe, which is very unusual. The achieved final burnups are different for the three rods and have intermediate values. The fractional density of the densified fuel, obtained according to the prescribed procedure, is 0.9663, 0.9665 and 0.9608 for the rods **a**, **b** and **c** respectively.

Because of the insufficient computer resources, only the simplified power histories for all the three cases, FUMEX 1, 2 and 3, were used.

3 PIN-Micro Versions

3.1 Standard Version Models and Options

The PIN-micro code is an integral, axisymmetrical, 1.5 dimensional code for thermal and mechanical analysis of oxide fuel rods. This code describes steady-state performance and slow transients. But it is not able to predict adequately the fuel rod behaviour at fast transients and regimes leading to PCMI and it has no cladding burst criteria.

Table 1 Main features of the experiments

ID	Central Hole	Pellet Outer R [*mm*]	Clad Outer R [*mm*]	Fuel Stack Length [*mm*]	Free Volume [cm^3]	Filling Gas	Filling Gas Pressure [*MPa*]	Fuel Enrichment [%]	Fuel Density [% TD]	Open Porosity [%TP]	Grain Size [μm]	q_l [*kW/m*]	q_l Ramp [*kW/m*]	Gap [μm]
FU1	Yes, $^1/_8$	4.045	4.750	810	8.2	He	10.	3.5	94.	62	10	~20	~20	155
FU2	No	2.96	3.51	443	3.1	He	10.	13.	94.3	10	7 - 10	40 - 44	No	50
FU3A	Yes, $^1/_2$	5.35	6.25	140	3.8	He	1.	10.	95.	10	3.4	~37.5	~62.5	50
FU3B	Yes, $^1/_2$	5.35	6.25	140	3.8	Xe	1.	6.	95.	10	20	~32	~44	50
FU3C	Yes, $^1/_2$	5.375	6.25	140	3.8	Xe	1.	10.	95.	10	3.4	~32	~44	25

The fuel thermal conductivity correlation in the PIN-micro code is rather close to the Lyon's correlation [5]:

$$\lambda = \lambda^{95} \cdot f$$

$$\lambda^{95} = \frac{1}{(0.0285 \cdot T + 3.77)} + 1.1 \cdot 10^{-6} \cdot T + $$
$$+ 1.01 \cdot 10^{-13} \cdot T^3 \cdot \exp\left(7.2 \cdot 10^{-4} \cdot T\right)$$

$$f = \left(\frac{1.025 \cdot \rho}{0.95 \cdot [1 + (1 - \rho) \cdot 0.5]} \right)$$

where: λ^{95} [*W/m* K] - fuel thermal conductivity for 95% TD; T[K] - temperature; ρ - fuel fractional density.

The swelling model is borrowed from the COMETHE-III-J code [6]. In this model the empirical swelling rate of the fuel with theoretical density (10.97 g/cm^3) depends on the fuel burnup and density, connected with some geometric factors:

$$\frac{dV}{V} = 0.1492 \cdot (1 - \rho)^{0.254} \cdot \left[\exp\left(-F \cdot \frac{dV}{V} \right) - 1 \right] +$$
$$+ \rho \cdot b_{100} \cdot Bu$$

where: dV/V - swelling rate; Bu [*MWd/t* UO_2] - burnup; b_{100}=1.82·10^{-6} - swelling rate of fuel with theoretical density; F=100 - geometric factor.

The fuel densification model is an empirical model, similar to the one used in the GAPCON-THERMAL-2 & 3 codes [2]:

$$dr = dr_m \cdot 0.217147 \cdot \ln\left(\frac{Bu}{Bu^*} \right)$$

where: dρ =ρ-ρ_0; ρ_0 - initial fuel density; ρ_m - maximal density of the densified fuel; Bu [*MWd/t U*] - current burnup; Bu^* - burnup which determines the end of densifications.

The pellet radius change is calculated from the density change:

$$\left| \frac{dR}{R_0} \right| = \left(1 + \frac{d\rho}{\rho_0} \right)^{-\frac{1}{3}} - 1$$

The model of the grain growth (restructuring model) is from MATPRO 10 [5]:

$$G_s = G_{s0} + 7.0 \cdot \exp\left(-\frac{6.18 \cdot 10^{13}}{R \cdot T} \right) \cdot t$$

where: G_s [μm] - current grain size; G_{s0} [μm] - as fabricated grain size; T [K] - temperature; t [*s*] - time.

The fission gas release model used in PIN-micro is that of Weisman [7] supplemented with the model of gas sweeping by grain boundaries during the grain growth. The latter is applied as follows:

$$F = 1 - \left(1 - K'\right) \cdot \left[1 - \exp(-Kt) / Kt\right]$$

$$K = 0.25 \cdot \exp\left(-\frac{14800}{T} \right)$$

$$K' = \min\left[1, \exp\left(A_2 - \frac{A_1}{T} - A_3 \cdot \rho \right) \right]$$

$A_1 = 6917$; $A_2 = 33.95$; $A_3 = 33.8$

where: F - fractional fission gas release; K - fraction of gas atoms achieving the grain boundaries; K - the probability that captured gas atoms will be released; $K = K' \cdot K''$; T [K] - temperature; t [days] - irradiation time; ρ - fractional fuel density.

For varying power history:

$$dn_i = n_i - n_{i-1} =$$
$$= P_i \cdot \left[dt_i - \left(1 - K_i\right) / K_i \left(1 - \exp\left(-K_i dt_i\right)\right) \right] +$$
$$+ C_{i-1} \left[1 - \exp\left(-K_i dt_i\right)\right]$$

where dn_i - number of gas moles released per time step with duration dt_i; P_i - fission gas generation rate; C_{i-1} - concentration of fission gas captured by trap during the preceding time steps. In this case the total gas release is determined as:

$$F = \frac{\left(\sum_{i=1}^{m} dn_i \right)}{\left(\sum_{i=1}^{m} P_i dt_i \right)},$$

where m is the full number of time steps with constant power lewel.

The fission gas model is supplemented by a gas sweeping model, accounting for the additional fission gas, released during the fuel restructuring:

$$f_{i+1} = \frac{n_i . x_2 + (1 - f_i) . n_i . x_3 + f_i . n_i}{n_{i+1}} . (1 - \Delta f_{gg,i}) + \Delta f_{gg,i}$$

where:

$$f_{gg,i} = 1 - \left(\frac{D_i}{D_{i-1}} \right)^3 \; ; \; x_3 = 1 - \exp(-K_1 . \Delta t_i)$$

$$x_4 = \begin{cases} 1. when\ K_1 . \Delta t \le 10^{-4} \\ x_3 / K_1 . \Delta t > 10^{-4} \end{cases} \; ; \; x_2 = [1 - (1 - K_2)] . x_4$$

$$K_1 = 0.25 \exp\left(-\frac{14800}{T} - 9.575 \right)$$

$$K_2 = ghy\left[1, \exp\left(33.95 - \frac{6917}{T} - 33.8\rho \right) \right]$$

f_i - local fractional FGR after the i-th time step

Δt_i [*s*] - i-th time step

Δn_i [*mol*] - fission gas generated during the i-th time step

n_i [*mol*] - fission gas generated at the end of i-th time step

D_i [μm] - grain size at the end of i-th time step

$\Delta f_{gg,i}$ - local fractional FGR during the i-th time step due to sweeping by grain boundaries movement

ρ - fractional density.

3.2 Modified Pre-Test Version

The code capability to calculate only up to 200 time steps was extended for calculations of up to ~1300 time steps.

The Jens-Lottes heat transfer correlation, as supposed by the Halden Reactor Project, was introduced to simulate the cooling of the fuel rods under heavy water boiling conditions.

Although it was recommended not to pay big attention to the cladding thermomechanical properties and behaviour for the first three FUMEX cases, it was necessary to introduce specific zircaloy thermal expansion coefficients to calculate properly the cladding axial elongation.

The fuel thermal conductivity, originally dependent only on the temperature in PIN-micro, was not changed to be dependent also on the burnup, because its degradation with the burnup increase is partly compensated through the rather simply radial flux depression model, which underpredicts the flux depression and thus overpredicts the fuel temperatures.

The suggested radial flux depression correlation and the recommendation to use models for the rim effect are not applied, because of insufficient knowledge and experience of the working group.

As in the PIN-micro code it is not possible to model axial nodes with different length and different central void length as well as fuel volume changes due to open porosity and dishing, two variants of the code were prepared, based on some engineering estimations: the first one without central void and the second one with central void. Both variants were applied for each case with central void and dishing (FUMEX 1 and 3). Thus, the local parameters (temperatures) were taken from the variant with central void for the thermocouple, and the fuel rod global parameters (FGR, pressure) were averaged from the results of the both variants.

A special model for the cladding elongation as a consequence of the thermal expansion and irradiation growth under non-axisymmetric conditions was applied. It is based on the experimental evidence, that after the first reactor start-up (ramp) some partial local PCMI already occurs and consequently causes residual axial elongation [8]. The Russian model [8] was modified for short fuel rods (such as in FUMEX).

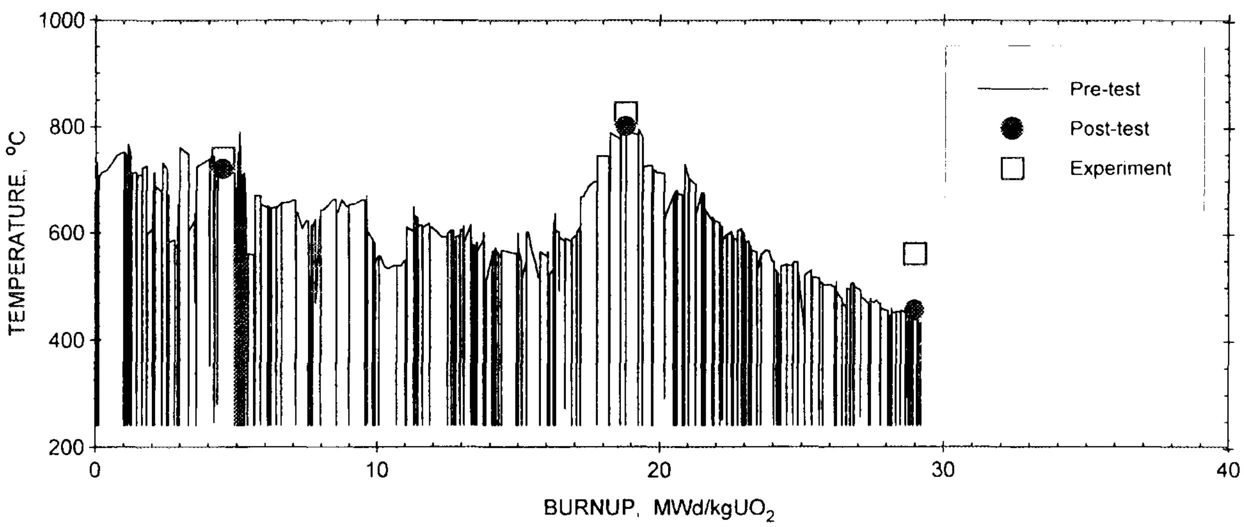

Figure 1 FUMEX 1: Center line temperature vs burnup

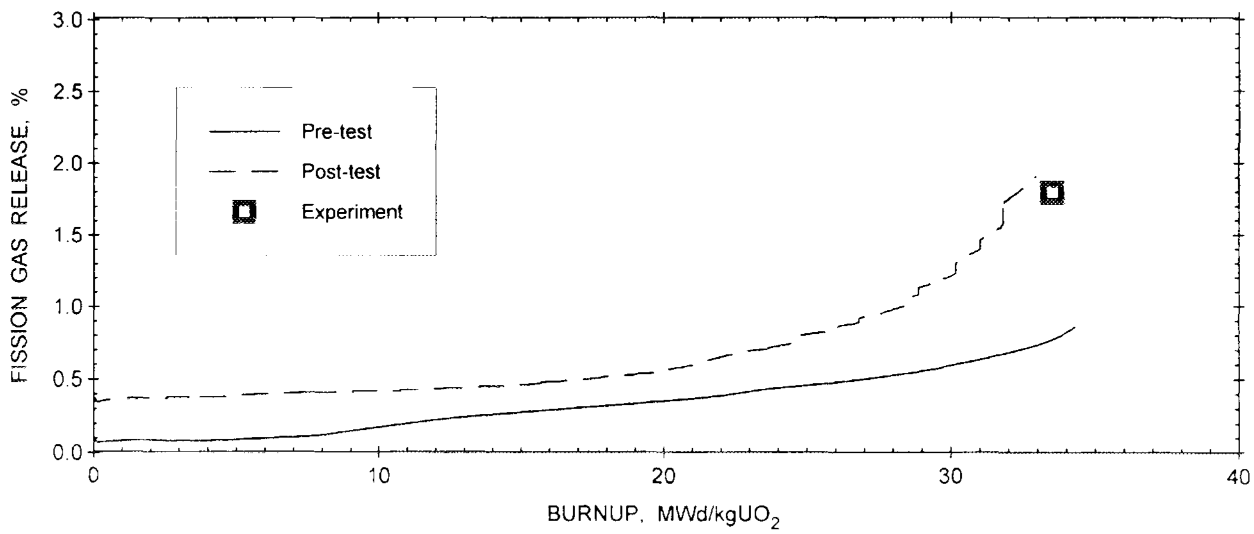

Figure 2 FUMEX 1: Fission gas release vs burnup

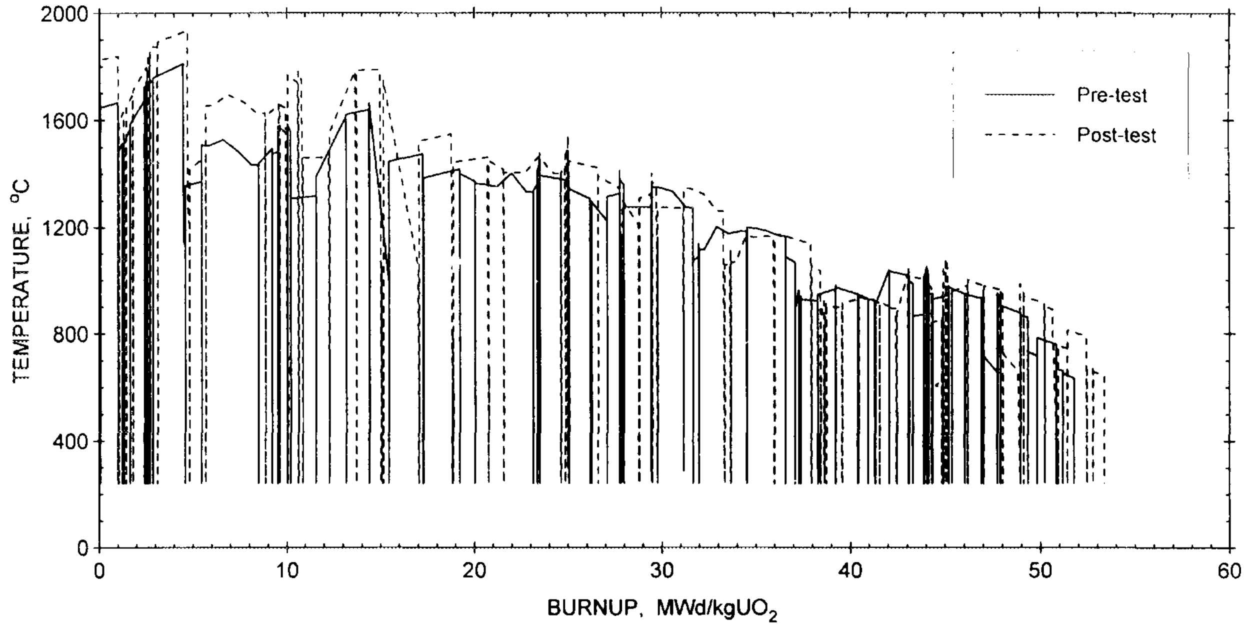

Figure 3 FUMEX 2: Center line temperature vs burnup

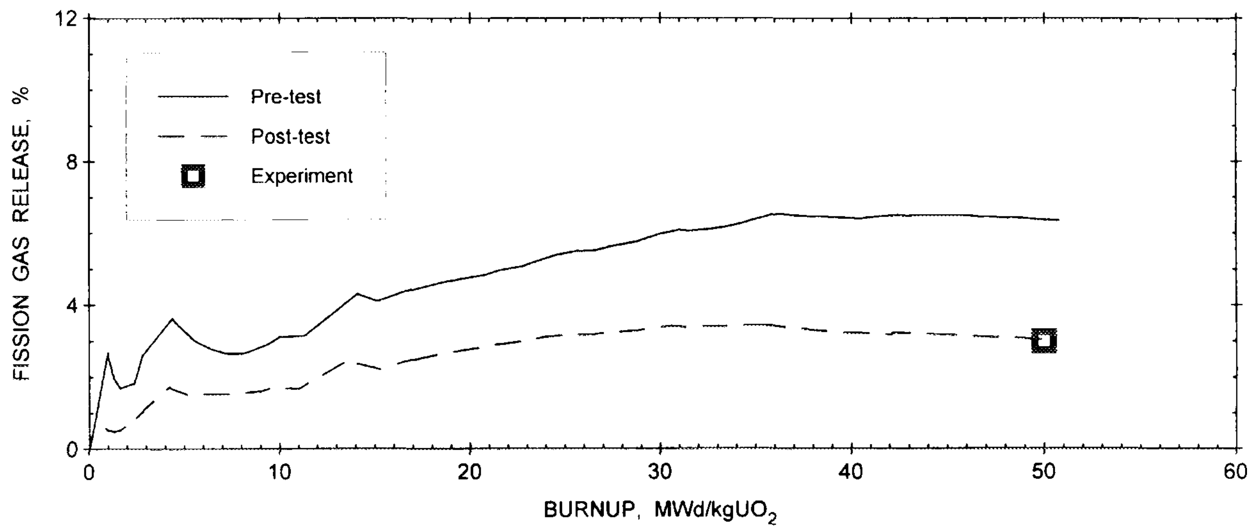

Figure 4 FUMEX 2: Fission gas release vs burnup

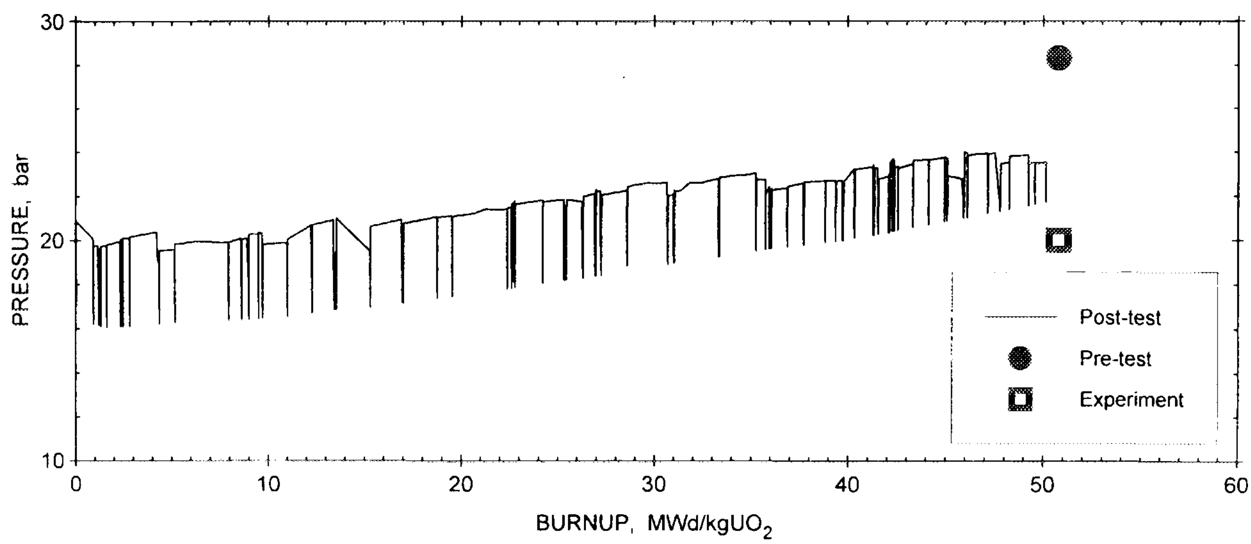

Figure 5 FUMEX 2: Pressure vs burnup

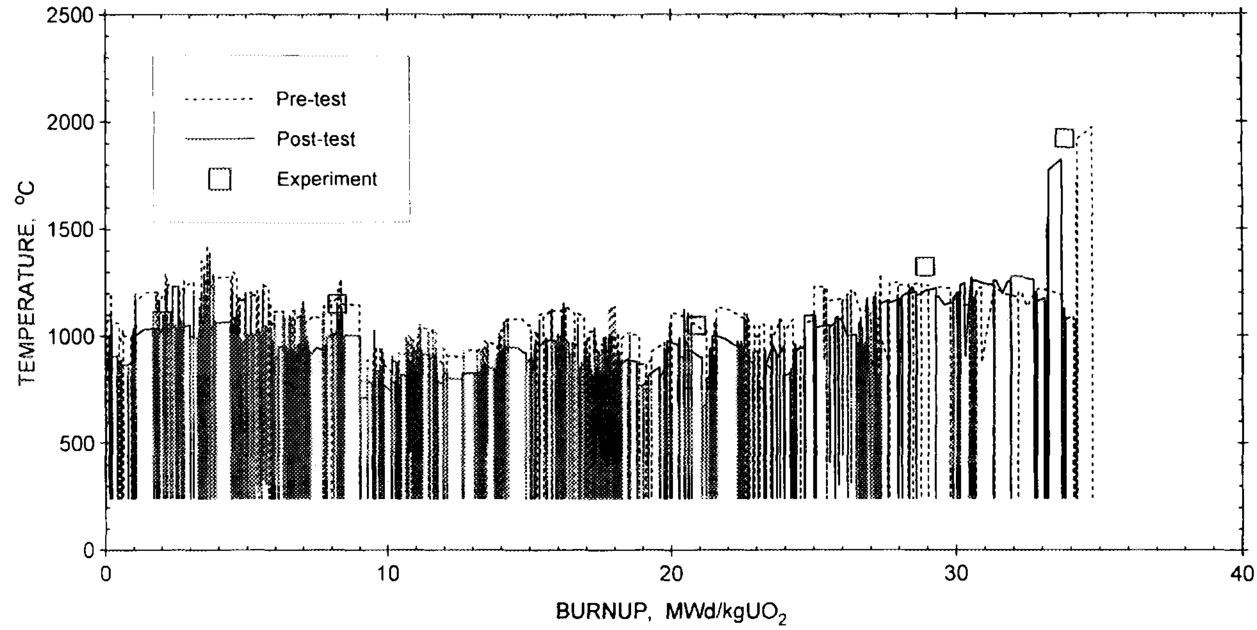

Figure 6 FUMEX 3a: Center line temperature vs burnup

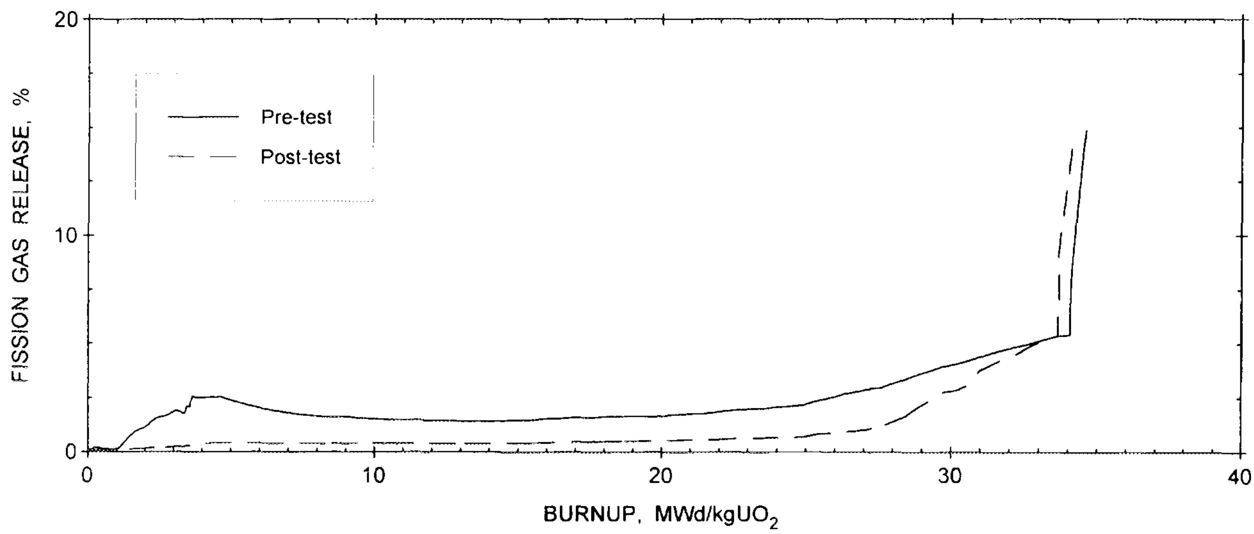

Figure 7 FUMEX 3a: Fission gas release vs burnup

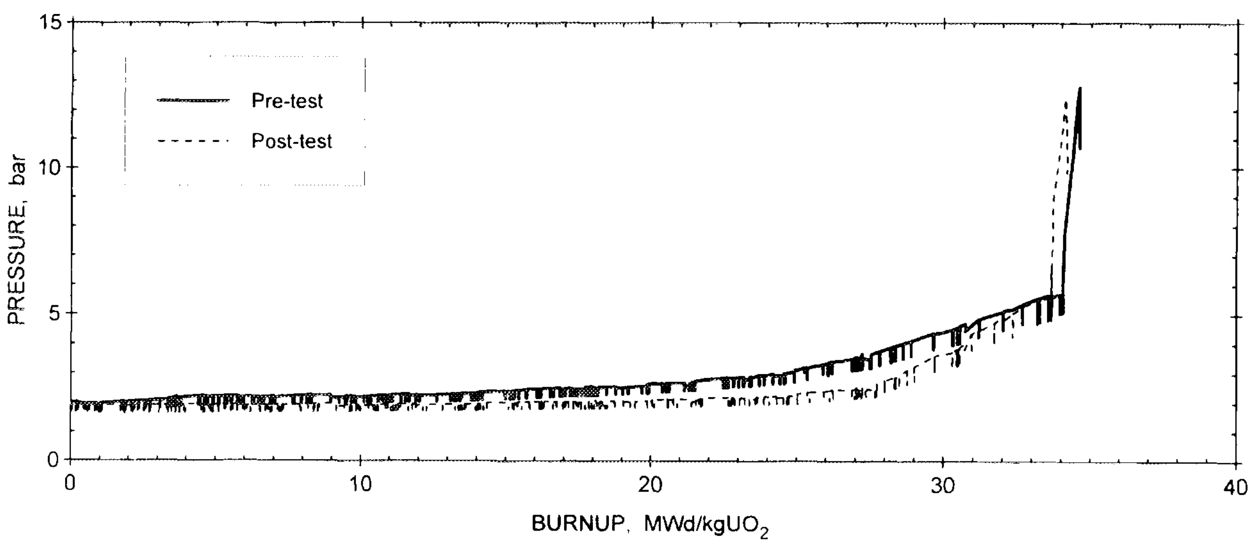

Figure 8 FUMEX 3a: Pressure vs burnup

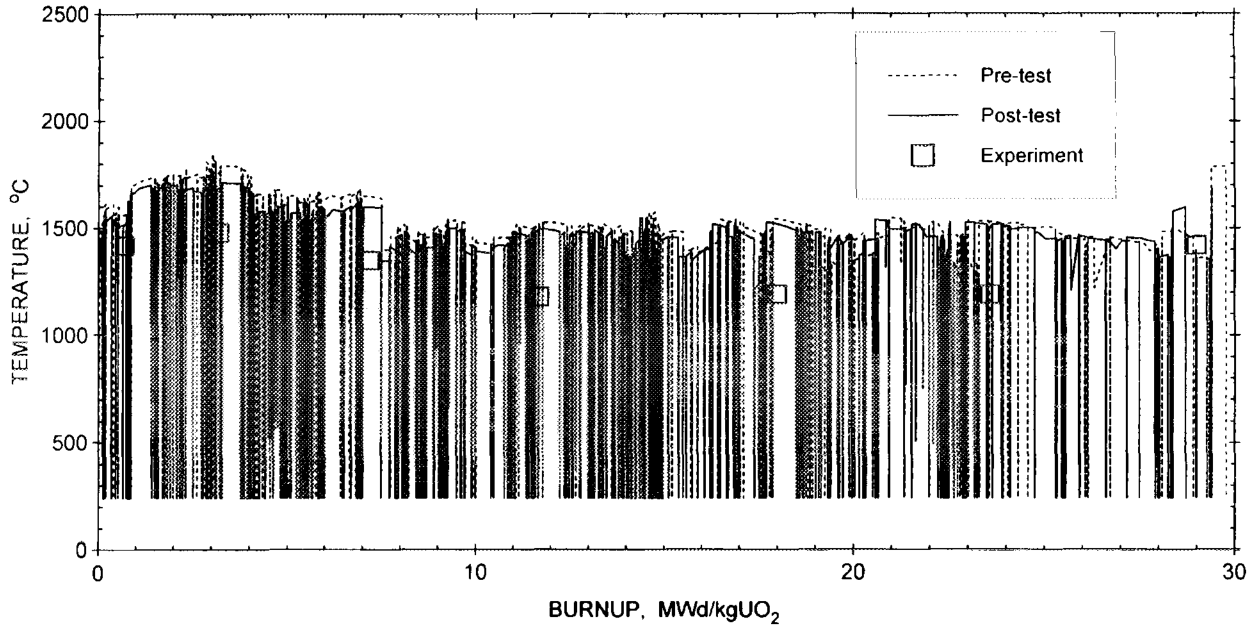

Figure 9 FUMEX 3b: Center line temperature vs burnup

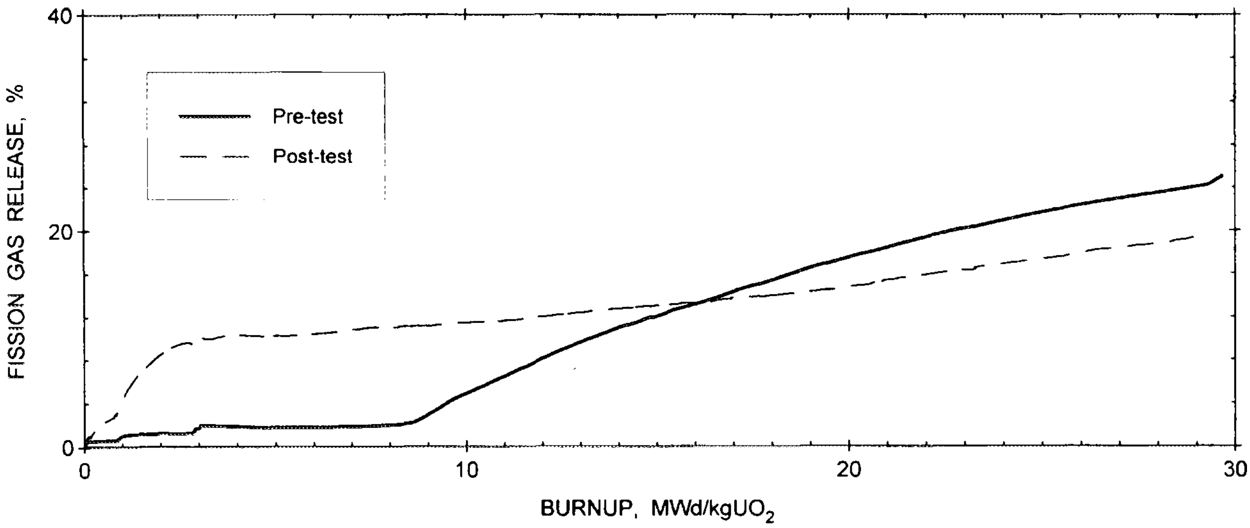

Figure 10 FUMEX 3b: Fission gas release vs burnup

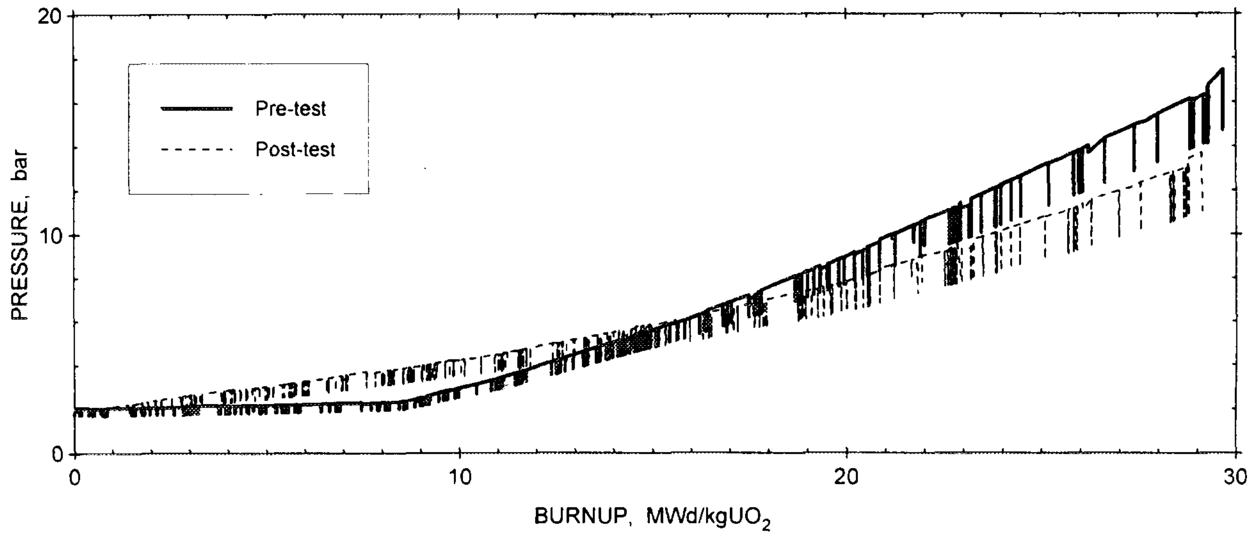

Figure 11 FUMEX 3b: Pressure vs burnup

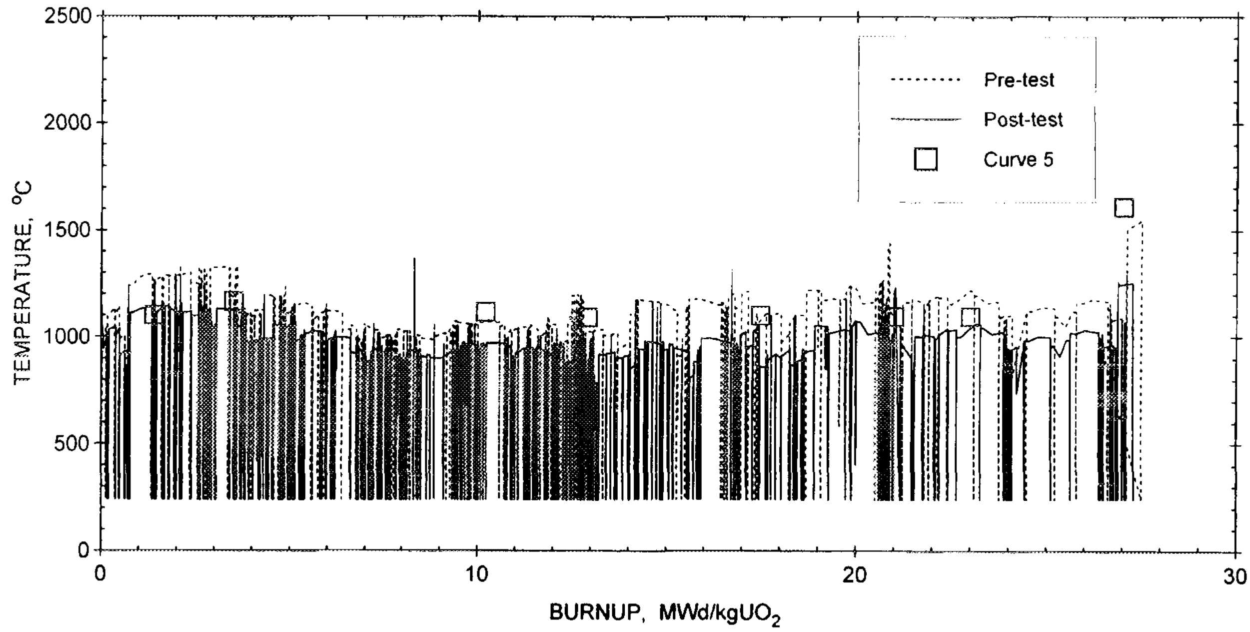

Figure 12 FUMEX 3c: Center line temperature vs burnup

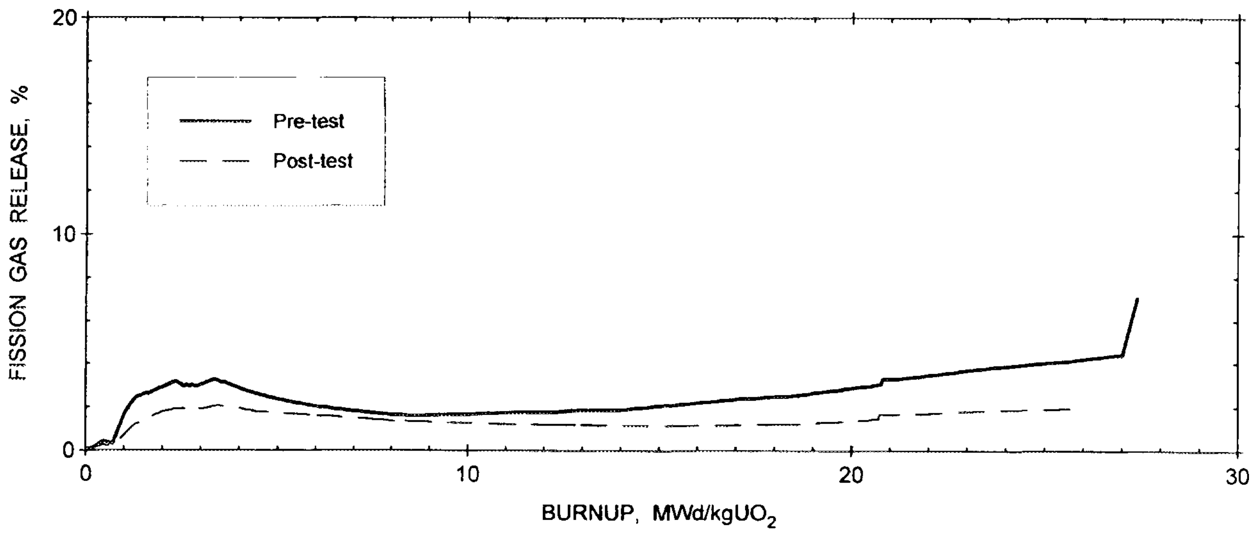

Figure 13 FUMEX 3c: Fission gas release vs burnup

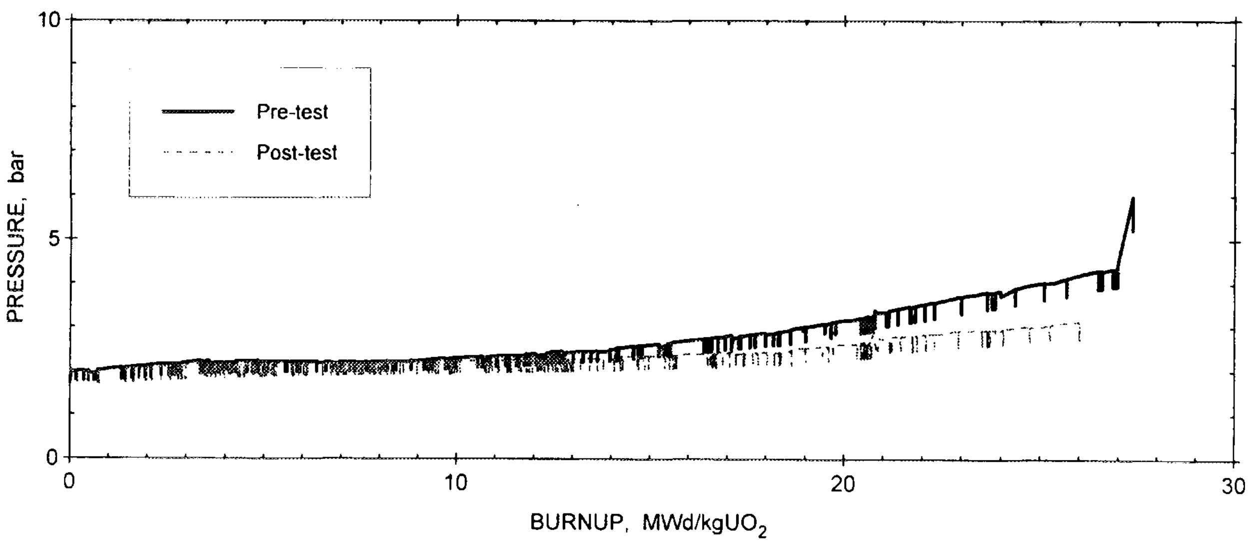

Figure 14 FUMEX 3c: Pressure vs burnup

The diffusion constants for the FGR model and the FGR enhancement through the sweeping of the fission gas on the grain boundaries were redefined and fixed according to the given limits for the gas pressure in the FUMEX 2 fuel rod.

The relocation model was modified to account for the creep of the fuel during the irradiation, which is not modelled originally and to prevent PCMI in FUMEX 2.

3.3 Modified Post-Test Version

The Post-Test version includes the following modifications: FGR model for the initial period of performance and for extended burnup; relocation model accounting for fuel creep; gap heat transfer model for open gap and for contact between fuel and cladding to account for surface roughness.

The approach to the burnup determination is modified in accordance with the possible pellet chamfering, dishing etc., on the base of data measured for initial fuel debris. The burnup accumulation between two time steps in the modified post-test version does not depend on power during the next, but during the previous time step. The radial flux depression correlation is also modified. The fast neutron flux determination dependent on burnup is based on the correlations from Ref. [8].

4 Calculated Pre-Test and Post-Test Results

4.1 FUMEX 1

The given rod average power is relatively low. The final burnup is ~34 *MWd/kg* UO_2. The centre line temperature *vs.* local burnup in Fig. 1 follows the trend of the power curve, because the gap never closes. Fuel central temperature at low burnup is well predicted, but at higher burnup the temperature is too low because of the lack of fuel thermal conductivity dependence on burnup and possible inadequate relocation and radial flux depression models. The predicted FGR in Fig. 2 (pre-test results) is lower than the measured one as well as the final value.

4.2. FUMEX 2

The rod average power decreases gradually during the whole irradiation time from the level of ~44 *kW/m* to ~16 *kW/m* achieving ~50 *MWd/kg* UO_2 burnup. The fuel centre temperature in Fig. 3 follows the trend of the power curve up to the middle of the irradiation and after about 35 *MWd/kg* of UO_2 decreases more intensively due to the gradual gap closure, which nevertheless remains open until the end. The calculated fuel centre temperature is too low at the end of life because of the mentioned reasons. The FGR (Fig. 4) is slightly higher at lower fuel temperatures and increases gradually and reaches ~6.5%, causing pressure increase, Fig. 5, up to 29 *bar* (27 *bar* at zero power and 220°C). For the post-test the FGR model was modified and the calculated gas pressure is in the frame of the measured values.

4.3. FUMEX 3

The rod 1 average power remains more or less constant during the whole irradiation, except for the last start-up where it increases about 70%. The fuel centre line temperature in Fig. 6 follows the trend of the power curve almost to the middle of the irradiation up to about 30 *MWd/kg* UO_2, where thermal contact between pellet and cladding occurs. The fuel centre temperature is slightly overpredicted at low burnup and afterwards exhibits the tendency to underprediction. The temperature varies between ~900°C and ~1300°C, except for the last ramp where it reaches ~2000°C. The dependence of the fuel central temperature on burnup at different constant levels shows that the temperature is well predicted only at low power levels. There is an overprediction at higher levels in the beginning and underprediction at the end of the irradiation. The power ramp is predicted very satisfactorily. The FGR in Fig. 7 is relatively constant ~2.5% and rises at the last power ramp up to 15%. The gas pressure, Fig. 8, is also relatively constant at ~2 - 3 bars up to 27 *MWd/kg* UO_2, then increases gradually to 5 bars and to 13 bars during the last ramp.

The rod 2 fuel centre line temperature, Fig. 9, does not change very much (1500°C - 1800°C). This temperature is significantly overpredicted. The temperature ramp has been predicted relatively well. The FGR in Fig. 10 remains constant at ~2% and begins to rise gradually at 8.5 *MWd/kg* UO_2 burnup reaching the highest value of ~25%. The gas pressure, Fig. 11, reaches ~18 *bars* (~15 *bars* at zero power) including the last ramp.

The rod 3 fuel centre line temperature, Fig. 12, does not change very much (1000°C - 1300°C), because the gap size is low and a thermal contact between pellet and cladding may be also considered. The fuel centre temperature is systematically overpredicted excepting the final ramp. The FGR in Fig. 13 does not change significantly and remains at ~2 - 5%. At the last ramp it rises to ~8.5%. The gas pressure, Fig. 14, is also relatively constant and reaches ~6 *bars* after the last ramp.

5 Conclusions

Compared with many recent computer codes developed and applied to investigate the sophisticated processes in a fuel rod during irradiation, the relative simple and old-conception steady state PIN-micro code with its restricted range of application caused significant difficulties to be used as an appropriate tool for the FUMEX calculations and to obtain reasonable results.

These difficulties were partially surmounted through different model modifications and corrections, based on special engineering estimations. Nevertheless, the results obtained do not seem unreasonable.

In all calculated results the FGR values appear to be troublesome low for the corresponding gas pressure increase.

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Modelling of VVER-1000 Fuel: State and Prospects

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1 Introduction

Significant experience of the VVER-1000 fuel operation indicative about high reliability of fuel elements is at present accumulated. In the last years the level of fuel failures on Russian NPP units does not exceed $(1.5 - 2.0) \cdot 10^{-5}$. Post irradiation examination of the fuel pins operated up to design burnup have confirmed their good condition without any attributes, capable to limit their further operation.

This result permits to consider a question on further increase of fuel burnup.

The fuel reliability is provided with the project of an active core, perfection of a design and fabrication technology of fuel pin. The important role in maintenance of necessary reliability of fuel belongs to computer methods of modeling of fuel pin behavior in real conditions of operation.

Code START-3, with application of which fuel pin VVER-1000 are developed is based on representative volume of experimental data on study of material properties, processes, post irradiation examination of experimental and standard fuel pins.

At the stage of verification and correction in accounts are used characterized parameters of experiments. In design work the reasonable conservative approach, assuring the satisfaction to design limits with necessary margins is applied.

2 VVER-1000 Fuel Pin

The main features of a design of a fuel pin VVER-1000 are stipulated by application of alloy Zr-1%Nb for cladding, pellets with a central hole and flat end faces, availability sufficiently large plenum.

The fuel is operated in 3-year cycle with average burnup of unloaded assemblies of 43 *MWd/kgU.*

Post irradiation examination have shown following results on a condition of fuel pins.

Corrosion of cladding

The oxide film thickness on the external surface does not exceed 4 - 8 μm and does not grow with burnup. It weakly changes on a length of a fuel pin (gain on 1 - 2 μm from a bottom to a top).

On an internal surface there are only local sites with thickness not more than 10 μm.

Cladding hydriding

There are insignificant quantity of hydrides with random or tangential orientation. The content of hydrogen does not exceed $(5 - 6) \cdot 10^{-3}$%.

Mechanical properties of a cladding

It is observed an usual radiating hardening, which does not hereinafter depend on burnup. Uniform and general lengthening is not less than 4 and 15 % accordingly.

Cladding deformation

The elongation of fuel pin linearly depends on burnup and does not exceed 0.37%.

The reduction of a diameter depends on burnup and at ~ 45 *MWd/kgU* approaches ~ 1%.

Fuel pellets

The axial gaps in a fuel column are away. The fuel structure corresponds to initial one. It is observed a usual cracking pellet picture. Swelling is less than 3%.

Fission gas release

Fission gas release as an average one in assembly makes 0.7 - 3% up to burnup of ~ 47 *MWd/kgU.*

The examination results testify to a good condition of fuel pins down to maximum design burnup, that is confirmed by experience of operation. The level of refusals of fuel pins does not at present exceed $(1.5 - 2) \cdot 10^{-5}$.

Characteristics of VVER-1000 fuel pin

Cladding	
Material	Alloy Zr-1%Nb
Outside diameter, *mm*	9.1
Internal diameter, *mm*	7.72
Fuel pellet	
Outside diameter, *mm*	7.57
Internal diameter, *mm*	2.4
Enrichment, %	3.6 - 4.4
Density, g/cm^3	10.4-10.7
Fuel pin	
Length, *mm*	3837
Length of a fuel column, *mm*	3530
Helium pressure, *MPa*	2

3 Code START-3

The models, used in the code, are based on experimental study of material properties, processes, post irradiation researches of experimental and standard fuel pins. They include such labor-consuming works, as in-pile research:

- non-steady creep of zirconium claddings,
- fuel creep,
- irradiation fuel densification.

The code includes following main selections:

- thermo-hydraulic account,
- thermo-physical account,
- mechanical account.

Table 1 **Parameters of experimental VVER-1000 fuel pins and measured gas release levels**

Pin No.	Irradiation time	Gap size	Average fuel density	Burnup, *MWd/kgU*		Gas release	Maximum of heat rate
	h	*mm*	*g/cm³*	Average	Max	%	*W/cm*
1 2	9091	0.19 - 0.32	10.55 10.60	27.32 28.53	32.57 36.23	30 46	400/250 450/280
3 4	10799	0.19 - 0.32	10.53 10.53	33.5 28.6	45.3 40.0	58 45	460/397 407/350
5 6	18288	0.19 - 0.32	10.65 10.64	37.6 52.76	49.63 71.23	16 54	300/210 435/305

The results of thermo-hydraulic calculations are temperatures of an outside surface of a cladding, which are boundary conditions for determination of temperature fields in a fuel pin.

In thermo-physical accounts are defined:

- non-stationery temperature fields in sections of a fuel pin,
- gas composition and pressure inside a fuel pin,
- swelling and irradiation densification of a fuel.

Taking into account:

- cracking and radial moving of pellet fragments,
- pellet restructure (growth of columnar and equiaxial grains),
- fuel conductivity degradation in dependence on burnup,
- non-uniformity of burnup and heat rating on pellet radius, owing to Pu building and availability of an integrated absorber,
- zirconium cladding oxidation.

As a result of mechanical account the fields of stresses and deformations in a fuel and cladding are defined. Also the degree of cladding damage is estimated in the form of an accumulated depth of stress corrosion crack.

Taking into account:

- thermal, elastic, plastic, creep deformations, volume changes, irradiation growth;
- pellet cracking and healing.

Whole fuel pin is divided into sites in a longitudinal direction (up to 30 and more). The load history is described by parameters in a given set of basic points, between which parameters linearly approximated. The quantity of basic points is not limited and in practice is limited only to acceptable time of the account. The code is used on computers PC 385, 486, 586 and includes ~ 5600 lines.

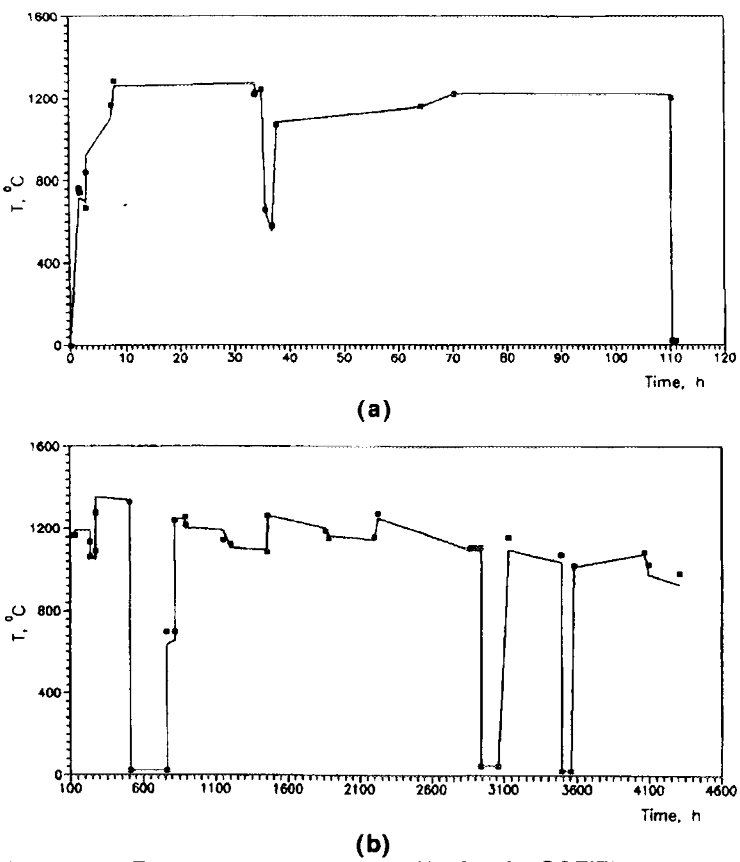

Figure 1 **Fuel center temperature (fuel rod 3 SOFIT): calculation (cont. line) and experimental data**

4 Accounts of Standard and Experimental Fuel Pins

The code capability was checked up by comparison with experimental evidence on standard and experimental fuel pins. Some results of check on a domestic data are here indicated.

For check and updating of temperature accounts were used. In particular, results of Russia-Finland experiment SOFIT on test rods of type VVER, equipped with thermocouples.

The accounts have confirmed a possibility of good enough forecasting with the help of a code START-3 of temperature fields in a fuel pin. As an example on Fig. 1 the comparison of calculated and measured fuel center temperatures in the rod No. 3 irradiated up to the average burnup of 8.8 *MWd/kgU*.

Fission gas release was considered in a wide range of magnitudes: from low sizes, characterized for a standard fuel VVER-1000 up to high significance's, achieved in experimental fuel pins, irradiated in reactor MR.

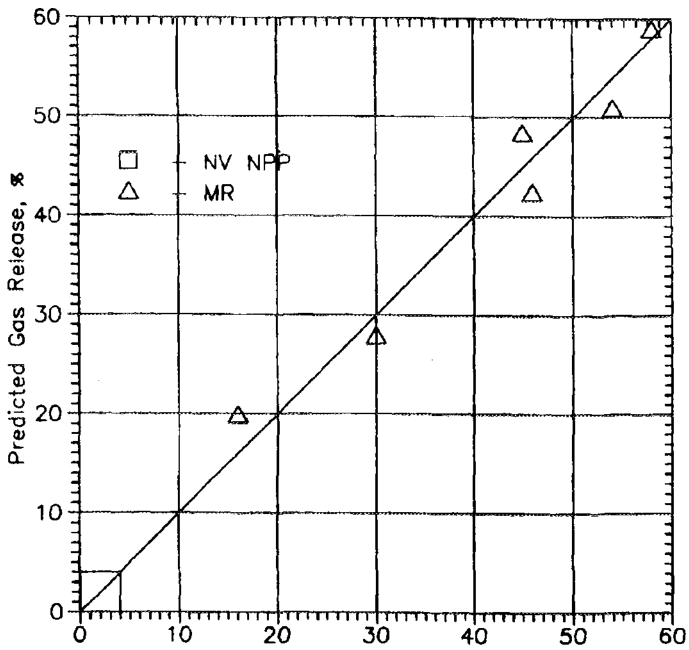

Figure 2 Comparison of predicted and measured gas release

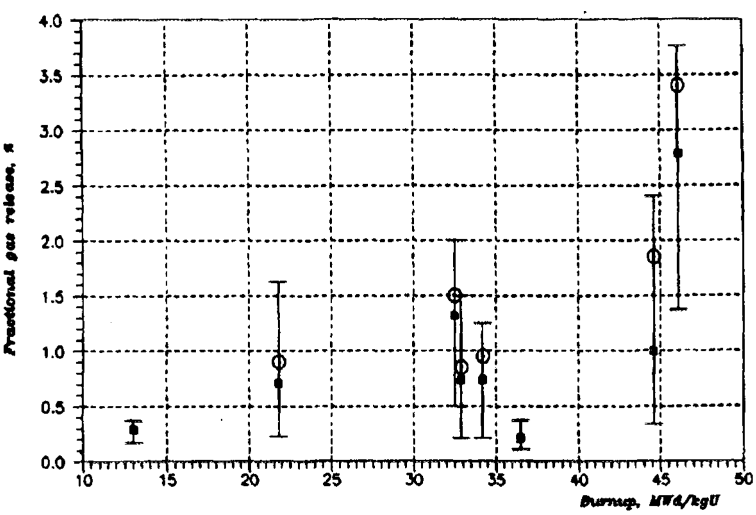

Figure 3 Gas release vs. burnup for VVER-1000 fuel

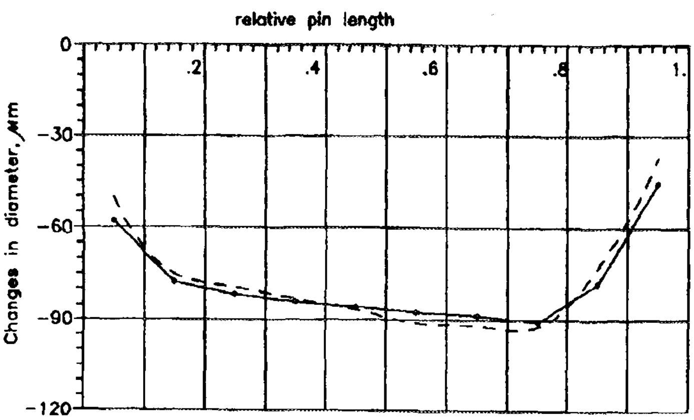

Figure 4 Changes in fuel pin outer diameter at the end of campaing

In Table 1 the characteristics of experiment in MR are indicated and on Fig. 2 the comparison of calculated and measured FGR is shown.

The fuel VVER-1000 is characterized relatively small FGR as it is shown on Fig. 3. Appreciable spread given for fuel pins within the assembly is thus observed.

On this picture the accounted significance's are shown two. As it is visible, the calculated significance's are within the limits of natural spread of the data.

In design works, that VNIINM conducts on development and substantiation of fuel VVER-1000 with the use of code START-3 the calculations are carried out for parameters, ensuring forecasted FGR at a level not below maximally observable significance's. It guarantees unconditional fulfillment of the requirements on design limits and margins.

For maximum accounted design parameters given with certain margins the calculated evaluations show that FGR in a fuel pin during 3 year campaign does not exceed 12%, gas pressure less than 1.1 *MPa*.

As it is visible for a design of a fuel pin VVER-1000 fission gas release is not the restrictive factor for achievement extended burnup.

On Fig. 4 the estimated and measured change of fuel pin diameter for standard assembly VVER-1000 after burnup of 44.7 *MWd/kgU* is shown.

Over the set of accounted and experimental data it is possible confidently to conclude that the fuel pin has sufficient margins of serviceability, capable to provide the further increase of design burnup of fuel VVER-1000.

As it is shown the code START-3 reasonably good forecasts behavior of fuel pins VVER-1000 in normal operation conditions. Certain confirmation of capability of the code are also results of accounts of the experiments within the framework of the program FUMEX (see Table 2).

Conducted at the first stage of our work the accounts of cases 1, 2 and 5 have shown reasonable conformity with experimental data and results of other codes. At the same time the analysis of results has shown that at preparation of a code for given

Table 2 Participants in FUMEX Blind Problem

N	Country	Organization	Code
	Norway/OECD	Halden	Experiment
1	Argentina	CNEN	BACO
2	Bulgaria	INRNE	PIN Micro
3	Canada	AECL	ELESIM.MOD11
4	Finland	VTT	Enigma 5.8f
5	France	EdF	TRANSURANUS EdF 1.01
6	France	CEA/DRN	METEOR - TRANSURANUS
7	CEC	ITU	TRANSURANUS
8	India	BARC	PROFESS
9	India	NPC	FUDA
10	India	BARC	FAIR
11	Japan	NNFD	TRUST 1b
12	Japan	CRIEPI	EIMUS
13	China	CIAE	FRAPCON-2
14	Romania	INR	ROFEM-1B
15	Swiss	PSI	TRANSURANUS -PSI
16	Czech Rep.	NRI Rez	PIN/W
17	UK	BNFL	ENIGMA 5.2
18	UK	NE	ENIGMA 5.8D
19	Russia	IIM	START 3

accounts some discrepancies were allowed. After corresponding corrections the more precise results for specified cases were received and the calculation of other rods were fulfilled.

Number of these results are presented on Figures 5 - 17 in comparison with experimental data and results of other codes.

It is possible to note their reasonable conformity. At the same time, submitted by Halden project the very detailed experimental data present the good basis for further improvement of a code. Such work is scheduled within the framework of the program FUMEX.

5 Prospects of VVER-1000 Fuel Modelling

The majority of codes created at present reasonably simulate a fuel behavior at design burnup, that is confirmed by results of accounts, executed by the participants in the program FUMEX.

At the same time, the planned increase of fuel design burnup puts additional problems before the developers of the codes. The decision then is connected with calculating and experimental researchers on such directions, as:

- refinement of FGR models under extended burnup;
- fuel pin behavior in maneuver mode of operation at increased burnup;
- fuel conductivity degradation under burnup taking into account the complex enough nature of this phenomenon;
- the features of fuel pin behavior in conditions of dense contact between pellet and cladding, including heat transfer and mechanical interaction;
- development of secondary defects in leak fuel pin;
- study of formation process and properties of external porous pellet rim concerning to its influence on fuel temperatures and FGR.

This rim occurs in a zone increased local burnup on pellet external edge, which is stipulated by plutonium formation there. On Fig. 18 the calculating radial distributions in the pellet of fuel pin VVER-1000 are shown at various average burnups. As it is visible the non-uniformity is increased with burnup growth and together with them a layer with increased local burnup is enlarged. The structure changes and the possible significance's of thickness of this layer are that the necessity of the account of this phenomenon at fuel modeling is obvious.

The results of mentioned above works will be used for further improvement of code START.

6 Conclusion

The comparison of calculated and experimental data shows ability enough of the code START-3 to simulate a fuel pin behavior in normal operation conditions.

The calculations confirm the experimentally observed evidence of an essential margin on serviceability of fuel pin VVER-1000 with three year operation cycle, that permits to increase a design fuel burnup in the nearest future.

The main directions of work on further development of modeling methods consists of the calculated and experimental researches of features of a fuel behavior at extended burnup.

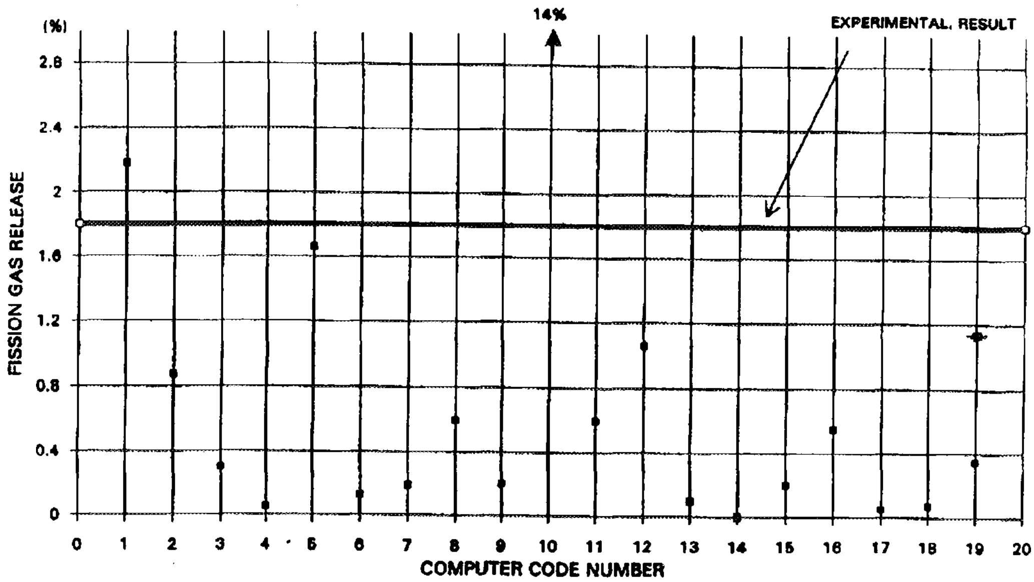

Figure 5 FUMEX 1: E.O.L fission gas release

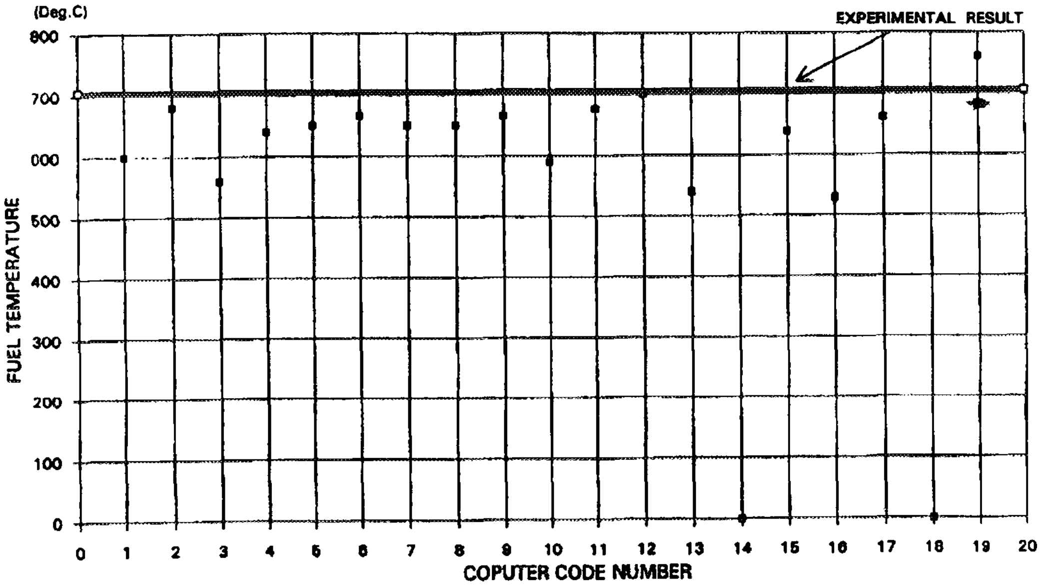

Figure 6 FUMEX 1: Fuel central temperature at 5 MWd/kg and 15 kW/m

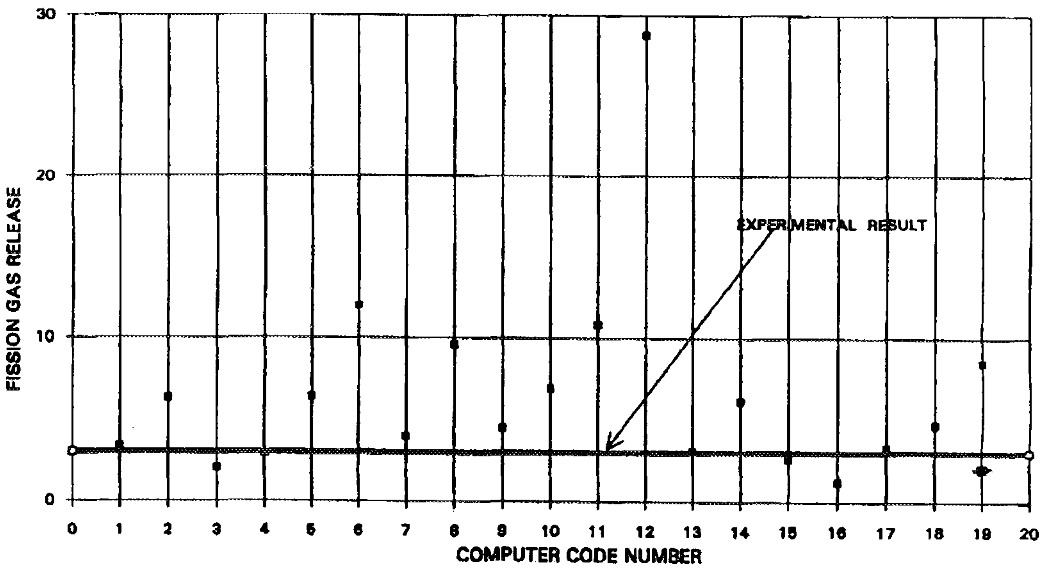

Figure 7 FUMEX 2: E.O.L fission gas release

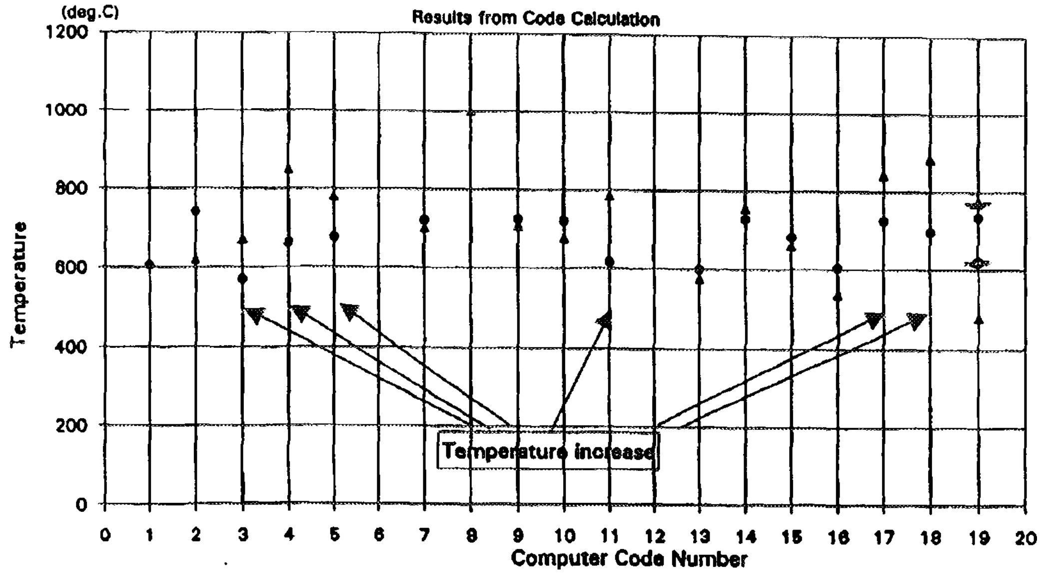

Figure 8 FUMEX 2: Fuel central temperature at 5 MWd/kg, 15 kW/m (●) and E.O.L (Δ)

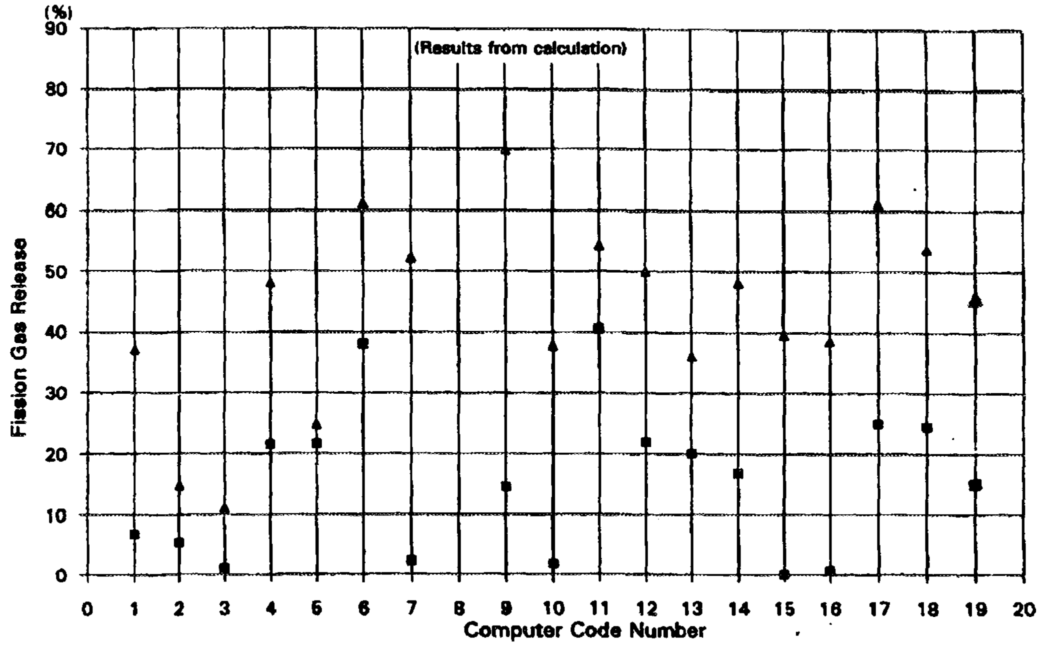

Figure 9 FUMEX 3.1: Fission gas release just before (■) and after (Δ) ramp

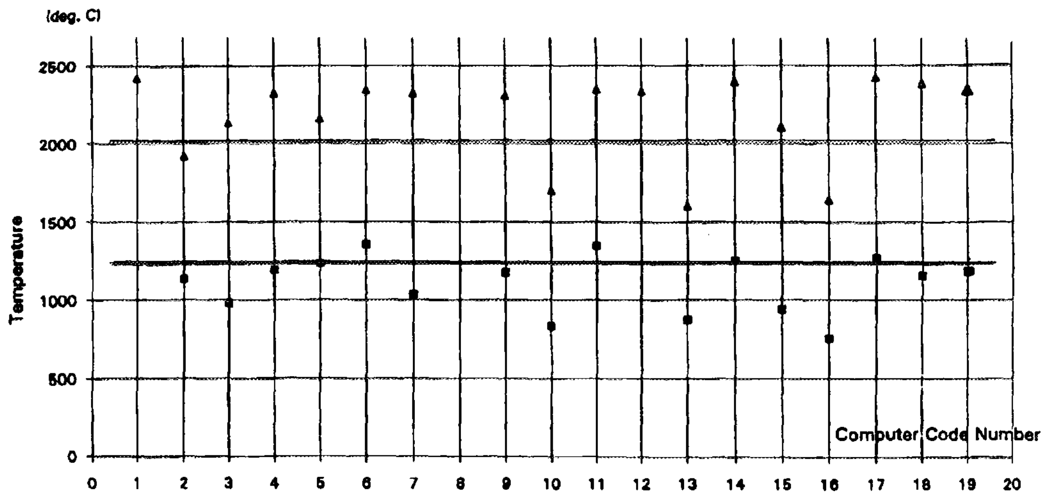

Figure 10 FUMEX 2: Fuel central temperature just before (■) and at top (Δ) of the ramp; experimental data: ━ before and — after ramp.

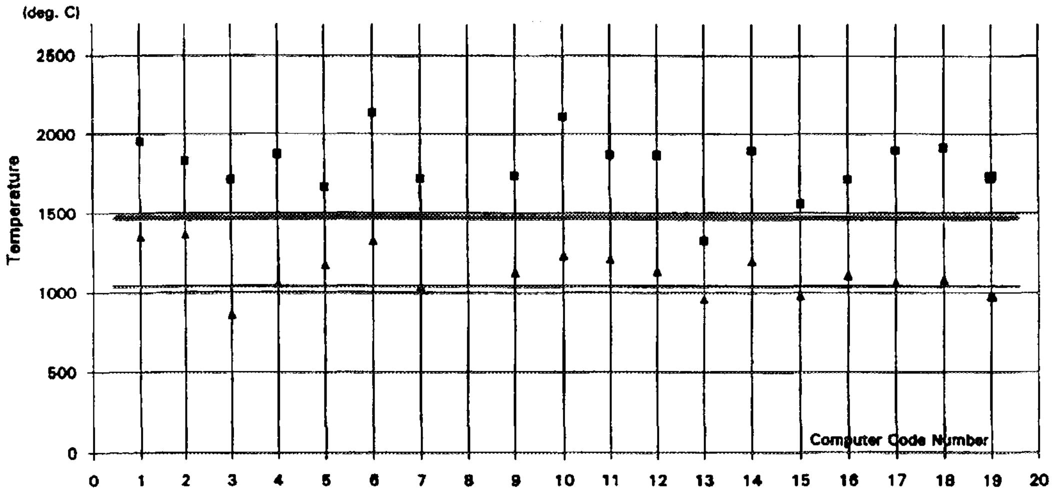

Figure 11 FUMEX 3.2: Fuel central temperature at 25 kW/m, before (Δ) and at the top (■) of the ramp; experimental data: — before and ▬ at top of the ramp.

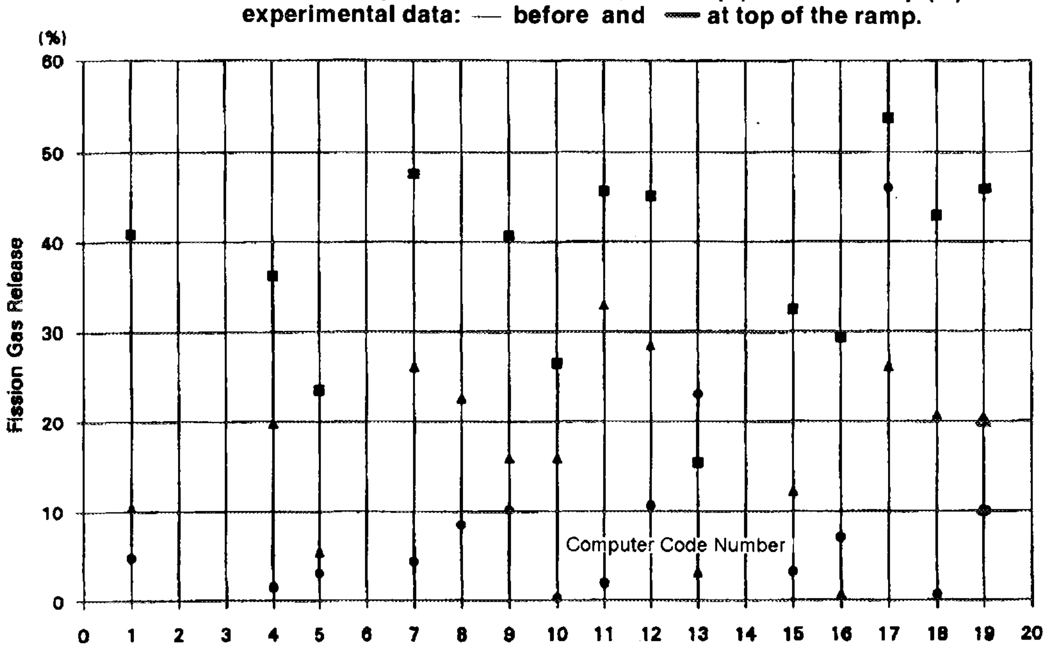

Figure 12 FUMEX 4A: Fission gas release before (●), after (Δ) ramp and E.O.L (■)

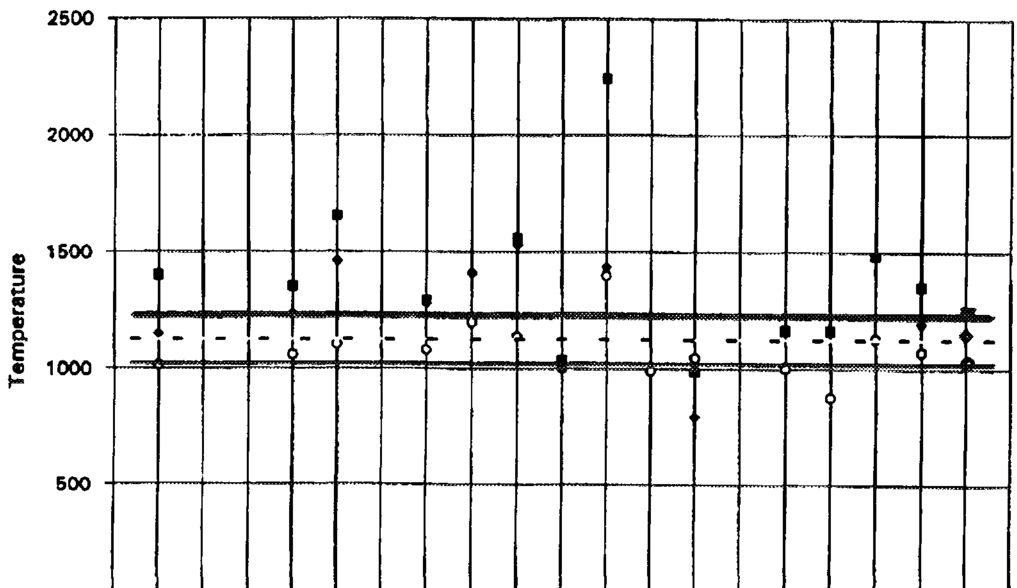

Figure 13 FUMEX 4A: Fuel central temperature at 30 kW/m, start-up (O), ramp (◆) and E.O.L (■); experimental data: — start-up, – – ramp and ▬ E.O.L.

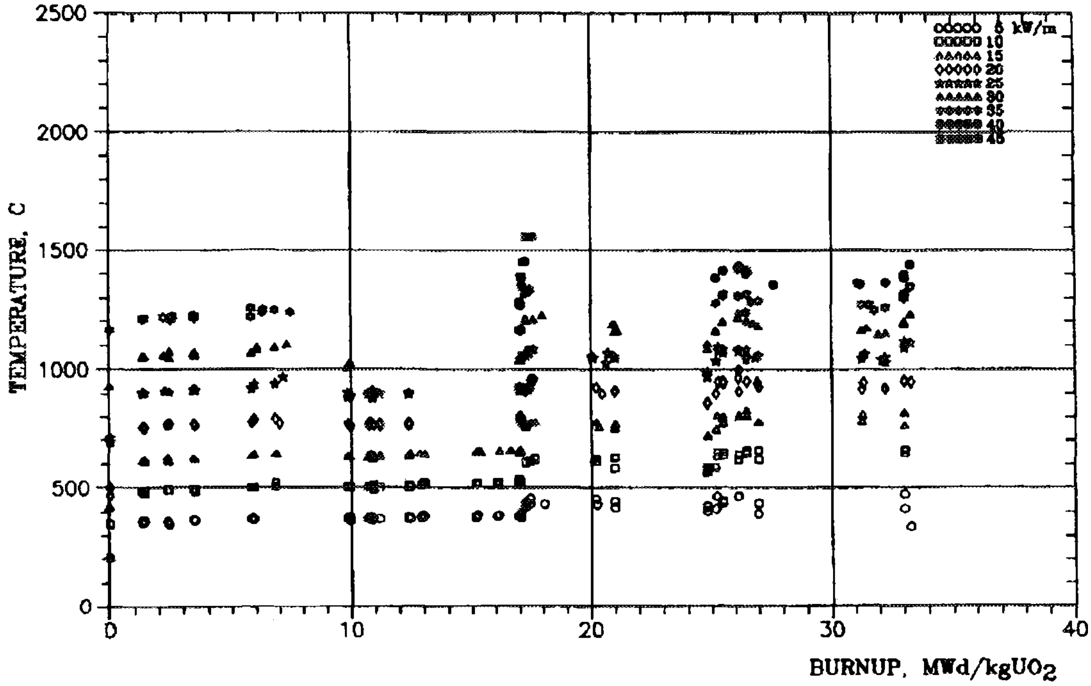

Figure 14 FUMEX 4A: Fuel central temperature vs. burnup

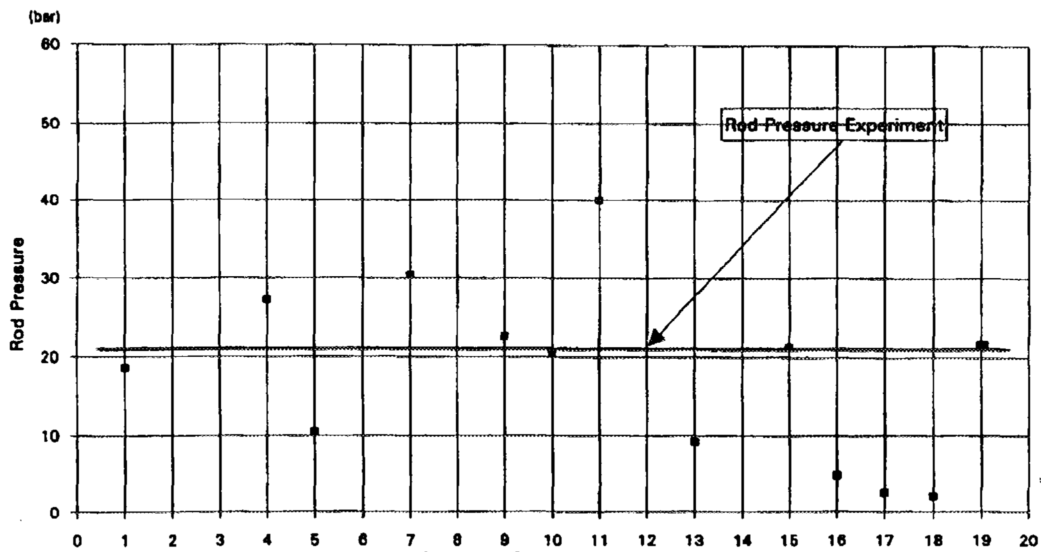

Figure 15 FUMEX 4A: Rod pressure after ramp (hot conditions)

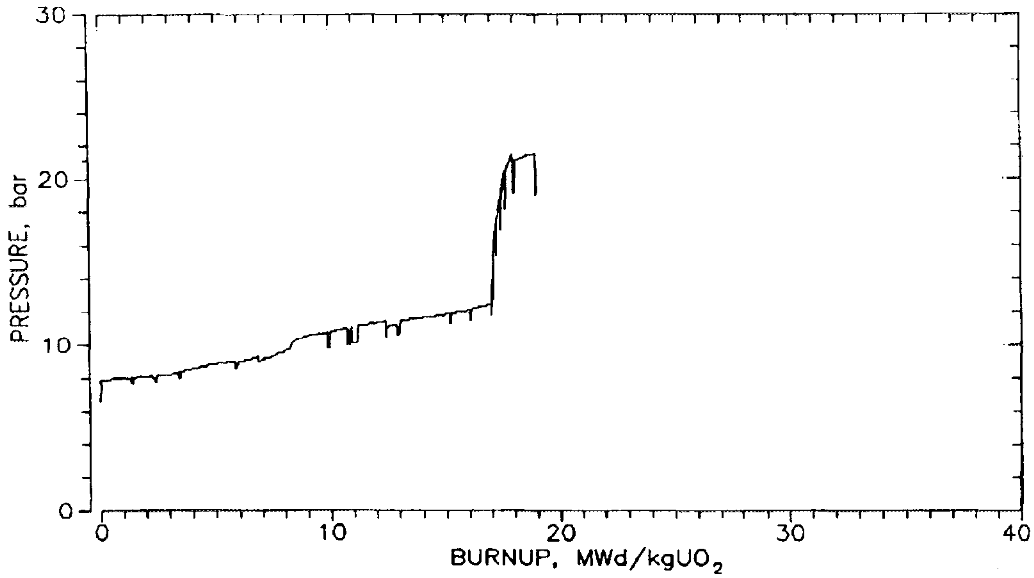

Figure 16 FUMEX 4A: Pressure vs. burnup

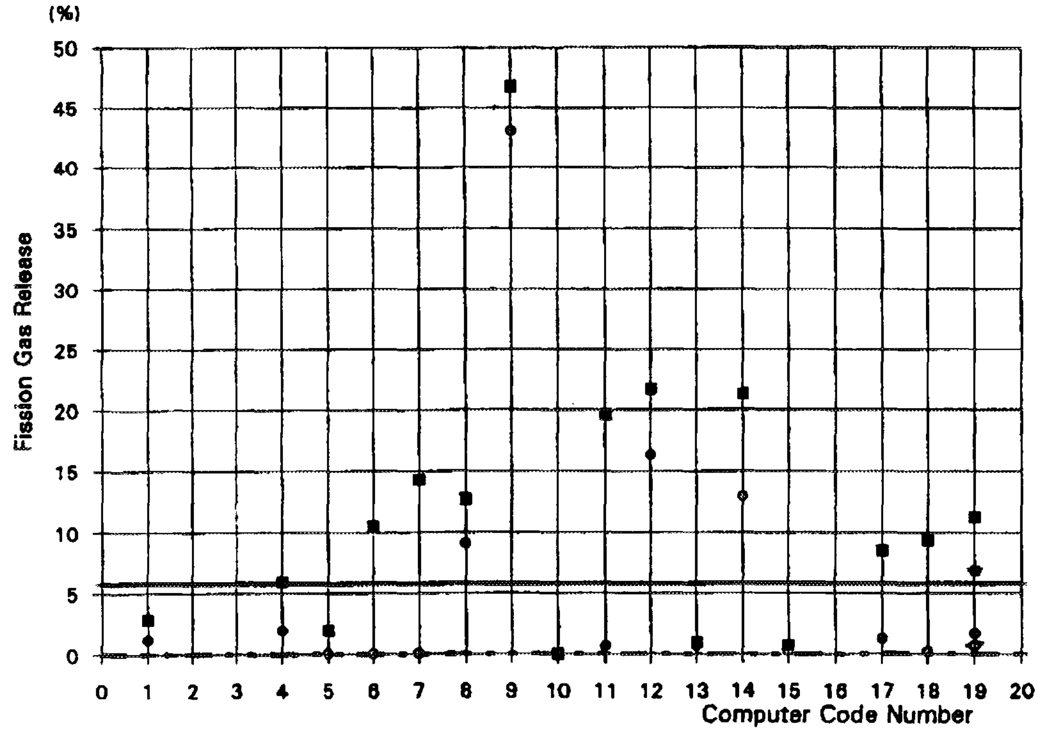

Figure 17 FUMEX 5: Fission gas release at start of ramp (O) and E.O.L (■); experimental data: – – start of ramp and ▬ E.O.L.

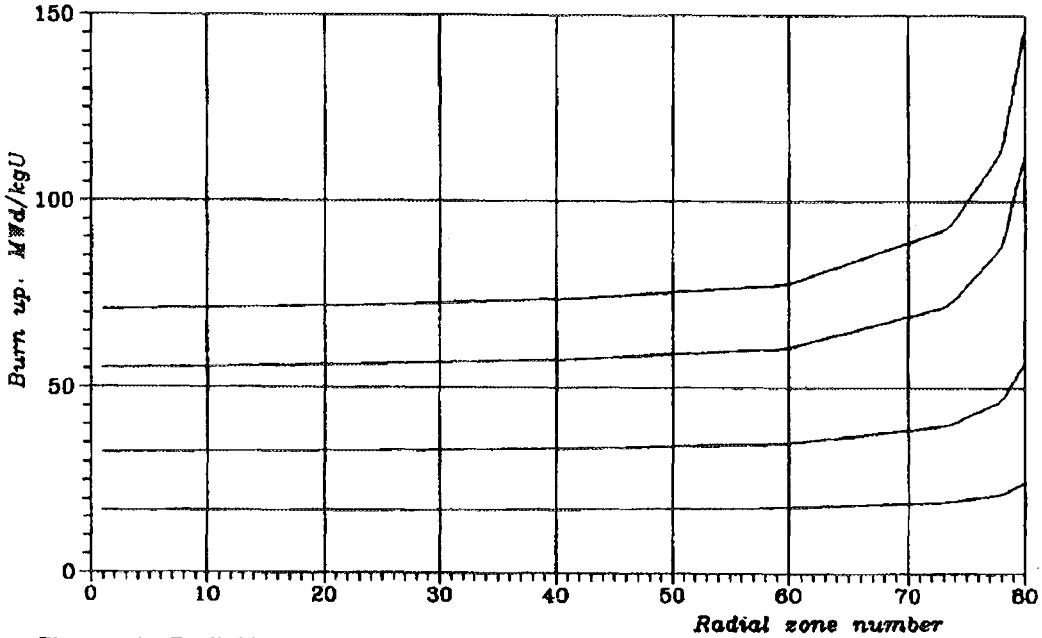

Figure 18 Radial burnup distribution for several average fuel pellet burnup levels

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Main Examination Results of VVER-1000 Fuel After Its Irradiation in Power Reactors

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1 Introduction

The main purpose of the VVER-1000 fuel examination was to study the efficiency of their operation at steady-state regimes during two or three cycles and to assess the potential for burnup and operation length extension. For this purpose the examination of the fuel assembly members was performed which enabled us to specify the properties of the materials and their influence on each other in power reactor operating conditions. On the basis of this examination the fuel assembly design was improved and the calculation codes were verified. Furthermore, the reasons and consequences of fuel failures were studied.

To date, nine fuel assemblies have been examined, one of which was - reference, two others - unsealed. All of them were discharged from the reactor without achieving the project burnup (Table 1).

2 Examination Program

All nine fuel assemblies were investigated according to one program including the following methods:

- visual inspection;
- measurement of overall dimensions;
- eddy-current test;
- gamma-scanning;
- X-ray and neutron radiography;
- analysis of gas pressure and composition inside fuel rods;
- ceramography/metallography;
- mass spectrometry;
- microanalysis and electron microscopy of fuel and fuel claddings.

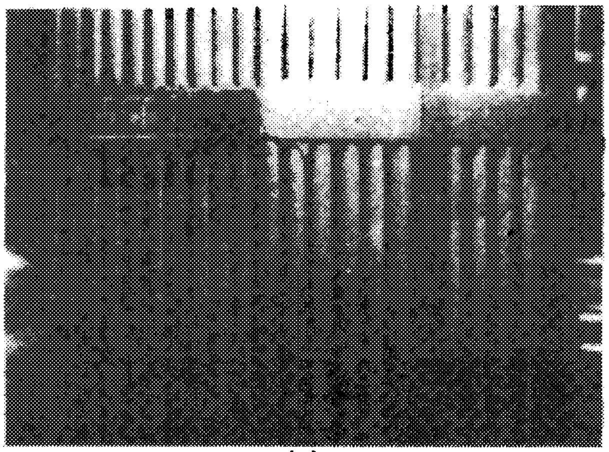
Using these methods the values of the main parameters were obtained characterizing the condition of fuel assembly and its structural members. The main of them are:

- change of fuel assembly and its members configuration;
- distribution of oxide film thickness along the fuel rod length;
- distribution of fuel cladding gap along the fuel rod length;
- state of fuel stack, welding, spacer grids, assembly, spring block and guides of absorber channels;
- distribution of fuel burnup, fission products and actinides along the height and radius of the fuel rods;
- distribution of material properties along fuel assembly volume, amount and character of hydrides in zirconium alloys;
- availability and character of defects.

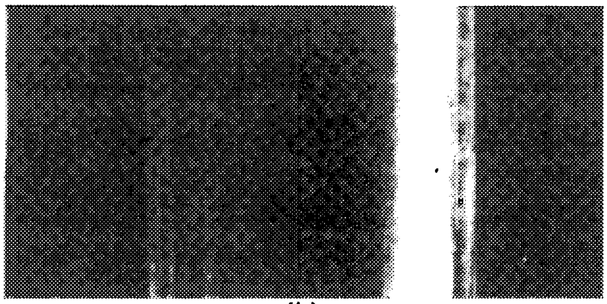
Additionally each fuel assembly was investigated by other methods determined by the specific research tasks, such as fuel refabrication with its further testing in the research reactors or in the in-cell stands.

Table 1 List of the Examined VVER-1000 Fuel Assemblies

No.	FA	Reactor, NPP	Operation period	Duration eff. days	Average burnup	Initial fuel enrichment
1	0068	5-NVNPP	31.10.83 - 09.05.85	491	21.7	2.4; 3.0; 3.3
2	0007	5-NVNPP	12.09.82 - 09.05.85	825	32.6	3.3
3	0106	1-SUNPP	22.11.82 - 14.08.86	878.4	36.7	3.3
4	1114	5-NVNPP	25.06.84 - 25.06.87	923.1	44.7	3.6; 4.4
5	1565	2-Kal.NPP	05.11.85 - 02.07.88	838.7	34.7	3.3
6	1568	2-Kal.NPP	05.11.85 - 02.07.88	838.7	32.9	3.3
7	0623		10.10.88 - 17.06.89	236.8	19.5	3.6; 4.4
8	4108	5-NVNPP	05.07.86 - 11.07.90	881.4	46.2	3.6; 4.4
9	0619	1-Kal.NPP	10.10.88 - 23.11.91	859.5	42.4	3.6; 4.4



(a)



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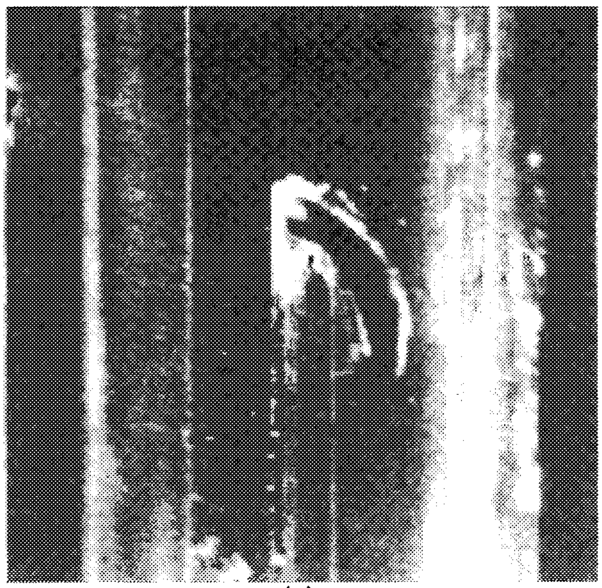
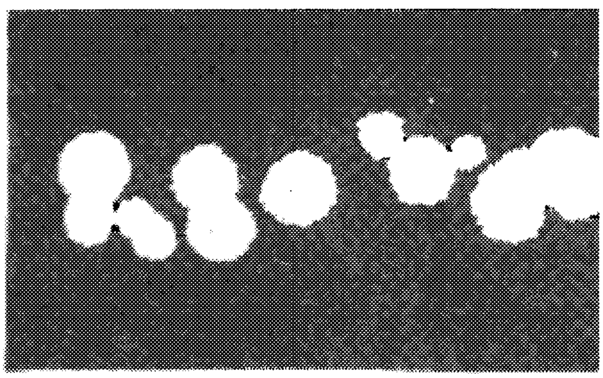


(c)



(d)

Figure 1 Fuel assembly (a), fuel element (b), fuel cladding defect (c) and nodular corrosion (c).

3 Examination Results

3.1 Visual Inspection

The visual inspection testified that there were no substantial changes in the fuel assembly design nor defects or configuration modifications of its structural members. The surface of cladding and wrappers was of dark color, typical for the autoclaved zirconium alloys (Fig. 1). The failed fuel rods were found to have defects caused the interaction of cladding with solid coolant particles. Moreover, some fuel cladding defects were observed which could be explained by spacer grid fretting-corrosion effect (Fig. 1). The nodular corrosion was found on the fuel rods of the South-Ukrainian assembly which was discharged after the disrupture of the main circulating pump and penetration of graphite into the coolant (Fig. 1). It could be caused by the specific microenvironment formed under the deposits of graphite compounds on the cladding.

3.2 Dimensional Changes of Fuel Assembly and Its Fragments

In contrast to the VVER-440 the majority of VVER-1000 units use the wrapperless fuel assemblies. The bend of these fuel assemblies is of intricate shape and for the fuel assemblies represented in Table 1 achieves 4.8 *mm*. No evidence of the post-reactor elongation of the wrapperless fuel assemblies was noticed.

A substantial scattering of fuel rods elongation was observed even in one fuel assmeby (Fig. 2a). But an averaged elongation (Fig. 2b) had a linear correlation with fuel burnup.

The typical axial distribution of fuel rod diameter is given in Fig. 3. Taking into account the initial was observed of fuel pellet and inner cladding diameter it should be expected that in some cases the fuel-cladding gap will disappear even at burnups of 35 *MWd/kgU*. But the examination shows that on the average the decrease of fuel cladding diameter was observed at burnups up to 50 *MWd/kgU* (Fig. 4).

In the process of operation the channel guides of the safety and control rods take the shape of the fuel assembly. The change of length, outer or inner diameter was not detected. The measurement of spacer grid cells diameter revealed the increase of distinction between fuel rod diameter and that of grid cell being freed from the fuel rods (Fig. 4).

3.3 Fuel Rod Cladding Condition

As a rule, after operation the inner surface of the VVER-fuel rod claddings was covered by dark uniform oxide film of 4 to 8 μm thick which increases by 1 - 2 μm in the upper part and practically does not depend on burnup (Fig. 5a). On the inner cladding surface the oxide film thickness modifies in the range from 0 to 10 μm and it can by essentially changed along the cladding perimeter (Fig. 5b).

Sometimes claddings are found to have small quantities of lamellar form hydrides, the size of which does not exceed 50 μm. In this case the arrangement of the hydrides has random or tangential orientation (Fig. 6). The hydrogen content in claddings does not exceed (5 - 6)$\cdot 10^{-3}$ *mass%*.

The mechanical properties of material claddings were practically the same for all the fuel assemblies that have been tested and are independent of the sample place and fuel burnup (Fig. 2). The yield strength at operating temperatures is no less than 300 *MPa*. As for the uniform and total elongation, they are respectfully no less than 4 and 15% at room temperature.

All types of VVER-1000 cladding failures which caused the premature discharge of fuel assemblies can be referred to the interaction of fuel rods with solid foreign particles. Some fuel assemblies were found to have essential cladding failures that did not result in fuel rod unsealing (Fig. 7).

3.4 State of Fuel Stack

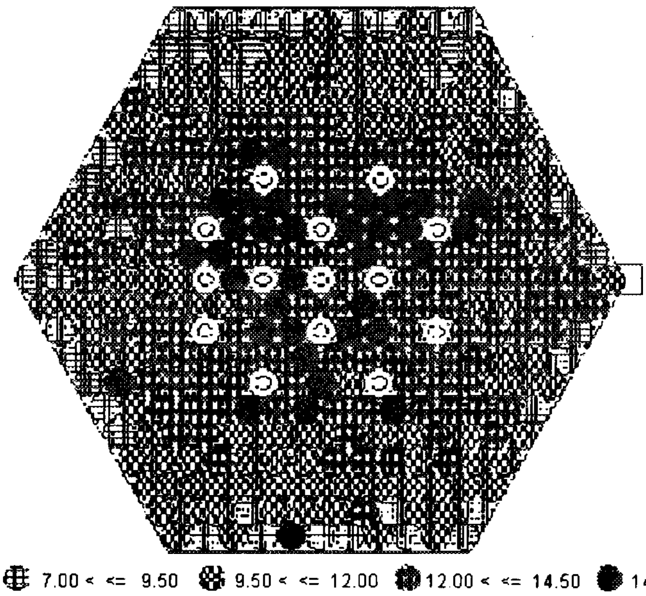
The dimensions of fuel pellet central hole do not change in the burnup range covered (Fig. 8). The change of the grain size goes slowly (Fig. 9) and that points to the low (lower than 1500°C) fuel temperature. The maximum dioxide uranium swelling does not exceed 4%. Fuel pellets are mainly fragmented by 3 to 5 parts preserving therewith its cylindrical shape. The fuel column length in some cases increases up to 58 *mm*.

The distribution of all radionuclide fission products (Fig. 10) points to the absence of the axial migration among them and confirms the low fuel temperature in the operation process. In the areas of spacer grids the decrease of radionuclide concentration is observed

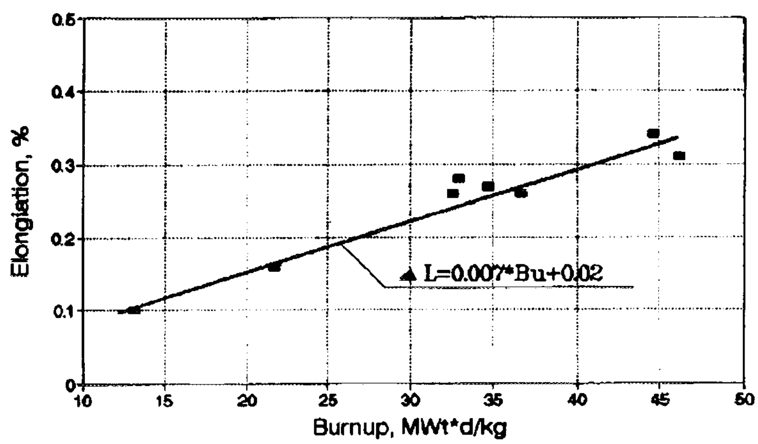
The radial distribution of fission products and actinides also points to the absence of their noticeable migration.

3.5 Pressure and Fuel Gas Composition

In the process of fuel burnup the fuel gas pressure slowly increases and for those fuel assemblies which have been tested did not exceed 30 *atm* (Fig. 11). The pressure increase was conditioned by the noble gas fission products release form fuel. The dependence of gas fission products release from the size of fuel grains has been found.



(a)



(b)

Figure 2 Elongation of fuel elements of VVER-1000:
(a) elongation of fuel elements into fuel assembly;
(b) dependence of fuel rod elongation on average fuel burnup

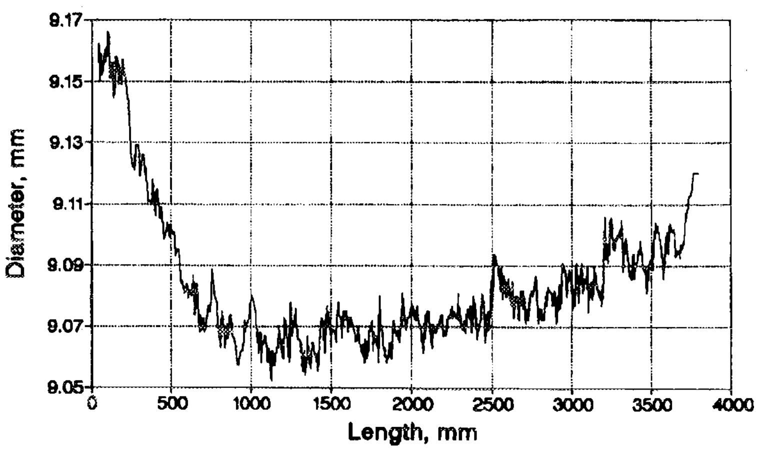
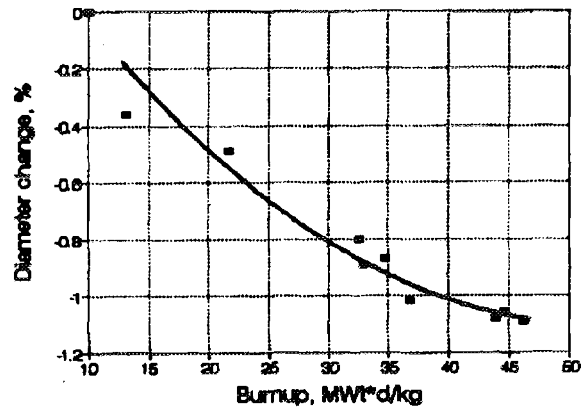


Figure 3 Distribution of fuel rod diameter size along length of the VVER-1000 fuel element

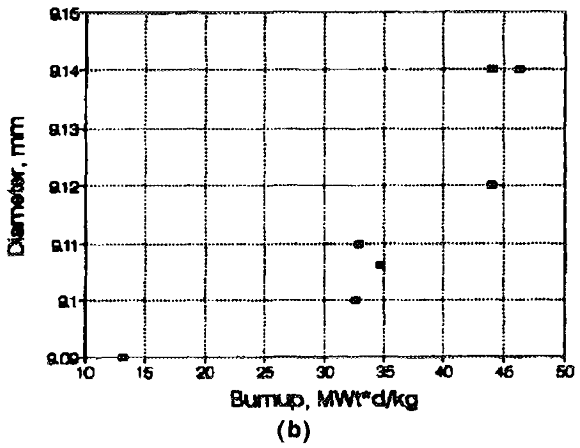
4 Conclusion

The examination results suggest that the VVER-1000 fuel spent at steady-state operating conditions up to 50 *MWd/kgU* of burnup is in satisfactory condition. The examination of all the types of the significant fuel cladding failures indicates that the reason lies in the interaction of cladding with coolant solid impurities. The nodular cladding corrosion of fuel assembly discharged from the South-Ukrainian NPP is caused by the graphite compounds deposited on the fuel rod. Those depositions were a result of the circulating pump damage and beard an accidental, non-typical character. Some of the fuel rods were found to have a small cladding "fretting" of the spacer grid cell material.

The values of the majority of parameters determining the fuel efficiency allow an assumption that there is a potential for further extension of fuel burnup and operation length.



(a)



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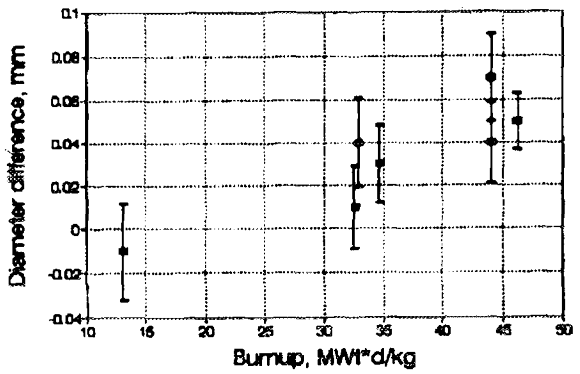
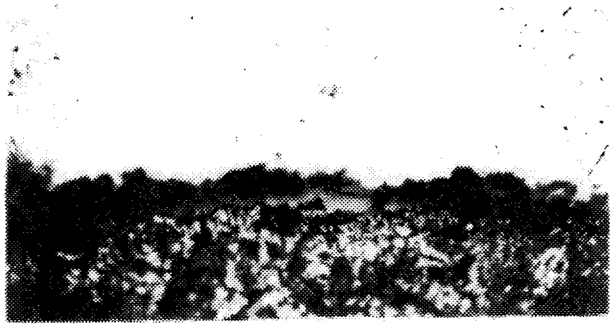


(c)

Figure 4 **Dependence of fuel pin diameter change (a), spacers hole diameter (b) and diameter difference of spacer hole and fuel pin (c) on fuel burnup**

(a)



(b)

Figure 5 **The oxide film on the outer (a) and inner (b) surfaces**

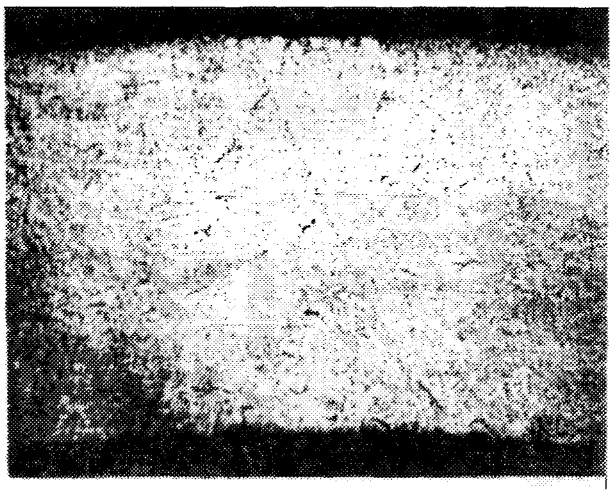


Figure 6 **Characteristic structure of irradiated claddings**

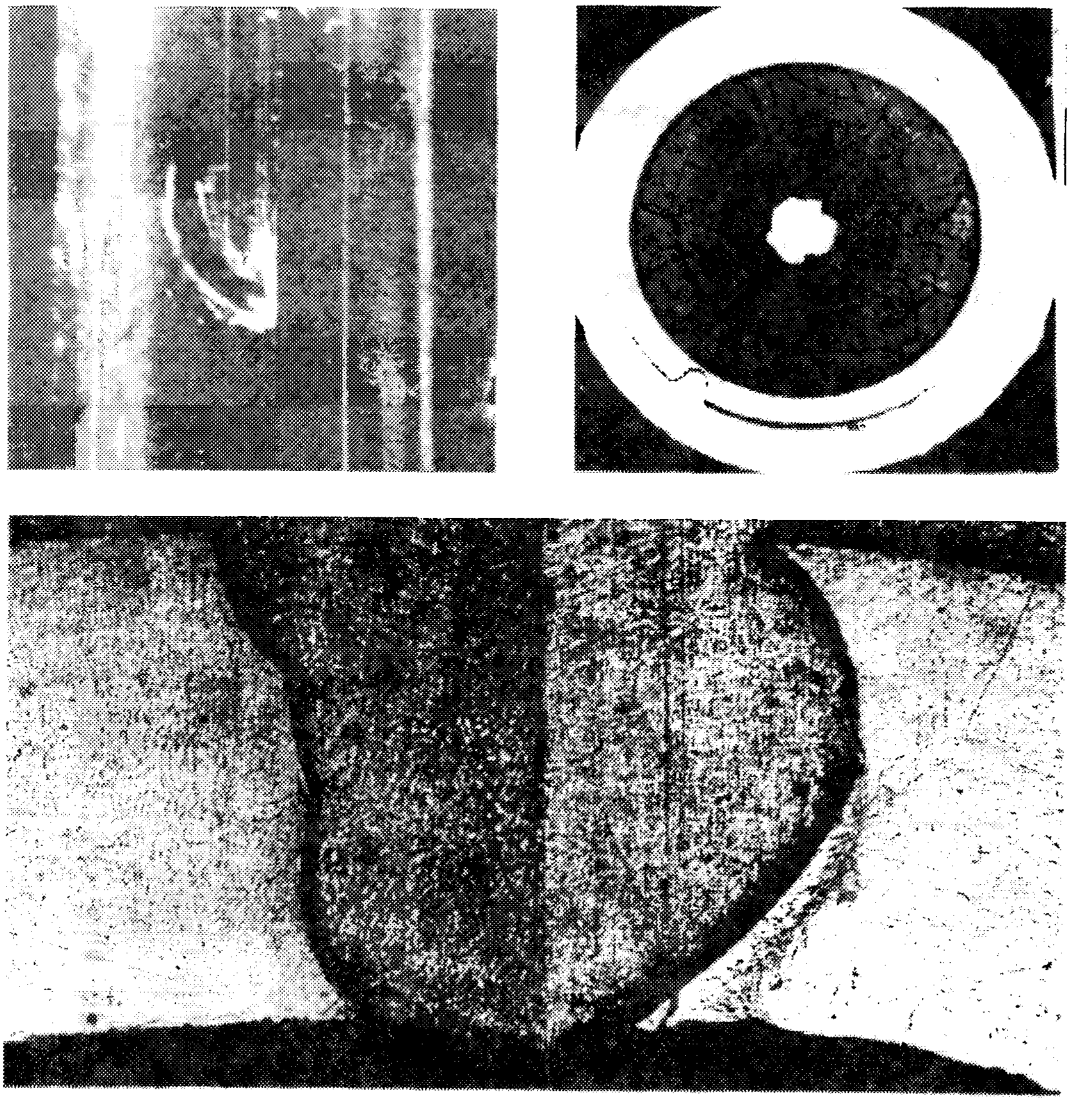


Figure 7 Cladding defect

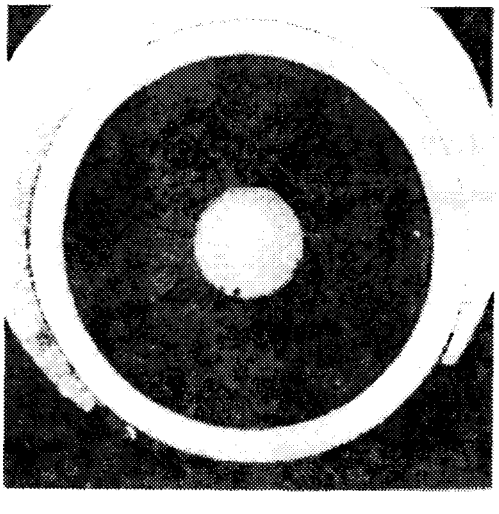


Figure 8 Fuel element macrostructure

Figure 9 Fuel microstructure

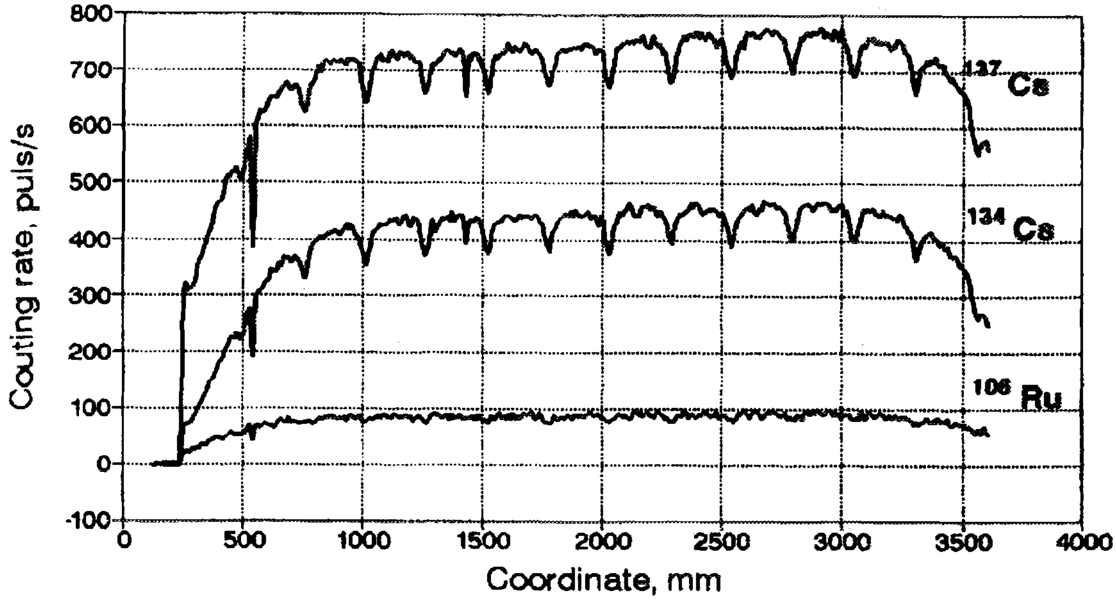


Figure 10 Distribution of fission products gamma activity along the fuel rod length

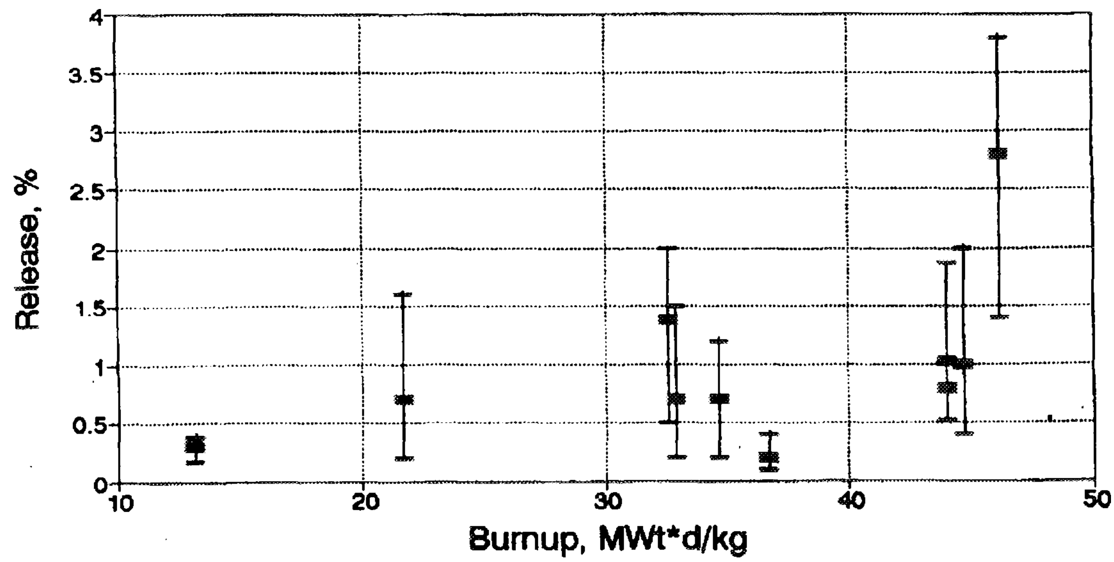


Figure 11 Dependence of fission gas release on burnup

First Qualitative Analysis of Fuel Irradiation Results Carried Out in the MR Reactor on VVER-1000 Fuel

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1 Introduction

Two Russian papers on VVER fuel behaviour at high burnup and high linear rate were recently published separately. Although the irradiation parameters are not always representative of the reactor standard functioning, it was found interesting to put together these results to improve and disseminate the knowledge of VVER fuel behaviour with central hole and also to contribute to the general information on high burnup, highly rated water reactor fuel behaviour. In the future, this first interpretation would need to be improved by some modelling based on more accurate fabrication and irradiation data.

In this paper results on VVER 1000 fuel rod behaviour reported in Ref. [2] are examined, Four experiments which were carried out in the MR reactor are reviewed. They are numbers 13, 19 and 21 (c and d) [1]. The main goal of these experiments was to assess the influence of burnup and the size of the pellet central hole on the fuel temperature and thus on the fuel swelling and the fission gas release. These experiments were carried out at different linear rates and burnups.

In any case the aim of this paper is not to make a judgement on the VVER fuel or a comparison with western PWR fuels. The main reason is that functioning requirements of VVER fuel are different (higher linear rate) and that they are designed accordingly (plenum for gas expansion). It has to be noticed also that the linear rate of some of these experiments is above the linear rate experienced in VVER power stations, therefore these experiments can only be considered as extreme cases to explore the limits of utilization or safety cases.

2 Fabrication and Irradiation Parameters

In all cases the outside cladding diameter and wall thicknesses were 9.05 - 9.18 *mm* and 0.68 *mm* respectively. Other fabrication and irradiation parameters are given in Table 1 [1, 2], they are not always complete or sufficiently accurate to make a complete assessment, however it has been considered that dimensional and physical parameters were in the same specification range, making possible the comparison between these experiments. The higher helium pressure in the irradiation 19 and 21/d (see Table 1) compared to 13 and 21/c probably caused a decrease in the beginning of life temperature but the influence of this parameter on the irradiation results at high burnup has been considered as minor. The beginning of life (BOL) and end of life (EOL) linear rate of the fuel rods was recalculated from the power histories of each irradiation and burnup. This is not a rigorous method, therefore these values can only be considered as an order of magnitude.

From the view point of burnups, experiments 13 (small central hole) and 19 (large central hole) were irradiated to close levels (41.5 - 48.7 and 37.6 - 43.3 *MWd/kg*) but at different linear powers, reaching 510 and 440 *W/cm* respectively.

Experiments 19 (large central hole) and 21 (large and small central hole) experienced close levels in linear powers at the beginning of life (440 and 460 *W/cm*) but reached average burnups of 43 and nearly 80 *MWd/kg* respectively.

3 Fuel Examination Results

Post-irradiation examinations showed that VVER-1000 fuel rod claddings demonstrated the good

Table 1 Main fabrication and irradiation characteristics

Experim. No.	Fuel density g/cm^3	Central hole diam., *mm*	Gap diam. *mm*	Initial He pressure *MPa*	Irradiat. time, *hours*	Average burnup in a rod, *MWd/kg*	Max. linear heat rate *W/cm*	
							BOL	EOL
13	10.40 - 10.70	1.4 - 1.6	0.19 - 0.29	2.0	17658	41.5 / 48.7	432 / 510	169 / 200
19	10.50 - 10.70	2.3 - 2.5	0.19 - 0.32	2.4	22949	37.6 / 43.3	382 / 440	179 / 206
21 / c	10.56 - 10.72	2.2 - 2.4	0.25 - 0.29	2.0	40164	61.8 / 78.8	361 / 460	133 / 170
21 / d	10.40 - 10.60	1.4 - 1.6	0.19 - 0.32	2.5	40164	58.6 / 79.9	336 / 458	124 / 169

Table 2 **Experiment No. 13 : Main examination results (small central hole and high linear rate, He: 2.0 MPa, except A12)**

Rod No.	Burnup *MWd/kg*		FGR %	Length changes *mm*	Diam. changes *mm*	Max. linear rate, *W/cm*		Cs shifting
	Average	Max.				BOL	EOL	
A1	48.7	60.4	41	0.5	-0.021	507	199	yes
A2	48.3	59.9	35	0.5	-0.027	503	197	yes
A3	46.3	57.4	38	0	-0.036	482	189	yes
A4	44.6	55.8	32	0.4	-0.014	469	184	yes
A5	44.5	55.6	32	0.3	-0.028	467	183	yes
A6	41.7	51.7	37	0	-0.033	434	170	yes
A7	41.5	51.4	22	0.3	-0.026	432	169	yes
A8	42.6	53.2	39	0.3	-0.030	447	175	yes
A9	42.8	52.7	38	0.5	-0.019	443	174	yes
A10	46.5	57.2	34	0.5	-0.019	480	188	yes
A11[1]	43.1	51.8	-	-	-0.015	435	171	yes
A12[2]	48.2	60.7	38	-	-0.055	510	200	yes

[1] Failed fuel rod.
[2] The initial He pressure in this rod was 0.1 MPa.

Table 3 **Experiment No. 19: Main examination results (large central hole, He: 2.4 MPa.)**

Rod No.	Burnup *MWd/kg*		FGR %	Length change *mm*	Diam. change *mm*	Max. linear rate, *W/cm*		Cs shifting
	Average	Max.				BOL	EOL	
F1	42.4	57.2	11	0	-0.004	431	202	no
F2[1]	42.3	57.1	11	0	0.016	430	201	no
F3	43.1	58.2	14	0	-0.017	438	205	no
F4[1]	40.8	55.1	10	1.2	0.012	415	194	no
F5	40.1	54.1	12	1.0	-0.007	408	191	no
F6[1]	38.5	52.0	12	0.3	0.014	392	183	no
F7	37.6	50.7	14	-0.2	-0.020	382	179	no
F8[1]	38.0	51.3	12	-0.3	-0.007	386	181	no
F9	41.0	55.3	13	-0.1	-0.012	416	195	no
F10[1]	42.4	57.2	13	0.2	0.007	431	202	no
F11	43.3	58.4	14	-0.3	-0.008	440	206	no
F12[1]	42.9	58.0	9	0.2	0.016	437	205	no

[1] Cladding of these fuel rods was made of a new alloy.

corrosion and mechanical properties after irradiation during 6 years to maximum burnup of 93 *MWd/kg*. Oxide film thickness did not exceed 15 microns, hydrogen content was not more than 100 *ppm* and the claddings had high strength and rather high ductility. Experimental data were reported in Ref. [1, 2].

The other results of destructive and non-destructive examinations are given in Tables 2, 3, 4 and 5. They include: burnup determination by gamma spectrometry, caesium spectrometry, fission gas release, fuel rod diameter and length change and macrographs showing the central hole size and fuel morphology after irradiation.

Destructive examinations were performed on a few rods from each experiment, the main results are summarized in Table 5. Data on fission gas release and caesium shifting are plotted in Fig. 1.

3.1 Fission Gas Release (Figures 1 and 2)

In Fig. 1, fission gas release rate is plotted according to the average burnup for all fuel rods.

Comparing experiments 19 and 13, a large increase of FGR has been observed which is probably due to the large difference in linear rate (440 and 510 *W/cm*), however, the significantly less fission gas release observed in experiment 19 (9 - 14%) than that recorded in experiment 13

Table 4 Experiment No. 21: Main examination results (21/c large central hole, 21/d small central hole)

Rod No.	Burnup *MWd/kg*		FGR %	Length changes *mm*	Diam. changes *mm*	Max. linear rate, *W/cm*		Cs shifting
	Average	Max.				BOL	EOL	
C1	75.1	88.6	41	0.3	-0.010	438	162	yes
C2	66.3	78.2	29	0.9	-0.010	378	143	no
C3	62.0	73.2	31	0.3	-0.040	362	134	no
C4	61.8	72.9	17	0.9	-0.020	361	133	no
C5	69.6	82.1	21	0.6	-0.030	406	150	no
C6	78.8	93.0	33	1.0	-0.020	460	170	yes
D1	79.9	92.6	38	1.8	0.030	458	169	yes
D2	75.6	87.7	41	1.5	0.040	434	160	yes
D3	64.4	74.6	24	1.0	0.010	369	135	yes
D4	58.6	67.9	17	2.4	0.010	336	124	no
D5	68.4	79.3	29	0.8	0.010	392	145	yes
D6	78.4	91.0	33	0.8	0.030	450	166	yes

(22 - 38%) in rods at a similar beginning of life linear rates (430 - 450 *W/cm*) can be explained by the beneficial influence of a large central hole on fuel temperature.

In experiment 21, the larger scatter of the FGR rates should be noted. No correlation can be found between the size of the central hole and fission gas release, both experiments 21/c and 21/d show a similar release. The large scatter of results could be the reason for this, in this case, more data would be needed to appreciate the difference, if any. A possible explanation is that the beneficial effect of a large central hole has been compensated by the higher He pressure in experiment 21/d with small central hole. Other possible reasons could be: that new phenomena appear at high burnup, for instance that the fuel densification at beginning of life, at higher temperatures with a small central hole, is beneficial for fission gas release at high burnup; or that the benefit of a larger central hole is attenuated at high burnup perhaps because of the fuel swelling effect.

In the same experiments but in the rods irradiated at low linear rates (340 - 360 *W/cm*) the FGR rate was limited to 17% at 60 *MWd/kg.*

In experiment 19 at moderate linear rate and burnup, a relatively high fission gas release was found (10 - 15%), part of it could be due to the cracks appearing in pellets at reactor start up and shut down, however, on the one hand, no such

Table 5 Results of destructive and non-destructive examinations of irradiations 13, 19 and 21

Experiment No.	Rod No.	Av. Burnup, *MWd/kg*, & Max. linear rate, *W/cm*	FGR %	Cs shifting	Central hole diam., *mm*, after/before irradiation	Fuel restructuring	Figure No.
13 small central hole	A1	48.7/507	41	yes	0.6/1.5	Restruct., col. grains	4a
	A7	41.5/432	22	yes	0/1.5	Restruct. cent. hole disappeared	4b
19 big central hole	F11	43.3/440	14	no	2.5/2.4	No restructuring radial cracks	5a
	F6	38.5/392	12	no	2.5/2.4	No restructuring radial cracks	5b
21/c big central hole	C6	58.8/460	33	yes	2.0/2.3	Restruct., col. grains,	5c
	C4	61.8/361	17	no	2.3/2.3	No restructuring radial cracks	5d
21/d small central hole	D1	79.9/458	38	yes	1.2/1.5	Restruct., col. grains	4c
	D4	58.6/336	17	no	1.5/1.5	No restructuring radial cracks	4d

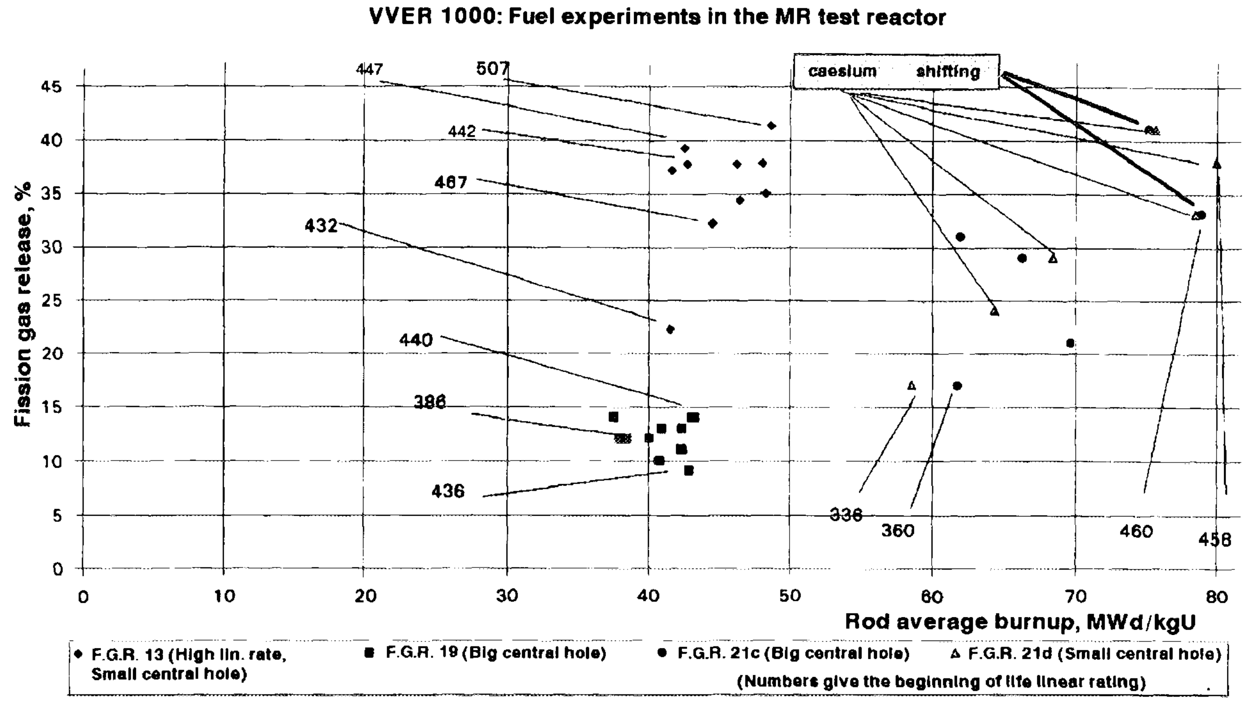


Figure 1 Fission gas release rate as function of rod average burnup for all MR experiments considered

phenomenon was shown in other reactors experiencing similar power changes (Halden), and on the other hand, it is known that the thermal process takes place when the fuel is irradiated at linear rate as high as 440 *W/cm*, therefore it seems that the contribution of the FGR thermal process is already important for these rods.

Comparing experiments 19 and 21/c (large central hole, close to an initial linear rate of 440 and 460 *W/cm*) it can be seen that FGR rates increased greatly between 40 and 80 *MWd/kg* average burnup. This release increase follows a polynomial function (Fig. 2) which can be expressed as follows:

$$\mathrm{FGR}(\%) = a + bx + cx^2 + dx^3$$

where: a = 0.1885; b = 0.31464; c = 0.003479; $d = 6.953 \cdot 10^{-5}$; x = rod average burnup in *MWd/kg*

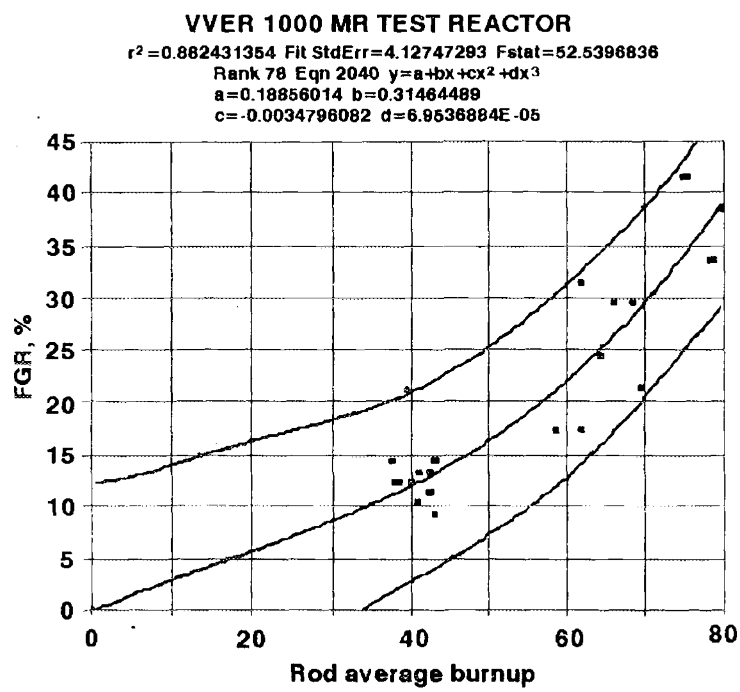


Figure 2 Fission gas release rate according to rod average burnup for experiments 19 and 21

3.2 Caesium Shifting (Figure 3)

From observations made on FBR fuel, Cs shifting along fuel column and Cs accumulation at the end of the fuel stack is interpreted as a sign of fuel overheating. The post-irradiation examination results show that Cs shifting occurred in rods where uranium dioxide restructuring, with formation of recrystallized grains, columnar grains or new central hole, are observed (as in Figures 4a-c and 5c). As can be seen from Tables 2 - 4, Cs shifting has affected all fuel rods with small central hole irradiated at initial linear rates above 340 *W/cm* and has not affected fuel rods with large central hole irradiated at initial linear rates less than 440 *W/cm*. Thus, if the beginning of life linear rate is the right parameter responsible for overheating, the large central hole would bring a global gain of about 100 *W/cm*.

3.3 Macrographic Observations (Figures 4 and 5)

Many fuel fractures can be observed on all macrographs, some are radial, others are circumferential and the crack healing is sometimes observable.

The fuel rods with small central hole were irradiated in experiment 13 at initial linear rates from 510 to 430 *W/cm* to an average burnup of 48.7 - 41.5 *MWd/kg*. All fuel rods show Cs shifting (Table 2) and fuel restructuring. As seen from Figure 4a, in the pellet irradiated at initial linear rate of 507 *W/cm* columnar grains and central hole 0.6 *mm* in diameter were formed and in the pellet irradiated at 432 *W/cm* central hole was completely filled by uranium dioxide (Figure 4b).

In the experiment 21/d the fuel rods with small central hole were irradiated at lower initial linear rate (458 - 336 *W/cm*) than rods irradiated in experiment 13 but reached a higher average burnup (79.9 - 58.6 *MWd/kg*). Fuel restructuring was observed in all fuel rods except in 21 (D4) irradiated at the lowest linear rate of 336 *W/cm*. As seen from Figure 4c, the pellet irradiated at initial linear rate of 458 *W/cm* shows more intensive restructuring (columnar grains and a new central hole 1.2 *mm* in diameter) than the pellet irradiated at 507 *W/cm* to lower burnup (Figure 4a). Thus, despite the large drop in linear rate (from 458 to 169 *W/cm*) during the irradiation time, one can conclude that at higher burnup restructuring takes place perhaps due to decreasing thermal conductivity of fuel and thus increasing fuel temperature.

The fuel rods with large central hole were irradiated in the experiments 19 and 21/c at comparable linear rates (from 440 to 460 *W/cm*) to average burnups of 37.6 - 43.3 and 61.8 - 78.8 *MWd/kg* respectively. In contrast to the small central hole, the big hole never disappeared. The diameter of the large hole decreased to 2.0 *mm* in the worst case (Table 5). No restructuring was found in experiment 19 even in the most rated fuel rod irradiated at 440 *W/cm* to maximum burnup of 58.4 *MWd/kg*.

Intensive restructuring with columnar grain appearance took place in the pellet irradiated in the central part of the fuel rod C6 (experiment 21/c) at 458 *W/cm* to 93 *MWd/kg* (peak). Noticeable restructuring was also observed in pellets located near the ends of this fuel rod and irradiated at about 350 *W/cm* (point) to 70 *MWd/kg* (Figure 9a from Ref. [2]). If the small difference in the size of the initial central hole of experiments 19 and 21/c can be neglected, it can be concluded that fuel restructuring took place also at high burnup in rods with a large central holes.

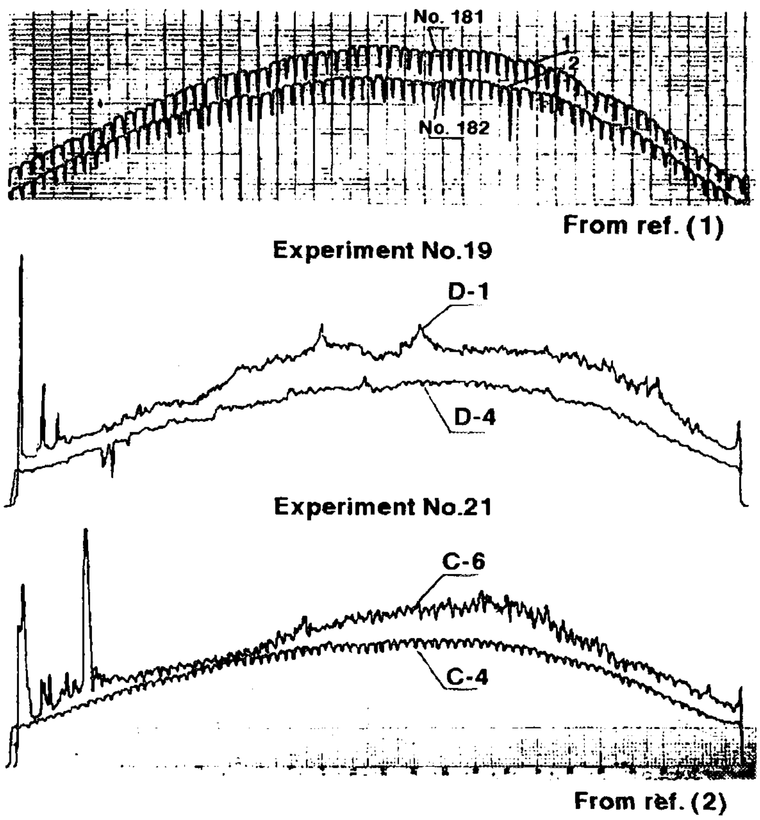


Figure 3 **Curves of distribution of total activity in rods with (D1, C6) and without (D4, C4) Cs shifting (experiments 19 and 21).**

4 Summary of the Main Findings

At intermediate burnup, when comparing experiments 13 and 19 in which fuel with small and large central holes was used respectively, the benefit of large central hole on Cs shifting, fission gas release and fuel restructuring was shown.

Considering Cs displacements, the benefit brought by a large central hole at high burnup can be roughly evaluated at around 100 *W/cm* (beginning of life linear rate).

In experiment 13 the high FGR (32 - 40%) is probably due to the high linear rate (470 - 510 *W/cm*), however, the significantly less fission gas release observed in experiment 19 (9 - 14%) than that recorded in experiment 13 (22 - 38%) in rods at similar beginning of life linear rates (430 - 450 *W/cm*), can be explained by the beneficial influence of a large central hole on fuel temperature.

In experiment 19 run at 440 *W/cm* it is suspected that the fission gas release by thermal process had already taken place.

In experiment 21 fuel rods with small and large central holes show similar fission gas release (17 - 41%) during irradiation to high burnups (average 60 - 80 *MWd/kg*). In rods with initial linear rate below 360 *W/cm* at the beginning of life fuel structures are preserved at 60 *MWd/kg* average burnup. These rods show a fission gas release rate limited to 17%.

From observation of fuel structure, Cs spectrometry and fission gas release, a large degradation of fuel thermal conductivity can be identified at high burnup. If the fuel burnup is the right parameter to be considered, burnup limits identi-fied by Cs. shifting (overheating), fuel structures (columnar grains and new central hole) and high fission gas release is as below:

- 70 to 75 *MWd/kg* for rods with large central hole (maximum linear rate decreasing from 440 to 160 *W/cm* during the irradiation).
- 58 to 64 *MWd/kg* for rods with small central hole (maximum linear rate decreasing from 370 to 140 *W/cm* during the irradiation).

5 General Conclusion

As stated before, the aim of this paper is not to make a judgement on the VVER fuel or a comparison with western PWR fuels, because of the differences in functioning and design of VVER fuel. Another reason is that linear rate and burnup of some of these experiments is above the actual licensed values in VVER power stations.

Irradiated beyond the usual linear rates at high burnup, these experiments are of interest. Up to now the fuel life limiting factor was cladding corrosion when using Zircaloy-4. As the cladding corrosion situation improves, the next life limiting factor to be met could be the fuel itself. The decreasing thermal conductivity int his fuel is probably of prime importance and should be further studied and modelled in the light of studies already carried out on the subject (rim effect...).

The last remark is on the importance of study carried out here on pellets with central hole of different size. Because the definite benefit of a bigger central hole has been demonstrated in this study, an assessment of hollow pellets should be pursued and modelled as an effective means to decrease the fuel temperature and improve correlated parameters. Fuel with central hole has been used for FBR reactors and has shown its efficiency.

References

[1] V. Koliadyn et al., "Operability of VVER-1000 Experimental Fuel Rods at High Burnups in the MR Reactor". Proc. ANS/ENS Int. Top. Meeting on LWR Fuel Performance. 21-24 Apr. 1991, Avignon, France, p. 567.

[2] Yu. Bibilashvili et al., Toward High Burnup in Russian VVER Reactors and Status of Water Reactor Fuel Technology. Proc. Int. Top. Meeting on LWR Fuel Performance. 16-19 Apr. 1994, West Palm Beach, USA.

[3] K. P. Dubrovin et al., The results of pest irradiation examinations of VVER-1000 and VVER-400 fuel rods. *J. Nucl. Mat.* **178** (1991) 306-311.

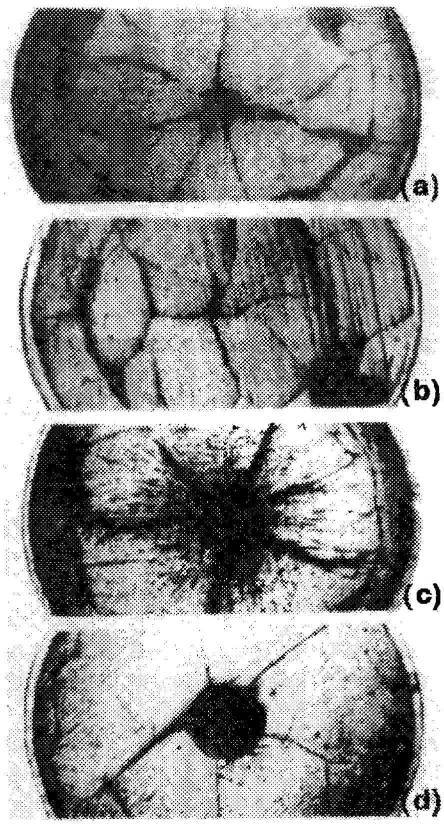


Figure 4 **Macrostructures of fuel pellets with small initial central holes cut from the central zones of the fuel rods A1 (a), A7 (b), D1 (c), D4 (d) - see Table 5.**

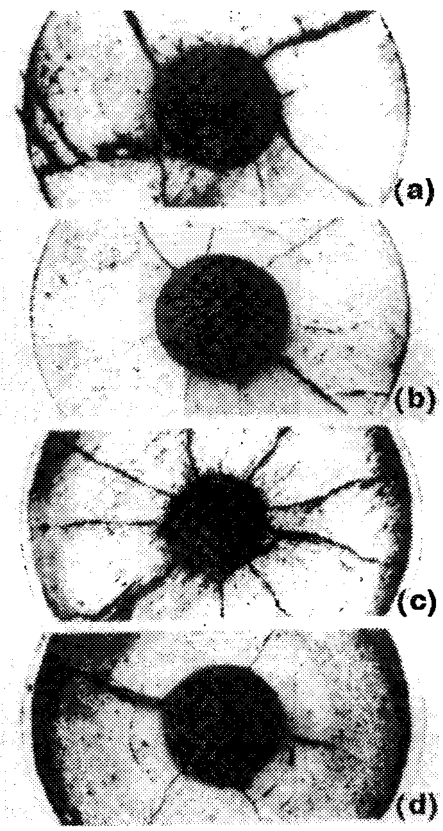
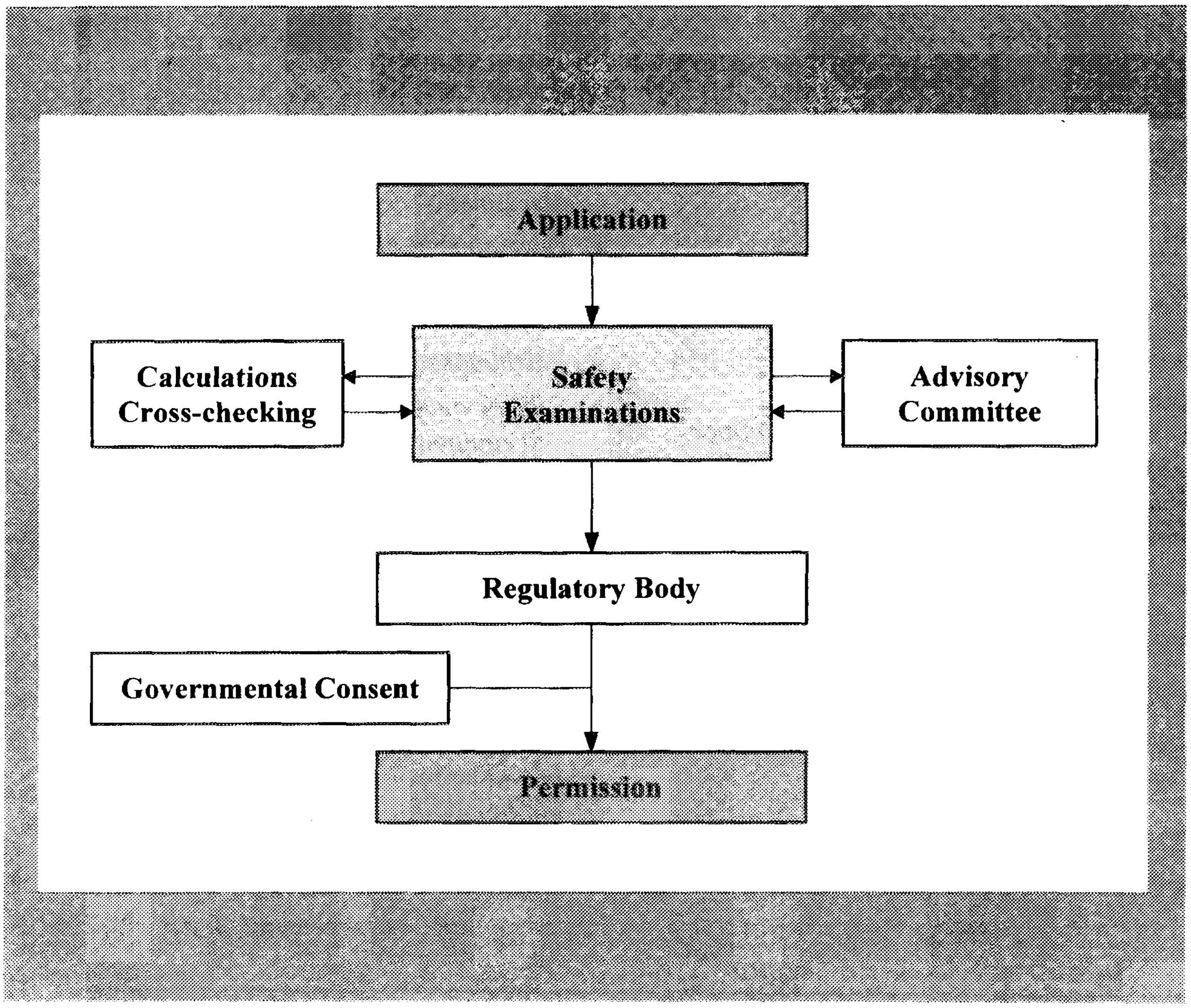


Figure 5 **Macrostructures of fuel pellets with large initial central holes cut from the central zones of the fuel rods F11 (a), F7 (b), C6 (c) and C4 (d).**

Part 3

Licensing of VVER Fuel and Fuel Analysis Codes



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- *Industrial system reliability and safety assessment*
- *Development of risk management programs based on cost-benefit analysis*
- *Earthquake engineering*
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Licensing of VVER Fuel and Fuel Analysis Codes

Panel Discussion Report

Panel Chairman: P. Chantoin

IAEA, Vienna, Austria

1 Licensing Procedures in Different Countries

The "Safety Body", established by law, is responsible for the application of Safety requirements based on safety guides. To license a load, reload, or a design modification (burnup, cladding....), the utility has to demonstrate that the safety requirements are met. This demonstration is based on fuel and core calculation modelling.

Reliable prediction constitutes a basic demand for safety based calculations, for design purposes and for fuel performance predictions. Therefore, computer codes are developed.

Even with the most simple codes, adequate operating limits can be derived to ensure safe operation provided sufficient margin is built-in to cover shortcomings.

To be used in safety calculations, core calculation and fuel codes have to be qualified and then licensed by the safety authority. This procedure is based on the results of experiments and reviewed by panels of experts according to the models, the discipline involved, or the kind of codes reviewed (QA procedure on the code structure, its content and its documentation).

During the meeting, it was shown that this general licensing scheme was followed by Japan, France, Finland and that the same process was underway in the Czech Republic and in Slovakia. There is a wish to follow this direction in Bulgaria but there is no clear regulation yet in the area of fuel and code licensing. In Russia fuel and code qualification and licensing have been performed according to normal world accepted practice. From the content of discussions it seems that in Russia, the demonstration that safety requirements (U.S Nuclear Regulatory Commission type) are met, is the task of the chief designer.

2 Other Needs for Licensing

Once qualified and licensed, the codes can be used for modelling providing that necessary data on fuel fabrication are available. Before 1988, Finland was examining a fresh fuel assembly each year to extract the necessary data and satisfy the safety authority requirement. Since 1988 quality checking is carried out directly with the help of a Russian audit company. It has to be noted that all VVER users can have access to fabrication data on request, free of charge, when the fuel is ordered.

The need for co-operation in the code and qualified fuel data, was also put forward. The IAEA underlined the Agency's efforts in promoting CRPs like FUMEX and the regional technical co-operation programme on modelling of VVER fuel.

3 The Temelin Issue

The question of the Temelin NPP was raised by the Russian side. The argument was that, for the first time, the fuel of this power station was not provided by the NSSS supplier and therefore some shortcomings could occur. The IAEA replied that the safety authority in the Czech Republic was up to now working in the right direction, moreover, that safety missions were regularly visiting the utility; consequently we did not have the right to doubt their ability to resolve their licensing tasks in a way satisfactory for the overall safety of the plant.

BG9600376

Nuclear Fuel Licensing Procedures in Bulgaria

Yordan Harizanov

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1 Nuclear Energy in Bulgaria

There are 8 main nuclear facilities in the country, as follows: six power reactors, an Away-From-Reactor Storage and a Research Reactor. All six reactors are located at the only operating nuclear power plant site: Kozloduy NPP.

The site is in the north-western part of the country on Danube river. Four of the reactors are of VVER-440 type and two of them - of the VVER-1000 type. The commercial operation of units 1- 6 started in 1974, 1975, 1981 and 1983 for VVER-440 reators and in 1987, 1989 for VVER-1000 respectively. Our second NPP site have been selected and construction of second power plant has started but due to a number of different reasons including political and economic changes, ecological organisations pressure and shortage of money, the work have been postponed.

The only research reactor IRT-2000 is cited at the territory of Sofia city. Due to the same reasons it was temporary put out of operation in 1989.

The existing AFRS is located at Kozloduy NPP site.

2 Structure and Role of the Safety Authority

The management of the use of atomic energy and the state control on its safe utilisatio is assigned to the Committee on the Use of Atomic Energy for Peaceful Purposes (CUAEPP).

CUAEPP is headed by a State Committee, established under the Low on the Use of Atomic Energy for Peaceful Purposes and composed by the member of the Government and Heads of Institutions and other administrators with responcibilities regarding the use of atomic energy. The State Committee meets twice a year for reviewing the activities of CUAEPP, approving regulatory documents created by CUAEPP during the past period and approving the interdepartmental schedule of work.

Under the Chairman of CUAEPP, there are several offices. The major one iis the Inspectorate on the Safe Use of Atomic Energy (ISUAE). ISUAE in its turn have two divisions: "Safety of Nuclear Facilities" and "Safety of Sources of Ionising Radiation".

3 Main Legislative Documents

The basic legislative document is the Low on the Use of Atomic Energy for Peaceful Purposes (LUAEPP) adopted by the Bulgarian Parliament (Decree No. 3300/1985). It stipulates the essential principles in the use of atomic energy and the state control on the nuclear and radiation safety.

The Governmental Regulation for the Application of the LUAEPP works out the details on its implementation, the organisation of the state control on nuclear and radiation safety, licensing procedures, etc.

Based on LUAEPP, eight Regulatory Documents have been created and adopted by CUAEPP since 1985. Regulation No. 3 (1987) defines the main safety criteria for nuclear power stations during design, constuction and operation phases.

Regulation No. 5 (1988) addresses the licensing of the use of atomic energy. The document specifies the necessary documentation and terms for initial licensing and for licence renewals.

4 Licensing

According to the Article 2 of the existing LUAEPP, the nuclear material in Bulgaria belongs to the Government. Other sources of ionising radiation could be owned by other organisations and companies. Point 2 of the same Article states that the nuclear material could be given for temporary use to other organisations. Article 22 of the Low says that all activities concerning the use of atomic energy must be done under the licence issued by the ISUAE. It includes the import, export, operation, storage and transportation of nuclear fuel.

As it was mentioned already, Regulation No. 5 describes the procedures, documentation and terms for licence issuance or renewal. For example, to apply for licence for spent fuel shipment abroad, a company or organisation must submit to ISUAE number of documents, including:

- agreement/contract between shipper and receiver of the nuclear material;
- certificate of the package;
- protocol for readiness of the lifting equipment wich wil be used;
- protocol for readiness of the transportation equipment;
- protocol for readiness for shipment of the packing for shipment;
- protocol for the test for hermeticality of the fuel assemblies;
- table with some technical data of the package;
- map of the fuel location in the baskets;
- protocol of licensing of the staff involved in the shipment;
- protocol of the radiation measurements of the package;
- protocol of the readiness of the roads to be used;

- guides for nuclear and radiation safety during the transportation;
- guide for the actions of the staff during the emergency;
- protocol for security provisions.

We do not have specific criteria for fuel licensing so we use the requirements of Regulation No. 5 of the CUAEPP and of course the safety limits and conditions established in the technical specifications. In case when the operator apply for licensing of new operational regimes of the fuel or other changes of the design characteristics we require detailed calculations and explanations not only by the apllicant and the manufecturer but from independent organisations as well. Some basic requirements on nuclear fuel safety, settled in the IAEA documents, can be used if appropriate. The main principles included in the guides and regulations of certain countries which are manufecturers of nuclear fuel, can be considered as well. But the main criteria for licensing in case of any changes in fuel characteristics and operation conditions are formulated in Regulation No. 3 of the CUAEPP. Under design of the reactor core, acceptable limits for fuel failures during normal operationmust be setled (by thre quality and rate of degradataion) as well as the correspondding levels of radioactivity of reactor coolant.

Due to the political, economical and ecological changes which took place during the last years in our country, many changes have to be done in the field of licensing of nuclear fuel. Nowadays it is getting more amd more difficult to licence some activities keeping in mind all the changes, e.g. correlating the new international treaties and conventions which Bulgaria has signed recently with the existing legal base in the area.

We have started to change our legislation but this process is difficult and time-consuming. The new Atomic Low is under consderation in Bulgarian Parliament and we expect it to be adopted by mid-1995. After it's adoption almost all existing regulations will have to be changed in order to meet the new requirements.

Nuclear Regulatory Guides for LWR (PWR) Fuel in Japan and Some Related Safety Research

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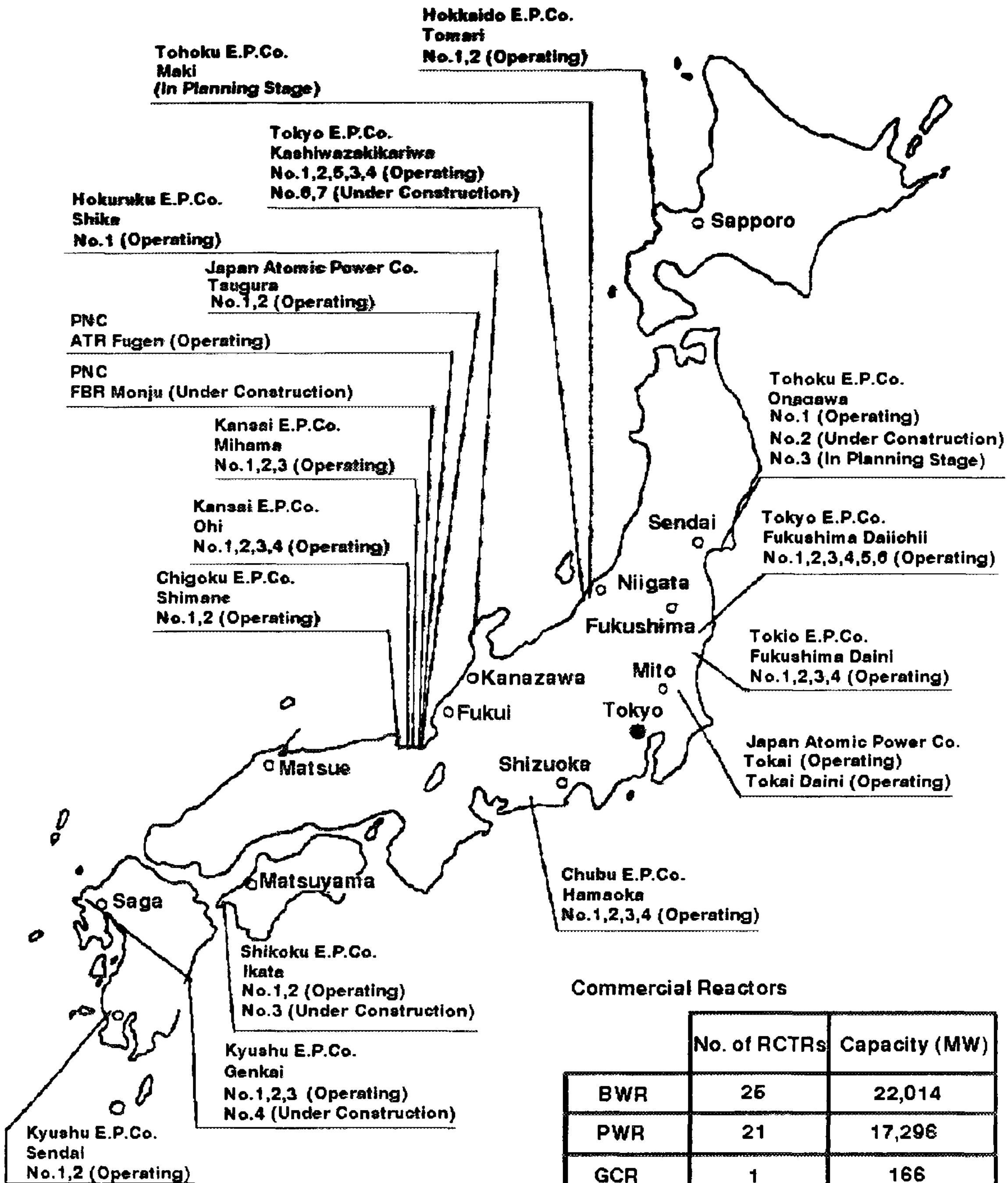
1 Introduction

More than 25 years have passed since the first nuclear power plant started operation in Japan. At present 47 commercial nuclear power plants (NPPs) are in operation. Their total generating capacity is 39.5 *GWe* which represents approximately 20% of the total generating capacity in Japan. In addition 5 commercial nuclear power plants including two ABWRs are under construction. The sites of these NPPs are shown in Fig. 1.



Commercial Reactors

	No. of RCTRs	Capacity (MW)
BWR	26	22,014
PWR	21	17,296
GCR	1	166
Total	47	39,476

Figure 1 Sites of nuclear power stations in Japan

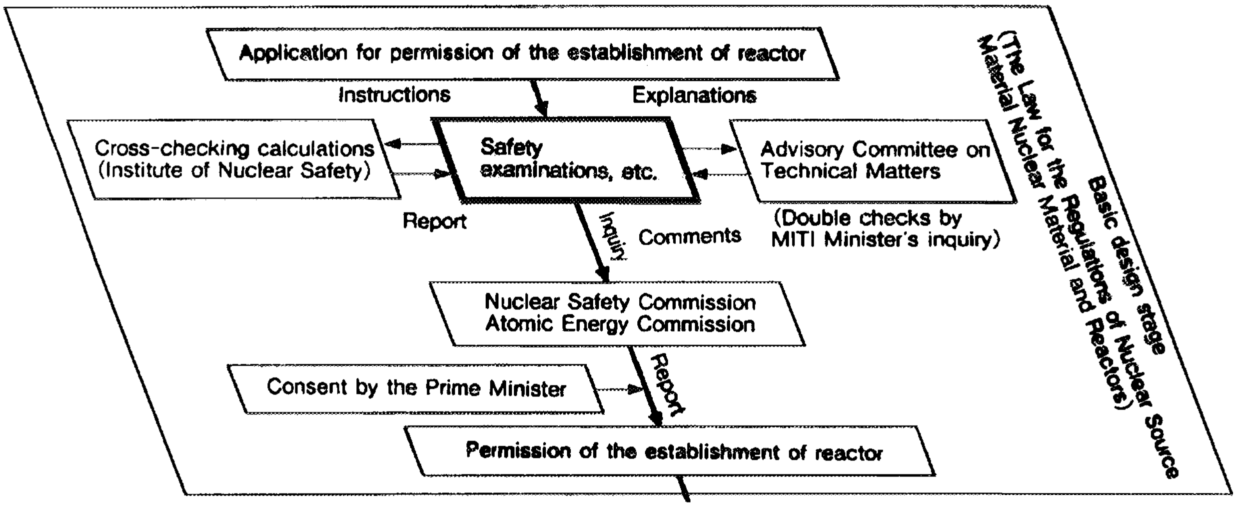


Figure 2 Flow of nuclear power plant's examination

Based upon the recognition of "No nuclear energy in use without assuring nuclear safety", the regulatory authorities enforce strict safety regulations in each stage of design, construction and operation of the NPPs.

In the present paper it is intended to introduce the general licensing procedure in Japan and regulatory guides and expert committee reports closely related to PWR fuel, together with some major results of rector safety research experiments at NSPR (Nuclear Safety Research Reactor of JAERI), which have been used for establishment of related guide.

Because of the limited space of the paper it is not possible to describe these guides and reports in detail. It, however, is intended to describe in some detail on the item which may be of interest to the fuel people in the meeting.

2 General Situation of Licensing of NPP and Regulatory Guides in Japan

2.1 Regulatory Processes for NPP

All the processes of the siting, design, construction and operation of NPP are subject to the control of competent government agencies. The Ministry of International Trade and Industry (MITI) is responsible for the regulation of commercial NPP, and the Science and Technology Agency (STA) is responsible for the regulation of research reactorsand of reactors under development. On the other hand, as an advisory organization for the prime minister, the Nuclear Safety Commission (NSC) reviews the regulatory processes and administrative processes conducted by the competent agencies (MITI, STA) from safety point of view and make necessary recommendations to the government.

The major licensing processes of the commercial NPP are divided into three stages; application for installation permit, application for construction permit and inspection during construction and operation phases. In the present paper, only the first stages are described generally as follows.

In the application for installation permit for commercial NPP an applicant who intends to install (or modify) NPP must submit to MITI the documents that includes the description of the site conditions, basic plant design and relevant safety evaluation. The MITI conducts the first review of the application with the aid of Technical Advisory Committee on Nuclear Power, MITI.

After the approval by MITI the NSC independently conducts the second review with the aid of Committee on Examination of Reactor Safety, NSC. The conclusion of NSC's review is reported to MITI. If the applicant is eventually approved by NSC, the MITI issues the license indicating the acceptance of the basic design of the plant for the proposed site.

The general correlation of the regulatory processes as described above is shown in Fig.2.

2.2 Major Regulatory Guides and Related Committee Reports on LWR Fuel [1]

The major regulatory guides which are closely related to LWR fuel are as follows:

(1) Guide for Safety Design of Light Water Nuclear Power Reactor Facilities (1990)

(2) Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities (1990)

(3) Guide for Evaluation of Emergency Core Cooling System Performance in Light Water Nuclear power Reactors (1981, Rev. 1988)

(4) Guide for Evaluation of Reactivity Initiated Events in Light Water Nuclear Power Reactor Facilities (1984)

In the Guide (1), the major item which is related with fuel is as follows (14. Fuel design):

(i) The fuel assemblies shall be designed not to loose their integrity despite various unfavorable factors that may take place during their use in the reactor.

(ii) The fuel assembly shall be designed not to be excessively deformed during transport or handling.

In the Guide (2), the major items which are related with fuel are as follows:

(i) The minimum heat flux ratio or the minimum critical power ratio shall be lager than the acceptable limit.
(ii) Fuel cladding shall not be mechanically damaged.
(iii) Fuel enthalpy shall not exceed acceptable limit.
(iv) Pressure on the reactor coolant pressure boundary shall not exceed 100% of the maximum allowable working pressure.

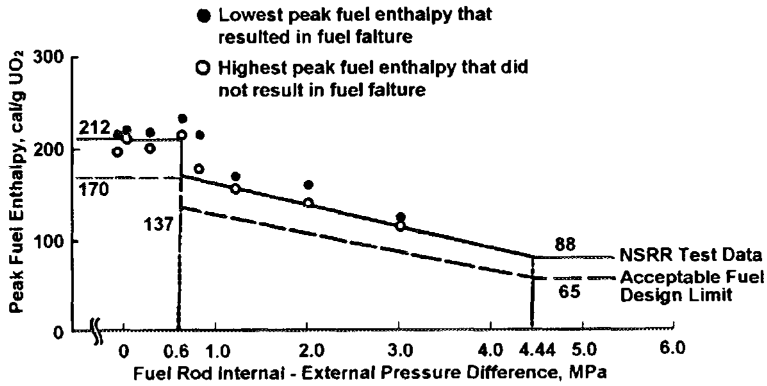


Figure 3 Fuel failure thresholds as a function of fuel rod internal-external pressure difference

For the criteria on "Accident":

(I) The core shall not be damaged considerably and adequate coolable state of the core shall be maintained.
(II) Fuel enthalpy shall not exceed the specified limit.
(III) Pressure on the reactor coolant pressure boundary shall not exceed 120% of the maximum allowable working pressure.
(IV) and (V) are skipped as they are not much related with fuel.

In the Guide (3), the major items which are related with fuel are as follows:

(i) The calculated maximum fuel cladding temperature shall nor exceed 1.200°C.
(ii) The calculated stoichiometric amount of oxidation of the fuel cladding shall not exceed 15% of the cladding thickness before significant oxidation.

The general content of Guide (4) is described in the next chapter.

There are several expert committee reports which are resolved or approved by NSC. The reports which are closely related with licensing of the LWR fuel are as follows:

(1) On the 17×17 Fuel Assemblies used in PWR (1976)
(2) On the Fuel Design Methodology of LWR (1988)
(3) Clarification of "Fuel Cladding Should not Fail Mechanically" (1985)

The general content of these reports is described in the chapters 4 - 5.

3 Regulatory Guide "Evaluation Guide for Reactivity Initiated Events in Light Water Power Reactor" (1984)

3.1 Objective

This guide aims to confirm the integrity of reactor core and reactor pressure boundary by evaluating fuel enthalpy increase due to rapid insertion of positive reactivity which results in reactor power increase in the reactor core under or near the critical conditions.

3.2 Position and Range for Applications

This guide is to evaluate the reactivity initiated events of PWR and BWR following the regulatory guides "Guide for safety design of light water nuclear power reactor facilities" and "Guide for safety evaluation of light water nuclear power reactor facilities". The present guide shall be reviewed when knowledge like design improvements, accumulation of experience, etc. are obtained.

The basic idea of the present guide may be applied to the evaluation of reactivity initiated events of the light water reactors or heavy water reactors which have similar fuel structure and compositions.

3.3 Judgment Criteria

3.3.1 During anticipated operational transient

(i) The maximum fuel enthalpy shall not exceed which are decided depending on the differential pressure between fuel internal pressure and coolant pressure as shown in Fig. 3 (dotted line).
(ii) The pressure on the reactor coolant pressure boundaries shall be 1.1 times maximum working pressure and below.

3.3.2 During the accident

(i) Maximum fuel enthalpy shall not exceed 230 *cal/g* of UO_2.
(ii) The pressure on the reactor coolant pressure boundaries shall be 1.2 times maximum working pressure and below
(iii) During anticipated operational transition and accident, shock pressure from the burst of water logged fuel shall not result in any damage to the reactor scram ability and integrity of reactor pressure vessel.

3.4 Requirements for the Analyses

The initial condition of reactor states, dynamic characteristics calculation, fuel behavior analyses, pressure surge calculations and mechanical energy generation by the burst of waterlogged fuel are described. The details are not given here.

3.5 Required Documents for Evaluation

Several documents including calculation code, major input and sensitivity analyses of the calculation programs are required. The details are omitted here.

4 Report by the Committee on Examination of Reactor Safety "On the 17×17 Fuel Assemblies Use in PWR" (1976)

4.1 Introduction

The purpose of this report is to investigate generally the basic design of Westinghouse 17×17 fuel assemblies to be used in PWR. The investigation includes structural, nuclear and thermohydraulic design of the assemblies. Considering the interest in the present meeting the emphasis on the description of the report is placed on the structural design.

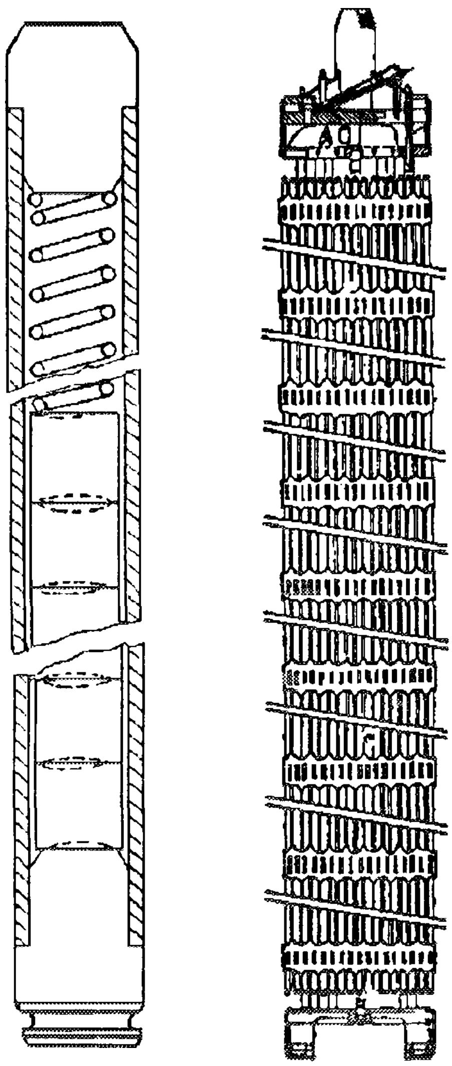


Figure 4 General design of PWR fuel rod and assembly (17×17 type)

4.2 Design of the 17×17 Fuel Assemblies

4.2.1 General

The design of the fuel rod is shown in Fig. 4 and the fuel rods are held by grids. Fuel assembly is composed of these rods and a skeleton which is formed by 24 control guide thimbles, 9 grids and upper/lower nozzles. The general design of 17×17 fuel assembly is show also in Fig. 4.

4.4.2 Design criteria

To assure the integrity of the fuel during normal operation conditions and anticipated operational transients, three criteria for structural design, nuclear design and thermohydrauliv design are established.

(a) Structural design criteria

Structural design criteria are designed as follows:

(i) Maximum fuel temperature shall be below melting point of uranium dioxide.

(ii) Inner pressure of fuel rod shall be within the operating coolant pressure (157 kg/cm^2g). This criteria has been changed in 1988 to allow the internal pressure to exceed the outer pressure under certain conditions (described in 5.4.1)

(iii) Stress in the cladding shall be within the proof stress of the Zircaloy-4.

(iv) The change of the circumferential tensile strain of the cladding during each transient shall not exceed 1%.

(v) The cumulative number of fatigue cycles shall not exceed the design fatigue life limit.

In addition to the above statements, fuel failure due to cladding fretting corrosion, cladding flattening, etc., shall be prevented.

Concerning the control guide thimbles, grids, upper/lower nozzles and the places close to these parts, the following criteria are established to ensure the integrity and avoid the influence of the surrounding structures:

(i) In principle the stress due to static and cyclic load in the normal operation and anticipated operational transient shall be evaluated based on ASME Section III.

(ii) Concerning the load during transportation and handling, significant deformation shall not take place for 6G (G denotes gravity acceleration).

(b) Nuclear design criteria

Six criteria are described, including peaking factors, maximum fuel temperature (below melting point of uranium dioxide), Doppler coefficient, maximum reactivity insertion. The details are not given here.

(c) Thermohydraulic design criteria

Four criteria are described including minimum DNB ratio (above 1.30), maximum fuel temperature (as mentioned above), coolant flow (above 95.5% of design value) and prevention of thermohydraulic instability. The details are not given here.

4.3 Result of Investigation

4.3.1 Structural Design

(a) Fuel design code

The code performs analyses for full length of fuel rod in 2D (R, Z). The code contains various submodels concerning fuel behavior and can calculate fuel temperature, pellet/cladding gap size, fission gas release, rod internal pressure, cladding deformation, cladding stress etc. as a function of burnup. The code is considered appropriate for the use of design of 17×17 fuel, judging from the comparison with the measurements and large amount of records of performances of the fuel designed and manufactured based upon the code.

(b) Fuel internal pressure (criteria has been changed)

(c) Cladding stress

For the evaluation of the cladding stresses, equivalent stress of the volume average are calculated and compared with proof stress of Zircaloy-4. The cladding stresses thus evaluated have been confirmed that they do not exceed the proof stress of cladding during normal operation and anticipated operational transients and have sufficient margins.

(d) Cladding strain

The 1% strain criterion is based on the irradiation performance records and judged appropriate. The evaluation of the cladding average circumferential strain during anticipated operational transient confirms that the calculated strain is sufficiently satisfy the 1% criterion The cladding strain due to swelling for long term operation is small enough and will not cause any problem at the end of fuel life.

(e) Cladding cumulative fatigue

The cumulative fatigue of the cladding has been investigated for the failure possibility using Langer-O Donnel correlation. Integrity of fuel is judged sufficiently assured against the power changes expected during the fuel life time.

(f) Stability of shape and sizes

The results of the investigation on the stability during transportation, rod growth, interaction with rods and nozzles, strength tests of assemblies, assurance of flow are confirmed that the stability of shape and sizes are maintained during the operation.

(g) Fretting corrosion

Fretting corrosion of 17×17 type rods were compared with the existing 14×14 and 15×15 type fuel rods and it was confirmed that the fretting corrosion of 17×17 rods is slightly less than the existing one.

(h) Fuel rod bowing

As the countermeasure to the fuel rod bowing of Westinghouse design fuel, the present design applies shortening of grid spans and reduction of restraint forces of the grids. The contact between rods is considered to be practically avoided by the new design.

(i) Behavior of 17×17 fuel under LOCA conditions

Westinghouse has simulated the behavior of fuel cladding during LOCA by a series of a single rod burst tests. behavior of cladding during simulated LOCA id characterized by swelling and burst temperature. Statistic comparison of the burst behavior of 17×17 type fuel cladding with that of 15×15 shows that both behave similarly.

(j) Others

Concerning densification and flattening of Westinghouse type fuel and general hydriding failure, the countermeasures have been taken and they are almost solved presently.

4.3.2 Nuclear Design

Nuclear analysis codes, linear heat rating of fuel rod, reactivity coefficients and constant axial offset control operation are described. The details are not given here.

4.3.3 Thermohydraulic Design

Thermohydraulic design codes, minimum DNB ratio, maximum fuel temperature and thermohydraulic stability are described. The details are not given here.

4.4 Others

Records of the Performance and Quality Controls been described in a separate chapter. Data, however, are rather old. Therefore they are not described here.

4.5 Conclusion

PWR new type fuel committee has conducted investigation on the 17×17 type fuel and concluded that the use of this fuel in PWR will not result any problem.

5 On the Fuel Design Methodology of LWR (1988)

5.1 Introduction

This report describes the results of investigation on the appropriateness of the revision of the current methodology for extension to the high burnup use of fuel assemblies.

5.2 Background

The current fuel design methodology has been described in the Committee on Examination of Reactor Safety Report "on the 17 × 17 fuel assemblies for PWR" and has a long term positive achievements. However, as the methodology is based upon conservative criteria and evaluation methods, there could be cases where merits of the design improvements are not properly evaluated.

On the other hand, evaluation codes can predict thermal/mechanical behavior of fuel rods with better accuracy at high burnup region (above 39 000 *MWd/t* for PWR) reflecting recent knowledge. And a new criterion on the internal pressure of the fuel rod has been proposed. Considering this background and based upon investigations on the recently developed/revised codes, the criterion has been examined.

5.3 Results of Investigation

Concerning the internal pressure criterion, the current criterion "fuel rod internal pressure shall be within the operating coolant pressure (157 kg/cm^2g) has been changed". The new criterion is "fuel rod internal pressure shall not exceed the pressure which results the increase of pellet/cladding gap due to outward creep deformation of the cladding under normal conditions".

The acceptance of this criterion is decided appropriate by the following reasons:

(i) Fuel rod internal pressure is restricted by the cladding stress criterion against mechanical failure. From the stand point of prevention of excessive increase of fuel temperature, generation of thermal feedback will be prevented by the new criterion.

(ii) The fuel design code to be used for evaluation is able to predict fuel rod behavior under irradiation and uncertainties of the prediction are well evaluated quantitatively by the fuel irradiation and other data.

It is expected that the review will be conducted according to the new information.

5.4 Appendix. PWR Fuel Design Methodology

5.4.1 Fuel Rod Internal Pressure Criterion

The revised PWR design criterion has adopted new fuel internal pressure criterion considering the conservatism of the current criterion in comparison with the cladding stress criterion.

(1) Prevention of Thermal Feedback

During normal operating conditions the fuel internal pressure is below the operating coolant pressure (outer pressure). Cladding diameter decreases due to the inward creep deformation and the cladding comes to contact with pellets. Hereafter internal pressure increases due to fission gas release and accumulation and the internal pressure may exceed the outer pressure at high burnup region. In this condition the once closed pellet/cladding gap may open again due to the outward creep deformation of the cladding, which results the increase

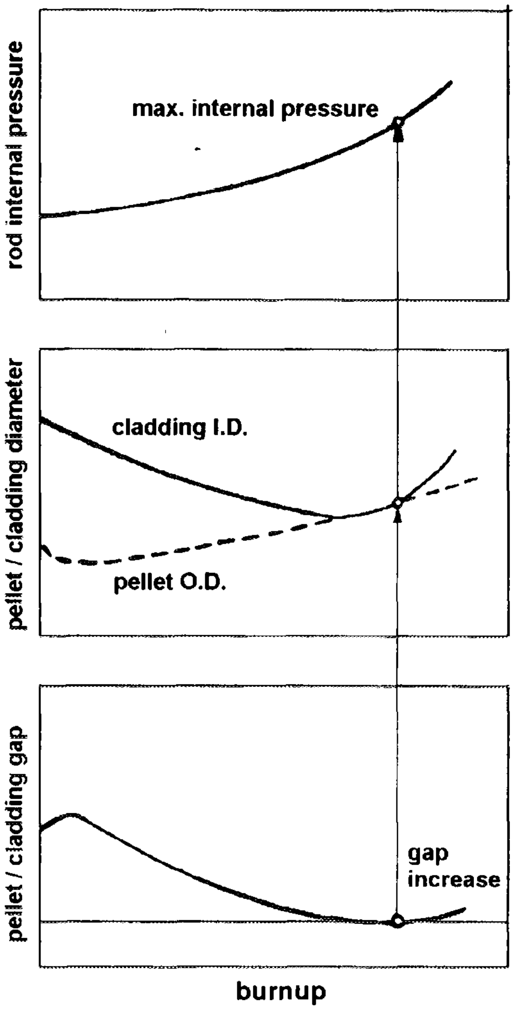


Figure 5 Method to determine the maximum internal pressure [1]

of cladding diameter. Gap conductance may decrease and fuel temperature may rise due to the opening of the gap. This may result further release of fission gas and then internal pressure continues to rise. This may result the further opening of the gap. This phenomena is called thermal feedback and the use of the fuel may result the excess increase of fuel temperature. The new criterion for fuel internal pressure is established to avoid the thermal feedback.

(2) Method to Decide the Internal Pressure Standard

(a) Analysis conditions
Fuel rod power histories are chosen among power histories to be used in the practice to give the severer fuel internal pressures.

(b) The maximum internal pressure which does not result gap increase (Fig. 5)
From the calculation of pellet/cladding gap change by the fuel rod design code the time of gap increase (or reopening of once closed gap) is determined. The fuel rod internal pressure of this point is the maximum internal pressure.

(c) Fuel rod internal pressure standard
By the method described in (b) the lower envelope of the maximum internal pressure for each

power histories is determined. In addition the uncertainties of the analysis (uncertainties arising from design code and fabrication parameters) are taken into consideration and herewith determined value is called as the fuel rod internal pressure standard.

5.4.2. Fuel Design Code

(1) Improvement of the design code

PWR fuel rod design codes have been improved by the high burnup data covering the PWR operating conditions to obtain better accuracy.

Concerning the revised Mitsubishi fuel design code, fission gas release model has been changed to be more burnup dependent by applying the model in which the fission gas release becomes significant after the fuel burnup exceeds given threshold at high burnup region. Also creep model of cladding and swelling/densification model of pellet have been improved.

Concerning revised Nuclear Fuel Industries fuel design code, diffusion constants of the fission gas release has been revised and pellet relocation model has been used.

(2) Verification of the improved fuel design codes

The verification of the codes are based upon wide range of data, including data from Halden reactor, post irradiation examination of power reactor fuel, international cooperation such as Overramp project and others. By these verification activities it has been confirmed that the calculations by improved codes give better agreement with the measurements. In the evaluation of the design codes it is appropriate to consider that the differences between calculation and measurements are taken as the uncertainties of the codes.

6 Clarification of the Statement "Fuel Cladding Should Not Be Mechanically Damaged" (Expert Committee Report, 1985)

In the Safety Evaluation Guide it is required that "fuel cladding shall not be mechanically damaged" in the case of anticipated operational transients. In the safety evaluation of LWR fuel design, the above statements are embodied by the following criteria:

- the circumferential average strain of fuel cladding shall be less than 1% (equivalent of 180% of the surface heat flux) (BWR);
- the maximum fuel centerline temperature shall be less than the melting point of uranium dioxide.

Therefore in the safety evaluation it is confirmed that the surface heat flux (BWR) or centerline temperature (PWR) fulfill these criteria. In this committee report it is investigated that the appropriateness of judging of mechanical damage by the thermal indexes and also the requirement of the expression "fuel cladding shall not be mechanically damaged" is clarified.

6.1 The Requirement of "Fuel Cladding Shall Not Be Mechanically Damaged"

This statement means that "penetrating damage shall not systematically be generated due to mechanical loading on the claddings".

6.2 Embodied Criteria (PWR)

(i) During the anticipated operational transients of the current reactor types and fuels, damage modes which may cause penetrating damage to the cladding are discussed. To prevent the systematic damage during anticipated operational transient, it is concluded that the corresponding criteria to "fuel cladding shall not be mechanically damaged" can be represented by the condition to prevent overstrain failure of the cladding, i.e. "the circumferential averaged strain of fuel cladding shall be less than 1%". In the case of PWR during the anticipated operational transient, "the maximum fuel centerline temperature shall be less than melting point of uranium dioxide". has been confirmed by the dynamic analyses. By satisfying this condition, the circumferential average strain of fuel cladding is less than 1%. Therefore it is concluded that this evaluation methodology is appropriate.

(ii) Recently the cause of fuel failure during the power ramping tests is identified as stress corrosion of cladding of Zirconium alloys. It is well known that Zircalloy cladding may fail depending on the power histories even with strain less than 1% or with centerline temperature below melting point. The investigation has been conducted concerning this point and the conclusion is that the possibility of systematic fuel failure during the anticipated operational transients for the current reactor type and fuel is considered sufficiently small. Therefore it is concluded that there in no need to change the current fuel failure criteria.

7 Studies of Fuel Behavior During Simulated Reactivity Initiated Accidents (RIA) in NSRR ([2], [3])

7.1 Introduction

RIA research program in Japan was imitated in 1972 as the NSRR (Nuclear Safety Research reactor) program. The results of over 800 experiments with fresh fuel rods have been reflected on the establishment of Japanese regulatory guide "Evaluation Guide for Reactivity Initiated Events in Light Water Power Reactor", issued by Nuclear Safety Commission in 1984. Considering the tendency of high burnup use of LWR fuel, NSRR was modified to be able to handle the preirradiated fuel in 1988. At the same time the operation modes of NSRR were also modified so that various power histories can be given to fuel specimen.

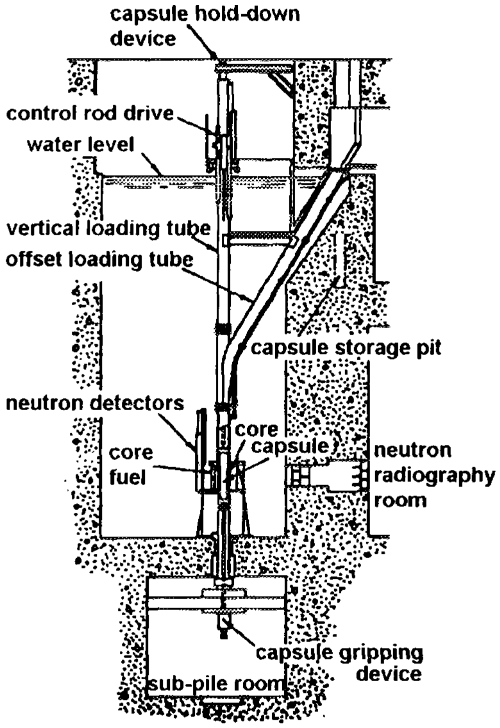


Figure 6 General arrangement of the NSRR

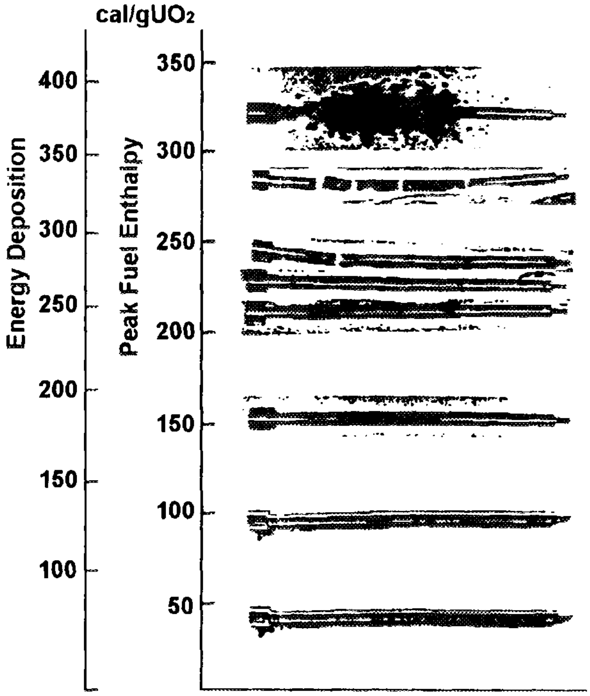


Figure 7 Appearance of post test fuel rods related with peak fuel enthalpy

7.2 Test Method

The NSRR is a modified TRIGA-ACPR (Annular Core Pulse Reactor). The general arrangements of the NSRR is shown in Fig. 6. The core structure is mounted on the bottom of 9 *m* deep open water pool. The pulsing operation is made by quick withdrawal of transient control by pressurized air. The maximum peak reactor power is about 21,000 *kW* in about 4 *msec*.

In typical tests (Standard tests), a test fuel rod is installed in a stainless steel capsule in the stagnant water at ambient pressure. The capsule and fuel rod are heavily instrumented, for example, by cladding thermocouples, fuel centerline thermocouple, strain gauges on the cladding, pellet/cladding elongation gauges, fuel internal pressure sensor and so on.

Tests in the pressurized conditions and under fuel bundle conditions have been conducted. Observations capsule with high speed camera to record the fuel outer appearance during tests have been developed and utilized.

7.3 Major Experimental Results with Fresh Fuel Rods

7.3.1 General Fuel Behavior

In order to study general failure mode of the fresh fuel (PWR type) standard tests rods were subjected to energy deposition of about 50 to 550 *cal/g* UO_2 or corresponding peak fuel enthalpies of about 30 to 450 *cal/g* UO_2. The photographs of typical post-tests fuel rods are shown in Fig. 7.

When the peak fuel enthalpy is below 88 *cal/g* of UO_2 DNB (Departure from Nucleate Boiling) does not take place and there is no change in appearance. After peak fuel enthalpy is beyond 110 *cal/g* UO_2 DNB is generated and cladding surface of the heated zone is oxidized. Fuel failure takes place at about 220 *cal/g* UO_2. Failure is caused by the embrittlement of the cladding due to oxidation and partial melting. When the peak fuel enthalpy exceeds 325 *cal/g* UO_2 fuel pellet melts and is ejected into water accompanying cladding failure. Molten fuel fragmentation into water causes generation of mechanical energy.

7.3.2 Influence of Fuel Design Parameters

The influence of major fuel design parameters, such as fuel internal pressure, gap sizes, gap gas composition, etc., on the fuel failure has been investigated. Among the parameters, the influence of fuel rod internal pressure was most significant. The data were already shown on Fig. 3. When the internal pressure is high, the failure mode changes form cladding melt to high temperature burst of the cladding.

7.3.3 Influence of Cooling Environment

Influence of bundle geometries, system pressure, cooling conditions, etc., on the fuel failure has been investigated. Under the bundle geometry failure threshold was found to be reduced by about 15% of the standard single rod test due to the reduction of heat conductance at the cladding surface. However, tests under simulated BWR and PWR cooling conditions gave similar failure threshold because the standard tests, as the cooling of the rod, were improved by forced cooling conditions.

7.3.4 Water Logged Fuel Failure

Water logged fuel generally fail when the fuel enthalpy reached about 100 *cal/g* UO_2 due to the burst of the cladding. In case of such failure, most of fuel pellets were released into water in fine particles and caused the generation of mechanical energy.

7.3.5 Generation of Mechanical Energy

When the fuel pellet is molten and ejected into water, molten fuel becomes fine particles and causes generation of mechanical energy by molten fuel/coolant interaction. The threshold enthalpy of this phenomena is found between 285 and 325 *cal/g* of UO_2.

7.4 RIA Behaviour of Irradiated PWR Fuel Rods

Experimental program with perirradiated LWR fuel rods as the test sample was started in 1989 in the NSRR. In this program transient behavior and failure initiation have been studied for PWR rods. Two types of rods were prepared. One is a commercial PWR fuel of 14×14 type (2.6% enriched, 95% TD) from power reactor and refabricated into segment rods (PWR-rods) at JAERI and the other is 10% enriched 14×14 type (95% TD) fuel segments (JM-rods) irradiated in JMTR of JAERI. The latter rods were used to study the fuel behavior at higher energy deposition. The peak fuel enthalpies for the former and the latter rods ranged 55 - 104, 96 - 164 *cal/g* UO_2, respectively.

In the PWR-rods, creepdown have taken place during irradiation and during the pulse irradiation in NSRR large PCMI was observed in comparison with the fresh fuel rods. Concerning JM-rods large deformation took place for the rods with fuel enthalpy above 200 *cal/g* UO_2. Fig. 8 shows the Ceramographs of the failed rod which is composed of 5% and 10% enriched pellets. Large diameter change occurred on 10% enriched pellet, due to swelling and crack in the pellet.

Fig. 9 shows fission gas release (FGR) from PWR-rods. From figure it is clear that FGR rate became larger when energy deposition was 113 (peak fuel enthalpy 85 *cal/g* UO_2). The cross section of the specimen revealed large number of hair cracks in the periphery of the pellet and it was considered that these hair cracks played important role in the fission gas release.

To summarize the results, energy depositions of each experiment are plotted as a function of fuel burnup and compared with all the available data from SPERT and PBF experiments in Fig. 10. Solid dots indicate fuel failure. More data will be produced in high burnup region in the future.

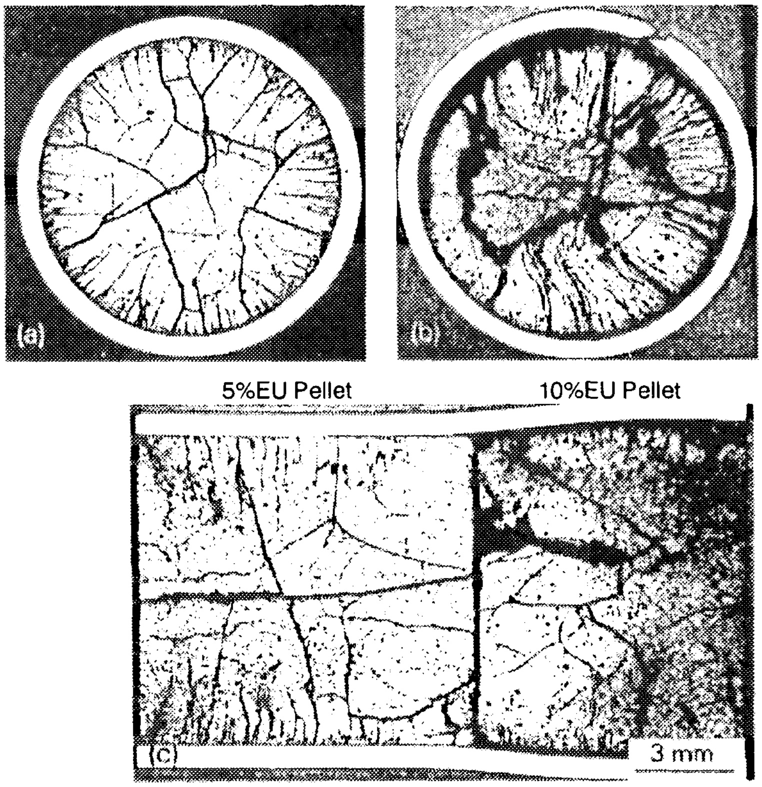


Figure 8 JM-rods after pulse irradiation. Cross sections at energy deposition of 113 cal/g (a) and 219 cal/g (b) and longtitudinal section at 219 cal/g (c).

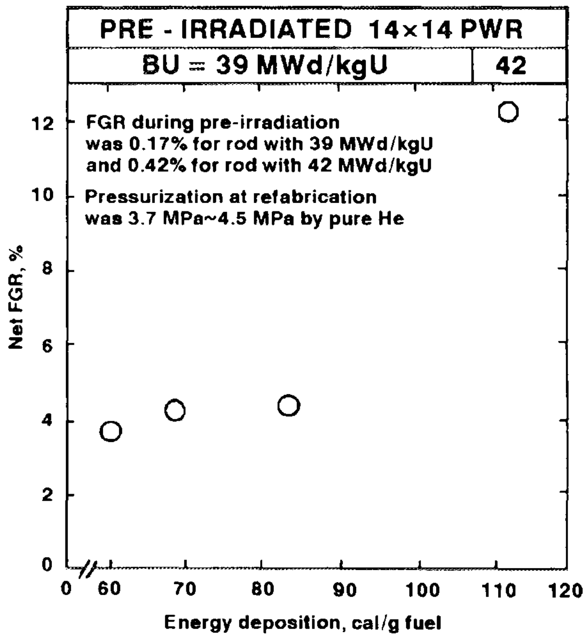


Figure 9 Relation of fission gas release and energy deposition of 14×14 PWR rod [3].

8 Application of the NSRR Results to the Licensing Giude

On the basis of the NSRR experiments, the Nuclear Safety Commission of Japan established a guide as stated above in 1984. For an anticipated transients the guide requires the acceptance design limit which is shown in Fig. 3 in dotted line to estimate number of failed rods. The acceptance design limit has been determined based on the single fuel rod test data with unpressurised and pressurized PWR type rods as shown in Fig. 3 in solid line. The difference between them include 16% reduction of failure threshold due to rod bundle effect and 10 *cal/g* UO_2 as a margin. The lowest bound of 65 *cal/g* UO_2 comes from the threshold enthalpy for the onset of DNB, as the high temperature burst of the cladding does not take place unless DNB occurs.

Concerning the generation of mechanical energy the highest peak fuel enthalpy at which fragmentation did not occur in the NSRR single rod test was 285 *cal/g* UO_2. The maximum fuel enthalpy for mechanical energy release as determined in the guide (230 *cal/g* UO_2) has been decided by reducing 15% and 10 *cal/g* UO_2 following the same manner as taken in determining acceptance fuel design limit in the case of anticipated transients.

Concerning the failure of water logged fuel, the highest fuel enthalpy at which rapture of the water logged fuel did not occur in the NSRR single rod tests was 90 *cal/g* UO_2. From this result the rapture threshold of the water logged fuel of 65 *cal/g* of UO_2 in the guide has been derived with reduction of 15% an 10 *cal/g* UO_2.

9 Concluding Remarks

In the present paper the general aspects of regulatory guides and related committee reports which are closely related PWR furl are described. Concerning the reactor safety research only the experiments at NSRR are described in connection with the guide for reactivity initiated events.

In Japan nuclear safety research, including reactor safety research has been conducted to support the regulatory activities by establishing/revising guides and preparing the necessary regulatory data, as well as improving nuclear safety. Nuclear safety research in Japan has been conducted in accordance with the national five years annual program.

Concerning LWR fuel safety research, fuel behavior in RIA and fuel integrity in normal operating conditions are major subjects of studies [4]. Perhaps it should be emphasized that fuel integrity has been the subject of safety research in Japan.

More efforts are needed to conduct nuclear safety research to ensure and to promote nuclear safety including reactor safety.

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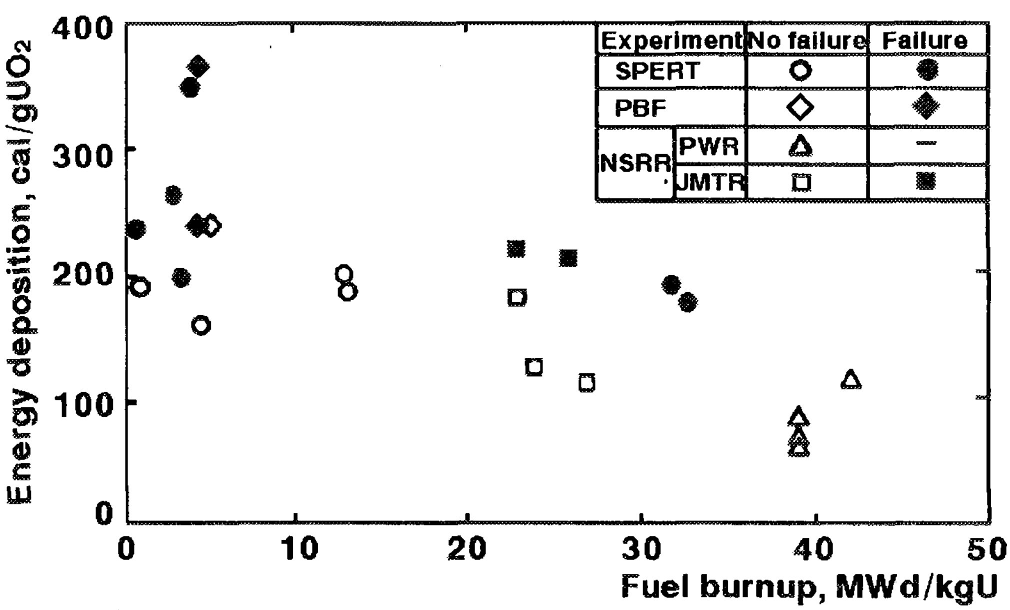

Figure 10 Results of SPERT, PBF and NSRR experiments with pre-irradiated fuels [3].

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Practice and Trends in Nuclear Fuel Licensing in France (Pressurized Water Reactor Fuels)

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1 Organization of the Governmental Safety Authorities

Prior to refering to the french practice and trends in nuclear fuel licensing, we should have a general understanding of the governmental authorities organization to check technical nuclear safety, which is one aspect of the regulatory action in this field.

The ministry in charge of industry and the ministry in charge of environment are responsible, within the govermental authorities, for matters relating to the safety of nuclear installations.

The main bodies involved in nuclear safety and working for these two ministries are the CSSIN (High Council for Nuclear Safety and Information, mainly in charge of supplying information related to nuclear safety to the population and the media), the CIINB (Interministerial Commission for Basic Nuclear Installations, which is consulted by the two ministers mainly for basic nuclear installation creation and license modification) and the DSIN (Nuclear Installations Safety Directorate).

DSIN is a specialized department of the ministry in charge of industry which works also for the ministry of environment. Its principal tasks are to handle licensing procedures for basic nuclear installations* , to organize and head the surveillance of such installations during all stages of their life (construction, operation, decommissioning), and to draw up and monitor the application of the general technical regulations.

DSIN also follows research and development work in the field of technical safety carried out by organizations attached to the ministry in charge of industry, particularly the CEA (Atomic Energy Commission) and Electricité de France (EDF), the French electrical power supplier and nuclear power plants operator.

In making its decisions and analysis, DSIN is mainly assisted by the IPSN (Institute for Nuclear Safety and Protection). IPSN carries out technical safety analysis making it possible to assess the provisions made by the operators of nuclear installations.

* "Nuclear installations" cover higly diversified activities as power stations, uranium enrichment, fuel production and reprocessing plants. In terms of regulations, they are classified in different categories which correspond to relatively restrictive procedures. "Basic nuclear installations" represent the category which covers practically all large installations corresponding of the conditions of the decree of 11th December 1963 modified by decrees of 1973, 1985, and 1990.

DSIN can also rely on the opinions and recommendations of groups of experts as the standing groups and the standing nuclear section of the central commission for pressure vessels.

The scope of the french nuclear power program and its development during the eighties led the governmental safety authorities to create additionnal means of checking technical safety. As these means were strenghtened, the need was felt to devolve the monitoring of nuclear installations to the DRIRE (Regional Directorate for Industry, Research and the Environment) so as to benefit from the efficiency resulting from geographical proximity to the installations.

2 Fuel Licensing Issues

2.1 The Formal Regulatory Context

Where nuclear fuel licensing is concerned, DSIN and its technical support IPSN are the two bodies mainly involved.

For a nuclear power plant operator, fuel licensing starts at the very beginning of the installation life. Indeed, the DAC (Authorization of Creation Ministerial Order) authorizing an operator to create a nuclear reactor within a nuclear power plant adresses important nuclear fuel issues and prescribes :

- the type of fissile material used (U, Pu);
- the close watch over core coolant water radioactivity in order to detect possible fuel leaks;
- conservative margins for fuel integrity in all plausible situations;
- nature of the material employed and quality of the coolant in order to limit corrosion
- that vibrations and other mechanical phenomenons must not affect the integrity of the internal equipments;
- that fuel handling and storage must be realized to avoid criticity risks and limit the risk of fall and overheating.

Apart from these general prescriptions, more specific licensing guidelines are specified in the document called RFS N°V.2.e (Fundamental Safety Rules related to General Rules applicable to the construction of fuel assemblies). The RFSs, issued by DSIN, constitute a set of safety rules on various technical subjects for ensuring the safety of nuclear installations and especially the pressurized water reactors.

The RFS N°V.2.e sets specific licensing guidelines and requirements for fuel licensing. Among others:

- licensee/operator must present the rules, codes, and norms used for conception, fabrication and implementation of nuclear fuel;
- licensee/operator must inform the govermental safety authorities (DSIN) and take the corrective actions necessary when he deviates from the codes, rules, or norms he specified for conception, fabrication and implementation of nuclear fuel;
- in order to facilitate the process, the licensee/operator must transmit the RCC-C (document describing the industrial practices for conception and construction of nuclear fuel) to the governmental authorities (DSIN).

2.2 Fuel Design and Manufacturing Requirements Related to Safety

Licensing guidelines for nuclear design are specified in the so called standard review plan (SRP), letter SIN N° 2323/78, november 13th 1978 and are primarily directed toward ensuring that various fuel design limits or acceptance criteria are not exceeded during normal operation or during anticipated operational transients. The major licensing guidelines and requirements for safety related reactor performance and major nuclear fuel manufactoring criteria related to normal, incidental and accidental situations and quality assurance rules on fuel design and fabrication are outlined in the previous section. These guidelines deal with thermal, mechanical, neutronic and thermal-hydraulic behaviour of the fuel and structures by the definition of limits concerning:

1) fuel system damage (e.g., stress, strain, and pressure acceptance limits);
2) fuel failure mechanisms (e.g., overheating, pellet cladding interaction (PCI) and fretting);
3) fuel coolability assurance (e.g., limits on structural deformations caused by cladding melting and fuel rod ballooning).

Initial guidelines are reviewed for areas affected by evolution of modes of fuel management (three or four batch loads) and particular operations modes (load or grids follow, frequency variation, operation at reduced or intermediate power, high burn-up, etc.)

In addition, materials properties and fuel system performance model typical of those which are used in the course of licensing calculations to meet the safety criteria are examined case-by-case for applicability to the different modes of fuel management and operations.

In France, generally the introduction of new design fuel assemblies into French reactor has been standard practice for almost fifteen years. The insertion of lead assembly is a practice proposed by the utility and endorsed by the safety authority on a case-by-case basis. There is no formalized licensing procedure requested for such lead assemblies. The document package from different fuel designers reviewed by the utility must be supplied to the French safety authorities for any loading of new type of fuel assembly in PWR plants.

These documents must be in full compliance with all requirements defined above.

Finally to check the accuracy and proper application of the design studies and safety criteria and quality assurance rules, prior authorization may be delivered, surveillance inspections are systematically performed by DSIN on the utility, the designer and the fuel manufacturer's units. The entire process is validated by a proposed post-irradiation examination programs from the beginning to the end of life of the fuel system.

The document package submitted by the utility is analysed by DSIN supported technically by IPSN. After the presentation and the discussion on the different technical aspects of the addressed case (lead fuel assembly (UO_2, MOX, $UO_2Gd_2O_3$, etc.), first core loading, operation modes, etc.), the final and generic authorisations delivered by the DSIN to the utility take into account the recommandations emitted by the standing groups (or Steering Committee) under the form of a letter from which are outlined all of "recommandations and requirements" which must be fulfilled by the utility before the fuel assembly could be loaded in the reactor core.

Up to now in France the generic authorisations released by DSIN to the industry for the fuel burn-up are limited to 47 *GWd/tU* (assembly average) for UO_2 fuel and 3 irradiation cycles for MOX fuel. These authorisations have been delivered for all types of fuel designs under the following considerations:

a) all design criteria are respected with regard to the thermomechanical behaviour of the fuel rod and the fuel assembly;
b) for the burn-up from 15 to 60 *GWd/tU* the risk of cladding rupture by PCI can be excluded during normal and incidental situations;
c) for burn-up above 35 *GWd/tU*, the load or grids follow operations and the extended burn-up have no effect on the fatigue and the endurance limits of the cladding;
d) commitment of the French utility on a R & D program to verify for high burn-up fuel:
    - for the reactivity initiated accident (RIA), the pertinence of the existing acceptance criteria based on the energy deposit in the fuel rod,
    - for the LOCA, the behaviour of the cladding under quenching condition.

Considering the RIA, experimental tests were performed on the CABRI reactor, in Cadarache, from 35 to 65 *GWd/tU*. Preliminary results obtained at 65 *GWd/tU* indicate the influence of the extended burn up on the stored energy threshold. At 33 *GWd/tU*, the test confirms the result obtained from SPERT and NSSR tests.

2.3 Limitation During Reactor Operation

Since the first fuel loading permit of the nuclear power plants, limited to a three batch core management with a 3.25% ^{235}U enrichment and a steady state (100% of nominal power), following evolutions have been proposed by the utility to improve the operating conditions of the nuclear power plants:

- load following, load regulation and frequency control, which have pointed out the importance and the limitations associated to PCI phenomena. This observation have led the French safety authority, for each new proposal from the utility, to require complementary studies on PCI phenomena not only in normal plant operation, but also for transient situations, to ensure rod integrity. Those special requirements lead to more restrictive limits for emergency shutdown (ΔT) and to a reduction of the operating range (P, ΔI),
- for 900 *MW* plants, the new hybrid pattern (3 batch MOX and 4 batch UO_2 - "GARANCE"*), has been implemented,
- for 1300*MW* plants, the four batch pattern (3.1% of ^{235}U enrichment) lead to a reduction of the operating range for the negative axial-offset (only one reloading).

Moreover, operating limitation are required in technical specification to ensure the safety of the plant during normal, incidental and accidental situations.

For example, these limitations concern the primary fluid activity, rod insertion limits, load following conditions.

2.4 Safety Goals and Associated Limits

Safety goals require that no damage or limited damage may occur on fuel rod during a transient according to its estimated frequency (P).

To fulfil this requirement, following criteria have been established for each condition of transients:

Condition 2 transients ($10^{-2} < P < 1$ /reactor year)

- DNBR > 1.17 or 1.30 (depending of the critical heat flux correlation),
- linear power density < 590 *W/cm*,
- non critical return (specially for cooling accidents for MOX fuel),
- fuel cladding stress less than yield strength.

Condition 3 transients ($10^{-4} < P < 10^{-2}$/reactor year)

- limited number of rod entering in DNB,
- molten fuel volume < 10% of the total fuel volume.

Condition 4 transients ($10^{-6} < P < 10^{-4}$ /reactor year)

- same as condition 3 transients and:

  Single rod ejection

  - maximum of energy deposit at the hot point < 225 *cal/g* (fresh fuel) and 200 *cal/g* (irradiated fuel),
  - molten fuel volume < 10 % of the total fuel volume,
  - cladding temperature < 1482 °C.

  Loss of coolant accident

  - max. cladding temperature < 1204°C,
  - max. cladding oxidation < 0.17 cladding thickness before oxidation,
  - max. hydrogen production shall not exceed 0.01 time the hypothetical amount which would be produced if all the cladding material enclosing the fuel (excluding the plenum cladding) reacted,
  - core geometry changes shall be such that core cooling is maintained,
  - long term cooling ensured.

The generic safety demonstration is based on the RCC-P (Règles de Conception et de Construction relatives au Procédé) and on accidents studies which demonstrate that all these criteria are met. This is the scope of the safety report package which must be agreed by the French safety authority before each new fuel loading.

3 Fuel Safety Documents for Reloading

Prior to loading fuel into the core of a reactor, the operator has to present the DSS (fuel reloading safety document) to DSIN for information. This document underlines that all the safety criteria associated with the fuel assemblies to be reloaded are respected (comparison of data calculated for the addressed cycle with the limits corresponding to the safety requirements). This document is specific to a reactor and to a bunch of assemblies.

The safety evaluation methods and key parameters used for establishing a specific DSS are justified in the DGS (generic fuel reload safety evaluation document).

4 Ongoing Research Program Related to Nuclear Fuel

From now, mean assembly fuel burn-up is limited to 47 *GWd/tU*. This authorization was given to EDF in 1988 and was subjected to the implementation of a research and development program on high burn-up fuels in order to confirm the authorized limit.

The CABRI-REP program, which started in 1993, was set up to better understand the effect of burn-up on the fuel (UO_2 and Mox) behaviour during a reactivity initiated accident.

Various experiments were realized in 1994, and the next experiment of this program is planned for early 1995.

In the meantime, DSIN examines, on a case-by-case basis, EDF requests to reach, for a limited number of assemblies (generally 1 or 2 in a core), a burn up higher than the authorized limit.

* GARANCE: new fuel management resulting of the reexamination of the CPY and CP0 reactors safety studies.

5 Conclusion

With the technical support of IPSN, DSIN carries out a thorough review of the adequation between the different types of fuel and the fuel operation modes, and the safety criteria. The authorizations DSIN delivers are based in particular on the technical analysis of the theoritical and experimental demonstration, by the utility, of the compliance with these criteria. The ongoing experiments in CABRI reactor, to determine the high burn-up fuel behaviour under reactivity initiated accidents until 65 *GWd/MtU*, are one of these.

DSIN asked the utility to stabilize and rank the fuel designs and managements modifications.

BG9600379

Fuel Utilization Experience in Bohunice NPP and Regulatory Requirements for Implementation of Progressive Fuel Management Strategies

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1 Introduction

At present, four VVER-440 type reactors are operational in Slovakia with total capacity of about 1760 *MW* electrical output at Jaslovské Bohunice site (V-230 and V-213 type). Four new units of VVER-440 are in different stage of construction in Mochovce site. The electricity production from nuclear power is about 50% of total generation capacity in Slovakia.

The VVER-440 reactors (V-230 and V-213 type) are operated either with a standard core consisting of 349 fuel assemblies, or with a reduced core consisting of 313 fuel assemblies.

36 shielding assemblies made of steel are placed at the peripheral positions of the core to reduce the flux of fast neutrons on reactor pressure vessel (230, starting from cycle No 6 of Units 2 and cycle No 13 of Unit 1). Independently of core size the nominal power (N_{nom}) of the reactors is maintained at 1375 *MW*. Required fuel cycle lengths are in the interval of 200 - 350 full power days (FPD) at present. The feed batch containing of about 90 to 121 fresh fuel assemblies are necessary for achieving of this requirements.

Starting from cycle 5 of Unit 1 (in 1983) low-leakage fuel management strategy was introduced insteed of traditional out-in fuel management scheme. Next improvement of the fuel cycle economy was achieved in 1987 by introducing of 4-year fuel cycles for part of fuel assemblies in reload batches. Fuel burnups expected at the end of Unit 4 cycle 10 are as follows:

- mean burnup: 38.3 *MWd/kgU,*
- maximal assembly: 39.7 *MWd/kgU.*

A biding process was launched in 1992 in former CSFR to diversity the fuel supply for VVER-440 reactors. This process is now in final stage.

A regulatory requirements for licensing process of new fuel, were formulated by Nuclear Regulatory Authority of Slovak Republik (NRA-SR) [1]. This document is now in the process of approval and should be issued at the end of this year. The document also specify the requirements for licensing of the recent fuel in different operational conditions in comparison with approved ones.

The paper summarizes the gained experience in fuel utilisation in 4-year fuel cycles and describes the basic requirements for fuel licensing process.

2 Fuel Residence Time Prolongation in VVER-440 Reactors

The original project of VVER-440 reactors supposed that the fuel assemblies will be operated during 3 reactor cycles with cycle lenght about 320 FPD. The feed batches for such a cycles are as follows [2]:

Cycle	Fuel assembly		Control assembly
	3.6%	2.4%	2.4%
Odd	102	0	13
Even	102	6	12

In the table: 3.6 % - enrichment 3.6 weight % of ^{235}U.

Reactor cycle length 320 FPD was recognized as nonoptimal for the national electricity grid. It was reduced to 290 FPD starting from the year 1984. This tendency was connected with lowering of mean batch enrichments i.e. with increasing of fresh 2.4 % FA portion in feed batches.

At the beginning of 80-s there was in progress an experimental utilisation of VVER-440 fuel in four-year cycles mainly in USSR, Bulgaria and former GDR. The reliability of the VVER-440 fuel burning 4 reactor cycles (4 years) in the core in the period of 1982-1989 in GDR was satisfactory (See Ref. [3]).

1212 FA were used at four reactor cycles. No adverse effect was found in comparison with standard 3-years utilisation of fuel. Maximally power loaded assemblies achieved burnup about 41 *MWd/kgU.*

Based on the experience, achieved with four year utilisation of standard VVER-440 reactor fuel, partial 4-year utilisation was accepted in former CSFR. The first utilisation started at Unit 3 of Bohunice NPP in 1987. 12 FA assemblies with 3.6% enrichment were left in the core for fourth cycle. The fuel assemblies were placed in the core periphery. The burnup of the 12 fuel assemblies after four cycles reached only 29.5 *MWd/kgU* i.e. smaller value in comparison with standard 3-year fuel exploitation [4]. During the following four years a limited portion of FAs were used in 4-year cycles. The maximum achieved burn-up at this period reached 36 *MWd/kgU.* It does not exceed the burn up reached at the beginning of the Bohunice NPP (Unit 1 and 2) operation.

Amount of 4-year fuel was limited not only by reactor operator precaution, but also by fuel cycle demands (long cycle following a short one). The amount of fuel bundles and the achieved burnup experience in Bohunice NPP are given in Table 1 (see ref. [4]).

Increasing 4-year FA utilisation can be seen in the tables starting from the year 1992 in connection with stabilisation of reactor cycle lengths. In 1994 1/4 of FA was left in Unit 3 and 4 for their fourth burn up cycle. The expected average burnup of FA operated fourth cycle in the year 1994 is 36.5 *MWd/kgU* with maximal assembly burnup 39.7 *MWd/kgU*.

Present experience from operation of the Bohunice NPP shows that no leakers were found among FAs operated through four burnup cycles.

3 Regulatory Approach to the New Fuel Utilisation Approval in VVER-440 Reactors

In the process of the new FA type utilisation approval or approval of present fuel utilisation in different conditions in comparison with licensed ones, the NRA SR started from recommendation of IAEA Safety Guide 50-SG-QA-11 and US-NRC-RG 1.70 Chapter 4 (and competent part of NUREG-0800).

New fuel is characterised by reasonable differences (in comparison with recent one) in fields as follows:

- construction (dismountable FA, differences in the grid structure, etc.);
- thermo-hydraulics (differences in fuel geometry and flow dynamics, coolant mixing, by-pass, etc.);
- neutronics (water-uranium ratio, enrichment, etc.);
- material design (exchange of the stainless steel grids by Zr grids, Zr 1% --> Zircaloy 4, etc.).

New conditions for fuel operation different from recent ones are accounted as follows:

- escalation of FA power above approved limit;
- extension of the burnup and residence time of the fuel in the reactor core above approved limit;
- load follow operation;
- changes in a water chemistry.

Table 1 Four-year fuel parameters at Bohunice NPP

Year	Cycle No.	Cycle length [FPD]	Full core: FA number	Full core: Mean burnup *MWd/kgU*	Full core: Max.FA burnup *MWd/kgU*	Core periphery: FA number	Core periphery: Mean burnup *MWd/kgU*	Core periphery: Max.FA burnup *MWd/kgU*	Core center: FA number	Core center: Mean burnup *MWd/kgU*	Core center: Max.FA burnup *MWd/kgU*
					Bohunice NPP - Unit 1 (reduced core starting from cycle No. 13)						
1987	8.	248+ 5									
1988	9.	247+ 6	12	33,1	33,1	12	33,1	33,1			
1989	10.	281+19	12	28,6	28,6	12	28,6	28,6			
1990	11.	294+12	6	28,7	28,7	6	28,7	28,7			
1991	12.	278+ 0	1								
1992	13.	257+12									
1993	14.	236+ 7	24	34,1	35,4	24	34,1	35,4			
1994	15.	257+14	36	36,4	38,1	24	35,6	35,7	12	38,1	38,1
					Bohunice NPP - Unit 2						
1987	8.	276+21									
1988	9.	289+ 8									
1989	10.	283+12									
1990	11.	292+ 8									
1991	12.	232+35									
1992	13.	201+13	6	32,2	32,2	6	32,2	32,2			
1993	14.	252+ 7	12	30,4	30,4	12	30,4	30,4			
1994	15.	260+18	24	36,7	37,2				24	36,7	37,2
					Bohunice NPP - Unit 3						
1987	4.	256+ 8	12	29,5	29,5	12	29,5	29,5			
1988	5.	292+ 0									
1989	6.	281+22	6	35,2	35,2	6	35,2	35,2			
1990	7.	283+12									
1991	8.	325+ 6									
1992	9.	274+21	12	34,4	34,4	12	34,4	34,4			
1993	10.	260+ 9	42	34,9	35,9	36	35,2	35,9	6	33,1	33,1
1994	11.	201+28	78	34,7	37,2	36	33,9	34,7	42	35,4	37,2
					Bohunice NPP - Unit 4						
1987	3.	231+10									
1988	4.	299+ 0	12	31,3		12	31,3				
1989	5.	303+ 4									
1990	6.	285+ 8									
1991	7.	291+34	36	32,7	33,5	36	32,7	33,5			
1992	8.	281+ 7	42	35,6	37,6	24	34,5	34,5	18	37,0	37,6
1993	9.	262+18	60	37,4	38,4	36	37,1	37,2	24	37,8	38,4
1994	10.	197+26	78	38,3	39,7	24	37,9	38,2	54	38,5	39,7

In accordance with the Slovak laws No 50/1976 and 28/1984 the responsible organization (NPP) is obliged to apply to NRA SR for approval of new fuel utilization or recent fuel utilization in new condition. Important part of the application is a rewised Safety Report which contains:

- calculational and experimental results which proves the thermo-mechanical, thermohydraulical and neutronics features of the fuel,
- safety assessment for the fuel in normal and accident condition.

Required content of the experimental and calculational results correspond with US-NRC-RG 1,70 Chapter 4. The safety analyses are required in the range defined in [5].

In connection with new fuel (or recent fuel in changed conditions) NRA has another additional requirement - confirmation of the safety features of the fuel in real conditions (active power reactor tests in similar power reactor). If such results are not available prototypein- reactor testing is required.

4 Conclusion

An experience was gained with utilisation of the recent fuel in 4-year fuel cycles. Among 504 fuel assemblies left for the fourth burnup cycle no leakers were found. Regulatory requirements were formulated for exploatation of new fuel (or recent fuel in changed conditions) in VVER-440 reactors operated in Slovakia.

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MAIN SUBJECTS OF ACTIVITY

- Activities associated with decommissioning of nuclear power plants, commercial and research reactors and other nuclear facilitites
- Disposition of stored radioactive wastes after obtaining the corresponding permissions
- Scientific research, design and technological support for nuclear power generation plants
- Reconstruction of NPP units
- Lifetime extension of NPPs, nuclear facilities, other power plants and energy-related facilities
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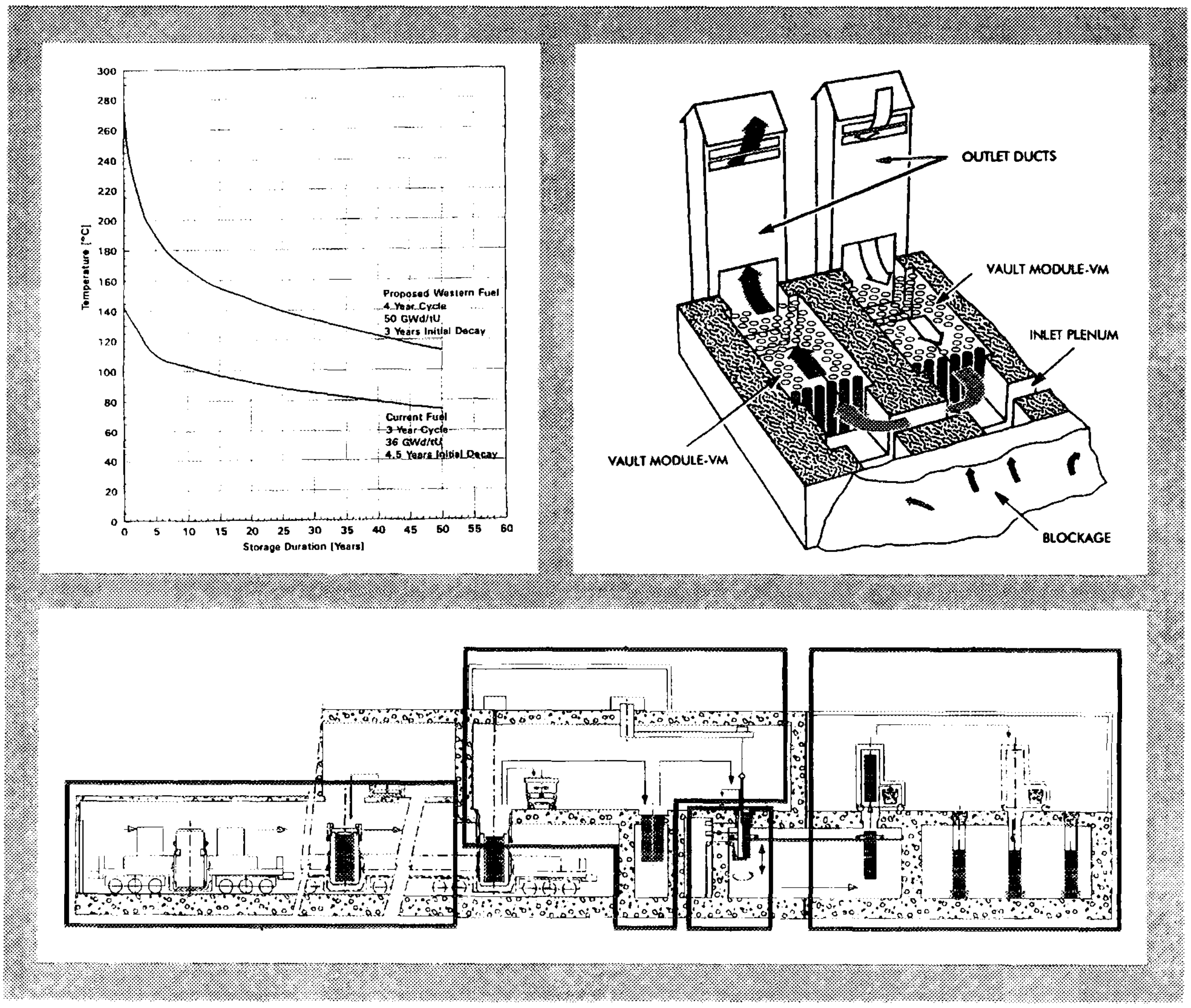
ОСНОВНЫЕ ВИДЫ ДЕЯТЕЛЬНОСТИ

- Деятельность в связи со снятием с эксплуатации атомных электростанций, промышленных и исследовательских реакторов и других атомных установок
- Ликвидация накопленных радиоактивных отходов после получения соответствующего разрешения
- Научные исследования и конструктивно-технологическое обеспечение атомной энергетики
- Реконструкция блоков атомных электростанций
- Продление срока службы атомных электростанций и установок, других электростанций и энергетических сооружений
- Модернизация энергетических сооружений
- Консультантская деятельность в области энергетики, включительно атомной энергетики

Part 4

Spent Fuel of VVER Plants



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RADIOCHEMISTRY AND RADIOECOLOGY STUDIES

RADIOCHEMISTRY

The activities of the INRNE staff in this field are centered on the following topics:

- fundamental research on the behaviour of radionuclides in carrier-free solutions;
- fundamental research on the electromigration of radionuclides;
- basic study of transition metal ions, lantanides and actinides complex chemistry;
- fundamental research on the migration of radionuclides in geological strata and prediction of their migration from radioactive waste repository.

Applied activities include radiopharmaceous production. Different radiopharmaceuticles labelled with Tc^{99m} and I^{131} are produced in INRNE hot cells and supplied to hospitals.

RADIOECOLOGY

The activities of INRNE staff are centered on the scientific basis of radioactive waste management and environmental radioactivity studies.

The activities in the field of scientific basis of nuclear waste management (treatment, conditioning and disposal) include:

- development of improved methods for treatment and conditioning of radioactive wastes from NPP and nuclear applications;
- safety assessment of radioactive waste disposal;
- development of improved engineered barriers against radionuclide migration.

Experience and knowledge gained in this field resulted in development together with other Institutes of the Bulgarian Academy of Sciences of a document entitled "National Radioactive Waste Policy", proposed to Bulgarian Government.

The environmental radioactivity studies include:

- radioecological assessment of the decommissioned nuclear facilities mainly from the uranium mining and milling industry;
- development of methods for analysis of radioactivity of samples from geological materials, soils, plants and natural waters of different origin (terrestrial and marine). A variety of methods are applied, including radiochemical analysis, gamma, beta and alpha spectrometry and neutron activation analysis.

Spent Fuel of VVER Plants

Panel Discussion Report

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Two topics were proposed for discussion, namely:

- the actual storage situation in eastern countries and
- to explore the alternative storage possibilities.

Topics discussed in the presentations were:

- Survey and prospects of spent fuel management in Bulgaria.
- Long term interim storage: dry storage in vaults.
- Spent fuel management.

During the discussion, the urgent need for enlarging present storage capacities or creating new storage capacities in Bulgaria, Slovakia, Czech Republic and Hungary, was expressed and repeated several times. For instance the Bulgarian capacity will be full by 1999 providing that the away-from-reactor storage facility is modified to be able to accept VVER-1000 fuel and that the technology of compaction of the fuel storage is used. This means that a decision to build an additional long term interim storage capacity in Bulgaria has to be taken by 1995.

The possibility of using available capacity in the region was explored. One available option would be to return the spent fuel to Russia, however this service, which was previously included in the fuel contracts, is now payable in hard currency. In addition the storage in Russia can only be accepted if it is intended for future reprocessing.

The subject of storage in vaults versus casks was also discussed. The conclusion drawn was that the cost evaluation, when comparing the two options, should be based on full life costs.

Finally, the financing of new storage facilities was discussed. It appeared that cash is scarce in east European countries making the project of building new storage capacities very uncertain. It is an area where international financing (CEC, G7, etc.) is urgently needed.

BG9600380

A Strategic Approach to Short- and Long-Term Irradiated VVER Fuel Management

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1 Introduction

Before I start, a few words about BNFL. From 3 main sites in England we provide nuclear fuel cycle services for all the British reactors and for reactor operators in Europe, Japan and the Rest of the World. We believe in the closed fuel cycle, with re-use of the uranium and plutonium separated by reprocessing.

Our new thermal oxide fuel reprocessing plant (THORP) has just come into active operation and we will be reprocessing 7000 *t* of thermal oxide fuel (PWR, BWR and AGR) over the next ten years. However, we recognise that reprocessing is not the only strategy for short-term or even long-term irradiated fuel management. There are alternatives, but these alternatives need to be compared on a consistent basis, taking account of all cost and other implications, not just short-term factors. You should also bear in mind that storage, even dry storage is not a long-term solution.

In conjunction with a European electric utility we derived a methodology for comparison. The key step is the setting of selection criteria. What defines the best option? The key stages of the methodology are:

- Evaluate current situation
- Identify list of options/possible solutions
- Cost/funding analysis
- Selection criteria
- Optioneering/evaluation
- Conclusions

and will be described in Section 2.2.

2 Methodology

2.1 Value of the Methodology

The value of this methodology is that all interested parties can take part in the analysis and together derive the basis for decision-making. Some of the questions picked out at random from those posed in the study we carried out with the European utility are shown below. We were able to answer all these and I will describe how we answered the second one at the end of the presentation.

- When can uranium and plutonium recovered by reprocessing be recycled cost effectively in VVERs?
- If reprocessing is not the short-term option chosen, is storage of irradiated fuel at the original licensed nuclear reactor site preferable to a separate storage-only site?
- Are modular vault dry stores and cooling ponds, which necessitate significant capital investment prior to deployment, more costly overall than other options?
- Should the most suitable form of irradiated fuel management be determined only by cost constraints?

2.2 Steps in the Methodology

2.2.1 Assess the Current Situation

What is the current situation on fuel arisings and storage capabilities, storage regime, fuel characteristics, storage locations? Who are the key personnel in the country concerned, utility men, regulators, operators etc? Who should form the small team along with the consultant? The consultant, or rapporteur in the jargon, acts like a catalyst. He or she simply accelerates a logical decision-making progress that could take years without assistance. Site visits are essential to gauge public feeling. And we found a questionnaire approach much the best way. We then had an immediate written record to discuss, clarify understanding and even on occasions compare differing answers.

2.2.2 Assess Priorities

How important is the spent fuel management problem? In what timescale does it need to be solved? How does the problem fit in with the other Government priorities? What are the criteria against which you will be judging?

2.2.3 Options/Possible Solutions

The technique used for generating possible solutions was brainstorming and we found it convenient to do this over a number of different time periods.

Possible options generated will not be a surprise but at least we knew we had a finite set to evaluate. Options were of course reprocessing (Russia, UK, France) and direct disposal (in country/international repository) but in the interim, storage. I stress again that storage systems are not a long-term solution.

2.2.4 Generic Storage Systems

Systems we divided into five generic types:

- pond system;
- one piece flask (metal);
- two piece flask (concrete);
- canister in vault;
- bare fuel vault system.

Examples are the 6000 *t* pond at Krasnoyarsk in Russia, the CASTOR iron flask for VVER 440/84 fuel used at Dukovany in the Czech Republic, Sierra Nuclear Corporation's cask with its MSB, multi-element sealed basket proposed for Zaporozhie, Ukraine. Canister in vault example is NUHOMS - probably the most widely-used dry storage system so far, although only in USA. Another example is the French CASCAD system. Bare fuel vault example is GEC Alsthom's as proposed for Paks in Hungary.

2.2.5 Cost/Funding Analysis

One of the major considerations is cost and to take account of the fact that expenditure would be taking place at differing periods in time we used the technique known as Discounted Cash Flow (DCF). This represents a mathematic way to derive a figure known as the Net Present Value (NPV) which is the amount of money that if invested now with an assumed rate of return would meet the costs as they arise. The option with the lowest NPV is clearly the most economic. This technique also allows the calculation of a budgetary provision, i.e. how much to allow over the remaining reactor lifetime for spent fuel management. The choice of discount will vary according to the particular situation of the utility.

2.2.6 Selection Criteria

Cost and technical factors are not the only criteria on which to make a decision. Our brainstorming session identified a number of important selection criteria. Likely ones will include:

- cost (life cycle);
- environmental;
- time to deploy;
- local content.

The choice of additional selection criteria will depend on a country's situation.

2.2.7 Optioneering/Evaluation

The way in which the options were evaluated was for each time frame, the study team ranked the options and rated them against each of the criteria. The host country or organisation must then decide on the relative weightings of the criteria. For example, is high local content crucial? It is then a straightforward task to use a matrix to devise weighted scores.

Without uranium and plutonium recycling the reprocessing option is the most costly but environmentally it is probably the best because of the protection systems in operation and the controlled and monitored way in which wastes can be disposed of safely. Local content in this option is minimal. In the study with the European utility, where costs, local content and environment were major issues, reprocessing was not the favoured option. In that particular set of circumstances when the totals were worked out in an objective manner, it was concrete casks that had the highest score.

2.2.8 Conclusions

The conclusions that can be reached from this methodical approach lead to firm recommendations based on objective assessments. The recommendations have ownership because of the involvement of the appropriate personnel at every stage. The conclusions are unique - just as every situation is unique. The methodology builds on existing expertise. It is not an imposed solution and there is an excellent exchange of knowledge and skills between the people involved.

3 Postscript

As a follow-up study we were asked to consider storage location, be it for casks, vaults or whatever. Three possible locations were considered and a similar marking and weighting system was used to derive the objective conclusion that the Nuclear Power Plant was in this particular country the best location, with economics and politics being the major issues in this case.

I would simply end by saying that when a country is choosing its spent fuel management options, I believe it should not just consider the short-term nor costs in isolation. I hope you share that belief.

BG9600381

Spent Fuel Generated by the Kozloduy Nuclear Power Plant within the Period 1974 - 1994

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Spent fuel management during the 20-year's operation of NPP Kozloduy can be divided in two stages based primarily on the storage time requirements of the spent fuel receiving country.

The first stage covers the short-term (3 years) storage of the spent nuclear fuel (SNF) in the reactor building spent fuel pits (SFP 1 to 4) with subsequent dispatch to the ex-Soviet Union and spans the period from 1977 when the first lot of fuel was sent back from Unit 1, up to 1988. Within that 11-year period, there were 21 shipments of SNF to the ex-USSR at no cost comprising a total of 3086 assemblies with various levels of enrichment and burnup. The basic means of transportation adopted was the Soviet-designed container TK-07 which accommodated one shipping cask holding 30 fuel assemblies with maximum burnup level 24 *MWd/kg* of U and total maximum heat capacity 8 *kW*.

The initially specified manner of transportation included a stretch of road (about 7 *km* from the NPP to the station's pier) by which 100-ton containers were transported on trailer trucks. They were transhipped on a propelled river barge with eight seats for vertically placed containers which was brought by thruster to the Reni port on the Danube. From there, after reloading on to railway platform cars they were taken by a special train to the reprocessing works. After the Vranca earthquake when the integrity of the Reni piers deteriorated, that manner of transportation was revised and options involving the Varna-llichovsk ferryboat line were proposed, however, due to lack of a railway from Kozloduy to Varna and the several transhipments required by these options, it was resumed as the only practical route. During that period, Bulgarian firm, the containers were transhipped by means of a 100-ton floating crane rented from port Ilichovsk every year. That complicated the transportation and increased its cost.

Another problem that has not been solved up to the present moment is leaks of SFP cooling water into the adjacent rooms. That is the main reason for which only half of the SFP capacity is utilized. Such leaks are due to design drawbacks, that is, sheet steel lining of the concrete walls permitting them to "respirate" during the regular filling and draining of the pits for refuelling of the reactors. The repairs of SFP 1 and 2 did not yield any significant results, and the proposed redesign of all the pits in order to improve their capacity and applying of a new lining were rejected by the design engineer in favour of an interim storage facility on the plant site. The latter solution was necessitated by the request of the spent fuel recipient to extend the time of SF storage from 3 to 5 years on the plant site before sending it back to the USSR. According to the initial design that spent fuel storage (SFS) had the capacity to accommodate the spent fuel from Units 1 - 4 in 10 years' operation, however, the design was subsequently amended to receive spent fuel from the new Units 5 and 6 VVER-1000 as well.

Due to obstacles with securing the VVER-1000 spent fuel transport vehicles (containers, trailer trucks and container tippers) and in order to increase the capacity of the SFP (spent fuel pits) the existing racks were replaced by new, higher-density ones. That permitted to extend the deadline time for completion of the SFS (spent fuel storage)and for developing the rest of the transport vehicles and handing equipment.

Another problem related to erection of the new type of reactors was taken up and left unsolved, namely, the vessels for shipping the SF containers on the Danube. The design of a self-propelled special-purpose river vessel developed by the Institute of Shipbuilding in Varna was not implemented due to shortage of finances. The advisability of its implementation is also questionable now after the complications created by the new conditions of sending back the SF, set by Russia, and the uncertainties connected with the transit of radioactive materials through the territories of Moldova and the Ukraine.

With regard to the complicated situation in the ex-USSR, we sought an opportunity to dispose of the SNF in certain West-European countries who offered reprocessing services. A high interest was demonstrated in that respect by BNFL of UK who undertook a study on the possible manner of transportation. For a number of reasons beyond their control the effort was not brought to an end but still that option can be considered among the feasible and well studied.

The second stage began where the last campaign of SF removal ended in the fall of 1988, and continues till the present day. That stage is related to the problems of commissioning the on-site spent-fuel storage (SFS) and demonstrates the most typical drawbacks of construction work management and licensing. Additional problems in that respect were created by the reassessment of the seismic risk on the NPP Kozloduy site. As a result of that, the calculated Maximum Design Earthquake of the plant site was adjusted one degree higher. That necessitated development and implementation of a program for additional seismic upgrading of the structures and installing more detecting and recording systems.

The changes which took place in the ex-Soviet Union led to significant changes in the approach to spent fuel management in the East European countries, including Bulgaria. The first offer for receiving spent fuel at international market prices was not acceptable to us and we had to make every possible effort to commission the SFS earlier than scheduled.

As a result of summit talks, there is now an oral agreement about sending back the spent fuel from the new reactor units which has been recorded in the inter-government covenants on their construction; however, the issue of sending back the spent fuel from Units 1 - 4 is still open.

Actually, it was found that certain incompletely defined electricity generation technologies have been offered as it is the case with nuclear generation, with a promise for later defining and still vaguer promises for receiving back the spent fuel or nothing at all is mentioned, as is the case with radioactive waste treatment.

These are some of the major problems of the nuclear power subsector, the lack of solution for which give rise to doubts about the expedience of developing this technology in countries such as Bulgaria which do not possess the required technological and research basis to support such complex and high-cost technology. This is especially acutely felt at the current moment when practically all relations with the equipment supplier have been severed.

In the critical financial condition of the Bulgarian power sector at the moment, the managers of this sector are not very enthusiastic about the option of sending the spent fuel back to Russia. That is why the option of building modular spent fuel disposal facilities after western dry technologies is considered the most acceptable.

The annual capital costs for the construction of this type of facilities are more reasonable than the costs of sending the spent fuel back to Russia forgetting, however, the fact that, sooner or later, the funds for reprocessing of that fuel will have to be provided. On such grounds it can be assumed that allocation of such funds should start at the moment of fuelling of the reactors, while actually that has never be done and nobody can answer the basic question "Where will the money come from for reprocessing such large inventories of spent fuel, and also for disposal of the high-radioactive waste generated upon that reprocessing?" That matter should be legally settled through the Law for Peaceful Use of Atomic Energy, the Procedure of its application, and the relevant Orders.

In our case, there is state monopoly in the field of nuclear power, and the Government is owner of all nuclear stocks, and provided that its responsibility for management of the generated spent fuel is not legally regularized, it is practically impossible to make one decision or another without referring it to the Government while the latter, making the decision on development of the nuclear energy, has let itself be misled by the promise of the nuclear generation technology supplier to accept readily the spent fuel throughout the service life of the plant without binding that with the respective international agreements as is the case with the spent fuel from the Kozloduy Units 1 to 4.

In 1992, there appeared the risk of shutting down Unit 5 (VVER-1000) due to filled up spent fuel pit. That made the managers of the subsector to seek urgently a way out of that situation by assigning to Energoproject the task to develop Terms of Reference for the purpose of inviting tenders for construction of a dry spent fuel storage. The ToR Document was sent out to nine world known firms with past experience in that field. After evaluation of the bids, our specialists selected the NUHOMS technology offered by Nuclear Pacific of USA as one of the most beneficial and quick to implement proposals for the site of Kozloduy NPP. No contract was concluded, however, because that required the approval of the Government which, in reply to our request, asked us to clarify finally our relations with Russia. None of the two countries seems to be in a hurry to settle that issue relying on the remaining few years before final exhausting of the SFP capacity.

In conclusion it can be said that, even in the event of refusal on the part of Russia to receive the spent fuel for reprocessing, we can depend on the near term construction of modular spent fuel disposal facilities which could postpone, by some time, the reprocessing or direct final disposal of the SF. Most professionals in Bulgaria tend to accept that decision in spite of the social risk connected with the uncertainly about the behaviour of the fuel elements in time. It shall be taken, however, before the end of next calendar year. That would permit to raise the necessary funds for spent fuel management which is imperative at this critical point of transition to market economy to which Bulgaria has come.

The things that still have to be done are to persuade the Government of the country and to develop the appropriate legal and normative regulations.

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Status and Operational Experience Report of Spent Fuel Storage Facility in Kozloduy NPP for the Period 1990-1994

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1 Background information

The idea of building a Spent Fuel Storage Facility appeared with Kozloduy Unit 1 start up in 1974 as an alternative of spent fuel transportation to USSR. Designing began after relevant contracts signing of in December 1978 and construction started in 1982. Start-up lining operations and functional tests were performed in the period 1988 - 1990. During this period all programmes, instructions and documents were coordinated with CPUAE.

In February 1990 after an inspection of the Bulgarian Regulatory Body a license for temporary operation of the Spent Fuel Storage Facility was obtained with the purpose of complete functional tests of the systems and the equipment and on 28.02.1990 the first basket with 30 spent fuel assemblies from VVER-440 was installed.

Immediately after that a Programme for safety upgrading of Spent Fuel Storage Facility was worked out and implemented to be in compliance with new regulating documents.

2 Spent Fuel Storage Facility Technical Characteristics

The Spent Fuel Storage Facility is designed for a long-term storage of 168 baskets (4920 spent fuel assemblies) which are generated by the NPP units over a period of ten years. The assemblies are stored in the Spent Fuel Storage Facility after 3 years storage in the reactor pools, the transportation is carried out by means of internal stationary transport container (ISTC) and a specialized auto trailer.

There are two operating conditions for the Spent Fuel Storage Facility:

- spent fuel receiving condition;
- spent fuel storage condition.

3 Operational Safety Aspects of the Spent Fuel Storage Facility

The Spent Fuel Storage Facility operational safety is ensured by a number of project technical decisions and actions:

- The assemblies are stored under a protective layer of demineralized water in transport baskets. The distance between the baskets is not less than 1600 *mm*. Having such an arrangement the nuclear safety conditions are with backup because according to these conditions any touching of the baskets is allowed.
- The spent fuel transportation is carried out either under a protective layer of water or in a special protected container.
- Residual heat removal is performed by water cooling system in the pool which ensures temperature below 45°C in the spent fuel compartments.
- The water used in the storage pool, as it has to meet special requirements, is decontaminated in time and quality controlled.
- Fuel compartments leaks are controlled and their detection and location is possible in time because linings are double - sandwich type.
- Continuous remote control of water temperature and level in the spent fuel pool is performed.
- Radiation dosimetric control is performed.
- Automatic mode of spent fuel storage pool feedwater and overflow is available as well as continuous control of the permissible high and low water levels.
- There is an additional free compartment, not filled with spent fuel which allows compartments preventive maintenance and repair during Spent Fuel Storage Facility operation.
- There are special stands for flushing and decontamination of transport and technology equipment.
- Separation of rooms into zones of organized access and dosimetric control of operating personnel in accordance with actual regulations and rules.
- Feed-water, cooling, cleaning and overflow piping of the pool are situated in its upper part which prevents occasional discharge of fuel compartments through "siphon effect" in case of a pipe break.

Two groups were assigned to perform two independent ecological investigations for proving the building safety.

- Spent Fuel Storage Facility ecological investigation for NPP by a group from "Risk Engineering" headed by Anton Boyadjiev.
- Complete report of the assessment of NPP Kozloduy Spent Fuel Storage Facility impact on environment by a group from the University "St. Climent Ohridsky" headed by Tsvetan Vasilev.

The conclusions of both groups are unanimous:

1. "The main conclusion in accordance with the purposes of the investigation is that the discussed project does not violate the clauses of

the Atomic Energy Law and Environmental Protection Law."

2. "Spent Fuel Storage Facility operation in normal conditions is not dangerous for its operating personnel as well as for the environment. The performed up to date radionuclides content in the pools water analyses results and the air of the working rooms confirm this statement. On these bases the investigation group proposes continuation of the Spent Fuel Storage Facility operation."

The documents pointed above were considered and accepted on experts council in the Ministry of Environment.

All the above mentioned proves that the Spent Fuel Storage Facility operation is not dangerous for the environment and the inhabitants health of the surrounding villages, regarding contamination.

Nevertheless aiming for safety operation the Spent Fuel Storage Facility work continuous in the following directions:

- improvement of systems and equipment reliability to decreasing the accident probability;
- improvement of seismic stability of Spent Fuel Storage Facility building structure;
- increase of personnel qualification.

4 Spent Fuel Storage Facility Operation Experience

At present in the Spent Fuel Storage Facility are stored 75 baskets with 2196 spent fuel assemblies from VVER-440. The operational experience shows that foreseen in the project, consecutive filling of the compartments is not suitable because of the unequal heat loading of the pool building construction. A programme for equal filling of the 4 compartments was worked out where the allocation was according to the residual release energy. This lessens to a large extent the possibility of micro cracks on pool walls and bottom and possibility of leaks.

In order to prevent linear expansion of the walls as result of the high temperature differences, the practice of cooling in small temperature intervals ($\Delta T \approx 5°C$) was adopted.

This also leads to better operation of the Spent Fuel Storage Facility.

The existing Facility for temporary storage of spent fuel under water when it is properly operated without accident situations is reliable enough, does not lead to contamination of environment and is not dangerous for the health of the operational personnel or the population.

Storage of Spent Nuclear Fuel: The Problem of Spent Nuclear Fuel in Bulgaria

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Introduction

The storage of spent nuclear fuel (SNF) is an intermediate stage of the nuclear fuel cycle - between the utilization of the fuel in a NPP and the reprocessing or final disposal. The storage of SNF requires special technologies because of the high radioactivity (~ 1 *Ci/g*, ~ 10^5 *n/s.kg*), the residual heat (~ 1 *W/kg*), the generated plutonium (5 - 10 *kg/tU*) and the hazard of radioactive contamination in case of fuel degradation.

The SNF is a special type of radioactive product generated during the work of nuclear power plants. The common feature between SNF and highly radioactive wastes (HRW) is that the planned final repositories are both for SNF after preconditioning procedures and HRW.

1 The Problem of SNF

Up to 1993 more than 120000 *t* heavy metal (*t* HM) of SNF from LWRs and HWRs have been generated. Only 5000 *t* HM have been reprocessed without the fuel for ~ 220*t* military plutonium. [1, 2] The expected capacity of the reprocessing plants in 1995 is ~ 5800 *t/year* at generation rate of ~ 10000 - 12000 *t* HM/ye*ar*. If the new plant in Russia is put into operation the reprocessing capacities will increase with two lines of 1500 *t/year* each. The plant in Krasnoyarsk is planned for VVER-1000 fuel but its construction is temporarily stopped (1994).

The amount of generated fuel worldwide is greater than the total capacity of all reprocessing plants but nevertheless the plants are not loaded - the reprocessing plant in Sellafield has commenced work at the beginning of 1994 although it has been in condition for operation for a long time. The reason for the postponement is that the contracts for reprocessing do not guarantee continuous and profitable process. The high reprocessing price makes most of the potential customers to defer the decision for reprocessing or disposal.

The estimated reprocessing capacity for LWR fuel is summarized in Table 1 [3] :

Approximately 50% of the total quantity will be generated in countries which have accepted or are close to accepting the option for final disposal without reprocessing: Canada, Sweden, Spain, Finland, USA.

Another group of countries support the reprocessing option: UK, France, Belgium, Argentina, China, Italy, Russia, Germany, Japan, etc.

Japan and other countries have invested in research projects for reprocessing of SNF with partitioning of Pu, Am, Cm and other transuraniums, and also of long living fission isotopes (Cs-137, Sr-90). Both options are investigated - final disposal of the separated isotopes and also transmutation of the long living isotopes - fissioning with fast and thermal neutrons for the actinides and photonuclear reactions for the Cs and Sr. Most of the efforts are for the plutonium isotopes [4, 5].

The existing stockpiles of plutonium are planned to be reused in MOX fuel. In Japan, France, Belgium and other countries there are facilities for production of MOX fuel. The estimations are that the mixed oxide fuel can reduce the rate of plutonium generation by 1/3 and the already separated plutonium can be "burned" in the beginning of the next century. 30 reactors in Europe have already been licensed for use of MOX fuel.

A third group of countries without reprocessing or repository capabilities store the SNF in wet or in dry storage facilities and have delayed the decision waiting for further development of the technologies and reduction of the cost of the back end. Approximately 14 countries have accepted the policy of "deferred decision", 9 of them together with the option for reprocessing or final disposal. (Argentina, Czech Republic, Finland, India, Italy, Rep. of Korea, Lituania, Mexico, Pakistan, Slovenia, South Africa, Spain, Ukraine)

Bulgaria in an IAEA report is attributed to the group of countries with deferred decision [6].

The greater part of these countries have constructed or planned facilities for storage of SNF for all the fuel which is expected to be generated until the end of the operation period of their NPPs. The conclusion from the present paragraph is that:

a) although the capacity of the reprocessing plants is more than two times less the generation rate of SNF, the plants are not loaded. Approxi-

Table 1 SNF Reprocessing capabilities, *t* HM

	1992	1995	2000	2005	2010
France	1200	1600	1600	1600	1600
UK	0	1200	1200	1200	1200
Russia	400	400	400	400	400
Japan	100	100	900	900	900
Total:	1700	3300	4100	4100	4100

mately only 5% of the total amount of generated SNF has been reprocessed;

b) the number of countries which have delayed the decision for the back end of the nuclear fuel is significant;

c) no big quantities of SNF have been diposed of in final repositories. The first repository is expected to start operation near the end of the centrury.

In other words: there is a worldwide demand for SNF storage capacity.

2 Technologies for SNF Storage.

The existing technologies are wet and dry storage of SNF [7, 8, 9].

The wet storage is an already developed technology and there are no communications for fuel degradation during wet storage (see footnote). For dry storage of uranium dioxide fuel inert atmosphere is necessary for temperature above 150°C since for failed fuel pins additional oxidation occurs and the cladding can be damaged.

In the last years new technologies were developed for dry storage of LWR fuel in inert atmosphere, double barrier, fuel control and passive cooling.

2.1. Wet Storage of SNF

The wet storage [7, 8] is only for intermediate storage of SNF. The projected terms of storage is in average 40 *y*, the maximal term is ~ 60 *y*.

The wet storage is considered to be an already developed technology. Some work is still continuing on consolidation of fuel, criticality calculations with K_{eff}=0.95 and further nuclear safety improvements. The quality of the water is maintained by:

- ion exchange systems;
- filters;
- systems for cleaning the surface of the water;
- sucking systems for the bottom of the pools;
- systems for scrubbing the walls of the pools, especially at the boundary air/water.

In most of the wet facilities there is a space between the liner and the concrete for leak control.

The fuel consolidation with boron absorbers between the assemblies is a practice both for AR and AFR pools.

The radioactivity release in the environment from wet facilities (AFR) is negligible and in some cases it is beyond the limit of detection.

The major radioactive wastes from AFR pools are ion-exchange resins.

The basic disadvantages of the wet technology are:

a) active systems for supply and purification of water;

b) possibility of loss of water for beyond project accidents. In case of loss of water the fuel can be damaged from overheating.

2.2 Dry Storage of Uranium-Dioxide SNF

Dry storage [9, 10, 11] with passive cooling with air has the inherent possibility for storage for more than a 100 *y*. The present technologies are for terms of 40 - 60 *y*. The declared terms of 40 - 60 y is due mainly to the lack of experience for long storage. There are communications for projects over 100 *y* [10].

At present the dry storage terms are comparable with those for wet storage. If the storage period is increased over 100 *y* that will be a strong advantage over the wet technology.

The developed technologies for dry storage make possible safe storage in regions with increased seismicity.

The major deficiency of the UO_2 fuel is the possible further oxidation to U_3O_8 when stored in air and high temperature. If there are defective fuel pins the cladding can be damaged since U_3O_8 has less density than UO_2. When the fuel is further oxidized the original crystal lattice changes and the ability to retain the fission fragments decreases rapidly.

When the fuel is oxidized to U_3O_7 the volume of the fuel is not changed since the densities if UO_2 and U_3O_7 are the same. When the temperature is lower and the content of oxygen is less the fuel is oxidized to U_3O_7.

For temperature below 300°C the rate of the reaction decreases by an order of magnitude each 30 degrees [12,13].

According to a 1990 research safe temperature for storage of defective fuel in air for 30 y is 135-160°C [10]. For inert atmosphere with 1% oxygen for 30 *y* the upper limit of the safe temperature is estimated 210 - 220°C. There are more general estimations that probably acceptable temperatures for which the oxidation of SNF will not lead to observable effects is below 200°C [9].

According to 1992 estimations non-defective fuel can be safely stored in air for 30 *y* at 330°C [11].

The possible solutions for dry storage of SNF are:

- storage in containers in inert atmosphere;
- storage of defective fuel in inert atmosphere;
- storage of the fuel below the oxidation threshold temperature.

The basic principles which have to be observed when facilities for dry storage are designed are:

- two independent barriers;
- possibility for fuel control;
- possibility for fuel retrievability.

The developed technologies for dry storage are divided into two major classes - double purpose technologies - for transport and storage, and technologies only for storage.

Typical representatives of the double purpose technology are the CASTOR and TN-AVR casks. The casks are filled with helium, have partial shielding for gamma-rays and neutrons. The seal is double with the an option for control of possible helium leak between them. The stored in vaults casks are cooled passively by air.

In the last years have been developed modular facilities with passive cooling for dry storage of LWR fuel with burnup up to 50 *MWd/tU*.

The modular facilities are divided into two types - vault types in which the increment is in large "portions" and facilities which consist of small modules for 1 - 4 canisters for 15 - 20 fuel assemblies each.

Modular vault facility is the storage facility built by GEC Alsthom for gas-cooled reactors. Similar facility is the one in Cadarache although not modular (the proposed by SGN facility for Kozloduy NPP is modular). In these facilities the fuel assemblies are sealed each in a separate tube in helium or nitrogen atmosphere and one or several such tubes in one common vertical larger sealed tube passively cooled by air. Such a system allows the control of possible helium leaks in the larger tube (if helium is used). GEC Alsthom Engineering Systems Ltd have a license for modular vault dry storage facility (MVDS) in Fort Saint Vrain (USA) and also a proposed project for Paksh. The operations in the MVDS are remote and automatic.

A typical representative of small modular facility with passive cooling with air is the NUHOMS system. In USA by NUTECH (beginning 1980 r., NUTECH, DOE, Electric Power Research Institute, Carolina Power&Light) a horizontal modular system for SNF has been developed (Horizontal Modular Storage - NUHOMS). The system is passive and consists of two modules - heavy shielding module - HSM and dry storage canister - DSC. Because of the modular type and cheap materials (concrete for HSM) it is considered that the system has economical advantages compared to other technologies. The modular type makes possible the building of a high capacity facility without considerable initial investments.

The dry canister is a stainless steel cylinder in which there is a gasket with sleeves for the assemblies. The leak tight bottom is guaranteed by the in-factory checked welding. The front end is sealed by two independent welds performed by automatic welding machine. During the fuel storage period the condition of the fuel is not controlled. The thermal capacity of the DSC is 20 *kW* or 24 fuel assemblies from VVER-1000 can be stored in the DSC after 5 *y* cooling in AR or AFR pools. The cooling gas is either helium or nitrogen. The approximate collective exposure is 10 *mSv.man* per cask transportation and loading in HSM.

There are communications (1993) for further development of the system in Japan for better seismic safety.

In 1993 a similar system was reported developed by AECL with vertical storage of canisters in concrete modules [11]. The MACSTOR system includes storage canister, transport cask with gamma and neutron shielding and modular facility for vertical storage. The system MACSTOR is designed for LWR fuel which is a new field for Canada and therefore Nuclear Fuel Handling Services, Transnuclear Inc. (TNI) are consultants for MACSTOR.

In 1994 there were reports for a development of a multi-purpose cask, MPC, for transport, storage and final disposal [14]. The DoE of USA supports the projects mainly because of incompatibility of different systems for storage and transport which requires additional fuel reloading, inefficient use of equipment and generation of contaminated and activated equipment. The additional fuel operations increase the cost of the fuel back-end.

The option for "control of the fuel" during storage is important because *after the intermediate storage the containers, casks or canisters will be opened* and there must be definite knowledge about the state of the fuel, otherwise the facilities for opening should be designed to handle fuel with massive fuel failures.

During the intermediate storage unexpected effects may occur e.g. more rapid interaction of the cladding with water or from inside interaction of the cladding with fission products.[1]

3 The Problem of SNF in Bulgaria

At the signing of the contracts for the nuclear power reactors in Kozloduy the Soviet partners took the obligation for returning back of the generated SNF for the life of the plants. These obligations were strictly observed for nearly 10 y and all generated spent fuel assemblies 3086 were returned to the Soviet Union. In 1979 the Soviet side insisted on building of a AFR facility which was to be put in operation in 1985. The purpose of this intermediate storage facility was to give time for the construction of the reprocessing plant in Krasnoyarsk. In 1987 the Bulgarian application for SNF export was denied since there was no free storage pools and the construction of the reprocessing plant had not yet started.

The AFR facility in Kozloduy was put in operation with a 3*y* delay and the Soviet Union accepted SNF as an exception in 1988.

At the moment (Nov.1994) approximately 1800 fuel assemblies are stored in the AFR facility and some 1060 in the AR pools (~ 330 VVER-1000 FAs; capacity of the AR pools - ~ 610 FA for VVER-1000 + a place for the transport container TK-13, ~ 700 maximal for double tiering for VVER-440 + a place for the transport container TK-6).

The short description of the problem shows that at present Bulgaria does not formally belong to the group of countries with "deferred decision". Deferred decision means that the decision for reprocessing or final disposal will be taken after a certain period e.g. 30 - 50 *y*. At present the national policy

[1] Increased hydrogen absorption from water and interaction of the cladding with fission products - communication from a Workshop on Storage of SNF in Vienna, October 1994: *I. Kadarmetov et al. Evaluation of Maximum Alowable Temperature of VVER-1000 SF under Dry Storage Conditions*; *I. Ivashov et al. Corrosion Tests of Construction Material of SF Assembly of VVER-440* (in Russian).

is to export from the county of the SNF (in our case for reprocessing). No decision has been taken for building of facilities for intermediate storage for all SNF which is expected to be generated till the end of the plant life.

There are certain economic advantages of the policy of "deferred decision" for a part of the SNF. A very careful analysis has to be performed for the various options of the "deferred decision", most probably with dry storage technologies, which in the end will include reprocessing or final disposal. The decision concerns approximately 12000 fuel assemblies for a term of 40 - 50 *y*.

When the technologies are estimated besides the nuclear safety features some additional parameters have to be considered:

- compatibility of the possible technologies for transport to the reprocessing plants or final disposal preconditioning facilities;
- minimization of the operations for reloading, especially for reloading under water after intermediate dry storage.
- participation of Bulgarian companies.

The existence of the AFR facility in Kozloduy is an advantage which has its place in the different schemes for fuel storage. The facility can be used for intermediate storage, loading and reloading operations, buffer storage capacity, intermediate storage before dry storage - after 10 cooling period in the AFR pools the residual heat decrease more than two time compared to 3 *y* cooling period.

4 Conclusion

The AFR facility is with limited capacity, it is designed only for VVER-440 fuel although within an year it will be technically possible to transport and store VVER-1000 fuel. The AFR facility has also licensing problems due to increased safety requirements. Work is going on for improvement of the facility in order to comply with the new requirements but nevertheless its operation will only postpone the problem with 3 - 4 *y*.

There is an additional requirement by the Bulgarian NRC (Committee on the Use of Atomic Energy for Peaceful Purposes) which makes these terms shorter - there must be a sufficient place in the AR pools for removing the core from the reactor in case of emergency.

There is very little time for solving of the problem of SNF in Kozloduy.

The developed in the last 10 *y* technologies for modular dry storage with passive cooling have significant advantage - minimal maintenance, increased level of safety, prolonged investment periods, reduction of the generated heat in time which is considered as a an argument for possible storage term extension. Those advantages of the dry storage modular technologies can help for the solving of the problem of SNF in Kozloduy.

The problem of SNF faces all countries which operate NPPs but each county has developed its national policy - final disposal, reprocessing, deferring the decision. In the last case there are storage facilities for all the fuel which is expected to be generated and also there are allocated funds for the back end.

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Spent Fuel Disposal Problem in Bulgaria

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1 Spent Fuel as a Radioactive Waste

The basic law regulating the use of nuclear power for peaceful purposes in Bulgaria was set up in 1985. Inspite of its late application the legislation does not cover sufficiently the complex processes of radioactive waste management and disposal.

The Bulgarian Academy of Sciences was asked by the Government to develope "National Concept for Radioactive Wastes Management and Disposal in Republic Bulgaria" in order to help the national authorities to clear the national policy in the field of radioactive waste management and disposal, important part of which is the disposal of spent fuel (SF) or vitrified high level wastes (HLW).

Spent fuel, if declared as a waste, and the vitrified high level wastes from the reprocessing, belong to Waste Category I, according to the IAEA recommended classification [1] - radioactive wastes with high, beta/gamma - emission, significant content of alpha-emitters, high radiotoxicity and high heat output, which causes special problems during their long-term isolation from the human environment.

According to the Bulgarian legislation - Regulation № 7 of The Committee for Uses of Atomic Energy for Peaceful Purposes "Collection, Treatment, Storage, Transport and Disposal of Radioactive Wastes in the Territory of Republic Bulgaria" from 1992, spent nuclear fuel, if considered as a waste, and the vitrified high level wastes belong to solid waste Category III - wastes with dose rate at distance 0.1 *m* from the surface higher than 10 $\mu Sv/h$, specific beta activity higher then $3.7 \cdot 10^9$ *Bq/kg* and specific alpha activity higher then $3.7 \cdot 10^8$ *Bq/kg*. The Bulgarian legislation does not distinguish the radioactive wastes with regard to their disposal. The highly radioactive heat generating wastes are in the same category with radioactive wastes with quite different characteristics.

Within the development of "National Concept for Radioactive Wastes Management and Disposal in Republic Bulgaria" review of the disposal principles and the available disposal options was performed [2] and the type and quantities of the wastes, which would be disposed of are estimated [3, 4].

2 Basic Principles in SF/HLW Disposal

According to the internationally agreed safety principles and criteria [5, 6], the spent fuel and the vitrified high level wastes, in suitable waste form could be disposed of only in deep geological formations.

The overlying objectives of the underground disposal of spent fuel and/or vitrified high level wastes are:

- to isolate them from the human environment over long time scales without relying on future generations to maintain the integrity of the disposal system, or imposing upon them significant constraints due to the existence of the repository;
- to ensure the long-term radiological protection of humans and the environment in accordance with current internationally agreed radiation protection principles.

The long-term isolation of SF/HLW relies on the overall systems approach and the multibarrier concept. As SF/HLW presents a potential risk over very long period, the safety of the system does not rest on a single component or barrier, but rather on the combined performance of several different engineered and natural barriers. If barrier fails to perform as designed, the overall system should still be sufficient to meet the safety objectives.

The necessary degree of isolation is provided by different barriers:

- the engineered barrier immediately surrounding the wastes - waste form itself, container, overpack, buffer, backfill and sealing materials, the underground structures;
- natural barriers - host rock, over- and underlying geological formations and the biosphere.

The waste form itself, has important role to prevent the radionuclides of reaching the human environment. The type of the waste form, which would be disposed of in deep geological repository, depends on the spent fuel management strategy - reprocessing or direct disposal, which is selected after careful consideration of wide range different factors, including economical, political, etc. Both waste forms - vitrified high level wastes as backend of the closed cycle, and encapsulated fuel in the case of once-through cycle - are considered suitable for long-term isolation [7 - 10].

3 Disposal Options

The purpose of the deep geological disposal is to provide long-term containment and isolation of spent fuel and/or vitrified high level wastes from the human environment.

Four groups of geological formations are currently considered for deep disposal: salt rock (bedded or salt domes), crystalline rocks (granite, basalt, gneiss), argillaceous sediments and tuff.

Rock salt (salt domes or bedded salt) is potentially suitable as host rock for disposal of high level heat generating wastes because of its favorable geological, hydrological, geochemical, physical and thermal properties.

The rock salt is characterized by: extremely low water content: (bellow 1 *w.*% in salt domes and a few *w.*% in bedded salt), very low permeability, low hydraulic conductivity (from 10^{-10} to 10^{-12} *m/s*). Its advantage is the widespread occurrence as massive, undisturbed units, stable over geological periods. The rock salt poses high plasticity and high thermal conductivity. Advantages are the high stability of mined openings, easy mining and the experience in storage of oil and non-radioactive toxic wastes.

Several countries consider the possibility of using salt formation for disposal of high level heat generating wastes - Germany, Denmark, France, Netherlands, Spain and the former USSR and their studies especially on the disposal of HLW in Gorleben salt dome show that rock salt could provide safe long-term isolation.

The hard or crystalline rocks (granite, basalt, gneiss) are considered by several countries as primary candidate for HLW/SF disposal - Canada, France, Finland, Spain, India, Japan, Sweden, Switzerland, former USSR Generally hard rock they demonstrate: long-term stability, high rock strength, high stability of mined openings, high chemical stability, reducing geochemical environment; moderate thermal conductivity, low porosity and resistance to heating and radiation. Additional advantage is the widespread occurrence in some countries as massive and homogenious rock bodies of little or no economic value and the existence of mining experience. Suitable for SF/HLW disposal is intact rock with low hydraulic conductivity (10^{-9} - 10^{-11} *m/s*), low porosity away from fractured zone.

Disposal of SF/HLW in argillaceous sediments is a preferred disposal option for Belgium and Italy.

Suitable for the disposal of SF/HLW are homogeneous clay formations with low porosity, very low permeability, low hydraulic conductivity (10^{-11} - 10^{-12} *m/s*) and high plasticity. Additional advantages of the clay formations as host rock for SF/HLW disposal are high sorption capacity of the

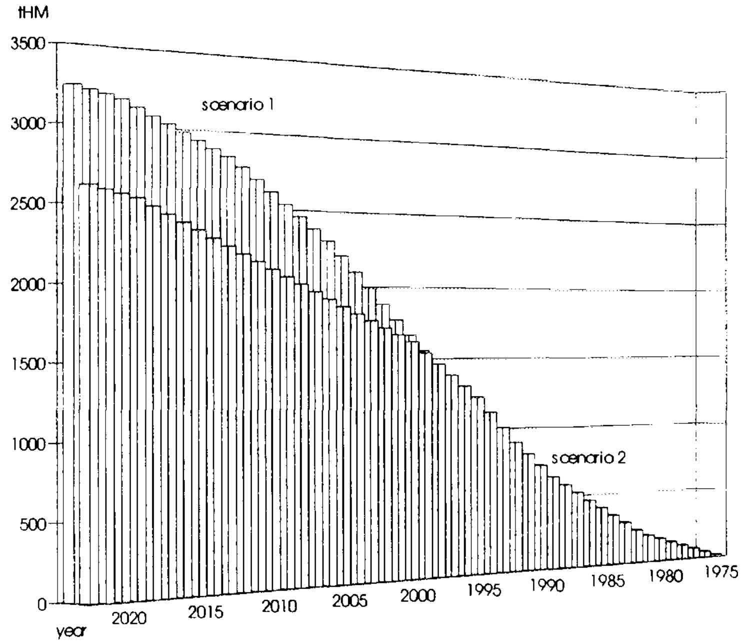

Figure 1 Projected cumulative amount of spent fuel in Bulgaria

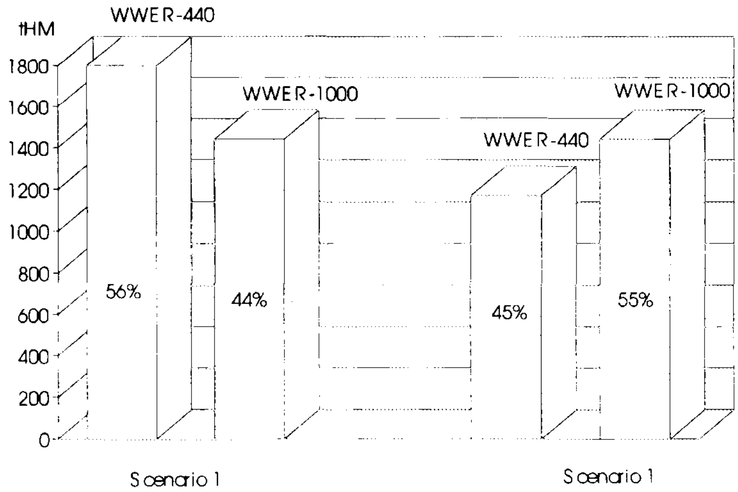

Figure 2 Quantity of spent fuel from NPP Kozloduy

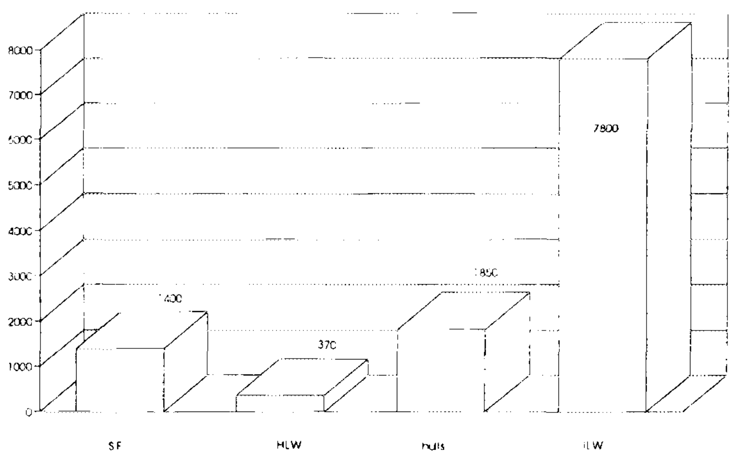

Figure 3 Radioactive waste production during the reprocessing of spent fuel

clay minerals and the low solubility. The relatively low thermal conductivity as compared with salt or granite requires long cooling periods (50 - 70 years) prior to transfer SF/HLW into the repository.

The only tuffaceous rock, which is currently being studied as host rock is at Yucca Mountain, Nevada. The disposal horizon is above the water table (about 300 *m*). The tuff is welded, devitrified, fractured (1 to 3 fractures per meter) with 10% porosity.

The fine grained sediments underlying the ocean floor are considered for disposal of SF/HLW, as they have low permeability, negligible pore water velocity and high sorption capacity, thus restricting the rate of migration of radionuclides from the waste to the seabed. Additional advantage is a dilution within the large volume of the ocean if any leakage out of the repository occurs.

4 Problem with the Disposal of Spent Fuel in Bulgaria

According to the agreement between Bulgaria and former USSR spent fuel was returned to the country of origin. With the change of the economical situation decision should be made about the type of the nuclear cycle. Bulgarian authorities have postponed the decision by building an away-from-reactor wet storage facility in Kozloduy NPP, which is capable to store the fuel for a 10 years period of operation of Units 1 to 4. That means that the disposal of spent fuel and the vitrified high level wastes has to be considered.

The possible alternatives for Bulgaria are to send the spent fuel for disposal in other counry, once-through cycle or closed fuel cycle. In the first case no disposal of the spent fuel should be considered. In the case of once-through cycle the direct disposal in deep geological formation should be considered. If decision for closed cycle is taken deep repository should be considered for vitrified high level wastes and for conditioned cladding hulls and low and intermediate level.

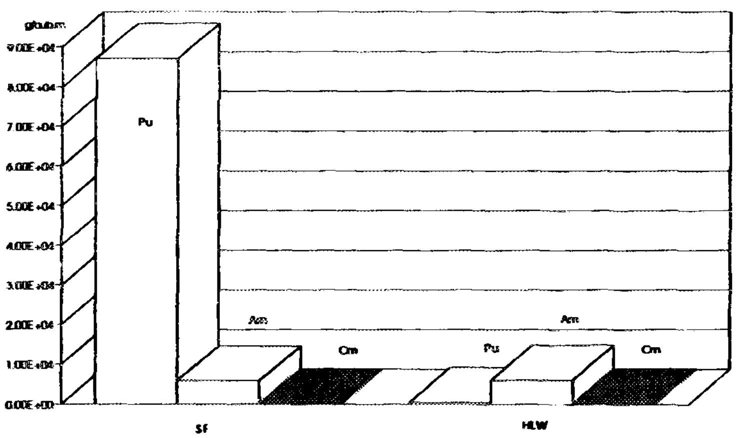

Figure 4 Concenration of Pu, Am and Cm in SF and HLW

The projected cumulative amount of spent fuel in Bulgaria, estimated within the development of "National Concept for Radioactive Wastes Management and Disposal in Republic Bulgaria", is given at Fig. 1 [4]. Scenario 1 represents the total amount which should be accumulated if the four VVER-440 and the two VVER-1000 units are operated till the end of their operational time. Scenario 2 represents the low estimate - if Unit 1 and 2 are out of operation at the end of 1995 year, Unit 3 and 4 at the end of 2000 year, and Unit 5 and 6 are operated till the end of their operational time.

At the first case total amount of 3 300 *t*HM will be accumulated. About 56% is the spent fuel from VVER-440 and 44% from the two VVER-1000 (Fig. 2). The total amount according to the second scenario is estimated to be about 2 700 *t*HM (45% from VVER-440 and 55% from VVER-1000).

Part of the spent fuel is returned to the country of origin. This means that the total amount of the spent fuel that will be disposed of in deep geological repository is about 2 800 *t*HM. The conservative approach considers the situation when all the spent fuel is disposed of in our country.

Within the development of "National Concept for Radioactive Wastes Management and Disposal in Republic Bulgaria" were estimated the types and quantities of wastes, which would be generated during the spent fuel reprocessing (Fig. 3). The total volume of the spent fuel is about 350 m^3. Its conditioning increases 4 times the volume.

The reprocessing of the total amount of 3 300 *t*HM leads to the production of about 370 m^3 vitrified high level wastes. Their radionuclide content differs from the radionuclide content of the original spent fuel. The main difference is that the vitrified HLW contains traces uranium and plutonium isotopes. When spent fuel is reprocessed with the currently used PUREX technology $\leq 0.5\%$ uranium and plutonium remainds in the vitrified HLW. The concentration of plutonium, americium and curium in the VVER-440 spent fuel and vitrified HLW is given at Fig. 4. In addition the volatile radionuclides are separated during cutting and dissolution of the spent fuel. From the point of view of long-term disposal special attention should be given on the separation and conditioning in appropriate waste form of iodine-129, krypton-85, carbon-14 and tritium.

Together with the high level wastes about 1850 m^3 cladding hulls and 7800 m^3 intermediate level wastes will be generated. The cladding hulls and the ILW contain long-lived alpha emitters and should be disposed of in deep geological repository. For some of the conditioned low level reprocessing waste shallow ground disposal with additional engineered barriers is possible.

The volume of the highly radioactive spent fuel is small when compared with the total radioactive waste production from the operation of all the nuclear reactors in Kozloduy NPP [3]. The estimated total radioactive wastes production, which will be accumulated till the end of the operational period of the six VVER units in Kozloduy NPP, including spent fuel is about 181 000 m^3 (Fig. 5):

- low and intermediate level evaporator bottom, spent ion exchange resins and HEPA-filters; conditioned, according to technology adopted in Kozloduy NPP - 96%;
- burnable and non-burnable solid operational wastes - 3.5%;
- high radioactive operational beta and gamma wastes - 0.2%;
- conditioned spent fuel - 0.8%.

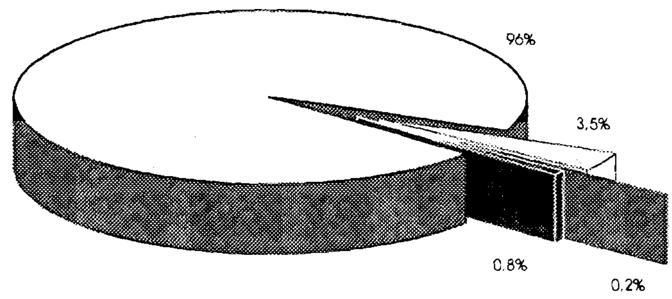

Figure 5 Total production of radioactive waste from Kozloduy NPP once-through cycle

The total nuclear wastes in production in a case of closed cycle is estimated to be about 190 000 m^3 (Fig. 6):

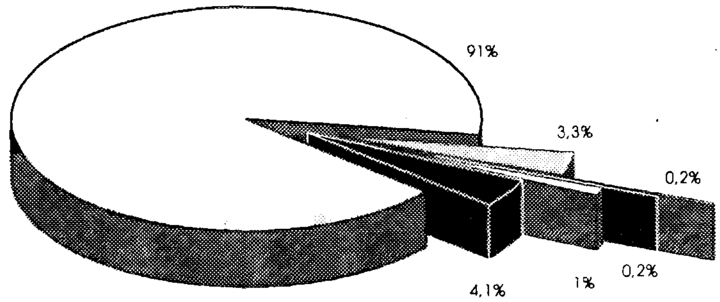

Figure 6 Total production of radioactive waste from Kozloduy NPP closed cycle

- low and intermediate level evaporator bottom, spent ion exchange resins and HEPA-filters; conditioned, according to the adopted in NPP Kozloduy technology - 91%;
- burnable and non-burnable solid operational wastes - 3.3%;
- high radioactive operational beta/gamma wastes - 0.2%;
- vitrified high level waste - 0.2%;
- cladding hulls - 1%;
- conditioned low and medium level long-lived wastes - 4.1%.

Within the development of "National Concept for Radioactive Wastes Management and Disposal in Republic Bulgaria" team of scientists from the Geological and Earth Science institutes of the Bulgatian Academy of Sciences under the coordination of Prof. Dr. Sc. Evstatiev and Prof. Dr. Kozhoukharov has categorized the territory of the country. The categorization is based on the existing detailed knowledge of the geological, geotechnical, hydrogeological, geographical, etc., conditions in the country.

As a result of this complex studies "Map of categorization of regions of the territory of Bulgaria" in scale of M 1:500 000 is drawn [11]. Three types of areas are (Fig. 7) are seen:

- unsuitable areas
- prospective regions for investigations
- regions offering favorable conditions for selection of suitable sites for additional investigations: a definite number of sites in these regions are selected.

The regions with favorable conditions are concentrated in the Northwest and Southeast part of the country.

The most promising areas in the Northwest Bulgaria are build from thick covers of well studied marls (more than 1000 *m* thick) some of which appear on the surface. They are practically dehydrated, not affected by tectonic movements in the last geologic ages and are situated in areas with seismic activity less than VIII according on the ten thousand years map.

Most promising in the Southeast Bulgaria are the Sakar granite pluton and some massifs built from gneisses, volcanic rocks and serpentinites with great thickness and enormous area of occurrence, but deep hydrological situation is not known.

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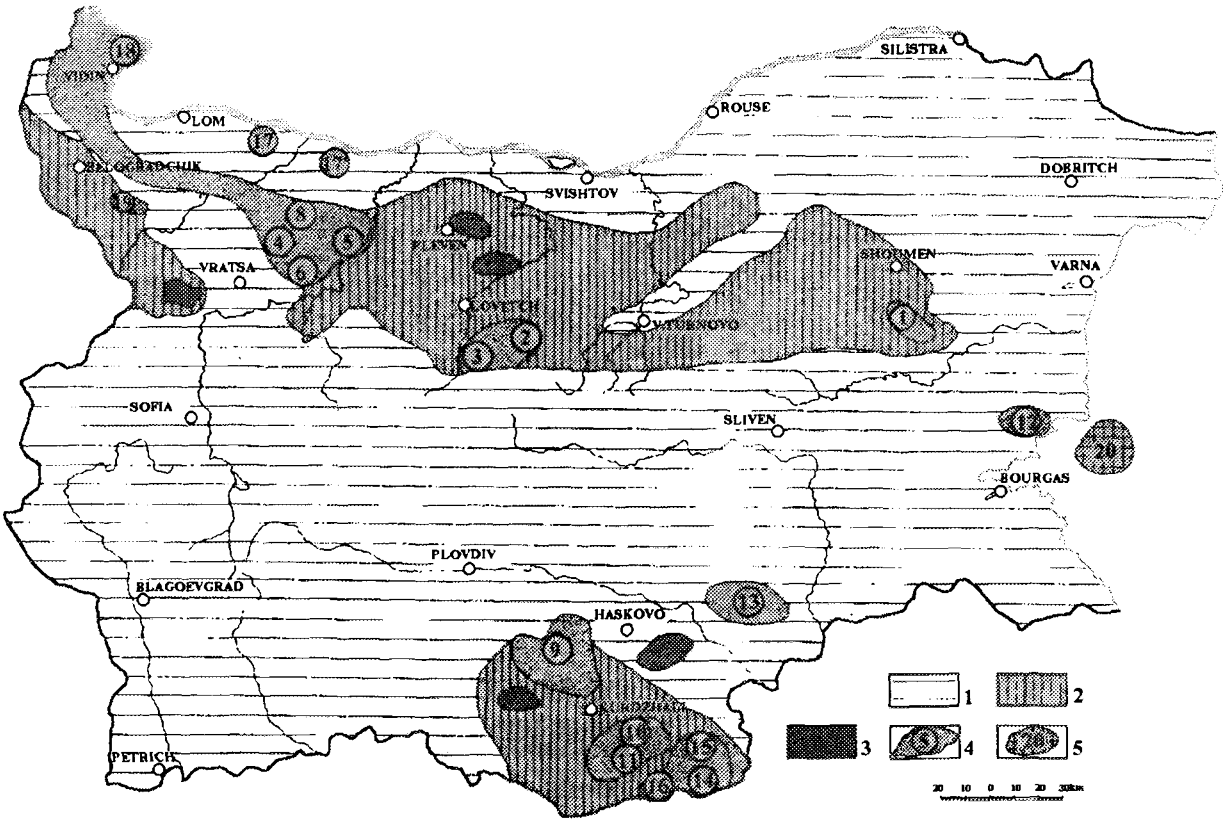

1 - Unsuitable area for RAW repositories; 2 - Prospective regions for investigation; 3 - Regions with suitable sites for additional investigations; 4 - A site and its number; 5 - A site in the Black sea

Figure 7 Map of categorization of regions on the territory of Bulgaria

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VVER Spent Fuel Storage

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1 Introduction

Each type of nuclear fuel is unique, initially by virtue of its physical size, material, and enrichment, and subsequently because of its operating history and burn-up.

If the fuel is to be stored after irradiation, then the storage system itself introduces its factors on to the fuel to add to the complexity. Such aspects as the fuel storage temperature and the geographical location all contribute.

All of these aspects need attention if the integrity of the fuel at the end of its storage is to be ensured, particularly as interim storage periods of 100 years may be required.

GEC ALSTHOM, has recently completed such an evaluation of VVER 440 fuel for the Paks NPP in Hungary. GEC ALSTHOM signed a contract with Paks NPP to provide a design and safety case for a Modular Vault Dry Store (MVDS) to accommodate 10 years of fuel arisings from their four -VVER 440 reactors. This amounted to a total of 5000 irradiated fuel assemblies (IFAs) which need to be stored.

This paper outlines the selection criteria used by Paks with specific reference to some of the more important technical, licensing and technology transfer issues.

2 Paks Dry Storage System Selection Criteria

Paks NPP having decided on a dry fuel storage system, had to determine which system was best suited to their situation. Selection is complicated by the fact that there are several dry storage concepts commercially available, with an even greater number of vendors.

To assist in the evaluation process Paks drew up a list of technical criteria which they considered to be important to ensure safety, reliability, flexibility and economy throughout the loading and storage period. Weightings were given to the more important criteria with rankings between 1 and 5.

Amongst the more heavily weighted criteria were:

- temperature of fuel in storage;
- sub-criticality assurance, (avoidance of criticality for fuel in the unirradiated condition without having to take credit for burn-up);
- assurance of decay heat removal;
- dose uptake to the operators and public;
- protection of the environment;
- volume of waste produced during operation and decommissioning;
- physical protection of stored irradiated fuel assemblies;
- IAEA safeguards assurance;
- storage system should not prejudice final disposal route;
- cost of construction and extent of technology transfer to Hungarian industry.

Some of the factors which affect the dry storage system's ability to satisfy the selection criteria are outlined below.

(a) Temperature of the Fuel in Storage

A major concern of Paks was that the storage system must not prejudice the final disposal route by permitting degradation of the spent fuel while it is stored. One of the most important factors is the temperature of the fuel whilst in storage. Virtually all of the mechanisms which cause degradation of the fuel are temperature dependent. There are published values giving maximum storage temperatures for different fuels.

For VVER fuel a recommended maximum value of 350°C in an inert atmosphere has been suggested. Paks adopted the policy that the lower the storage temperature for the fuel, the higher the resistance the stored fuel has to degradation during storage.

The MVDS offers the lowest storage temperature of any of the dry storage technologies.

(b) Sub-criticality Assurance

Some dry store systems take credit for fuel burn-up, in order to offer low capital costs for the facility. Such systems are open to scrutiny on account of the potential hazards associated with storing the wrong specification of fuel. A system which takes no credit for burn-up is inherently safer.

(c) Dose Uptake to the Operators and Public

Statutory requirements regarding permitted off-site radiation emission levels and dose uptake are likely to become more restrictive in the future. Any new on-site facility such as a dry store will contribute to the off-site dose and when the reactor was built, maybe 10 or 15 years ago, the allowable radiation emission levels were much higher. With the reduction in the allowable doses, the reactors contribution to the site boundary dose becomes an increasingly larger percentage of the allowable boundary dose. The result is that any new facility will require extremely efficient shielding.

In the UK where GEC ALSTHOM are building MVDSs for Scottish Nuclear at Torness and Hunterston, the MVDS contribution to the site

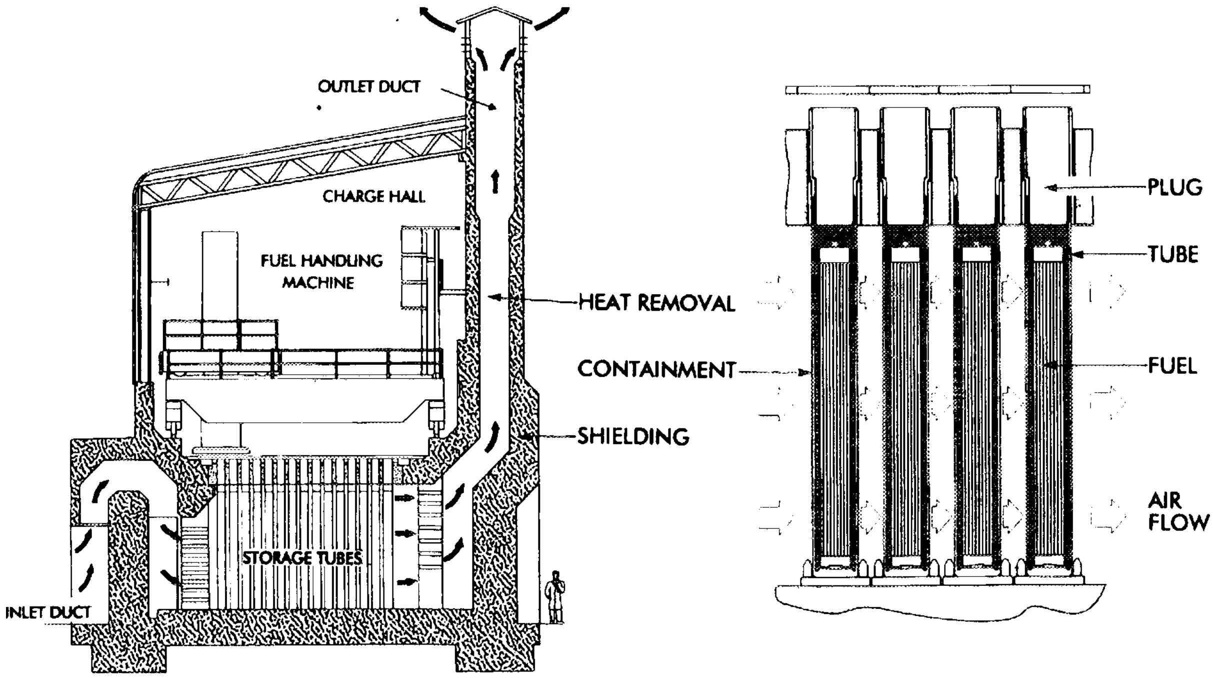

Figure 1 Modular vault dry store - cooling system

boundary dose is just 10% of the site allowable boundary dose. This is also the case at Paks.

It is axiomatic therefore that a system with very low radiation levels is required. Consideration must also be given to future extension and its contribution to the site boundary dose.

(d) Physical Protection

The facility must be able to withstand all the natural events, such as flood, very high winds, and earthquakes. Paks specified a seismic ground acceleration 0.35 *g*. Cognisance must also be given to national requirements, which might require such events as terrorist attack, aircraft crash, or explosive vapour clouds to be considered.

(e) Fuel Storage Attitude

VVER fuel assemblies are designed for producing thermal power inside a nuclear reactor. During that process the fuel assemblies are designed to be located and operated in a vertical attitude. When irradiated fuel assemblies are put into a dry store the attitude of the assembly over a long period of time can exert additional stresses and strains on the fuel pins. The condition of the fuel can be better justified if it is stored in the same attitude as it was designed to operate in the reactor.

Paks applied these criteria together with a number of others in their evaluation of the competing technologies, with a weighting against some of the criteria. As a result of this evaluation the GEC ALSTHOM MVDS system was selected. This does not mean that the GEC ALSTHOM MVDS scored the highest points in all categories, but against the Paks criteria, it was perceived as the system which most nearly satisfied their requirements.

3 The Paks Modular Vault Dry Store

3.1 Modular Vault Dry Store Concept

The MVDS is a passively cooled dry storage facility comprising three main areas: the Cask Reception Facility, used for the receipt of the transfer cask and transporter (and, if necessary, the drying of spent fuel if carried in wet casks); the Modular Storage Vault; and the Fuel Handling Machine. The three areas allow the MVDS to operate independently from the reactor facilities, even after these facilities have been decommissioned.

In the MVDS, spent fuel is contained in individual vertical sealed fuel storage tubes retained within a concrete vault, which can be constructed in a phased manner. The storage tube provides a high integrity mechanical barrier between the fuel and the cooling air. Cooling air is drawn by natural convection through a protective bird mesh and labyrinth, before entering the vault through concrete distribution louvres. The cooling air circulates around the storage tubes before discharging through the vault's exit louvres, up a concrete duct and out to the atmosphere after passing through a second bird mesh and a weather-protected duct top. The cooling air does not come into contact with the spent fuel and therefore does not become contaminated.

The concrete inner surfaces of the vaults are not exposed to contaminated air and remain clean structures throughout the life of the store. The cooling arrangement is illustrated in Figure 1.

The cooling system, which is passive, self-regulating and driven by the natural buoyancy of warm air, maintains a wide margin on acceptable fuel and structural concrete temperatures for all site specific environmental conditions.

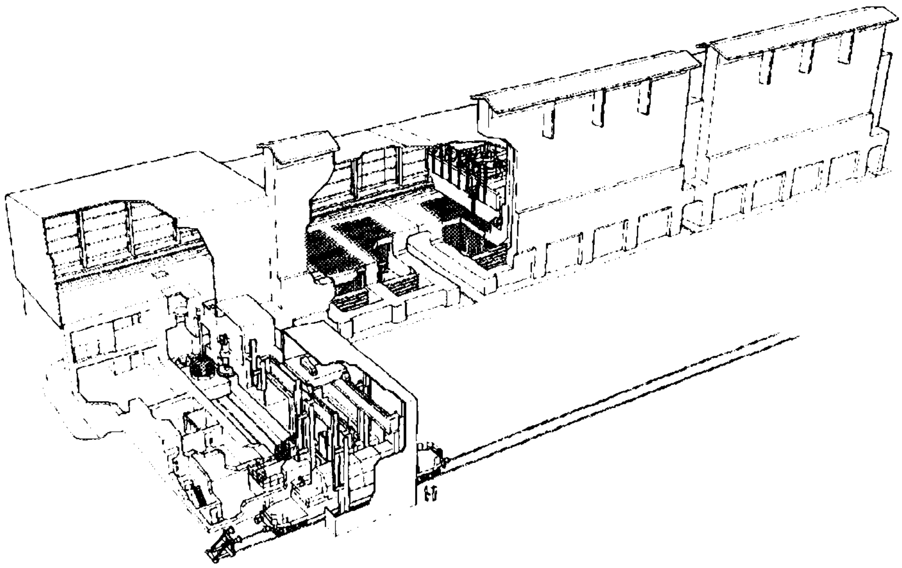

Figure 3 Paks NPP - Modular vault dry store

3.2 Description of the Paks MVDS

Paks has four - VVER 440 reactors and in each reactor are 312 fuel assemblies and 37 control assemblies. Annually about 120 spent fuel assemblies are removed from each reactor, giving a total arising of 480 fuel assemblies. In addition to the annual arising Paks wish to create some empty space within their storage pools and therefore require the on site dry store to receive spent fuel at twice the annual rate of arising (960 assemblies/year).

Paks are planning for an initial storage capacity of 4950 fuel assemblies equivalent to 10 years of operation of the reactors. The store will be built in 3 phases, the first phase will accommodate 1350 fuel assemblies (3 vaults) followed by two further phases of 1800 positions (4 vaults) each.

Provision has been made in the design for the store to be eventually extended to 14850 positions representing 30 years of reactor operation.

The Paks MVDS facility with three phases of construction is illustrated in Fig. 2. The VVER 440 fuel assembly is illustrated in Fig. 3.

The spent fuel arrives at the MVDS Transfer Cask Receipt Bay (TCRB) from the reactor building on a flat bed rail truck. The cask is the normal C30 cask used for off-site shipment so that no modifications are required to the reactor building storage pools. The cask is water filled and contains 30 assemblies. On arrival at the TCRB it is lifted off the rail truck by the overhead crane in the TCRB and lowered onto the Transfer Trolley, from where it is transferred into the inner area of the TCRB where the cask is off-gassed and the lid unbolted. On completion of this operation it is transferred onwards to the lid lift position and finally to the load/unload position, where the trolley is locked in place by means of a seismic lock.

The Fuel Handling Machine (FHM) is positioned over the cask at the load/unload port, where the load/unload port shield plug is removed into the

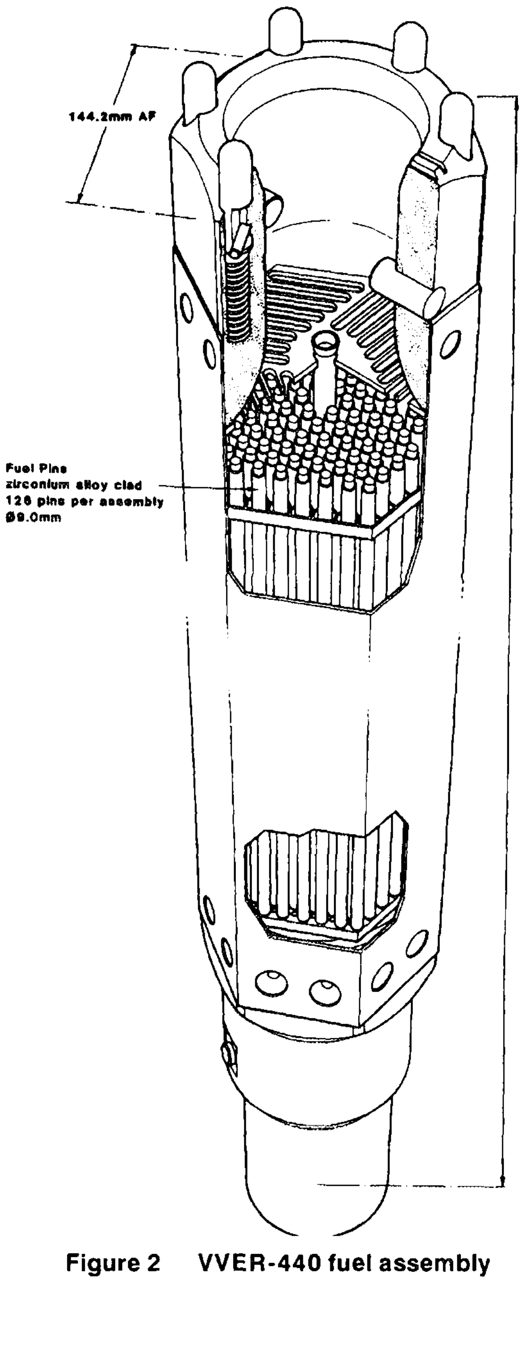

Figure 2 VVER-440 fuel assembly

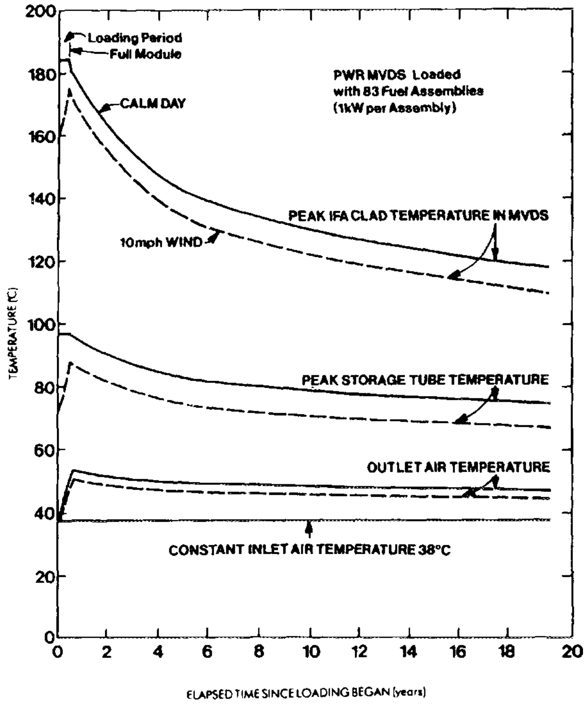

Figure 4 PWR MDVS - results of DADS assessment

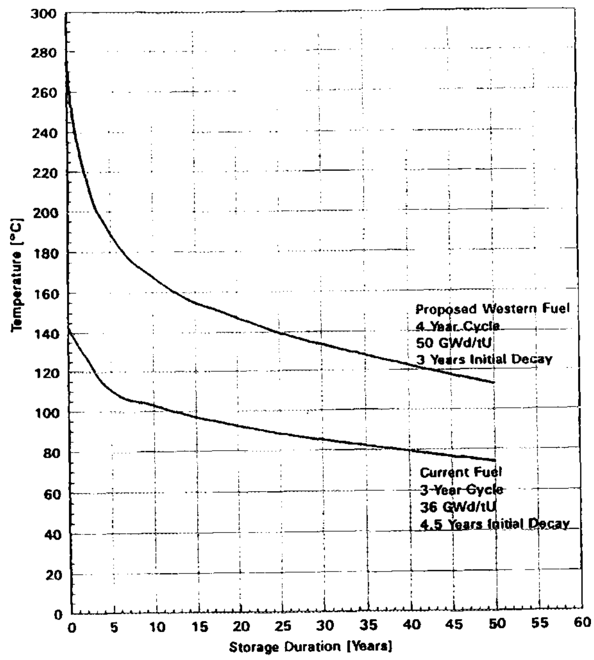

Figure 5 The peak local fuel clad temperature for maximum irradiated VVER-440 assembly during term dry storage in MVDS

FHM magazine. The FHM grab is lowered through the port to engage the fuel assembly in the cask and raise it into the drying tube. After dwelling in the drying tube for the requisite period with hot gas circulating around the fuel, it is raised into the FHM for transfer to the selected Fuel Storage Tube (FST).

At the FST the FHM removes the shield plug and lowers the fuel assembly into the tube. The plug is replaced and the FHM returns to the TCRB load/unload port to repeat the process. At the FST a service trolley is attached to the valve connection on the shield plug. The FST is drawn down to vacuum and back-filled with nitrogen. This operation is repeated a second time to ensure the required gas purity has been achieved. Finally the valve on the gas monitoring pipework to the FST is opened to allow the continuous monitoring of the integrity of the FST.

As described above a single VVER-440 fuel is stored in each storage tube. Because the storage tube radiates heat directly to the cooling airflow this gives an excellent heat removal capability, as shown in Fig. 4. The peak pin fuel clad temperatures, even for the most conservative case and the higher burn up fuels, are well below the 350°C temperature limit.

3.3 Safety Case

The MVDS system was licensed in America in 1988 by the US NRC for LWR fuel and this was accepted by Paks as the basis for their safety case, but with additional requirements to satisfy the various ministries and authorities within Hungary.

In the safety case the differences between PWR and VVER fuel and the site specifics of the Paks location had to be addressed. Some of the aspects addressed in the safety case are:

- Wind tunnel testing;
- Fuel temperatures;
- Criticality;
- Duct blockage;
- Fuel monitoring.

(a) Wind Tunnel Testing

As part of the assurance of cooling flow through the vault the effects of wind on the cooling flow have been investigated to ensure there is no cooling flow stagnation. This is not amenable to analysis and the only practical way it to model the store and the surrounding buildings and take measurements of pressure distribution across the store inside a wind tunnel.

The wind tunnel test where carried out with two models of the Paks MVDS having capacities of 4,950 storage positions and 14,850 positions respectively.

With the wind blowing from any direction, it must be demonstrated that a positive pressure differential always exists between the air inlet and the discharge.

In previous work GEC ALSTHOM investigated wind effects on the fuel storage temperatures. Figure 4 shows the effect of a modest breeze on the fuel temperature, the fuel storage tube temperature and the cooling air temperature. The MVDS design basis is the still-air condition.

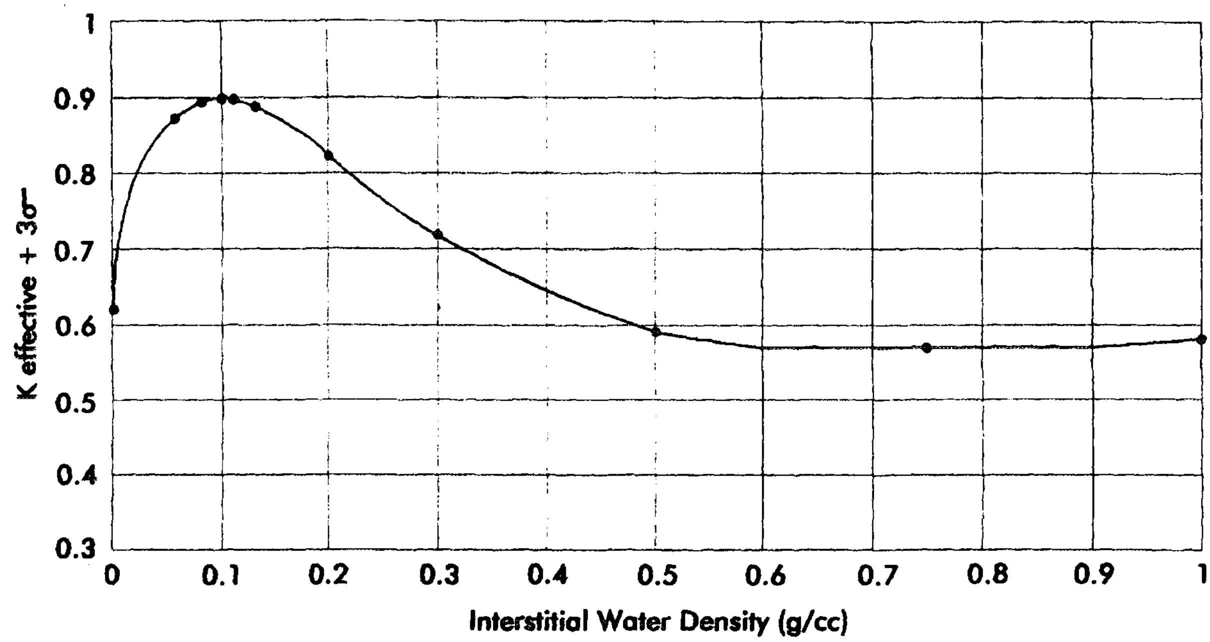

Figure 6 Variation in K effective with interstitial water density

(b) Fuel Storage Temperature

The graph shown in Figure 5 is taken from the Paks safety report and shows two normal cases for the VVER fuel.

Case 1 is based on the present operating cycle with fuel enrichment of 3.6%, 3 year cycle, 36 *GWd/tU* burn-up and 4.5 year initial decay before loading into the MVDS.The temperature shown of 140°C is the hottest point on the hottest pin.

Case 2 is the proposed future reactor operating cycle which is still based on fuel with 3.6% enrichment, but with a 4 year cycle, 50 *GWd/tU* and 3 years initial decay.

The peak temperature is 270°C, which is still well below the recommended peak temperature of 350°C for VVER fuel.

It is important when specifying the requirements of a storage system, that the future operating cycle of the reactor should be considered. As Figure 5 shows, it would take 25 years for the decay heat from the higher burn-up fuel to decrease to the level of the present operating cycle fuel.

(c) Criticality Analysis

Under US NRC regulation, when calculating criticality, all conceivable fault situations within the design basis must be considered. For the Paks safety case, this was extended to include two coincident faults. The first fault is the vault fully flooded, and the second fault is a fuel storage tube also flooded. Paks specified a sub-criticality value (K_{eff}) of < 0.95 for any condition, Figure 6 shows the results of the analysis.

It should be noted that no credit is taken for burn-up. The highest K_{eff} is with a vault flooded with a water density of 0.1. Even in this situation the K_{eff} is well below the Paks specified maximum.

(d) Duct Blockage

As part of the fault analysis, the effect of the cooling air duct blockage must be considered. This is an extremely unlikely event, as the inlet is 3 м high and runs the length of the store and an interconnecting passageway behind the inlet screen connects the vaults.

Figure 7 shows the effect of varying percentage blockage of the inlet duct, on the fuel pin temperature, the fuel storage tube temperature, and the outlet air temperature. It is not until blockage around 95% occurs that there is a significant temperature increase.

In fact if total blockage of the inlet duct did occur, then the condition shown on Figure 8 would be the result, thus maintaining a cooling airflow across the vault and allowing sufficient time for remedial action to be taken.

(e) Fuel Condition & Monitoring during Storage

The potential for consequential fuel degradation mechanisms in storage are increased if storage temperatures are high, moisture is present and there is an oxidising atmosphere. The low fuel temperature in storage achievable with the MVDS design will reduce if not eliminate completely, the potential corrosion mechanisms. Further loading operation for an MVDS storage tube requires evacuation of the enclosed air before back filling with nitrogen. This provides an inherent check on the dryness of each individual fuel assembly before storage giving additional protection against the presence of moisture.

Finally the ability to sample the gas within the storage tube enables the condition of the fuel throughout its storage lifetime to be monitored, giving confidence about the end of life condition.

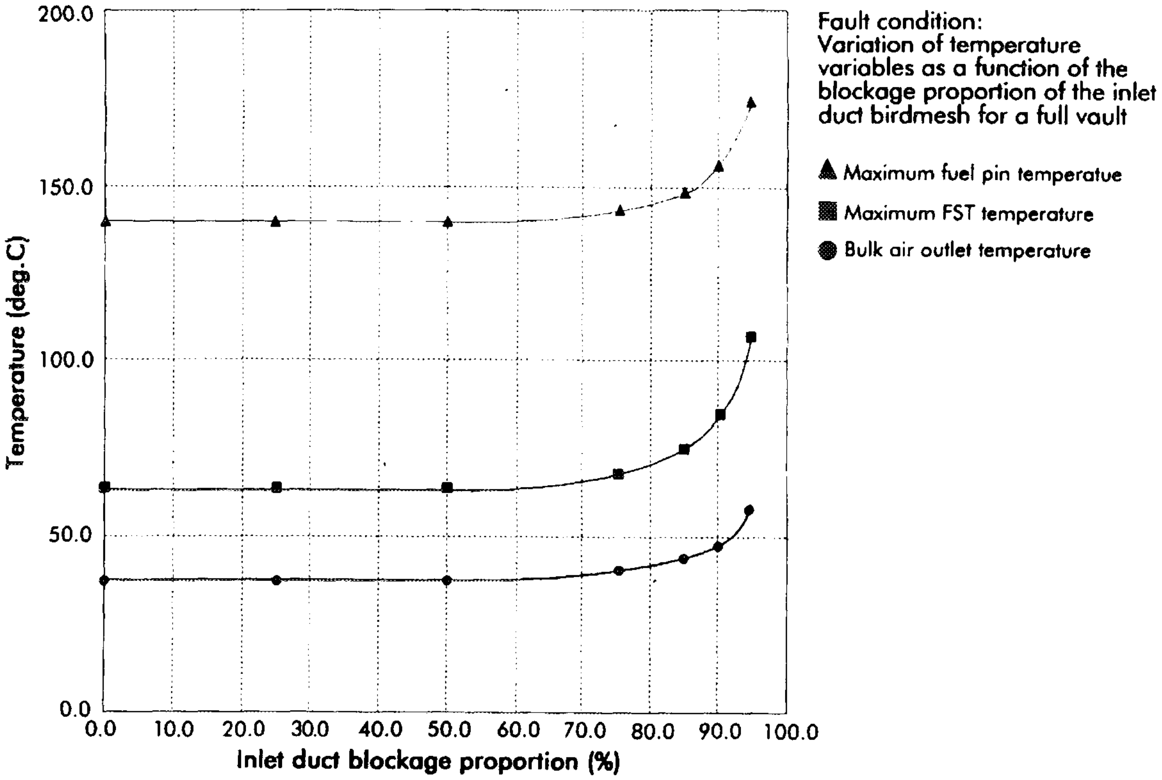

Figure 7 Effects of partial air blockage on fuel pin temperature

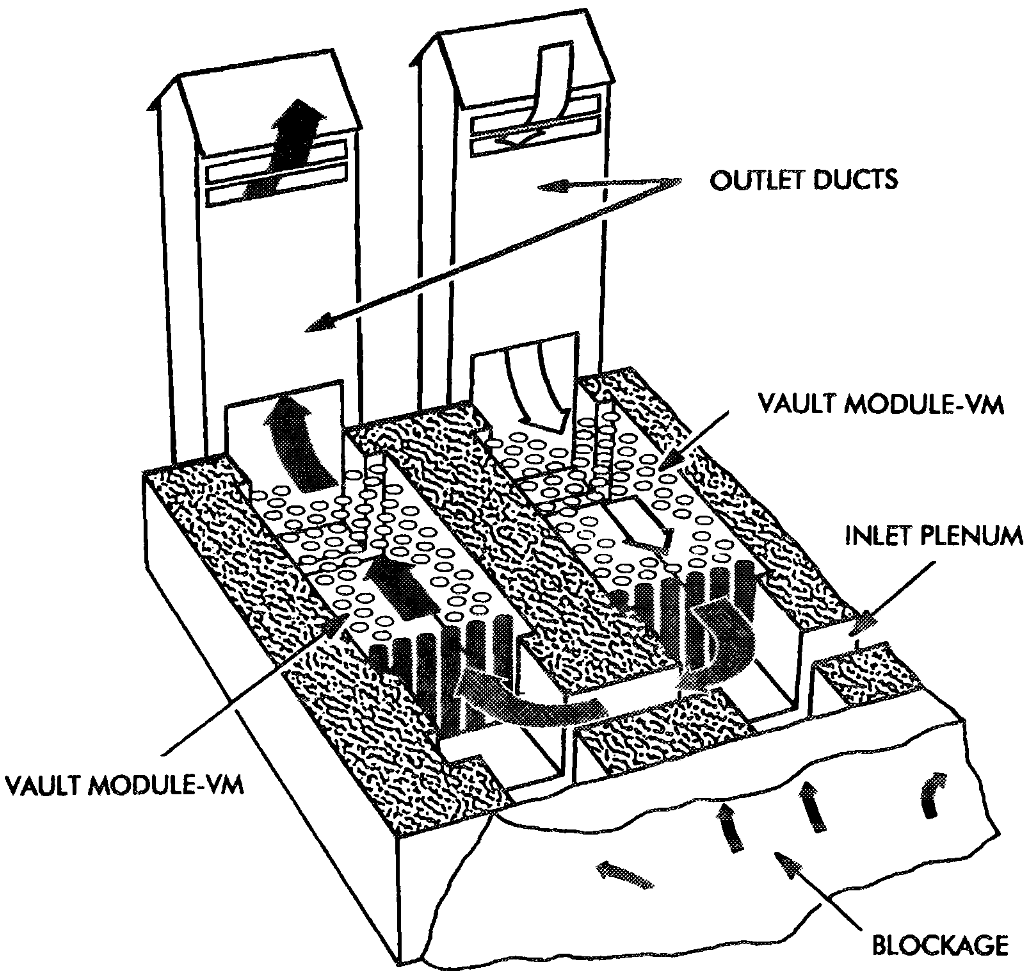

Figure 8 Cooling regime as result of duct blockage

Additional visual monitoring of the fuel cladding condition on entry to the store and subsequently during its storage, is provided by the Fuel Handling Machine. TV systems can be located in the machine to permit direct visual inspection of the fuel. The machine is fully shielded, dissipates the fuel assembly heat output naturally and allows direct manual recovery of defective fuel.

3.4 Licensing of Paks MVDS

GEC ALSTHOM proposed that the American Licensing and Safety Standards be adopted in Hungary for the licensing process of the MVDS. A non site specific design for the storage of LWR fuels had already been given US - NRC (Nuclear Regulatory Commission) licensing approval in 1988 and a site specific MVDS licensing procedure had been successfully completed in 1991 by the US Utility, Public Services of Colorado. This recommendation was accepted by Paks but with additional requirements imposed by various Hungarian Ministries and Institutions.

The safety and licensing standards which have been applied to the Paks MVDS are:

- American NRC Guide Lines - Licensing requirement for the independent storage of spent nuclear fuel - 10 CFR72;
- American Nuclear Standards - Design criteria for an independent spent fuel installation - ANS 57.9;
- Allowable doses in accordance with recommendations of the International Commission on Radiological Protection - ICRP 60;
- The standard format and content for the Safety Analysis Report for an independent spent fuel storage installation - US Reg. Guide 3.48;
- Additional requirements identified by the Hungarian Government departments of safety, health, ecology etc.

The completed PCSR was initially reviewed by a Paks appointed international review committee. This was followed by a Paks Jury made up of experts from the universities, the various ministries, design consultants and the institutions. The final stage is the submission of the documents to the Hungarian Licensing Authority. In parallel with this, the application for the site construction licence is being prepared.

3.5 Technology Transfer

The initial contract for the Paks MVDS was placed in September 1992 with GEC ALSTHOM Engineering Systems for the preparation of the Licensing Design, the Construction Design and the Pre-Construction Safety Report (PCSR), with support in presenting the Final Safety Case to the various Hungarian Committees. This was total technology transfer with Paks NPP operating as the Project Manager throughout and with the option to have all manufacture and construction work carried out by Hungarian industry.

In mid 1993 Paks took the decision to proceed with the implementation phase of the dry store project with a target completion in 1995. As the Project Manager Paks are now moving towards gaining approval of the PCSR, having selected the civil constructor and have commenced the procurement of long delivery items of equipment. GEC ALSTHOM's continuing role is to provide assistance and advice as required by Paks.

The Technology Transfer process has ensured the involvement of the Hungarian designers in the Paks MVDS detailed design. Manufacture and construction is being carried out by Hungarian Industry. Nonetheless, the procurement programme has provided GEC ALSTHOM with the opportunity to bid competitively against Hungarian Industry and successfully against Hungarian Industry for hardware supply. The manufacturing contract for the Fuel Handling Machine was placed with GEC ALSTHOM in 1993 for delivery in early 1995.

4 East European Perspective

Looking at Eastern Europe in general the selection of which spent fuel storage system is best for the reactor operator is complex and is invariably a compromise. No one system will fulfil all the requirements imposed by a utility and its licensing authority, so it is a question of establishing which system provides the optimum. The Paks approach, where a list of criteria was drawn up with a weighting against the more important parameters, is a very satisfactory method of sorting the increasing number of available dry storage technologies. To help clarify what is important, in addition to the priorities identified by Paks, some other key questions for consideration are discussed below.

a) Most Utilities are considering going to the higher burn-up fuels at some time in the future. The consequence of this is that the fuel will stay hotter longer and also emit an increased radiation level. The question to be asked here is whether the dry storage system will be able to accept the higher burn-up fuel or whether a considerably extended storage period in the fuel storage pool will be required? Will the intended system give adequate shielding to the higher radiation levels associated with higher burn-up fuels? The self-compensating, efficient, passive cooling system and the massive, low-cost concrete shield walls of an MVDS design provide considerable tolerance to higher burn-up fuels.

b) For the end of reactor life has consideration been given to the period of time that the fuel must remain in the storage pool prior to discharge into dry storage and the cost associated with this decay time? If the system can accept more recently decayed fuel,then significant savings can be made by the earlier withdrawal of the nuclear licence for the reactor buildings.

c) How self-contained is the system? If the storage period is 50 to 100 years, is the security and integrity of the system such that no matter what problem arises, the resolution of that problem is within the storage system control?
d) At the end of the storage period, if it were decided to directly dispose of the fuel to a deep repository, how easy would it be to remove the fuel from its present storage system and repack it into the final disposal container and how much would these transfer facilities cost?

5 Conclusion

Paks went through an extremely comprehensive evaluation of the available systems, selected the MVDS, and are making good progress towards licensing and construction of the system. They are now approaching their goal of a secure and safe route for the discharge of their fuel from the reactor pools and will have the next 50 years to consider its eventual disposal route.

Key Technical Issues Relating to the Safety of Spent Fuel Dry Storage in Vaults: CASCAD System

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1 Introduction

Proven short term industrial solutions today available to utilities for the management of their spent fuel arising, are:

- reprocessing which is intended to recover fissile materials for further use in new fuel and vitrify high level waste,
- interim long term storage which allows for a delay in the final decision.

The purpose of this paper is to discuss main technical issues relating to the safety of interim long term storage based on the technology of dry storage in vaults which has been developed in France for the operating CASCAD Facility at the Cadarache site.

2 Technical Issues Relating to the Safety of Interim Long Term Storage

The safety of the facility is defined as its ability in normal, incidental and accidental conditions, not to exceed the set thresholds for radiological effects on staff, population and environment which is achieved through ensuring:

- fuel decay heat removal,
- subcriticality control,
- radiological protection.

Issues relating to fuel retrieval are also of great importance when assessing the safety performance of such a facility. Maintaining spent fuel elements in their as received condition in order to enable post storage operations, is therefore an important safety issue.

Finally, as in any other nuclear facility, the design and operational experience feedback is also part of the safety assessment and should therefore be considered as a safety issue.

3 CASCAD System General Description

As specified in the introduction the discussion is based on the technology of dry storage in vaults as implemented in the CASCAD system. The purpose of this paragraph is to provide the reader with a general description of the system before addressing the safety issue.

The operating CASCAD Facility at the Cadarache site was commissioned in May 1990. Fuel is received in tight canisters which are transferred to storage pits in the vault where they are scheduled to be stored for up to 50 years. Canistering operations are performed in a cell of the reactor building. However, such cells are usually not available in standard commercial reactors, so that canistering operations are included in the CASCAD system as shown in the system synopsis (Fig. 1). The design of the cell for fuel unloading and canistering operations is based upon the design of the unloading cell of the TO Facility at the La Hague plant which has been commissioned in September 1986.

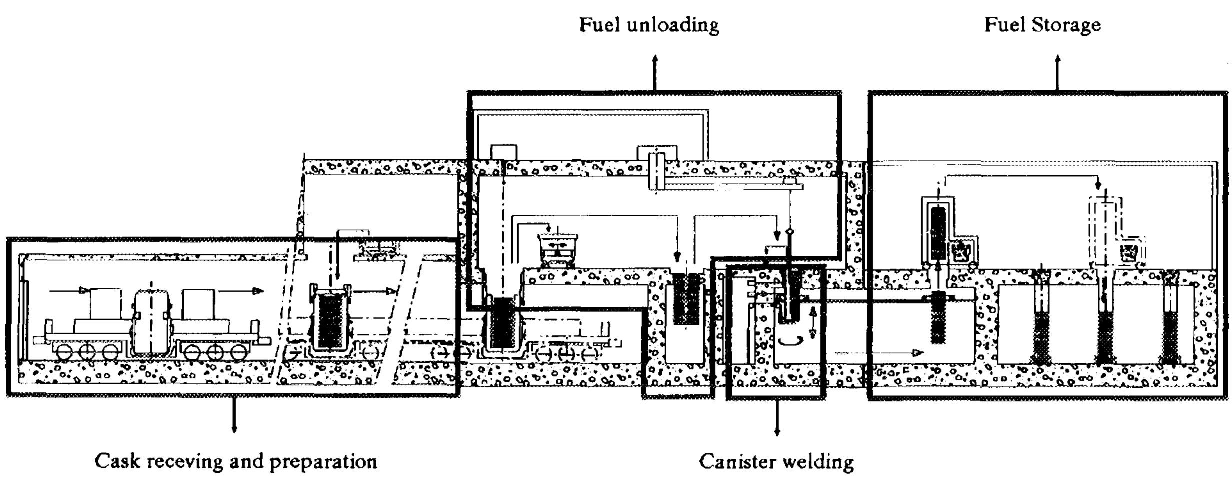

Figure 1 Operation diagram of the CASCAD system for the interim long term storage in vault

3.1 Cask Receipt and Shipping

The cask on its carrier, which may be any cask available on the site, is introduced into the facility where irradiation surveys and cask preparation for unloading are performed. once the cask is set; it is transferred to the fuel unloading unit. After fuel has been unloaded, the empty cask is returned to the preparation area and undergoes radiological survey before it is shipped back to the reactor or any other specific destination.

3.2 Fuel Unloading

Fuel is transferred from the transportation cask to the canister through a hot cell.

The cask and the canister are tightly connected to the cell as to ensure the continuity of containment and to prevent contamination of cask and canister outer surfaces. The design is so that it can be customized to adapt to different types of transportation cask.

The cask plug is connected to the cell plug and both plugs are simultaneously removed so as to prevent contamination of both plug surfaces which are in contact with the environment or areas where staff may be present.

Fuel handling in the cell is remotely performed by the mean of handling crane. Radiation shielding during unloading operations is ensured by the cell concrete structure.

3.3 Fuel Conditioning

Direct spent fuel handling only takes place in the cell where it is canistered so that fuel transfer to the storage vault is performed by handling a tight, contamination-free canister. The canister is welded and the tightness of the weld is checked before transfer to the vault.

Fuel conditioning operations carried out in the cell also cover fuel preparation for storage:

- fuel is dried in case of wet transportation,
- fuel dryness is checked and completed if not correct, canister internal cavity is filled with inert gas.

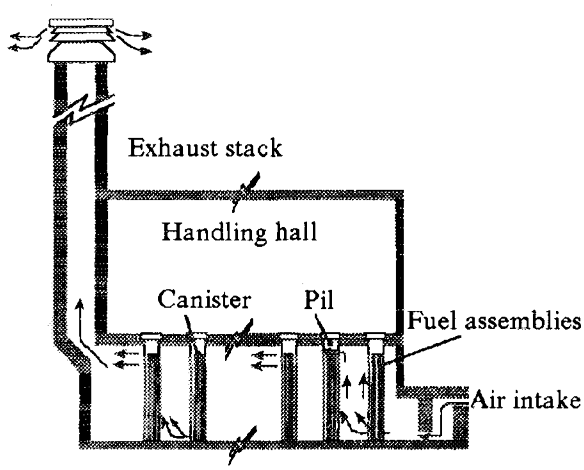

Figure 2 Principle of storage in vault with cooling by natural air convection

3.4 Canisters Emplacements in Storage Location

Canisters are transferred from the cell to the vault by mean of a trolley. In the vault, handling operations are performed using either a crane or a shielded transfer machine. Depending on the use of the crane or the shielded machine the hall may or may not be assessed during canister handling. When using a crane radiation shielding is ensured by the hall concrete structure while it is ensured by the machine structure when using a shielded transfer machine.

3.5 Fuel Storage

Storage of the fuel elements is entirely based on passive systems for maintaining the fuel in appropriate conditions. As a consequence the operations performed during storage are related to the monitoring of the storage conditions and the control of the confinement barriers.

3.6 Fuel Retrieving and Shipping at the End of the Storage Period

The cell allows operations for retrieval of stored fuel elements without requiring the existence of a reactor pool or any other facility.

Canisters are withdrawn from the pits and transferred to the cell using the same handling equipment as those used for loading. Reverse operations are performed to transfer furl elements from canisters to transportation cask which may or may not be the cask used for loading.

3.7 Operation

The operating of the system can be split in two phases:

- the vault loading and unloading campaigns which cover cask receipt, fuel element unloading and canistering, canisters setting in storage position and fuel retrieval,
- the storage period which include storage monitoring.

Operation organization is based on:

- during the storage period: the permanent attendance of a limited operating team and of technical support personnel (maintenance...). Operating information related to the storage system can be carried over to an another nearby facility so that there is no need for permanent attendance of personnel at the storage facility itself,
- during the loading and unloading campaigns: the occasional attendance of a more important operating team.

4 Decay Heat Removal

Decay heat removal has two main purposes:

- maintaining rod cladding temperature below a set limit in order to maintain the fuel in its as received condition,

- maintaining structures and equipment performing a safety function below the design temperature.

During fuel handling for loading and retrieving operations, decay heat removal is performed by the ventilation system of the buildings which is designed considering appropriate normal, incidental and accidental conditions.

The decay heat removal process in the storage vault is entirely passive and therefore inherently safe. The inner cavity of the vault is connected on one hand to an air vent for intake and on the other hand to a stack for exhaust. This induces natural convection of air and consequently air flowing through the vault when fuel loading starts (Fig. 2).

Structures important for the cooling performances are designed taking into account nominal storage conditions as well as conditions related to loading and unloading phases or meteorological abnormal and exceptional conditions.

Setting Limits for Rod Cladding Temperature

The purpose by limiting the fuel rod temperature is to ensure that the rod is maintained in its as received condition over the storage period, by preventing mechanisms that could damage the fuel.

IAEA reports (see Ref. [1]) recommended for LWR fuel a temperature limit of 350°C for fuel in inert gas.

Recent studies and tests have shown that the allowable temperature depends on the fuel characteristics and cooling time and can be calculated for a given storage period. Such calculations are used to verify if the limit of 350°C is indeed conservative for VVER fuel to be stored and to estimate the safety margin in the design by taking into account this limit.

5 Subcriticality Control

The features for sub-criticality control in the storage vault are such that sub-criticality in normal and accidental conditions is provided by the arrangement of pits in the vault (pit pitch). The integrity of the vault is ensured by the concrete structure that is designed to withstand extreme environmental conditions.

No credit is taken for burnup in the calculations.

As the cooling flow through the vault is derived from atmospheric air, moisture levels in the vault associated with various atmospheric conditions can not be discarded, so that water density considered for defining configuration studies covers the full range of densities from 0.0 to 1.0 g/cm^3.

6 Radiological Protection

Radiological protection is based on limiting collective and individual yearly dose equivalent to ALARA (As Low As Reasonably Achievable) levels and ensuring that they remain in any case below the set limits.

Exposure should be understood as exposure due to both contamination and irradiation. So that radiological protection consists in:

- confinement of radioactive material for protection against its dissemination,
- radiation shielding for protection against irradiation.

In addition a zoning of the facility according to contamination and irradiation risk defines access control to the rooms and personnel individual and collective exposure are monitored.

6.1 Protection Against Dissemination of Radioactive Materials

In application of the defense in death principle, the safety of the storage depends upon maintaining multiple barriers to prevent the escape of radioactive materials to the environment. The consensus on the implementation of the successive lines of defense is so the two confinement barriers satisfy the safety objectives.

Confinement by Multiple Barriers

The fuel matrix and cladding are often considered as a confinement barrier in dry storage system. However, as in general fuel cladding is not checked for potential damage after storage in the pool two barriers in addition to rod cladding seem necessary to comply with the objective set above. Those barriers are:

- the fuel cell structure and ventilation for the unloading and retrieving phases,
- the canister and the pit for the storage period.

The canister and the pit tightness are periodically controlled. The control allows for detection of an eventual failure of in the confinement and the identification of the defective barrier so as to take appropriate steps.

6.2 Protection Against Irradiation

The vault structure (walls and upper slab) ensures both protection of staff and public against irradiation risk due to the fuel elements. According to the ALARA principle radiation shielding is designed taking into account the type and duration of the operations to be performed as well as team organization.

7 Design and Operational Experience Feedback

The technical solutions here presented are based on or adapted from proven technologies used in operating facilities in France or abroad. The design not only benefits from the experience of SGN in the design, construction and start-up of facilities for fuel or high level waste handling and storage, but also from the experience of the CEA and COGEMA groups in operating such facilities.

Main reference facilities are:

- TO facility at the La Hague plant for spent fuel dry unloading,
- CASCAD Facility at Cadarache for storage of spent fuel in vault cooled by natural convection of air,
- R7 and T7 Facilities at the La Hague plant for high level waste handling and storage in air cooled vault,
- AVM Facility at Marcoule for high level waste handling and storage in air cooled vault.

Reference

[1] Final Report of IAEA Co-ordinated Research Programme on Behavior of Spent Fuel and Storage Facility Components During Long Term Storage (BEFAST II), IAEA-TECDOC-673, IAEA, Vienna (1992).

Conversion of Highly Enriched Uranium in Thorium-232 Based Oxide Fuel for Light Water Reactors: MOX-T Fuel

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Introduction

A problem which has emerged in the recent years is the conversion of the existing stockpiles of weapon-grade highly enriched uranium (HEU). According to the estimations the 50 000 warheads in the United States and the Russian arsenals contain some 1000 *t* of HEU and 220 *t* of plutonium [1] . The most straightforward solution for the HEU is blending it with natural uranium. The argument in favor of the blending is that the process is irreversible and such a technology helps the observation of the Non-Proliferation Treaty. An argument against this solution is that a lot of energy has been used for the enrichment - the ratio of the separation work units for HEU and uranium with ~ 4% enrichment is roughly 60 [2] and mixing the enriched uranium with ^{233}U has to be considered as the last option. The enrichment should result in improvement of the quality of the existing nuclear fuel, creation of new types of fuel or improvement of the fuel cycle.

Another problem of the contemporary nuclear fuel cycle is the inevitable generation of Pu and other higher long living actinides. The long-living actinides need to be conditioned as high level radioactive waste (HLRW) in case of fuel reprocessing and the separated plutonium reused in mixed oxide (MOX) fuel, or it has to be safely stored. For the rest of the actinides - americium and the even isotopes of plutonium extensive research programs exist for transmutation (fissioning) with fast neutrons. In the case of final disposal of spent nuclear fuel the actinides are the main source of heat generation after 150 *y* of cooling time and the main concern for estimation of the repository status after thousands of years.

The aim of the present paper is to discuss the feasibility of an idea for conversion of HEU without mixing with natural uranium and the utilization of Pu in a fuel in which no new plutonium is generated.

Fuel from Conversion of HEU and Pu Utilization: Principles

The proposition is to use ^{235}U (HEU) for a fissile isotope and ^{232}Th as a non-fissile isotope in a mixed oxide with thorium fuel for light water reactors. For most of the LWRs the percentage of the fissile isotopes is approximately 4% in the form of dioxide and for the rest 96% ThO_2 is proposed. Plutonium can also be used in the proposed fuel as a mixture with ^{235}U [3].

The main advantage is that weapon-grade HEU can be utilized by reversible blending with Th, a mono-isotope element which is at least as abundant as uranium. The possibility for reverse extraction of ^{235}U and the generated ^{233}U is also a weak point of the proposed fuel because of safeguard reasons.

The second advantage of the proposed thorium based fuel is that no ^{239}Pu and other long living actinides are generated. If the enrichment of the HEU is 93%, the presence of only 0.35% ^{238}U for a fuel of approximately 4.5% enrichment means that the generated ^{239}Pu and other higher actinides are expected to be 400 - 500 less than in conventional fuel. That is a great advantage in case of reprocessing of the fuel or final disposal. After 10 *y* of cooling in intermediate storage facilities the residual heat will be generated mainly by ^{137}Cs, ^{90}Sr and some 10 - 15 g/t_{HM} of transuranium actinides instead of 5 - 8 kg/t_{HM} of actinides.

The third advantage is that the generated ^{233}U after neutron capture by ^{232}Th is a very good fissile isotope which will be generated at 3 times higher rate than ^{239}Pu from ^{238}U. The high rate of production of ^{233}U and the high value of secondary neutrons per one absorbed neutron of ^{233}U can make the fuel cycle with very high fuel conversion coefficient or even with breeding of fuel under special conditions.

The fourth advantage is that in case of fuel reprocessing all fissile material (^{233}U + ^{235}U) can be chemically extracted and no ^{235}U will be lost.

Up to now the thorium cycle has been considered only for breeding or obtaining a fuel cycle with fuel conversion coefficient greater than 1. Most of the research has been done for a thorium cycle with fast breeders or crossed cycles - breeders reactors with thermal neutrons. The proposed fuel cycle is expected to be with fuel conversion coefficient less than 1 but with much higher coefficient than that of LWRs with ^{235}U + ^{238}U fuel. The additional energy generation will result in cheaper nuclear energy.

Estimation of Expected Properties of the Mixed Fuel

There is enough experience in the utilization of ^{232}Th and generation of ^{233}U. The first experimental data have been obtained in the BWR Indian Point. The experiments showed that except in breeders high conversion ratio or breeding can be achieved in thermal reactors with low capture cross-section in the moderator and coolant - e.g. HTGR, AVR, CANDU. The fuel elements of HTGRs contain two types of micro spheres - the first type is either ^{233}U or HEU ^{235}U in the form of UC_2 covered with carbon and silicon carbide and the second type is made of thorium dioxide covered with carbon [4].

The cross-sections for neutron capture and scattering of thermal neutrons of ^{232}Th and ^{238}U are comparable but since the capture cross-section of ^{232}Th is ~ 3 times greater than the capture cross-section of ^{238}U, the rate of generation of the fissile isotope ^{233}U will be 3 times greater than the generation of ^{239}Pu in conventional fuel.

The generation chain of ^{233}U-233 is very similar to that of ^{239}Pu: see Fig. 1(a). The 27 *d* isotope ^{233}Pa has a relatively high capture cross-section for thermal neutrons (21 *barn*, see [5]), or a small fraction of neutrons will be lost until ^{233}Pa decays to ^{233}U. In terms of reactivity the negative contribution from ^{233}Pa will be compensated by the lower cumulative yield of ^{135}Xe. The ratio of the cumulative yield of Xe-135 from ^{235}U and ^{233}U is 1.41 [6].

The values for the cross-sections for capture and fission with thermal neutrons of ^{233}U, ^{235}U and ^{239}Pu are very close or no great changes in the concentration of the fissile isotopes are necessary.

The relatively low value of the capture cross-section of ^{233}U increases the effective value of the number of secondary neutrons (4) per one absorbed neutron (2.28 [4]). That high value also increases the possibility for fuel breeding.

The approximate calculations for a homogenized core of VVER-1000 type for a 4.4% ^{235}U + Th show that the burnup can be increased by approximately 1/3 for the usual reloading schemes because of generation and burning of ^{233}U. The additional positive reactivity is generated during the operation and there is no need for compensation of the excess reactivity in the "fresh" core. For a burnup of ~ 40 *MWd/tU* the initial enrichment of U can be reduced by ~ 1%, or the ^{235}U+Th fuel reduces the price of the nuclear fuel.

The value of the fraction of delayed neutrons of the generated from thorium ^{233}U is greater but close to that of ^{239}Pu, or fuel of only Th + ^{233}U requires a modification of the control system. The presently assumed ratios of ^{235}U and ^{239}Pu in the MOX fuel are acceptable for the proposed thorium fuel with fissile isotopes ^{235}U and ^{233}U.

The generated ^{233}U and the residual ^{235}U can be chemically separated from the rest of the thorium fuel. Except fission fragments, the long living isotope ^{232}U ($T_{1/2}$ =72 *y*) is generated by (n, 2n) reaction: see Fig. 1(b).

^{232}U and its daughters are radioactive but the decay periods of the daughters are shorter. The daughter isotope with the longest decay period is ^{228}Th (1.9 *y*).

Technological Compatibility of the Thorium Based Fuel

The most important technological difference of the thorium dioxide compared to uranium dioxide is the specific density [7]. The obtained density of UO_2 is less than the theoretical one by 0.4 - 0.6 *g/cm³* and that will be the case most probably with ThO_2 too. The lower density means that approximately 10% less fuel will be loaded in the reactor core. The higher melting point of ThO_2 is an advantage of the proposed fuel.

Except as a mixture of UO_2 and ThO_2 the U-Th fuel can be made of physically separated substances - e.g. UO_2 coating over ThO_2 cylinder.

Conclusion

The advantages of the use of HEU in LWRs in mixed ^{235}U - Th fuel are:

- no generation of long living plutonium and americium isotopes. In the case of reprocessing the HLRW will contain only fission fragments and U, in the case of final disposal of the spent nuclear fuel the repositories should be designed for shorter periods (e.g. 500 *y*);
- the high conversion ratio of Th extends the expected burnup (by approx. 1/3) without higher initial enrichment. The same initial enrichment simplifies the problem for compensation of the excess reactivity in the beginning with burnable poison and boric acid;
- the high conversion ratio of Th makes possible the utilization of fuel with less initial enrichment (by approx. 1/3) for the same burnup. Thus less excess reactivity has to be compensated after reloading. -- in case of fuel reprocessing all fissile material (^{235}U+^{233}U) can be chemically extracted.

The proposed fuel or the Mixed OXide with Thorium - MOX-T fuel, actually extends the limits of depletion of fissile isotopes since the use of Th can slow down the rate of burning of uranium. The total amount of 1000 *t* HEU can supply fuel for

$$^{232}\mathrm{Th}+\mathrm{n}\rightarrow{}^{233}\mathrm{Th}\xrightarrow{22.1\mathrm{min}}{}^{233}\mathrm{Pa}\xrightarrow{27\mathrm{d}}{}^{233}\mathrm{U}$$

$$^{233}\mathrm{Pa}+\mathrm{n}\rightarrow{}^{234}\mathrm{Pa}\xrightarrow{1.2\mathrm{min}}{}^{234}\mathrm{U}$$

(a)

$$^{232}\mathrm{Th}\,(\mathrm{n},2\mathrm{n})\ {}^{231}\mathrm{Th}\rightarrow{}^{231}\mathrm{Pa}+\mathrm{n}\rightarrow{}^{232}\mathrm{Pa}\rightarrow{}^{232}\mathrm{U}$$

(b)

Figure 1 Generation chains of ^{233}U (a) and ^{232}U (b).

3.5 - 4 years for the reactors all over the world. The projected rate of dismantling is 2000 USA warheads annually and 1500 - 2000 Russian warheads, or 30 to 40 *t* of HEU are expected to be available per year. Depending on the type of the reactor, 50 to 80 reactors can be refueled with uranium from HEU conversion.

The proposed fuel can utilize also weapon-grade plutonium and plutonium from reprocessed spent nuclear fuel in a U + Pu + Th composition without generation of new plutonium.

If large quantities of HEU from nuclear warheads become available that will be a very unique possibility to start a large scale fuel conversion of Th within the clean fuel cycle with U in light water reactors and to improve the economics of the nuclear energy.

The further work on the problem includes calculation of optimal loading and reloading schemes, theoretical calculation of the thermal properties of fuel pins with ^{235}U+Th fuel and which is the most important and the most difficult problem - manufacturing of several test fuel assemblies and observation of their behavior in a reactor core.

References

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[4] G.Kessler, Nuclear Fission Reactors. Potential Role and Risks of Converters and Breeders, Springer-Verlag, 1983. (Russian edition, Moskva, Energoatomzidat, 1986)

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[7] B.R.T.Frost, Nuclear fuel elements, Pergamon Press, Oxford, 1982 (Russian edition - Moskva, Energoatomizdat, 1986)

Supplement

VVER Fuel Literature Review

5. Pool-Side Inspection of Spent Fuel

5.6.

Pekka Losonen, Kari Ranta-Puska. Pool Side Inspections and Performance Evaluation of VVER Fuel Rods. ANS LWR Topical Meeting, West Palm Beach, Florida, April 1994, p.113-118.

The results of non-destructive examinations of 24 VVER fuel rods irradiated in Loviisa-2 reactor are presented. The fuel rods are precharacterised and operated for three or four cycles to burnup up to 49 *MWd/kgU*. The pool side examinations, carried out in the spent fuel storage, involve visual inspections, rod length and gap size measurements, clad outside profilometry, etc.

VVER Fuel Literature Review

Collection of abstracts of VVER fuel data at manufacturing, reactor operation, experiment, post-irradiation examination and pool-side inspection

S. Stefanova[1], *M. Manolova*[1], *M. Vitkova*[1], *V. Simeonova*[1], *G. Passage*[1], *P. Chantoin*[2]

[1] Bulgarian Academy of Sciences, Institute for Nuclear Research and Nuclear Energy, Sofia, Bulgaria
[2] IAEA, Vienna, Austria

The abstracts below are included in 7 sections, according to their main topics, as follows:

1. *Fabrication Data and Other Fuel Characteristics*
2. *MR, MIR and Other VVER Experiments*
3. *VVER Fuel Post-Irradiation Examination*
4. *Fuel Failures*
5. *Pool-Side Inspection of Spent Fuel*
6. *Fuel Improvement*
7. *Other Papers*

Cross-reference is provided in each section to papers, included in other sections, but containing also information on the current topic (if any).

1 Fabrication Data and Other Fuel Characteristics

1.1. Scheglov A. S. et al. Thermophysical characteristic of VVER-1000 FR at Unit 5 of Novovoronezh NPP. *Atom. Energ.* **74,** p. 450 **(1993)**.

For thermophysical calculation a FR with maximal burnup 49.4 *MWd/kgU* from FA, irradiated during 3 fuel cycles (from 24.06.1984 to 25.06.1987). The basic characteristics with fabrication deviations are following:

inlet coolant temperature, °C	271-289
difference between outlet and inlet coolant temperature, °C	16-38
cladding diameter, *mm*	
inner	7.72-7.8
outer	9.12
pellet diameter, *mm*	
inner	2.2-2.4
outer	7.51-7.56
fuel density, g/cm^3	10.4-10.8
fuel stack length, *mm*	3524
fuel mass in FR, *g*	1460-1465
free volume under the cladding, cm^3	39.9
initial He pressure at 20°C, *MPa*	2.0
oxide film thickness on the FR outer surface, μm	5

The thermophysical characteristics of this FR, calculated on the basis of the average technological FR parameters, using the PIN-mod1 (improved version of PIN-04M) and RET(TR) codes, are compared and discussed.

1.2. Scheglov A. S., Proselkov V. N., Enin A. A., Statistical processing of the VVER-1000 FR constructional and technological parameters. *Atom. Energ.* **71,** p. 503 (1991).

In this paper results of statistical processing of measurement data (for fuel pellets, fuel stacks and fuel claddings) are presented. The measurements are carried out in 1989 (94 tubes and 5000 pellets) and in 1990 (160 tubes and 1500 pellets) on the random chosen tubes and fuel pellets, gone successfully through the fabrication control. Namely, the probability of initial diametrical gap distribution (on the measurement data of inner diameter of 6500 fuel pellets and inner diameter of 254 tubes), as well as the probability of initial effective diametrical gap distribution (on the measurement data of inner diameter of 6500 fuel pellets and inner diameter of 254 tubes) are given. The results obtained show that the most probable initial diametrical gap is about 200 μm.

See also abstracts:

2.1, 2.3, 2.4, 2.5, 2.6, 2.9, 2.18

3.4

6.5, 6.6

7.6

2 MR, MIR and Other VVER Experiments

2.1. A.V. Smirnov, V.I. Kusmin, V.P. Smirnov, K.P. Dubrovin, Y. K. Bibilashvily, B.A. Saletnih. VVER-1000 and VVER-440 Fuel Operation Experience. ANS LWR Topical Meeting, West Palm Beach, Florida, April 1994.

In this paper statistical data are given on fuel performance in VVER-440 and VVER-1000 reactors. Results of non-destructive examination of both types VVER fuel are reported up to burnup of 37.3 and 44.7 *MWd/kg* respectively.

Data presented shows that VVER fuels have satisfactory performance and are still, at these burnups, operated below their technological limits. On the basis of experience and examination design improvements have been proposed.

Up to now 33 VVER-440 and 19 VVER-1000 units were put into operation in Russia, Ukraine, Bulgaria, Hungary, in the former Czechoslovakia, in Finland and Germany. The total operation time of these units is about 530 reactor-years. More than 60000 FAs containing around 10 million FRs have been fabricated and loaded.

The paper considers statistical data of FAs utilised in all VVER units built in the former USSR and results of PIE of some FAs operated in VVER-440 and VVER-1000 reactors during 2 to 4 fuel cycles.

Some technological characteristics of the both types VVER FRs are given. The number of charged and leaking FAs between 1969 and 1991 is presented. The data for FAs manufactured in 1972-1976 show abnormal number of failed fuel with nearly 6% of them affected. This was originated by local internal hydrating of the cladding caused by absorption of moisture on low density pellets. After revision of the specifications and the manufacturing procedure of the fuel pellets and rods. Now the rod failure rates of VVER-440 and VVER-1000 are about 0.008%.

The following results of the PIE examination are presented:

a) Visual inspection - does not reveal marked mechanical or corrosion defects.
b) Global deformation - wrappers do not show any essential dimensional changes. The maximum elongation of the wrappers is 2 *mm* for the VVER-440 FAs and 4 *mm* for the VVER-1000. The bending deflection and twisting angle are 2.4 *mm* and 0.4 degrees for VVER-440 and 3.6 *mm* and 0.13 degrees for VVER-1000.
c) Fuel rod length change - despite the wide spread of data for the average elongation, the VVER-440 and VVER-1000 rod length change is proportional to the burnup with an average value of 0.009% per *MWd/kgU.*
d) Fuel rod strain evolution - as a result of the cladding creep down due to the coolant outer overpressure the VVER fuel rod diameters decrease. The mean value of the diameter decrease for the middle part of all investigated VVER-440 and VVER-1000 FAs irradiated up to the average fuel burnup range 33 - 45 *MWd/kgU* is 0.05 ÷ 0.06 *mm*. The data indicate that at a burnup of about 30 *MWd/kgU* the cladding begin to get in contact with fuel and the minimum rod diameter decrease is reached.
e) Cladding corrosion and hydrating - the VVER fuel rods are operated in the ammonia-boron-potassium water coolant. As a rule the outer surface of all investigated irradiated VVER-440 and VVER-1000 fuel rods are covered with a dark uniform oxide film about 5 micron thick. In some FAs nodular corrosion was found at the lower level of the fuel rods with oxide film thickness in some nodules up to 100 micron. On the inner cladding surface an oxide film varying from 0 to 10 microns in the same cross section was found. Sometimes 15 μm thick layer of UO_2 was stuck on it. Small amount of lamellar zirconium hydrides with random or tangential orientation not exceeding 100 microns were also found. The hydrogen content determined by spectral-isotope method ranged from 30 to 60 *ppm* for tight fuel rods.
f) Cladding mechanical properties - are determined before and after irradiation by transverse tension of annular samples of 3 *mm* high cut, carried out at temperatures 20 and 350 - 380°C. The mechanical properties after irradiation are characterised by high strength and sufficient plasticity, the yield stress not less than 300 *MPa* at operation temperatures. Uniform and total elongations are not less than 4% and 15%, respectively, at 20°C.
g) Fuel pellet structure - the investigations showed that the sizes of the central hole of the VVER-440 and VVER-1000 fuel pellets did not change at any burnup investigated. There was no marked recrystallization of the UO_2 grains in the centre of the pellets, it proves that the temperature did not exceed 1500°C in the centre part of the fuel. The porosity and the pore size distribution changed from 6% and 0.8 microns at the periphery to 3% and 1.1 microns in the centre of the pellet. The adjacent cladding layer of 10 microns is enriched in fission products, e.g. Cs and Xe.
h) Fission gas release - from the VVER-440 fuel it is 0.5% on the average in the burnup range of 34.1 to 36.8 *MWd/kgU* and the gas pressure inside does not exceed 0.3 *MPa* after irradiation during 3 cycles and 0.5 *MPa* during 4 cycles (the initial helium pressure for these rods is 0.1 *MPa*, and after 1986 the initial pressure is 0.5 *MPa*). For the VVER-1000 fuel rods the FGR is about 1% on average. The initial helium pressure is 1.9 to 2.6 *MPa* and it increases to 2.25 and to 2.8, respectively, after irradiation under normal conditions. FGR from the VVER-1000 fuel depends little on fuel burnup within the range 22 to 45 *MWd/kgU.*
i) Failed FAs examination - many of the VVER-440 and VVER-1000 fuel rods show minor traces of contact between rod and spacer grids.

The statistical data on the VVER-440 and VVER-1000 fuel performance in the NPP reactors and the PIE demonstrated a satisfactory performance of the FAs operated under basic load conditions up to an average burnup 45 *MWd/kgU.* The PIE also showed the necessity for modernisation of the VVER FA design to eliminate the deformation of the upper spacer grid rims and prevent the fuel rod cladding failure due to debris fretting.

2.2. Yu.K. Bibilashvili et al. Methodology of in-pile experiments including experimental studies of VVER-1000 refabricated fuels at the MIR research reactor. Proc. IAEA TCM on fission gas release and fuel rod chemistry related to extended burnup, Ontario, Canada, 28 Apr.-1 May 1992, IAEA-TECDOC-697 (1993) 130-132.

This paper gives some indications on the programme of reirradiation of VVER fuel which has been initiated in the experimental reactor MIR. It gives also a description of the process used for refabrication of fuel rods from power reactors.

Investigations were carried out to support MIR experiments with programmed heat rating variations in VVER-1000 type fuels. The maximum linear heat rates for fuels 4.4% enriched can reach 750 *W/cm* - 300 *W/cm* in the burnup range about 10 - 30 *MWd/kg* while for fuels of 6.5% enrichment designed to study fuel behaviour at extended burnup the maximum linear heat rates can rise to about 500 *W/cm* at about 45 *MWd/kg.*

Aside from experimental fuel rods enriched to 4.4% and 6.5% the whole irradiation and testing cycle of which takes place in MIR presently investigations have been started to study experimental fuels refabricated from standard NPP fuel elements irradiated to extended burnup.

At present pilot batches of refabricated fuels have been produced. Using fuels of about 36 *MWd/kgUO$_2$* burnup preliminary experiments with power increases were conducted.

In the research reactor MIR experimental investigations were started to asses VVER fuel behaviour under various operating conditions, including also extended burnup. A complex of work on design-physical modelling and instrumented support of VVER fuel studies are carried out.
Process of experimental fuel fabrication from standard irradiated fuels of NPPs has been worked out; it allows practically a retention of a filler gas and distribution of fuel fission products during operation. In-pile experiments with power ramps indicated their feasibility under specified conditions of programmed fuel loading in a wide range of fuel burnup.

2.3. Bibilashvili Yu. K. et al. VVER Fuel Operation Under Normal Conditions And Analysis Of Damage Causes. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russia, May 1992.

The paper analyses the experience gained in VVER-440 and VVER-1000 fuel operation both in design fuel cycles and on conversion to more efficient four- and three year fuel cycles. The VVER fuel operational conditions are discussed.
The results of the fission gas activity measurements of the primary circuit coolant and the control of the fuel cladding tightness in a shut-down reactor were used to analyse the loss of tightness by fuel elements and assemblies at normal operational condition sand fuel burnup.
The data given are mainly from the Third Unit of Kola NPP with VVER-440, and the Fifth Unit of Novovoronezh NPP and the First Unit of Kalinin NPP with VVER-1000 reactors. The analysis shows that the VVER fuel is highly reliable, the amount of leaking fuel rods (e.g. less than 0.01% of the total number for VVER-1000) meets the world-wide accepted level. No increase in the amount of damaged fuels with increasing burnup up to ~50 *MWd/kg* is observed.
The most important technological and geometrical characteristics of the VVER FA and FR are given, as well as some fabrication data and amounts of produced fuel.
The VVER-440 and VVER-1000 major performance parameters and steady-state fuel cycles are described. Experiments with load follow operation are mentioned as being under way.
An overview on the operation experience and increase of VVER-440 and VVER-1000 fuel burnup is presented. For VVER-440 the FA remain operable for 4 - 5 years reaching ~48 *MWd/kg* average burnup without exceeding 0.007% getting untight. For VVER-1000 the FA retain high operability level during three year cycle and reach the batch averaged discharge burnup ~ 42 - 44 *MWd/kg* and the untight ones are not more than 0.008% of the total amount. A considerably higher reliability has been demonstrated by the three year fuel cycle compared with the two-year one.

2.4. Goncharov V.V., Dubrovin K.P., Ivanov E.G., Koliadin V.I., Platonov P.A. and Riazantcev E.P. Testing of experimental fuel elements VVER-1000 in MR at deep burnup of the fuel., *Atom. Energ.*, **72**, p. 116 (1992).

Because of the high productivity and reliability of the fuel elements at two year operation and with average fuel burnup 28.5 *MWd/kg* (the number of the unhermetized fuel elements is less than 0.02%) investigations are made in order to enlarge the fuel duration time up to three years.
The parameters of the fuel elements for 2 and 3 year operation are given in table 1. The main differences are in the gap diameter (at 2 years - 0.19 - 0.32 *mm*; at 3 years - 0.15 - 0.26 *mm*) and in the central hole diameter (at 2 years - 14 - 1.6 *mm*; at 3 years - 2.3 - 2.4 *mm*). 25 fuel assemblies with approximately 300 fuel elements are studied. The experimental assemblies are studied in the common work conditions of VVER-1000 with the exception of 25th assembly studied in "manoeuvre" power conditions. The measurements show insignificant change of the fuel element's dimensions in the irradiation process (1 - 2 *mm* lengthening, diminishing of the diameter with 0.01 - 0.03 *mm*), and in the fuel element with modern construction (with enlarged diameter of the central hole - 2.3 *mm*) and the fission gas release is less than that in the fuel elements destined for 2 years operation (with diameter of the central hole 1.4 - 1.6 *mm*). The thickness of the oxygen layer of the outer surface of the studied fuel elements is less than 5 *mm*. The contents of hydrogen is less than 0.008 mass per cent in the cladding.
With the increasing of the minimal power over 380 *W/cm* in the fuel elements with central hole of 1.4 *mm* begins a recrystallisations of the grains of UO_2 around the central hole. In the process the release of UO_2 is increased in the central hole which is filled up when the minimal power increases.
In the fuel elements with central hole diameter 2.4 *mm* up to lineal power 510 *W/cm* the diameter remains almost without change. The change of the mechanical properties of the fuel element clad is given in table 3.
The results for 25th fuel element placed under the power cycling conditions are given in table 2 [5, 6]. The maximal burnup for it is 63.5 *MWd/kg* and the maximal linear power - 490 *W/cm*.
In order to study the boundary possibilities of the fuel, a fuel element after continuous irradiation is moved near the centre of the reactor core and is studied under power cycling conditions. The results of the experiment show a possibility for the fuel element to work under these conditions at average depth of fuel burnup above the planed for 3 years operation. The fuel elements remain whole with a small geometric changes, and with notable oxidation and hydration of the cladding. Fission gas release under the cladding is insignificant.
At minimal power 600 - 700 *W/cm* and the bigger burnup, considerable structural changes in UO_2 are observed (usually at little burnup these value is 800 - 900 *W/cm*). These conclusions coincide with similar investigations carried in Mole (Belgium).

2.5. K.P. Dubrovin et al. The results of post-irradiation examination of VVER-1000 and VVER-440 fuel rods. *J. Nucl. Mater.* **178** (1991) 306-311.

The present paper deals with the relatively detailed post-irradiation examination of the fuel rods irradiated in the material science reactor (MR) of the Kurchatov Institute of Atomic Energy, as well as at the 4th Unit of the Novo-Voronezh Nuclear Power Plant (NVNPP). Some examination results are analysed using the PIN-micro code.
The results on six experimental fuel assemblies, tested in the MR reactor under the VVER-1000 operation conditions, are presented. The fuel rods had different fuel-cladding gaps, initial pellet densities, pellet inner diameters and initial helium pressure. The experimental and calculated gas release data were compared. The gas volume in the fuel rods was determined by punching the claddings under water and collection of the gas into a measuring flask. The fission gas release was calculated

for some rods of the assemblies under consideration, by using the PIN-micro code. The relative power of the rods were determined on the basis of gamma-scanning curves. In the calculation the condensed power histories, as well as the largest pellet-cladding gaps were used.

The PIE results for the fuel rods from VVER-440 assembly are presented too. This assembly has been operating at the NVNPP during three fuel cycles (963 effective days) to the average burnup 32.4 *MWd/kgU*. The average linear rate in each fuel cycle was 116, 152 and 124 *W/cm* (maximum 250 *W/cm*). The post-irradiation investigations showed no defects on the end parts, case or rod bundle. All rods remained tight. The corrosion characteristics and mechanical properties of the cladding were satisfactory and did not differ from those for VVER-1000 fuel rods. For the examination results analysis three rods with different fission gas release were chosen. The averaged results on change in the radii of the rods with high and low fission gas release were compared with calculation results on cladding creep in these rods under the action of the coolant pressure without taking into account the axial fuel-cladding interaction. The poorer agreement between the calculated and experimental data obtained for the rods with lower gas release is likely to be due to a great contribution of the axial fuel-cladding interaction to change in the diameters of these rods.

Comparison of the calculation results with those of post-irradiation examination of experimental and standard VVER-1000 and VVER-440 fuel rods irradiated in the MR reactor and NVNPP shows that PIN-micro code can be used reasonably well for modelling the behaviour of the VVER under quasi-stationary conditions. The code was used for determination of the characteristics of the VVER-1000 and VVER-440 fuel rods, developed for operation during three and four fuel cycles, respectively up to mean burnup about 45 *MWd/kgU*.

2.6. V. Kolyadin, K. Dubrovin, P. Platonov, E. Ryazantsev, V. Yakovlev. Operability of VVER-1000 Experimental Fuel Rods at High Burnups in the MR Reactor. Int. Topical Meeting on LWR Fuel Performance. Avignon, France, 21 24 April 1991.

The results of tests in MR reactor at high burnups and post-irradiation examinations of VVER-1000 fuel rods are presented. The results of testing of the fuel rods under power cycling are given. The experimental fuel assemblies are tested under normal VVER-1000 base conditions.

Twenty five fuel assemblies containing about 300 fuel rods are tested. The main features of the MR tests are higher (than in VVER-1000) heat rates, larger number of transients due to reactor shutdowns, as well as a lower fast neutron flux.

The data on thirteen fuel assemblies in which the average and the maximal burnup exceed the rated values for the three-year fuel cycle are given. Nine of them are examined in the hot cells. The influence of various design and technological factors, such as gap size, initial helium pressure, ratio of plenum volume to fuel volume, pellet form etc., is investigated. The distribution of fuel burnup is studied using the gamma scanning method.

In the fuel rods with improved design a lower level of fission gas release is recorded. The matallographic analysis shows oxidation and hydrating of the cladding and structure changes in the fuel. The increase of the linear heat rate above 380 *W/cm* leads to recrystallization of the UO_2 grains and filling of the central hole with surrounding dioxide. During irradiation at heat rates up to 510 *W/cm* the hole diameter remains practically unchanged.

The examination of the mechanical properties of cladding shows that irradiation leads to increase of the strength characteristics of the cladding by about 50%.

2.7. Pytkin Yu.N. et al. VVER-440 physical experimental results for higher enriched fuel. *Atom. Energ.* **70**, p. 225 (1991).

At the VVER-440 reactor, operated at Kola NPP, physical experiments with different part of higher enriched fuel is performed. For the reactor 3/4 charged with 4.4% enriched FAs the experimental and computational results are compared, namely: temperature dependence of reactivity coefficient, Xe^{135} poison, differential and integral control rods efficiencies. The comparison shows good agreement except for the differential control rod efficiency increasing.

The fuel integrity of 4.4% enriched FAs, irradiated during 3 fuel cycles up to 35 *MWd/kgU* burnup, after reactor shutdown was controlled. As a result defective FAs are not detected and the average I^{131} activity of a sample has a value of $1.3 \cdot 10^{-6}$ *Ci/l*. At the operating reactor the total water activity of the primary circuit does not exceed $2 \cdot 10^{-5}$ *Ci/l*. One FR with cladding microdefects is found.

2.8. Proselkov V.N., Simonov K.V., Pushin V.B., Grishakov A.V., Pytkin U.N, Panin M.V. Conclusions from 4 years operation of highly enriched fuel of VVER-440. *Atom. Energ.* **71**, p. 209 (1991).

An experiment has been carried out in the 3rd reactor of Kola NPP for 4 years operation of highly enriched fuel. By routine loading of the reactor. 78 working fuel assemblies and 12 control fuel assemblies, with enrichment of 4.4% and 3.6% respectively, have been introduced. The helium pressure is the fuel element is 0.5 *MPa*. The new fuel has been placed at the periphery of the reactor core. The duration of the cycles is 246, 336, 300 and 378 effective days. The average burnup of 4.4% enriched U is 43.4 *MWd/kgU* and the maximal burnup in 12 fuel assemblies is 46.2 *MWd/kgU*. At the end of the 4th year of irradiation the average burnup in the most loaded 4.4% enriched fuel assembly is 51.5 *MWd/kgU* and the maximal burnup is 58.1 *MWd/kgU*.

The PIN program for evaluation of the thermomechanical behaviour of the fuel element was applied. The maximal centre temperature of the fuel rod, ~1165°C, has been reached in the beginning of the first year of irradiation and decreases to 600°C at the end of irradiation. The low temperature and linear power of the fuel element (< 325 *W/cm*) allow insignificant FGR because of the high pressure of the He under the cladding. At the end of the 4th cycle, even with most unfavourable combination of geometrical and technological parameters, the gas pressure doesn't exceed 3 *MPa*. Under this conditions a PCI contact is possible at the end of the 3rd or at the beginning of the 4th year. As a result of the PCMI the inner cladding radius increases with 21.3 μm, but the integrity of the fuel element is preserved. The temperature of the cladding outer surface does not exceed 309°C. The coolant contamination is within the prescribed limits.

The experiment confirms the possibility of transition to a 4 year operation of highly enriched fuel. Thus, the efficiency of fuel utilisation increases with 10.7%.

2.9. Yu.K. Bibilashvily. Relation of Fuel Rod Operational Parameters and Design Requirements to Requirements for Commercially Produced Fuels and Their Components. KFK and IAEA Sem. on Advanced Methods for Fuel Characterisation and Quality Control 1990. Karlsruhe, Germany, 1990.

Using as an example the VVER-1000 fuel design the paper discusses the approach to the choice of the main fuel design parameters and the development of the requirements for a commercially produced fuel rod and its components (cladding, fuel core). The interrelation between the design parameters, the operational conditions and the fuel life time is shown.

The experience gained in the Soviet Union allows the formulation of the following criteria that are used in fuel designing:

1. Power margin to melt-down, the ratio of the specific LHGR under which the fuel melts down to the maximum permissible LHGR, must be less than 1.3.
2. The end of cycle gas inner pressure must not exceed the coolant pressure.
3. Crack propagation in the cladding by the mechanism of stress corrosion cracking (SCC) must not exceed 10 microns since it is established that 60 microns crack is critical for the Zr + 1% Nb alloy while the ultimate value of the fabrication defect is 50 microns.
4. The yield point for a loss of stability of the cladding due to pressure drop on its wall, taking into account the anisotropy of the fabricated tube, is taken to be the condition under which the effective stress (σ_{eff}) attains the yield strength in the axial direction ($\sigma_{0.2}$) : $\sigma_{eff}=\sigma_{0.2}$.
5. The criteria for the operational reliability of the core are: the number of gas untight fuel rods allowable during operation must not exceed 0.1% and the number of leakers with direct contact between fuel and coolant must not exceed 0.01%.
6. Under MDBA conditions the maximum cladding temperature must not exceed 1200°C and the maximal local thickness of the cladding oxidation must not exceed 18% of the initial value.
7. Under RIA conditions the delivered maximal local enthalpy must not exceed 200 *cal/g*.

Results of the design studies on the ellipse evolution (cladding D_{max} - D_{min}) affected by the external overpressure of the coolant and the hoop stresses during the VVER-1000 cycle are presented. The stability of the cladding is ensured with the wall thickness not less than 0.85 *mm*, but because of the neutron-physical reasons the helium prepressurisation is recommended. The initial helium pressure under the VVER-1000 cladding is 2.0 (+0.35, −0.25) *MPa*.

Experimental results on the dependence between the stress-strained condition of cladding and the strength of the UO_2 particles of different sized fractions are given. The minimum size of the UO_2 particles that without fracture are capable to transfer forces to the cladding exceeding the ultimate value is 1 *mm*. The ultimate value is the contact force affected stress equal to the yield strength of the Zr + 1%Nb alloy. UO_2 particles less than 1 *mm* in size fracture due to thermal expansion long before stresses close to the yield strength developed in the cladding and thus they produce debris that escapes to the radial gap. In case the pellets are jammed with fuel debris more than 1 *mm* in size, high loads can develop in the cladding and it will accumulate damages and axial strains each time the reactor goes to power. That is why it was decided to apply chamfered pellets.

Quantitative results on the initial stage of irradiation induced densification of the fuel pellets are also given. The choice of the gas compensating volume size is discussed.

Because of the big importance and fundamental significance of the requirement on the permissible discontinuity between the pellets in the VVER-1000 fuel stack, special experimental work was performed in the MIR reactor using precise manufactured mock-up fuels. The results showed that in zones 15 - 20 *mm* long adjacent the discontinuities in a fuel column a power density peak is observed, both in the fuel rod with the discontinuity in the fuel and in the surrounding rods. The maximum peak reaches 1.3 for a 16 *mm* discontinuity. After these experiments the maximum single gap size between the pellets of the VVER-1000 FRs is established to be not more than 3 *mm* and the sum for a fuel stack not more than 8 *mm*.

The minimal VVER-1000 fabricated fuel density has changed from 10.4 g/cm^3 to 10.5 g/cm^3 - 10.6 g/cm^3 and thus the total moisture content is within the range 0.0003 - 0.0004 *mass%*.

The quality control of the produced claddings show 7 -10% defected tubes. Special experiments are carried out to investigate the interrelations between some characteristics of the fabricated cladding tubes on the stress corrosion cracking nucleation and propagation. The results for the tube rupture showed that the crack evolution proceeds in a drastically non-linear way. The critical crack depth beyond which accelerated propagation begins is 60 microns.

Special experiments using active species ^{131}I and ^{137}Cs showed that iodine accumulates in a defect at the inner cladding surface up to 95% of the total amount. Even under conditions at low temperatures and temperature gradients iodine and cesium accumulate at a lower temperature site.

The importance to maintain the values of the FR component parameters in an adequately narrow range is emphasised. The allowable values of the initial VVER-1000 FR geometrical and technological parameters and their spread are given. Special measures reducing the effects of some operational conditions which may impact the FR tightness are recommended.

2.10. Strijov P. et al. The improved version of the pin code and its verification, IAEA TCM on Water Reactor Fuel Element Computer Modelling in Steady-State, Transient and Accident Conditions, Preston, England, 19-22 September 1988, IAEA-TC-657/3.4.

A description of a PIN-04 code intended for the light water power reactor fuel rod behaviour modelling under quasi-steady-state operating conditions is given. The code version includes new dependencies over previous one for thermal conductivity of a fuel produced by home technology and for Zr + 1%Nb cladding creep. To verify the code prediction ability the comparison was performed of calculation results with the data of in-pile and post-irradiation research carried out at the Kurchatov Institute of Atomic Energy in MR reactor.

Two fuel rods with a 1.6 *mm* diameter central hole and initial gap values, corresponded to minimum and maximum standard gaps for VVER fuel rod are used to compare the centreline temperature.

The fuel rod No. 1 had the initial diametrical fuel cladding gap - 0.13 *mm* and helium pressure 0.5 *MPa*, the fuel rod No. 2 - gap 0.27 *mm* and helium pressure 2.0 *MPa*. The results of comparison between calculated and experimental values of the centreline fuel temperature for these fuel rods during the first start-up are given. The discrepancy between calculated and experimental values even at maximum thermal loads amounts to several degrees at the level of 1700°C.

To study changes of an internal gas pressure during irradiation two fuel rods were irradiated in MR reactor (number 10 and number 13). The fuel rods had an outside diameter 13.65 *mm* and differed from one another only by values of initial diametrical fuel-cladding gap. For the fuel rod 10 - 0.18 - 0.22 *m*, for fuel rod 13 - to 0.34 - 0.38 *mm*. The relative power of fuel rod 13 was about 7% lower than that for fuel rod 10. Measurements of pressure were conducted during two years and the maximum burn-up in this case was about 28.3 *MWd/kg* (FR 10).

Two versions of calculation were carried out: the first one with an initial gap of 0.22 *mm* and grain size 15 μm, the second one with an initial gap of 0.18 *mm* and grain size 30 μm. The helium initial pressure of 0.1 *MPa* amounts to 3.0 *MPa* at the end of the irradiation. The calculated values obtained with the PIN-04M code are rather close to the experimental results.

The comparison of calculation results with the data of the post-irradiation investigations of VVER-440 was carried out with a standard assembly of the Novo-Voronezh fourth unit. The chosen assembly has been operating during three full cycles up to a mean burn-up of 32.4 *MWd/kgU*. 93 fuel rods contained a fuel of relatively low density (10.30 - 10.35 *g/cm*) and the other 33 - fuel with relatively high density (10.50 - 10.60 *g/cm*). The performed gamma-scanning of all fuel rods gave the following results: "Saw-like" curves of gamma-scanning were obtained for 30 fuel rods, being characteristic of those cases when increased temperatures of fuel (> 1500°C) and caesium migration take place. For 96 fuel rods "smooth" curves were obtained, when lower temperatures of fuel (< 1500°C) take place and caesium migration is absent.

The "saw-like" gamma-scanning curves were obtained only for fuel rods with low fuel density, located in peripheral rows of the assembly and having an increased power in comparison with the inner rows. two fuel rods situated near the central tube and having also somewhat increased power were characterised by "saw-like" curves of gamma-scanning due to a neutron flux elevation in this part of the assembly. For all fuel rods, containing a fuel with high density only smooth gamma-scanning curves were marked. Fuel rods characterised by "smooth" gamma-scanning curves had the elongation over 5 *mm* and fuel rods characterised by saw-like curves had a less elongation. 18 fuel rods were punctured and the fuel rods with high density fuel showed a very low fission gas release under normal conditions.

The fuel rods with low density fuel have indicated a fission gas release 130 - 576 cm^3 (Kr and Xe).

The comparison of calculated and experimental results obtained for three fuel rods of the assembly is given:

Fuel rod number	120	112	15
V_{exp}, cm^3	20	485	576
V_{calc}, cm^3	18-22	413-763	548-586

Calculations were carried out for two values on initial diametrical gap 0.195 and 0.270 *mm*.

PIN-04M code together with submodels input in it describes correctly the processes taking place in VVER type fuel rods. The PIN-04M code can be used to optimise an initial internal gas pressure in the fuel rod of the increased power VVER-440 reactor. The most optimal value of the initial gas pressure for fuel rods of this type is the value of 5 - 7 *bar* and a further increase of the pressure does not lead to an improvement of thermal characteristics of fuel rods.

2.11. V. Yakovlev, P. Strijov, V. Murashov, A. Senkin, R. Terasvirta, P. Liuhto, J. Moisio, O. Tiihonen, S. Kelppe, K. Ranta-Puska. Qualification and Interpretation of MR Test Reactor Irradiation Data on VVER-440 Type Fuel Rods For Fuel Thermal Model Validation. IAEA TCM on Water Reactor Fuel Element Computer Modelling in Steady-State, Transient and Accident Conditions, Preston, England, 1988.

The report gives the results from the examination of the thermal and mechanical behaviour of the VVER type fuel rods by the irradiation program SOFIT, including a series of characterised and instrumented test fuel bundles. The irradiations and the examinations are performed in the Kurchatov Institute's MR reactor and hot cells. The main objectives of the program are to support measures to enhance the flexibility of the use of the fuel and accordingly verify for the VVER annular fuel and Zr + 1% Nb the performance codes from the PIN series in the USSR and the GAPCON-THERMAL-2 code in Finland.

A bundle of 18 rods is irradiated for 160 full power days. Six of the rods are instrumented with fuel centre line thermocouples. Fuel to cladding diametrical gap is varied between 0.150 *mm* and 0.280 *mm* and the helium initial pressure from 0.1 *MPa* to 1.5 *MPa*. The power determination is based on the calorimetric method. The maximum linear heat rates are above 400 *W/cm*. The discharge burnups of the fuel rods are around 10 *MWd/kgU*, local maximum 16 *MWd/kgU*. The maximum temperature is 1670°C for the rod with large gap (0.270 *mm*). The measured temperatures show dependence on the as-fabricated gap size and weakly on the helium fill pressure.

The results of the calculations obtained by the existing steady state fuel behaviour computer codes are compared with the experimental results (the temperatures). Study and verification of the initial (start-up period) relocation model in the GAPCON-THERMAL-2 code is performed. The final density is 98% TD.

The amounts of recovered gases indicate low or moderate FGR. For the rod with high power and large gap (0.280 *mm*) the measured FGR fraction is 7 ± 1%, which is well predicted by the G-TH-2 code.

The cladding Zr + 1% Nb specific dimensional changes are modelled on the base of the Kurchatov Institute irradiation experiments and out-of-pile tests for fully recrystallized cladding tubes. The general diametrical change in the experiments is relatively large (-25 to -45 microns), which is due to the compressive stresses (90 to 100 *MPa*) and the cladding temperatures (320 to 400°C). The in-pile results have been qualified and analysed. PIE work is in progress.

2.12. Proselkov V.N. et al. Experience of the VVER-1000 FR and FAs operation at Unit 5 of Novovoronezh NPP at the switching to 3-year fuel cycle. *Atom. Energ.* **65**, p. 3 (1988).

The VVER-1000 reactor of the Novovoronezh NPP Unit 5 is the basic one for the FAs, FRs and two- and three-year fuel cycle regimes examinations. From the first to the fourth fuel loads the unit operated in a two-year mode at reload enrichment 3.3%. Beginning with the fifth fuel load the switch to three-year mode of operation was started, using FAs with 4.4% enrichment, consisting of FRs with burnable absorbers.

Up to now experience has been gained in operation of 368 FAs (116656 FRs) at the 5th unit of the Novovoronezh NPP. 104 from these FAs were operated during 3 fuel cycles (up to 783.8 EFPD) at the maximum burnup 37.4 *MWd/kgU*. The total iodine activity in the coolant during five fuel loads did not exceed 10^7 *Bq/kg*. The total iodine activity, as well the I^{131} activity decreases from the first to the fifth fuel load, especially after leakers discharge in the second fuel load. There were 5 leaking FAs, i.e. 1.35% from the total number of FAs (368), irradiated at the reactor. On the basis of the activity data analysis can be concluded, that the leakers appear mainly during the first fuel load.

All FRs irradiated during three fuel cycles (104 FAs), as well as 4.4% enriched FRs, have kept the cladding tight.

2.13. Strijov P.N. et al. The PIN-04M code verification on the experimental data. VANT, Ser. Atomnoe Materialovedenie, No. 2, p. 39 (1988).

In this paper a short description of the PIN-04M code, used for VVER FR behaviour modelling in quasi steady-state regime of operation, is given. In the new code version correlations for the fuel thermal conductivity and Zr + 1%Nb cladding creep are included. The code is verified on the basis of in-pile and post-irradiation examination data, obtained at the MR reactor. Calculational results of initial He pressure optimisation at increased VVER-440 power are presented. These results show, that for this kind of FRs the optimal initial He gas pressure is 5 - 7 *bar*.

2.14. Pazdera F. The PIN code for termomechanical calculation for fuel elements in the VVER reactors and check in of the program. VANT, Atomnoe materialovedenie, No. 2, p. 18, (1988).

The short description of the PIN computer program designed for termomechanical calculations of fuel elements is given. The main parameters are the fuel temperature and fission gas release. The numerical results are compared with the experimental data obtained in the MR reactor in the Institute of atomic energy "Kurchatov".

The pressure increase of two FAs, No. 10 and 13, has been experimentally investigated during the irradiation. They differ by the gap size (No. 10 - 90 μm, No. 13 - 190 μm) and by the linear power (No. 10 has ~7% larger relative power than No. 13). The irradiation duration has been 2 years and a burnup of 25 *MWd/kgU* has been reached.

For the evaluation of the fuel rod properties at transient conditions in the MR reactor, the experiment is performed with FRs, supplied with sensors for measurement of the fuel and cladding dimensions (fuel rods No. 1 and 2). The fuel rods 1 and 2 differ by the initial gap size (No. 1 - 95 μm, No. 2 - 190 μm) and by the linear power (No. 1 has ~20% higher linear power than No. 2). The elongation is measured during about 2 month; the maximal burnup reached is 4.9 *MWd/kgU*.

The results obtained by calculations with the PIN code for these four FRs are discussed.

The results show a strong dependence between the fuel temperature and fission gas release.

For the FR No. 10 the maximum calculated fuel temperature is 2250°C and the calculated fission gas release is 43% (the experimental value is 20%).

For the FR No. 13 the maximum calculated fuel temperature is 2170°C and the calculated fission gas release is 60% (the experimental value is 25%).

It can be seen that the experimental results are within the calculation tolerance caused by the inaccuracy of the initial data.

The experimental and the calculated values for the gap size are shown. For the FR No. 1 (with initial gap 95 μm) the calculations shows that the gap closes at 60 *kW/m* linear power. The experimental axial interaction between the fuel and the cladding is observed at 40 *kW/m*. For the fuel element No. 2 (with initial gap size 190 μm) the PIN code predicts a gap size of 50 μm at 51 *kW/m* heat release. The experimental axial interaction between the fuel and the cladding is observed at 40 *kW/m*.

Up to the interaction between the fuel and the cladding the PIN code describes satisfactory the process of the cladding deformation but the error concerning the fuel deformation is two times larger.

The obtained results confirm the necessity to develop a new heat transfer models for the fuel-cladding distance. The asymmetry of the fuel pills needed to be taken into account. operation

2.15. V. Yakovlev, P. Strijov, V. Murashov, J. Johansson, R. Terasvirta, O. Tiihonen, K. Ranta-Puska. Research Carried on VVER-440 Type Fuel Rods in the MR Reactor. In: *Improvements in Water Reactor Fuel Technology and Utilisation.* IAEA, Vienna, 1987.

A description of the research programme carried out on the in-pile behaviour of VVER-440 fuel is presented. The work is carried out as a joined programme between Soviet and Finnish organisations and irradiation was performed in the MR test reactor at the Kurchatov Institute. The first test bundle characteristics and irradiation conditions in the MR reactor are given. The temperature results from 2000-hour steady state irradiation period are presented and preliminary comparison between measured and calculated fuel thermal behaviour is made.

To obtain the maximum information on the effects of the separate design parameters, some fundamental parameters are varied among 18 fuel rods of a bundle. Rods are manufactured to three pellet diameters, corresponding to the average (0.210 *mm*) and the two additional gap sizes (0.270 and 0.150 *mm*) within the design limits. The initial helium pressure is 0.1, 0.5 and 1.5 *MPa*. The pellet have a central hole of 1.6 *mm* in diameter. The rods are instrumented with thermocouples to measure the fuel pellets temperature at different positions.

A careful percharacterisation of the test fuel was performed: by the supplier and by the user.

The local maximum linear heat rates are kept well above 400 *W/cm*, higher than in VVER-440 (325 *W/cm*), the average one is about 300 *W/cm*. The highest burnup of the rods is around 6 *MWd/kgU* at 84 days. The highest measured temperature is 1670°C due to the high linear power. The maximum temperatures along the irradiation decrease slightly.

Calculations for the start-up period are performed with the PIN and the GAPCON-THERMAL-2 codes and the temperature predictions are compared with measured ones. The FGR is only calculated, it is assumed to be less than 3%.

The results of the SOFIT programme will be used to verify the fuel behaviour computer codes used in the Soviet Union and Finland as well as to study possibilities of enhancing the economy and operational flexibility of fuel utilisation.

2.16. Ignatenko E. I. et al. An experiment on FA burnup extension of VVER-440 reactor. *Atom. Energ* **61**, p. 240 (1986).

VVER-440 reactors are designed for three-year cycle mode of operation with fuel lifetime about 7000 effective full power hours. The fuel utilization can be improved by switching VVER-440 to four-year fuel cycles with extended fuel lifetime.

In this paper an industrial experiment is described, aimed to switching VVER-440 to four-year fuel cycle. At the Kola NPP quality characteristics of operational safety margins of FR reliability in the case of extended fuel lifetime (1042 EFPD, 3.6% initial fuel enrichment, 34.3 *kg/tU* average FA burnup) are obtained. The three stages of organisation works are described. Experimental data on average and maximal FA burnup after four-year fuel cycles are presented. The average FA burnup of the discharged fuel from fuel cycle to fuel cycle is increased up to 36.2 *kg/tU* at steady-state regime. This value exceeds the designed one with 17%. The maximal FA (3.6% enrichment) burnup in this experiment was 39.7 *kg/tU*.

2.17. Schavelin V. *M.* et al. Friction characteristics determination of reactor materials. *Atom. Energ.*, **56**, p. 134 (1984).

In this work a method and an installation for investigations at laboratory conditions of the friction characteristics of friction-couples fuel-cladding are described. The results for friction coefficient between UO_2 and zirconium alloys depending on temperature, contact pressure etc. are presented.

2.18. Vihorev U. V. et al. Results from the operation of a 5th VVER 1000 reactor in Novovoronezh NPP. *Atom. Energ.* **54**, p. 163 (1983).

The main characteristics of the reactor, the reactor core, fuel rods and assemblies as well as the operational conditions of the fuel assemblies are given.

During the first period of operation of the reactor core half of the FAs have been reloaded every year. This way, a 2 years cycle with average burnup of 30 *MWd/kgU* and 3.3% enriched fuel has been applied. Later, a transition to a 3 year cycle using 4.4% enriched fuel and average burnup of 40 *MWd/kgU* has been achieved. The FRs are made of UO_2 pellets with central hole and Zr + 1%Nb cladding. The filling gas (He) pressure is 1.96 ÷ 2.45 *MPa* at room temperature.

Maximum cladding inner side temp.	350°C
FR linear power	
average	176 *W/cm*
maximum limit	525 *W/cm*
maximum operational	465 *W/cm*
Fuel assembly dimension	238 *mm*
Number of FRs in the FAs	317
Fuel stack height in operational state	3560 *mm*
Minimum fuel density	10.4 g/cm^3
Fuel rod outer diameter	9.1 *mm*
Cladding thickness	0.63÷0.68 *mm*
Gap size	0.19÷0.32 *mm*
Central hole diameter	1.4÷1.6 *mm*
Gas plenum volume	11 cm^3
Maximum peaking factors	
FAs	≤ 1.35
volumetric	≤ 2.0
expected	2.79

In the initial period of the operation, the 5th reactor undergoes a high number of transition regimes. No influence of the power change on the operability of the fuel elements was observed. The coolant activity has shown low degree of untightness of the FRs. Only one FA of 48 checked one proved to be untight.

2.19. Murashov V. N. et al. Non-contact method for investigation of FR thermal state during the irradiation. *Atom. Energ.* **46**, p. 118 (1979).

During the irradiation the FR thermophysical characteristics can be considerably changed. The investigation results show, that at the beginning of irradiation (the first 100-300 hours) the fuel centre line temperature increases, and in the next period of operation at the constant power it decreases. In this paper a method for non-contact investigations of FR thermal state during long-time operation is proposed. This method is based on using of the following unsteady-state regimes: power ramping changes; reactor shutdown; reactor scram; power oscillation etc. The experimental measurements of VVER-type fuel assembly in the MR reactor are carried out. The FA with 18 FRs was supplied with both: thermo-couples on the inlet and outlet of FA, as well as thermo-neutron detector for power changes monitoring. The experimental data show, that at the first 100 hours the FR thermal conductivity decreases, after this period it increases. These data are given for FR linear heat rating, averaged over FA, with the value of about 200 *W/cm*. An reactor scram after 380 *h* leads to a sharp decrease of thermal conductivity, after that, in the operation at constant power the conductivity increases up to a stable level. On the basis of the results (experimental and theoretical) it can be concluded that during the operation the relaxation period decreases with a factor of 1.4, which corresponds to the VVER-1000 FR center line temperature decrease from 2100 to 1600°C (at linear heat rate 500 *W/cm*). As a result of long time operation at high linear heat ratings the thermal resistance of the pellet-cladding contact becomes negligible comparing to the total thermal resistance.

2.20. Samsonov B. V, Sulaberidze V. Sh. Peak pressure rise in FRs with compact UO_2 fuel at the beginning of the reactor operation. *Atom. Energ.* **42**, p. 277 (1977).

Experimental data on FGR in the FRs with compact UO_2 fuel show, that at the first reactor coming out on operational regime the gas pressure under the cladding sharply increases and after few hours decreases to a stationary level. This pressure increase is due to the gas release from the fuel. This release of technological gases can decrease the thermal conductivity of the He filled gap between pellet and cladding at the first hours of FRs operation. The fuel is fabricated, using standard VVER-technology. In this paper some experimental results are given and discussed. The characteristics of 9 irradiated FRs (such as average linear rating, calculated fuel centre, fuel surface and volume-averaged temperatures) are shown. The pressure changes at power ramp increasing is given. The FGR characteristics for the same 9 FRs such as initial gas pressure, sort of gas, gas amount [10^{-3} $m^3/kgUO_2$] as well as the temperature at the beginning of FGR are given. It is shown that this peak pressure increase in the FRs with compact UO_2 at the beginning of reactor operation is due to the fuel surface desorption of technological gases. This release becomes remarkable at the volume-averaged fuel temperature 1050 - 1250 K. The total amount of gas released is about $2 \cdot 10^5$ $m^3/kgUO_2$.

2.21. Miloserdin U. V. et al. Creep of UO_2. *Atom. Energ.* **35**, p. 371 (1973).

Laboratory and in-pile reactor investigations have been carried out for the creep of the U fuel, for irradiation of the samples being used one of the channels of IRT 2000 with diameter of 180 *mm*. In-pile experiments are carried out at temperature 1200 ÷ 1600 K and a possibility of charging the sample up to 120 *kg*.

When the neutron flux affects the sample, the creep rate increases and depends linearly on the fission density. At maximal fission density of $0.625 \cdot 10^{12}$ fis/cm^3s in the range 1373 ÷ 1523 K and stress in the sample 3 and 4 kg/mm^2 the creep velocity increases 3 times when compared to the laboratory experiments.

2.22. Bohmert J. et al. Experiments on modelling a jump-like changes in the power in order to determine damage to VVER fuel elements.

Jumplike changes in the power of the reactor give rise to critical loads and are one of the main reasons for damage to fuel element. A simple way to obtain the first experimental evaluations of the power limits is to model the action of a mechanical load on the cladding of the fuel element under laboratory conditions. The changes of the equivalent stress over time in fuel element cladding of VVER-440 in the maximally stressed transverse cross section are calculated for a model jumplike change in the power with the help of a one-dimensional program for calculating the characteristics of fuel elements STOFFEL-1.

A change in the power form 160 to 400 *W/cm*, (which corresponds to a change from 40 to 100%), followed by operation at a constant power is chosen as a model of the jump. The rate of change of the power equals 1 *W/cm*.sec (fast jump) or 0.01 *W/cm.sec* (slow jump).

The test set-up creates in the sample an uniaxial tangential tension force (test for transverse stretching of ring-shaped samples). Samples with two different forms are used: smooth, untreated rings 2 *mm* wide (form A) and a ring 4 *mm* wide on whose perimeter two narrow cuts are placed separated by 180°C (form B).

The sample is placed in an electrically heated, air-tight test chamber. The experiments are performed under isothermal conditions at 623 K (in the model calculations, depending on the power of the fuel elements, the average temperatures of the cladding range from 610 to 648 K) in a flow of argon or a mixture of argon and iodine (the flow rate is 0.5 litres per hour). The partial pressure of iodine is set with the help of a heated iodine generator in the range from 130 to 1060 *Pa* (under standard experimental conditions the pressure equals 800 *Pa*).

The samples are studied in the unirradiated state and after irradiation up to a fluence of fast neutrons approx. 10^{21} cm^{-2} in the VVER-2 reactor "Reinsberg".

The experiments gave the following results:

- unirradiated samples do not disintegrate in an argon-iodine atmosphere, but in subsequent tension tests the yield stress and the indicator of uniform elongation drop somewhat;
- in tests of irradiated samples in an argon-iodine atmosphere with a sufficiently high iodine concentration (the partial iodine pressure > 500 *Pa*) the sample disintegrates;
- the time for disintegration to occur and the disintegration stress depend on the rate of growth of the load and the iodine concentration; in the case of a slow power jump disintegration occurs at the stage at which the load increases with a stress of 150 *MPa*, while in the case of a fast power jump the sample disintegrates only at the second jump and after the maximum stress is exceeded;
- the structure of the surface of the fracture is characteristic for brittle fracture caused by iodine corrosion under stress;
- for partial iodine pressures < 400 Pa the sample does not disintegrate.

2.23. Moisio J. *M.* et al. VVER-440 fuel performance at Loviisa. Nuclear Europe Worldscan.

At Finnish IVO utility's Loviisa plant (two soviet type PWRs, VVER-440s) experience with the fuel has been in general, excellent. For fuel performance, the operating conditions maximum linear heat rate is to 325*W/cm*) have been favourable. High load factors, accompanied by a very stable reactor operation, have been achieved. By the 1989 overhaul, Loviisa-1 s cumulative load factor was 81.4% and Loviisa-2 86.4%. During recent years, annual load factors have exceed 90%. Collective irradiation doses have been low, e.g. in 1989 at 0.7 *manSv* for Lo-1 and 1.0 *manSv* for Lo-2.

During the two units' accumulated 23 reactor years 16 fuel rods have failed - a rate of 0.0039%, which is comparable to good international experience.

Higher than expected load factors and reduction of the reactor core in 1980 to lower the pressure vessel irradiation doses led to demand for higher fuel burnups. There was a clear need to rise the initial design rod burnup limit of 40 *MWd/kgU* to near 50 *MWd/kgU*. In the mid-1980s a programme was initiated to verify feasibility of the higher burnup limit. The development of the maximum rod and assembly burnup is shown.

To verify the acceptable fuel behaviour in burnup over 40 *MWd/kgU*, a high burnup programme was started in 1985; including irradiation and examination of two fuel assemblies for four cycles, instead of normal three cycles, to achieve the desired burnup level.

Four rods, selected from the three assemblies were further shipped to the Studsvik Nuclear AB hot cells in Sweden. Together with the findings on the one assembly examined by Studsvik in the first half of the 1980, these results form a good set of performance data from rod burnup range 32 to 46 *MWd/kgU*.

The creepdown values of the four rods examined by Studsvik showed an average diameter decrease of 25 μm to 60 μm and local maximum of 50 to 90 μm.

Gamma-scanning of individual rods with average rod burning up ranging from 41 to 48 *MWd/kgU* showed that: there was no caesium redistribution in any parts of the fuel rods; there were no visible gaps in the fuel columns.

The fission gas release fractions were nondestructively determined with gamma-method, developed by ABB Atom.

Punctured values ranged from 0.5% to 0.8% for the four rods with burnup from 43 to 46 *MWd/kgU*.

Pellet-cladding diametrical gap: Studsvik hot cell results indicated that after four cycles irradiation the diametrical gap is practically closed. For three-cycle fuel there seems to be some gap left.

Oxidation of the claddings: One fuel rod form the four, sent for PIE in hot cells was destructively inspected.
Microstructure of the pellets: The grain size varied somewhat from pellet to pellet, while no clear grain growth could be seen toward the pellet centre. Typically, some radial cracks existed in the pellets. The rim of the pellet sample had an increased porosity. Density measurements indicated pellet swelling of about 4% volume.
VVER-440 fuel at Loviisa NPS has performed well. The increase of burnup up to 48 *MWd/kgU* has not brought about a measurable fuel behaviour changes in, e.g. fission gas release, dimensions, mechanical integrity of rods, etc. The average failure rate has been comparable to international experience. Some factors that might have contributed to fuel failures have been eliminated with a new pellet fabrication method.

See also abstracts:
4.4, 4.7, 4.9
6.1, 6.6, 6.7
7.1, 7.4, 7.5, 7.6, 7.7, 7.9, 7.11, 7.12, 7.14

3 VVER Fuel Post-Irradiation Examination

3.1. A. V. Smirnov et al. Some experimental results of investigations on the nuclide composition and burnup fraction of VVER-1000 standard fuel. Proc. IAEA TCM on fission gas release and fuel rod chemistry related to extended burnup, Ontario, Canada, 28 April - 1 May 1992, IAEA-TECDOC-697 (1993) 133-144.

Fission and technological gases into the free fuel rod volume has been measured my means of μ-spectroscopy and mass-spectroscopy. Fuel rods for examination were taken from fuel assemblies with burnups ranging from 20 to 50 *MWd/kgU*.
It was found out that amount of fission gas release for intact fuel rods is not significant, e.g. the share of released Kr-85 has not exceeded 2.5%. In some cases the increased fission gas release was observed for defected fuel rods.
With the use of irradiation history data, information on fuel rod design and results of fuel rod parameters measurements in hot cells, calculations of fission gas release into rod free volume were carried out. Results of calculations and measurements are compared.
UO_2 with the initial enrichment 3.3, 3.6, 4.4% for U-235 from 2 fuel assemblies of VVER-1000 was investigated. Fuel assembly N4 has been operated in the fifth block of the Novo-Voronezh NPP up to design burnup 45 *MWd/kgU*, fuel assembly N6 has been operated in the first block of the South-Ukrainian NPP up to the design burnup 37 *MWd/kgU*. The time of fuel cooling in assemblies N4 and N6 made up 3.4 and 4.3 years, respectively.
To investigate the nuclide composition and to determine the burnup over the fuel rod height, fuel specimens, 10 *mm* long, were selected from different levels of fuel column. To measure the burnup and nuclide composition there were selected samples over the fuel rod radius from the fuel rod central part(level of 1970 *mm* from the fuel element bottom).
The experimental results of nuclides content and burnup over the fuel rod height and radius are presented. At the periphery the fuel burnup is 1.15 - 1.20 time higher than in the centre. Non-uniformity increases with growth of the fuel burnup. The investigation results for Pu accumulation over the fuel rod height and radius versus the burnup fraction were presented. The dependencies of Cs-137 accumulation and Cs-134/Cs-137 ratio upon the burnup were shown.

3.2. Elizarov V. P. et al. Experimental investigation of out-reactor methods of failed fuel detection. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russian Federation, May 1992.

Results are reported on test and comparison for all known methods of out-reactor failed fuel detection: water, water-gas, air-lift, vacuum and dry. The D-1 and D-2 experimental assemblies contained fuel elements with initial defects in claddings as well as standard assemblies with failed fuel elements after twelve years keeping were investigated. Experimental devices and methods of tests are described.

1. Water method.
   The stand water is agitated with the circulating pump and the sample of water is taken for determination of fission products activity with gamma-spectrometer of high resolution.
2. Water-gas method.
   The stand water is agitated with the circulating pump and the sample of water is taken for determination of fission products activity.
3. Airlifting method.
   Water is degassed by air entering into the lower sleeve of FFD can.
4. Vacuum method.
   Vacuum generated in FFD can stimulates the fission products release from the failed fuel elements.
5. Dry method.
   The FFD can is dried and blown air which is dispensed to the flow beta-radiometer. Process of fission gases release is intensified at the expense of fuel element heat-up.

3.3. Kanashov B. A. et al. Geometry variety of VVER fuel rods. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russia, May 1992.

The classified experimental data are presented for study of dimensional stability of standard fuel rods for the VVER type reactors with UO_2 fuel core in Zr - 1%Nb claddings. Dynamics of the length change, diameter and the gap between fuel and claddings was investigated for three fuel assemblies of the VVER-440 reactor and four fuel assemblies of the VVER-1000 in dependence of the fuel burn-up 22 - 45 *MWd/kgU*.
Characteristics of the tested fuel assemblies and elements are presented.
Three standard fuel assemblies of the VVER-440 reactors were selected for investigation together with four fuel assemblies of the VVER-1000 reactor. The main design parameters of the fuel rods are given.
In every fuel assembly a group of 30 - 60 rods was investigated in detail.
The general view of the measurements is characterised by the following:
- the greatest decrease of diameter is observed in central part of the fuel rod

- in the process of operation practically all fuel rods become oval and the period of ovality is differing from rod to rod
- on diagrams of the average diameter the periodicity is observed and it is multiple of the length of a fuel core (11 *mm*) for the fuel rod with burn-up more than 35 *MWd/kgU*.

Measurements of a diameter gap under the normal conditions between fuel and cladding in the zone of the maximum burn-up showed that in the range from 20 to 30 *MWd/kgU* its values decrease monotonously from 0.11 - 0.17 *mm* to 0.03 - 0.08 *mm* . When burnup was more than 35 *MWd/kgU* the diameter gap didnt exceed 0.02 - 0.04 *mm*. Estimation of the difference between thermal expansions of the claddings and fuel values in radial directions under a normal mode of operation showed that the value was 0.07 - 0.10 *mm*. It is 0.03 - 0.08 *mm* more than the minimum "cold" gap at the burnup exceeding 35 *MWd/kgU*.

The irradiation growth at the initial stage when the fuel burnup is 32.6 *MWd/kgU*, produces a small effect on the deformation of Zr - 1% Nb tubes in the fuel assembly of VVER-1000.

At higher burnup the swelling fuel cores can deform the cladding, that was observed in some fuel rods in fuel assemblies.

3.4. J. Moisio, R. Terasvirta, K. Ranta-Puska. Experience from examinations of fuel rods irradiated to high burnups in Loviisa reactors. Proc. IAEA TCM on the Fuel Performance at High Burnup for Water Reactors, Studsvik, Sweden, 5-8 June 1990. IAEA, 1991. p. 78-84

Destructive examination on selected high burnup fuel rods, irradiated in the Loviisa reactors, have been carried out at Studsvik Nuclear AB's hot laboratories since 1982. Together with the pool-side examinations and recorded operation information they provide a qualified set of performance data from a burnup range of 32 to 48 *MWd/kgU* for VVER fuel featuring annular pellets and Zr + 1%Nb cladding. Emphasis in the examination has been put on dimensional changes, rate of cladding oxidation, fission gas release, and other phenomena that may become limiting at high burnups. In the 14 punctured rods, the fission gas release fraction has been around 0.5%, inevitably due to the thermal processes only. Four of these rods and seven additional rods were non-destructively gamma scanned for rod plenum gas inventory. The two methods agree as to the low gas release. The measured rod average diameter decreases due to cladding creep ranged from 25 to 60 μm in the high burnup rods. The final gaps were generally closed and in parts of the rods fuel-to-cladding bonding was observed. Moderate rod length increases were recorded. Only minimal cladding external corrosion of 1 to 4 μm was seen. The results of calculations by Technical Research Centre (VTT) were compared with the measured dimensional changes and fission gas release. It is concluded that the low maximum linear power, not exceeding 300 *W/cm*, and relatively early gap closure keep the fuel temperatures below the gas release threshold.

Examination of one whole fuel assembly and the later examination of four rods from three high burnup fuel assemblies of Loviisa NPS, irradiated to high burnups, have shown large safety margins for this type of fuel design. The detailed hot cell examinations of fuel rods with burnups from 32 to 46 *MWd/kgU* together with the pool-side inspections of the fuel rods up to burnups of 48 *MWd/kgU* have shown very small water-side corrosion thickness and low fractions of released fission gases. Results imply that cladding oxidation should not become a limiting factor at extended burnups for these design and operational conditions. The small release fractions, supported by the code calculations, suggest low fuel temperatures.

The results of the examinations and studies carried out demonstrate that during normal operation there are sufficient margins to increase the allowed fuel rod burnup to 48 *MWd/kgU*.

3.5. P. N. Strijov et al. Computer and experimental VVER fuel rod modelling for extended burnup. Proc. IAEA TCM on Fuel Performance at High Burnup for Water Reactors, Studsvik, Sweden, 5-8 June 1990. IAEA, 1991, p. 167-172.

To improve the economy, four-year instead of three-year refuelling cycle has been started with VVER-440, and similar steps with VVER-1000 (three-year instead of two-year) are taken, both leading to considerable burnup extension. Fuel rod behaviour of these reactors with extended burnup was analysed with the PIN-micro code, developed in co-operation between Russian and Czech specialists. With the aim of further verification, calculations with this code were compared with results of irradiation of instrumented experimental fuel rods and post-irradiation examination (PIE) performed at the Kurchatov Institute of Atomic Energy, Moscow. As an example some comparisons with PIE of VVER rods irradiated to burnup 60 *MWd/kgU* are shown as well as comparison with two rods with inside gas pressure instrumentation irradiated to 32 *MWd/kgU*. The results of variant calculations by the PIN-micro code are presented for VVER-440 and VVER-1000 fuel rods for a four- and three-year refuelling cycle, respectively. To optimise the initial internal gas pressure in VVER-440 fuel rods for four-year refuelling cycle some variant calculations were performed, whose results show maximum value 0.5 - 0.7 *MPa*. It is shown how the calculations performed allowed improvements in the fuel rod design.

To study changes of an internal gas pressure during irradiation an experimental assembly was irradiated in MR reactor two fuel rods of which were instrumented with sensors for continuous pressure recording. These fuel rods N10 and N13 had an outside diameter 13.65 *mm* and were differed from one another only by value of the initial diametrical fuel-cladding gap (018 - 0.22 *mm* and 0.34 - 0.38 *mm* for fuel rod N10 and N13 respectively. The maximum burnup for rod N10 was about 28.3 *MWd/kgU*.

In connection with the fact that an initial size of a fuel grain varies within the limit of 15 - 30 microns two variants of calculations were carried out: with an initial gap 0.22 *mm* and grain size 30 microns; with an initial gap 0.18 *mm* and grain size 30 microns, which give an upper and lower evaluations of the existing scattering of parameters. As a whole the PIN-micro code predicts rather well the dynamics of increase in the internal gas pressure, which evidences indirectly the applicability of the fission gas release model used.

To verify prediction of fission gas release with the PIN-micro code for prepressurised fuel rods, comparison with post-irradiation examination with seven fuel assemblies (FAs) contained experimental VVER-1000-type fuel rods and irradiated in MR reactor was performed [5]. In five FAs numbering from A to E fuel rods contained pellets with a 1.4 - 1.6 *mm* central hole used in VVER-1000 fuel rods for a two-year refuelling cycle. In F and G FAs fuel rods contained pellets with a 2.4 - 2.5 *mm* central hole used in VVER-1000 fuel rods for a three-year refuelling cycle. A first group of fuel rods (A trough E FAs) is characterised by high FGR, central hole closing and fuel restructuring caused by high fuel temperature (> 1700°C). The fuel rods from second group (F-G FAs) are characterised by a lower FGR and absence of fuel restructuring. The PIN-micro calculation results confirmed qualitatively the difference in behaviour of fuel rods containing the pellets with initial central hole diameter of 1.4 - 1.6 *mm* and 2.4 - 2.5 *mm* with respect to FGR, attainable temperature level and fuel restructuring.

To compare the calculation results with experimental data obtained of full-scale length fuel rods, the results of post-irradiation examinations of a standard assembly of the Novo-Voronezh NPP, Unit 4 were used. The chosen assembly had operated during three years cycles up to an assembly mean burnup of 32.4 *MWd/kgU*. A specific feature of the given assembly was that its fuel rods contained the fuels of the various parties production, differing in the initial density value (low density 10.30 - 10.35 g/cm^3 and high density 10.50 - 10.60 g/cm^3). 18 fuel rods were punctured in order to determine the FGR. For three fuel rods a comparison between experimental and calculation results was carried out. On the basis of this results it can be concluded that the PIN-micro code describes sufficiently well the processes in VVER fuel rods. More detailed analysis of post-irradiation fuel rods examination of this assembly is given elsewhere.

The calculation results of extended burnup VVER-440 and VVER-1000 fuel rods in four- and three-year refuelling cycle, respectively, were presented. The thermomechanical characteristics were calculated for the most thermally-loaded fuel rods. A conservative model of fuel relocation was used. The relation between input parameters (fuel density, geometrical dimensions etc.) was also chosen so that one could get the most conservative estimates.

For VVER-440 (initial gas pressure 0.5 *MPa*) the following two most conservative variants were chosen: first, maximum possible initial gap of 0.27 *mm* and minimum possible initial fuel density of 10.4 g/cm^3, and second, minimum possible initial gap of 0.12 *mm* and maximum possible initial fuel density of 10.8 g/cm^3. The attaining of the highest fuel temperatures and relatively high FGR and internal gas pressure are typical of the first variant, while for the second one the specific features are early fuel-cladding gap closing and occurrence of fuel-cladding mechanical interaction. The variant calculations performed show that for a fuel rod of this design the attainment of dangerous conditions (such as high fuel temperature > 1700°C, rapid FGR or high degree of fuel swelling capable of causing untimely damage of fuel rod) is not expected.

For VVER-1000 three-year refuelling cycle fuel rods (He-filling gas initial pressure 2.0 *MPa*, initial diameter of central hole of 2.4 to 2.6 *mm*, minimum size of diametrical gap is 0.15 *mm*) variant calculations were performed too. The calculation results show that for a fuel rod of this design the attaining of dangerous conditions is not expected. For example a maximum attainable fuel centreline temperature has the value of 1815°C (1900°C with Xe oscillation) and this is yielded by the most conservative estimate and only in the very beginning of irradiation.

As is known, a certain increase of helium fill gas pressure improves fuel rods thermal characteristics. In order to optimise initial pressure value for VVER-440 fuel rods with four-years cycle, a number of variant calculations were performed. The results of this optimisation give the optimum initial gas pressure value of 0.5 - 0.7 *MPa*.

The results presented on calculated and experimental data comparison show that the PIN-micro code predicts quite well VVER-type fuel rod behaviour at least up to burnup of about 40 *MWd/kgU*. The analysis of fuel rod behaviour of four-year refuelling cycle VVER-440 and three-year refuelling cycle VVER-1000 reactors shows that fuel rods of this design are only capable of achieving rather moderate levels of fuel temperature, FGR and internal gas pressure even if conservative estimates are used. However, to stimulate behaviour of this type fuel with burnups above 40 *MWd/kgU* additional experiments and calculations are required, first of all towards improving FGR and fuel swelling model.

3.6. Ageenkov A. T. et al. Investigation of gaseous phase composition of the Novovoronezh NPP FRs. *Atom. Energ.* **40**, p. 203 (1976).

10 FRs of two FAs, irradiated at the second unit of the Novovoronezh NPP during 190 and 500 EFPD, at linear rating 155 and 142 *W/cm*, and cooled after discharge 2 and 1 year, respectively are investigated. The average fuel burnup was 7.5 and 19.1 *MWd/kgU*, respectively. The calculated tritium and Kr-85 accumulation, taking into account the decay, for the first FA is 0.1 ± 0.01 and 2.1 ± 0.2 *Ci*/FR, respectively; for the second FA is 0.24 ± 0.02 and 5.3 ± 0.4 *Ci*/FR, respectively.

The experimental method is based on the cladding puncturing and determination of the collected gas composition. The experimental data on gas amount and composition in 10 FRs of 2 FAs are presented. The comparison between experimental and calculational results show, that in the gaseous phase of the Novovoronezh NPP FRs negligible amount of tritium and Kr-85 is released (0.12 - 0.27 and 0.06 - 0.28%, respectively). This insignificant FGR is due to the relatively lower irradiation temperature. The referenced data [1, 2] show, that significant FGR begins at the temperature of columnar crystal UO_2 growing (> 1600°C). Gabeskirija V. et al [3] have investigated the Novovoronezh NPP fuel, and they have not detected columnar grains in fuel structure. This result shows, that the fuel temperature is significantly lower (< 1400 - 1500°C).

In this way the gaseous phase composition of FRs with burnup 7.5 and 19.1 *MWd/kgU* is investigated. The gas amount under cladding is 14.5 - 21.1 $n.cm^3$ (at 95% statistical interval). The gaseous phase components (over the volume) are as following: He (65 - 95%) and hydrogen-tritium (< 32%). Tritium exists in FR mainly as a water (99%). From unheated FRs (40 - 50°C) $(2 \pm 8) \cdot 10^{-4}$% tritium and 0.04 - 0.13 % Kr^{85} from total accumulation are released. At heating up to 200°C 0.05 - 0.1% tritium and 0.1 - 0.2% Kr^{85} are released.

See also abstracts:

2.1, 2.4, 2.5, 2.10

4.5, 4.6, 4.11

5.2

6.1, 6.5

7.6

4 Fuel Failures

4.1. Pazdera F. et al. Application of the PES-PEPA expert system at the Dukovany nuclear power plant. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russia, May 1992.

The report summarises practical experience obtained by the application of the expert system PES-PEPA developed fro VVER-440. type V230/V213 nuclear power plants. As an example, it shows its ability to identify, independently on the personnel activity, the presence of a damaged FE in the core, to optimise transients with the aim to limit radioactivity releases and to assess burnup of the damaged fuel assembly.

Since 1990, a simple Fuel Expert System has been in operation at all units of Dukovany NPP which enables to:

- optimise transients with the aim to minimise the probability of damage to fuel elements as a result of pellet-cladding interaction;
- identify independently on personnel activity, the beginning of dehermetisation of the FE cladding;
- quantify the number of damaged FE's and the degree of their damage by evaluating the measured values from on-line gamma-spectrometer monitoring primary coolant;
- optimise transients with regard to total released radioactivity from the damaged fuel element;
- estimate probable burnup of damaged fuel elements.

The results of Fuel Expert System application at the Dukovany NPP for the first and second units for the period Jan. 5, 1992 - March, 15, 1992 are given.

The determination of leaking a fuel element is based on two independent criteria:

a) stationary state primary coolant irradiation analysis - obtained by the KGO (cladding tightness check) procedure results (i.e. after more than tree days of stabilised unit performance characterised by zero output and pressure gradient).
b) identification of the spikes of Xe-133, Xe-135, I-131, I-132 at each transient.

Possibilities to influence the amount of radioactivity released during operation transients is also given.

The influence of an optimisation (during power changes) on the irradiation situation is demonstrated.

The results show:

- relatively small radioactivity release from the damaged FE to the coolant during power decrease. In accordance with the prediction, the influence of the chosen power rate change was negligible;
- significant decrease of radioactivity release as a result of the proposed power rate change for the transient. The actual decrease was estimated as an order of magnitude, which was in compliance with the predicted value.

The Fuel Expert System which predicts the probability of fuel damage and minimises radioactivity releases into the primary coolant has been developed and verified in operation of the Dukovany unit.

The most important result is the possibility to reduce the radioactivity of gaseous releases at least by 10% by regulating unit s power, and by at least 20% when employing primary coolant treatment.

4.2. Dubrovin K. P. et al. Data on leaking fuel assemblies in LWRs operated in the former USSR. Proceedings IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russia, May 1992.

The paper presents data on the quantity of leaking fuel assemblies discovered in all the operating NPP units in the USSR with VVER-440, VVER-1000, RBMK-1000 and RBMK-1500 reactors from the moment of start-up and up to the scheduled refuelling in 1991. The failure rate of fuel assemblies and fuel rods manufactured in different periods of time and operated in different NPP units is considered. A brief discussion is also given of post-irradiation examination performed to investigate the reasons for relatively poor reliability of fuel assemblies produced during period of transition from the extrusion technology of VVER-440 pellet fabrication to a technology of individual pressing. Some data on primary circuit coolant activities for VVER reactor NPP units is given .

The results presented in the paper indicate a satisfactory reliability achieved with modern VVER-440 and VVER-1000 fuel.

The given data shows that reliability of VVER-440 fuel assemblies manufactured in 1972-76 was poor. Failure rate of fuel assemblies produced in that period reached 6%. This was caused by a change in the pellet fabrication technology.

Historically is so happened that up to 1972 fuel for VVER reactors was produced in the USSR according to a unique procedure which differed from that applied all over the world. According to this procedure small amounts of oxides of metals (other than uranium) and plastifier were added to uranium dioxide powder and the resulting mixture was subjected to extrusion forming a long rod having central hole. The rod was then cut into extruded pellets about 30 *mm* long.

In 1972 an technology was developed for producing pellets about 12 *mm* long by means of individual pressing (without other oxide additions). In 1973 in the course of the scheduled refuelling 69 and 100 experimental assemblies with pressed pellets and 75 and 26 assemblies with extruded pellets were loaded at Novovoronezh NPP (NVNPP), Units 2 and 3, respectively.

It is shown that VVER-440 fuel assemblies produced in 1977 and later provided a satisfactory level of reliability. On the average, number of leaking assemblies was less than one percent and the number of assemblies that had to be discharged prematurely - less than 0.2%.

The data on leaking VVER-1000 fuel assemblies indicate a satisfactory reliability of VVER-1000 fuel. The average fuel rod-failure rate is 0.008%.

The data presented indicate a satisfactory reliability of the modern VVER fuel and that the RBMK fuel rod failure rate should be reduced, particularly through optimisation of the operating conditions.

4.3. Velyukhanov V. P. et al. Features of operating a VVER reactor core containing leaking fuel rods. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russia, May 1992.

The paper addresses criteria of performing in-service inspection of the leaking fuel rods (FR) of VVER-type reactors including assessment of the limiting content of iodine radionuclides in the primary circuit.

Experience of operating fuel assemblies, assessment of fuel assembly (FA) reliability indicators and dynamics of the primary coolant activity change during normal core charging have been summarised in the paper.

During the hole period of VVER reactors operation no unplanned reactor shutdown on exceeding the limiting coolant activity level has been reported. However, there were cases involving unscheduled fuel assembly withdrawal on a failure detected during reactor planned preventive maintenance outage.

Experience of operating reactor core containing leaking fuel assemblies as well as after-burning use of the leaking fuel assemblies is described in the paper.

The paper presents statistics on leaking fuel assemblies for VVER-1000 reactors during the whole operation period and dynamics of the primary coolant activity change during reactor cycle.

A number of examples involving operation of leaking fuel assemblies at the soviet NPPs are considered.

The authors discuss the possible causes of cladding leakage, namely, violation of the technical specification (for example as to power increase rate), inhomogenity of power density distribution, overheating of fuel rods as a result of deposits on the spacer grid etc.

In accordance with "Nuclear Safety Rules for NPPs" only < 0.2% of the fuel elements with the "gas leaks" and < 0.02% of the leaking rods with "pellet-water interaction" are permitted. Here the primary coolant activity should not exceed $1.5 \cdot 10^{-2}$ *Ci/kg* as to I isotopes. However, the first critical level (current level of the operational safety for VVER type reactors) is significantly lower, i.e. $1.0 \cdot 10^{-3}$ *Ci/kg*. Activity level is determined from the cleaning water flow rate of 30 *t/h* and 20 *t/h* for VVER-1000 and VVER-440 respectively. On exceeding the second level a unit has to be shut down.

Main causes of the VVER-reactor fuel leakage are:

1. Design and process deficiencies of fuel rods and assemblies.

   These causes mostly relate to VVER-440 FRs and FAs and to the first stage of operating this reactor type (1970s). One of the mentioned causes of fuel rod cladding cracking during operation is zirconium alloy hydrogenation from inside as a result of fuel pellets high humidity. Nowadays this cause is eliminated as a result of improved technology of fuel production. In some cases cracking cladding was the result of fretting-corrosion, caused by defects of spacer grids. Defects were observed as a result of a wrong mould of a fuel column (fuel pellet insertion with different fuel enrichment). All these deficiencies were eliminated and they had not been observed during VVER-1000 FA operation.
2. Latent process defects.

   Transition to flow-line automated production of fuel rods and assemblies of the type, in principle, has a potential for omitting the latent defects in FR cladding (inclusions, cracks) or weldments. However, experience of VVER fuel operation points to extremely limited number of FA with such defects.
3. Power bursts.

   More severe regimes for FR are those, accompanied by power bursts. Inspite of high plasticity of Zr + 1%Nb alloy after high fluence (d = 15% after $F = 10^{22}$ N/cm^2), fuel rods sensitivity to the transients of operation is caused by zirconium alloys inclination to brittle corrosion cracking in the presence of cracking stresses in corrosive medium (in this case it is the products of uranium dioxide fission - iodine, caesium, tellurium etc.)
4. Carbon-containing impurity ingress into the core.

   In 1990 during the operation of the 16h fuel load of VVER-440 reactor (60 effective days) at Kola NPP, unit 2, overheating of practically all the fuel assemblies of the third year of operation was detected. It was caused by coolant flow rate decrease (20 - 30%) in the FA.
5. Foreign objects in the core.

   During operation of Zaporozhye NPP, unit 4 (VVER-1000) in the fourth fuel charge (1991) the highest permissible value of the outlet water temperature was observed on several FA. It was considered to be a result of the flow rate decrease through the first loop. After that the reactor power was reduced up to 60%.

It can be concluded that:

1. Fuel operates rather reliably. The cases of FR depressurisation make up for VVER-440 - 0.005%-0.010%, for VVER-1000 - 0.005% - 0.009%. As a rule depressurisation has the character of gas leak.
2. The main causes of FA depressurisation involve non-optimum conditions of operation, connected with multiple defects of the primary and secondary equipment and violation of the technical specifications.
3. The possibility of after-burning use of fuel with micro leaks in the fuel rod claddings with the primary circuit water activity limitation was revealed.

4.4. Grachev A. F. et al. Experimental study of the VVER fuel behaviour in research reactors and shielded hot cells. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russia, May 1992.

Some experimental results of post-irradiation examination of VVER-1000 fuel are given . A complex of technical means and methods, which are used for investigation of all stages of fuel defect appearance and development, is presented. This complex comprises the SM-2, MIR and VK-50 reactors and the hot laboratory.

The comprehensive failed fuel investigation program involves the following types of fuel failure: changes of fuel element and assembly geometric dimensions and shape due to non-uniformity of irradiation strain, local thickness reduction of the fuel assembly wrapper or the fuel element cladding, caused by fretting corrosion, spacer grids geometry changes (crumpling, damage), formation of thick deposit layers on the fuel element cladding, local and extended corrosion areas, cracks and, finally, cladding perforation as a possible place for a fission products release.

Detection of defects begins with visual inspection of the fuel assembly wrapper (for instance, those of VVER-440), bundle and single fuel elements. Visual devices with a magnification of 30 are used for this purpose. They are connected with photo-, videocameras and computers. In most cases surface anomalies are successfully detected already at the visual inspection stage. As a rule, they represent external cracks, corrosion spots, deposit, etc.

An external damage of the fuel element claddings is often observed in the places where the cladding comes in contact with other objects. In this case the anomaly and defect size can be defined by means of profilometry.

Cracks, cladding hydrogenation, including those, connected with secondary defects, can be detected by the eddy-current flaw detection.

Cladding defects, that make fuel release from the fuel element possible are detected by gamma-scanning.

After the defect is detected, the program of further investigation is worked out in accordance with its type.

The results of this methods permits evaluation of cladding thickness, level of its hydration and oxidation, fuel-cladding gap size, fuel-cladding interaction. The fuel and cladding structure gives an information on the temperature distribution over the fuel element volume.

The MIR reactor with the core of 1000 *mm* height comprises several loop facilities. Their type and parameters are determined by the experiment tasks.

The SM-2 reactor with the core height of 350 *mm* comprises 3 water cooled loop facilities. The VP-2 and VP-3 loop facilities are used for testing of power reactor fuel elements.

4.5. Dubrovin K. P. et al. Some results of post-irradiation investigations of VVER-1000 unsealed fuel assembly. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russia, May 1992.

The results of post-irradiation tests of a fuel assembly are given. The fuel assembly is from Unit 1 of the South Ukrainian NPP. 300 μm depth local corrosion of cladding was identified.

In the beginning of South Ukrainian NPS operation (1 unit) a graphite bearing of the main circulating pump (MCP) failed. When MCP was repaired, the unit had been operating for a long time on a power close to nominal value. After eddy-current examination one-third fuel rods were identified as those having the surface defects of the cladding.

The visual inspection of the damaged fuel surface revealed:

- white contrast spots;
- dark grey graphite deposits of about 0.3 *mm* thickness, partially covering the white nodular formations;
- graphite deposits 0.5 *mm* thick in the bottom of fuel rods.

4.6. Dubrovin K. P. et al. Results of post-reactor investigations of VVER-1000 untight fuel assembly. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russia, May 1992.

Results of post-reactor investigations of VEER-1000 untight fuel assembly are given. The research results of structure, composition and properties of cladding are presented.

Growth of the iodine radionuclides activity in the coolant was observed during commissioning to rated power of the reactor of the 5th block at the Novovoronezh NPP after 3-weeks operation with 50% power. The untight fuel assembly found by the failed fuel detection (FFD) system was delivered to RIAR and investigated in a material science laboratory. The fuel assembly had been operated for 489 eff.days (4 and 5 fuel cycles) up to burnup 21.7 *MWd/kgU*. The maximum heat load on the fuel rods did not exceed 240 *W/cm*.

Maximum increase of the length and decrease of the fuel rods diameter made up 11.4 *mm* (0.3%) and 0.04 *mm* (0.41%), respectively. A fraction of inert gases (Xe and Kr) released from fuel did not exceed 2.3% of their quantity accumulated in the course of operation.

A uniform oxide film, 5 μm thick, was generated on the outer surface of all fuel rods. Hydrogen content in claddings did not exceed 5.10^{-3} % *mass*.

Mechanical properties of tight fuel rod claddings whereat a sufficiently high level. Uniform relative elongation changed within the range 3.5 - 4% and the total relative elongation equalled 16.0 - 25 %.

Maximum swelling of UO_2 was 1.2%. Absence of growth in U dioxide grains and negligible number of released gas fission products evidenced of the fact that the maximum temperature in the centre of the fuel did not exceed 1500°C.

Slight hydration and oxidation from the side of the inner cladding surface occurred around the lower defect (mark 3580 *mm*).

Intensive local hydrogenation and oxidation of the cladding from the inner surface side took place in the area of upper destruction at mark 500 *mm*. Hydrogen concentration in this region rose up to 0.39%. The thickness of the continuous hydrides layer achieved 250 - 300 μm.

4.7. Bibilashvili Yu. K. et al. Effect of inner surface preliminary oxidation on stress corrosion cracking susceptibility of Zr - 1% Nb tubing. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russia, May 1992.

Internally pressurised claddings of VVER-1000 fuel were tested under iodine SCC conditions at 653 K in a laboratory unit equipped with an automated acoustic-emission system. Zr - 1%Nb claddings were pre-coated with zero 0.5 - 10 μm thick. Cladding time to failure is given as a function of oxide layer thickness. Oxide film growth is shown to lead to a sharp reduction of time to failure of claddings subject to iodine corrosion. Areas of failure were studied fractografically and diagrams of recorded acoustic emission (AE) signals that accompanied major stages in crack evolution are interpreted. Also discussed is the influence of cladding inner surface oxidation on mechanism of SCC resistance degradation with fuel burnup.

Internally pressurised tubular Zr - 1%Nb specimens were investigated:

1. Tubes 150 *mm* long, 9.15 *mm* outer diameter and 0.7 wall thickness, cut from unirradiated VVER-1000 fuel cladding were subjected to air oxidation in a resistance furnace at 773 K. The tubes were arranged horizontally in a specially designed container.
2. Claddings as oxidised were grooved in the middle over 40 *mm* length to produce a higher stress area. The average thickness of the thinned cladding was 350 +/- 15 μm.
3. The end plugs were electron beam welded. To assure the needed quality of the welds the oxide layer near the tube ends at the inner and outer surfaces over 5 - 6 *mm* length was removed.
4. The specimens as pre-evacuated were filled with an inert gas (argon). Iodine as loose crystals was inserted in the amount of 0.2 *mg/cm²*.

The oxidation temperature of 773 K was chosen based on time of specimen holding that is needed for the maximum oxide layer of 10 μm to form. Some specimen were oxidised at T = 673 K. The difference in the oxidation temperature did not affect the results of cladding testing under SCC condition.

According to the appearance the resultant ZrO_2 can be divided into three types:

1) a black strongly adherent film up to 2 - 2.5 μm thick;
2) a grey more loose film likely to contain a larger amount of defects from 2 to 10 μm thick;
3) a spotted film more than 10 μm thick with light spots 1 - 3 *mm* in size against the background of the oxide.

The tests indicate a sharp reduction in the cladding time to failure from 56 to 4 *h* with an increase of oxide layer thickness at the inner surface. To identify the cause of this cladding behaviour in an iodine environment the use is made of the concept of failure time as a sum of two items: t_1 is the time of crack nucleation and t_2 is the time of its propagation.

The conclusions are:

1. Internally pressurised VVER fuel simulators were tested under iodine SCC conditions. Zr - 1%Nb claddings had a ZrO_2 layer of different thickness at their inner surface. Using AE-method and fractographic analysis it is shown that a crack has a three-stage propagation in cladding thickness beginning with transgranular shear in iodine the time of which constitutes the major fraction of the hole time until loss of tightness by cladding.
2. An increase of oxide film thickness at the inner cladding surface from 0.5 - 1.0 μm results in a sharp reduction of time to failure both in iodine and inert gas condition.

4.8. Panov E. A. et al. Analysis of trends in fuel rod depressurisation and determination of "gas leak" and "pellet-water interaction" type failures using irradiation monitoring techniques of fuel rod leak tightness. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russia, May 1992.

Analysis of fuel rod failures in the Light Water Reactor operation is presented. Analysis includes the mechanism of formation and development of fuel rod cladding failure until through-wall defects appear (welding defects; inner hydrating defects; pellet-cladding interaction; crude deposit - intensified corrosion) as well as factors that determine defects propagation after fuel rods depressurisation (metal condition in the vicinity of defect determined by the mechanism of formation and propagation defect; operational transients; degree of core cooldown after depressurisation during preventive maintenance).

Possibilities of in-service monitoring of fuel rod trough-wall crack propagation using normal tools of cladding back-tightness monitoring are addressed and used in the course of analysis. Characteristics and values are presented for irradiation parameters for fuel assemblies during propagation of defects with different degree of rod depressurisation, including "gas-leak", cladding crack and "open pellet-water interaction" with potential particulate fission product release from the damaged rods as well as after formation of recurring defects.

Based on experimental data on specific activity of different iodine isotopes in the primary coolant, a mathematical model to analyse defect propagation trends has been developed. The model describes the rate of radionuclide release from depressurised rods and the rate of nuclear fuel fission processes in the vicinity of defects. The model and results of analysis are illustrated by experimental and statistical data on depressurisation of VVER (PWR) and RBMK (BWR) reactor fuel rods.

It will be noted that research of the activity fission products on primàry coolant are contemporary for learning of FR failure process in constant and change conditions of operation (by the reactor power changing).

In particular, such measurement was made on NPP "Loviisa" and showed two manner of FRs failure development: 1 - quick increasing of activity and setting of activity level smaller of peak activity; 2 - quick increase of activity in 5/10 of magnitude and setting of activity level smaller of peak activity for 15/45 o'clock. It will be noted that sometimes the iodine and the gas activities were increase quick for time, in other case - the iodine activity was increasing only later the delay period.

In conclusion is added, that the FRs with the gas leak defects may be quite operating, that is provided of the them state monitoring possibility and of the defect development prediction possibility operation choosing. The FA operation with the greater cladding defects is limited and them unloading is as a rule regulated of standardised documents.

4.9. Shestakov Yu. *M.* et al. Operation and irradiation monitoring of the water cooled reactors with faulty fuel rods. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russia, May 1992.

Typical operation peculiarities of fuel assemblies (FAs) with faulty fuel rods of Soviet pressurised water reactors (VVER) and one-circuit pressure tube boiling water reactors (RBMK) are addressed. Operation of FAs with faulty fuel rods is considered for steady state plant operation, for transients and shutdowns. The methods of irradiation monitoring of rod cladding back-tightness in the course of operation, irradiation leak-tightness criteria and the creed for discharging failed fuel rods from reactor are considered. General reliability indicators describing fuel element cladding leak-tightness in the core and criteria of current plant irradiation safety are presented. Possibility to solve the problem of choosing optimum operation regime in the presence of failed fuel rods in the core is considered.

4.10. Scheglov A. S. Fuel cracking influence on the temperature field in the power reactor fuel rod. *Atom. Energ.* **73**, p. 158 (1992).

The initial radial gap between pellet and cladding (75 - 130 μm for VVER-1000 FR) is changed during the reactor operation. These changes are due to the decrease or increase of fuel ràdius (55 μm) through thermal expansion or cladding radius (8 μm), respectively, decrease by cladding elastic deformation (3 μm), and fuel cracking (10 - 50 μm). In the parenthesis the calculational results for VVER-1000 FR with initial radial gap 120 μm, linear power 403 *W/cm* at the first days of operation are given. The calculations are performed, using the PIN-04M code.

In this paper an evaluation of influence of the azimuthal fuel cracks on the temperature field in a given axial FR node is made. The calculations are carried out, using the MRZ code [6]. The calculational results for maximal and average temperature dependencies on the radius of crack inner surfaces are presented. Some results, obtained by PIN-04M code are presented too.

4.11. Reshetnikov F.G et al. Problems of development of VVER-1000 fuel rods for load follow NPPs with extended burnup. *Atom. Energ.* **64**, p. 258 (1988).

On the basis of the reactor operation experience and the reliability studies it can be concluded that the most important reason, limiting the FR operability is the thermomechanical interaction between fuel UO_2 pellet and Zr alloy cladding, arising during reactor transient regimes. The physical basis, identifying the FRs behaviour at transient regimes, is the fast changing of the temperature fields. This changing, under certain conditions leads to the arising of the additional high tensile stress in the cladding, and to accumulation of additional damages in it.

To predict the cladding damages and destruction a method is developed and investigations of the mechanical characteristics of Zr + 1%Nb alloy are performed. These investigations are carried out under the following conditions: up to 1000 *h* at tensile stress, typical for transient regimes of operation, as well as at iodine concentration on the inner surface of samples 0.1 - 0.2 mg/cm^3. Some data on the strength, characterising corrosion damage accumulation depending on the time, are given too. An applied method for crack growth by stress corrosion cracking, including the stage of crack initiation on the inner surface of the initially undamaged cladding is developed too. The calculational results, applying this method are compared with the experimental ones. The experimental results on the dependence of the iodine yield from the CsI resolving v/s He pressure under the cladding are given. Up to 350°C this dependence is linear, behind this value (at 360 - 380°C) iodine accumulation abruptly increases.

Depending on the influence of on the FR behaviour the transient regimes can be subdivided into three basic groups: power ramping, power cycling, power reconstruction. The most hard ones are the power ramping regimes. The calculational results (the cladding radial tensile stress depending on time, sec) for VVER-1000 using the START-2 code are presented.

To study the FR operational characteristics and to evaluate the permissible power deviations the calculational investigations of the cladding strength at the power reconstruction regimes after short-time reactor shutdown at different time point (the middle of the second year, the beginning, the middle and the end of the third year) during the fuel lifetime (3 years), as well as at different peaking factors K (K = 1.3, 1.2, 1.12) are carried out. K is the relative local linear rating for the given steady-state level during power regulation and suppressing the xenon oscillations. In all cases the maximum cladding loading appears at the third year of operation. At K = 1.3 the cladding is damaged with cracks up to 37 μm after three fuel cycles. More optimal regulating by K = 1.2 decreases the damage degrees, so that the crack depth do not exceed 4 μm. By K = 1.12 the maximal stress do not exceed 120 *MPa* and the claddings do not accumulate visible damages for given regime. But for the basic NPP the amount of such regimes can reach up to 20 - 30 per year, and for the manoeuvre NPP is more than 400 per year.

The most typical transient regimes of the manoeuvre NPP are characterised with the big tensile stress (about 50 - 100 *MPa*) arising in the cladding. The time of stress affecting is about 9 - 12 *h*. By large number of such regimes the cladding will be operated during long time, about (12 ÷ 15).1000 *h* under tensile stress for three-year cycle. This can influence the corrosion stability of the Zr alloys on the coolant side. To provide the reliable operation of VVER-1000 at extended burnup and load follow regimes, further improvement of the fuel rod construction is needed. At the present the work in this field is performed in two directions: the first one is using the cladding with enhanced stability in relation to cracking corrosion; the second one is decreasing the mechanical loading on the cladding. In connection with the first direction, for the cladding the bimetallic tubes with inner pure Zrl layer is used. The calculational distribution of the stress over the FR radius for the standard cladding and for the cladding from the bimetallic tubes in the case of three most hard transient regimes is presented. The second direction of FR improving includes the applying of oxide fuel with enhanced creep and plasticity characteristics. The investigations of the influence of different additions (Fe-, Al-, Nb-oxides) on the mechanical properties of UO_2 are carried out. The experimental results of the temperature dependence of the plastic deformation for different fuel pellet composition is given.

To study the VVER-1000 FRs behaviour at extended burnup (more than 50 *MWd/kg*), investigations at the experimental reactor MR are carried out. At the present the 21st pilot FA is tested. The three FAs had the maximal burnup over 75 *MWd/kgU*. The coolant pressure was 16 *MPa*, the coolant outlet temperature was 320°C. The FA No. 1 was especially tested with the aim to study the influence of power ramping (moving FA from the periphery to the centre of reactor core) on the FR operability. The initial linear power was 450 *W/cm* (FA was in the periphery of MR). During the operation the linear power is decreased (250 *W/cm* maximal linear power after 17800 *h*). After that this FA was removed in the centre of MR at maximal linear power 470 *W/cm* during 6 month. It was discharged after 6 month and it was tight. The PIE results show the following: the average fuel rod elongation is 1.2 *mm* (initial fuel stack length 1m), fuel rod diameter increased by 0.01 - 0.03 *mm*. Fractional FGR was 30% for the FRs with minimal linear power and 45% for the FRs with maximal linear power. The initial He pressure was 2.45 *MPa*. The oxide film depth on the outer surface of cladding did not exceed 3 μm. The hydrogen amount in the irradiated claddings was about 0.005% (*mass*). The mechanical properties of claddings before and after irradiation are presented.

4.12. Shtavelin V. *M.*, Kostachka A. V., Kuznetsov A. A., Golovnin I. S. and Bibilashvily U. K. Intra-reactor investigations of reactor materials friction characteristics. *Atom. Energ.* **61**, p. 175 (1986).

An equipment is developed and tested for measuring the friction characteristics of surfaces in contact inside the reactor. In the case the friction between UO_2 and compound of Zr + 1%Nb in the irradiation process in an experimental reactor is measured.

As fuel samples are used assemblies of UO_2 with outer diameter 7.5 *mm* and inner hole diameter approximately 1.5 *mm* with height ~10 *mm*. The density is 97 % from the theoretical and the diameter of the grains 15 ÷ 25 μm

The samples from Zr + 1%Nb have the form of a plate (80 × 12 × 2 *mm*)

All experiments are carried out in He atmosphere. In the process are registered the friction force and the shear stress and also the friction contact temperature. The method of defining the friction coefficient requires beforehand strain treatment of samples during 0.5 ÷ 25 *h*. The nominal contact pressure changes from 5 to 30 *MPa*, the relative shear stress velocity is $1.1 \cdot 10^{-6}$ and $1.3 \cdot 10^{-3}$ *m/s*. Figures 3 and 4 show the strong dependence of the friction force coefficient from the neutron density flux. The bigger the shear stress velocity at high density of neutron flux is, the faster is the increase of contact surface and respectively the faster decreases the pressure which leads to decrease of creep velocity and diminishing of the contact surface.

It is proved also an increase of the friction coefficient with the increase of the fluence, which is connected with the bigger strength of th Zr compound.

In the experiment is observed an unexpected diminishing of the shear stress coefficient.

4.13. Pshenichnikov B.V. FGR from untight FR after reactor shutdown. *Atom. Energ.* **39**, p. 400 (1975).

In this paper a mathematical model for FGR from untight FR to coolant after reactor shutdown is described. An evaluation of the quantity of gas under cladding is made. After reactor shutdown at the pressure decrease from operational to atmosphere more than 95% gaseous fission products are released in the coolant practically promptly following the pressure in reactor vessel.

See also abstracts:

2.1, 2.9, 2.12

3.2

5.2, 5.3

6.1, 6.2

7.16

5 Pool-Side Inspection of Spent Fuel

5.1. V. G. Dvoretzky, S. V. Pavlov, A. V. Mestnikov. Poolside Inspection of Fuel Rods From Experiments in Research Reactors. Proc. IAEA TCM, Lion, France, 21-23 Oct. 1991, IAEA-TECDOC-692

A complete set of equipment and procedure providing for inspection of fuel rods at the poolside of the research reactor are described. The complete equipment permits application of such methods as visual inspection, eddy-current flaw detection of claddings, geometry dimension measurements and leak testing.

A loop facility and a special test bench for investigation of irradiated mock-up fuel rods of VVER type in the cooling pool of the MIR loop research reactor, is used.

The following non-destructive inspection methods are applied:

- visual inspection with the TV-camera;
- eddy-current flaw detection of fuel rods;
- claddings with the encircling coil section;
- measurement of the fuel rod cross-section;
- measurement of the fuel rod length;
- leak testing of fuel rods from the availability of water under the cladding.

More than 50 mock-up fuel rods of the VVER reactor are inspected during operation of the test bench. Among them there are several fuel rods with abnormal shape changes. The defectogram of one of these fuel rods is presented.

The trial operation results indicate good serviceability of the benches and their separate units.

Some methods can be used simultaneously in order to reduce inspection time.

In the future, additional methods are supposed to be used at the test bench, such as: gamma-scanning, sampling of deposit from the surface of the fuel cladding gap, measurement of pressure of gaseous fission products etc.

The quantitative data for operating time and personnel estimates for the test bench are given. No other quantitative data are presented.

5.2. H. Lindroth, R. Terasvirta. Post-Irradiation Examination of High Burnup Fuel and Leaking Fuel from the Loviisa Nuclear Power Station. Proc. IAEA TCM, Lion, France, 21-23 October 1991, IAEA-TECDOC-692.

The high burnup program includes poolside inspections of three fuel bundles irradiated to 38...40 *MW/kgU* with maximum rod burnups 43...48 *MW/kgU*.

Several fuel rods are removed from the assemblies and subjected to dimensional measurements and gamma measurements for fission gas release assessment. Four fuel rods are selected for hot cell examinations in Studsvik, Sweden. to have more detailed information on fuel condition and to verify the capability of the poolside measurement equipment. Their average burnups are from 42.9 to 45.9 *MW/kgU*. The average rod growth is 0.4% of the total length.

The cladding creepdown of the fuel rods and their ovality are determined by the fuel rod profilometry. The average creepdown of 26 rods is 67 μm and the average ovality of 19 rods is 41.8 μm.

The cladding creepdown of rods irradiated for four cycles is higher than of rods irradiated for three cycles. The ovality is almost similar for the three and the four year cycle rods.

In the hot cell examinations the average cladding creepdown varied from 22 μm to 60 μm. The difference between hot cell and poolside results is caused by the difficulties in the poolside measurements.

According to both poolside inspection and hot cell results the gap is closed or almost closed in the central regions of the rods. FGR of examined rods is estimated below 2%.

The metallography and the ceramography showed that the inside oxide thickness varied from 0 to 13 μm and the outer oxide thickness is between 1 and 4 μm.

The hot cell examinations showed variation of grain size from pellet to pellet and fuel with some large irregular porosity. The fuel temperatures are low.

The total number of leaking fuel assemblies at Loviisa NPS is 17. In two assembles some peripheral fuel rods are pulled off to reveal the failed rods. These individual rods are inspected visually and their creepdown, ovality and diametrical gap between pellet and cladding are determined.

The post-irradiation examination results of three high burnup assemblies show that with the power histories at Loviisa NPS, the rod average burnup of 48 *MW/kgU* can be reached with large margins to the fuel design limits.

The visual inspection shows that nine of the ten assemblies had a deformed upper grid. Visible failure in cladding is found from eight assemblies. It is caused by design error in the upper grid of the fuel assembly. Although the analysis has shown that this error does not have essential consequences to the normal or accidental behaviour of the fuel, a new upper grid, eliminating this problem has been designed by the fuel vendor.

The pool side examination of two precharacterised Loviisa fuel assemblies has started in September 1992. The examination program will include all measurements and investigations.

The investigations of the upper grid behaviour will be continued in the next year. All future failed fuel assemblies are also planned to be examined in order to possibly reveal the reasons for the failures.

5.3. P. Vesely, F. Pazdera. Technical Aspects and Economic Assessment of Poolside Inspection, Repair and Reconstitution of VVER-1000 and VVER-440 Fuel Assemblies. Proc. IAEA TCM, Lion, France, 21-23 Oct. 1991, IAEA-TECDOC-692

This paper deals with the technical aspects of eventual poolside inspection, repair and reconstitution of VVER-440 and VVER-1000 fuel assemblies. Main differences between fuel assemblies of those reactors and typical PWR's are given.

- technical feasibility and
- economic effectiveness.

Detailed discussion of technical feasibility shows that the following main problems are to be solved:

- design of equipment for possible inspection, repair of fuel and RCCA control;
- new design of VVER-440 and VVER-1000 fuel assemblies (dismountable upper parts, new design of spacer grids).

The assessment of economic effectiveness has been made for the wide range of fuel cycle parameters, including natural uranium price, enrichment cost etc. The sensitivity study was performed for possible interest rates and burnups in the range of 30 to 60 *MWd/kgU*.

Technical feasibility of necessary modifications was demonstrated for both types of VVER fuel assemblies with the respect to the possibility of their poolside inspection, repair and reconstruction.

The actual technical solution of repair and reconstruction could be very similar to that of the PWR in the case of VVER-1000 and slightly modified in the case of VVER-440. The influence of the interest rate and the burnup level on the economic effectiveness is discussed, however the most important factor seems to be the difference between VVER-440 and VVER-1000, namely the influence of the fuel assembly size. Unless the failure rate in VVER-440 will increase, there will be no economic incentive for implementation of the repair or reconstruction technology. For VVER-1000, the incentive is about the same as for western reactors.

5.4. Dubrovin K. P. et al. Investigation programme for VVER-440 spent fuel assemblies and its implementation. Paper pres. to Soviet-Finnish Sem., May 1991, Institut Atomnoi Energii (IAE), Moscow.

The paper gives the basic principles of the investigation programme for VVER-440 spent fuel assemblies. The programme provides for the investigation of assemblies whose design, materials of manufacture, method of production and operating conditions have been purposely varied. In the first instance these variations include increasing the burn-up, changing the enrichment, increasing the operating period, etc.

The post-irradiation investigations of the VVER-440 assemblies are based on the complex of the hot cells at NIIAR (Nauchno-issledovatelskii Institut Atomnih Reactorov - Scientific Research Institute for Atomic Reactors), IAE (Institut Atomnoi Energii - Institute of Atomic Energy) and the Novovoronezh nuclear power station. Novovoronezh mainly use non-destructive methods of investigation, while IAE and NIIAR make wide use of the most varied and up-to-date methods and resources of material research. The assemblies for investigation are transported in standard TK-6 units and after investigation they are put into a special skips for salvaging.

The investigation programme for spent VEER-440 assemblies involves:

1. Investigation of assemblies in support of increasing the refuelling cycle.
   The change in VVER-440 operation from a 3-year refuelling cycle to a 4-year cycle required some changes in assembly design.
2. Use of zirconium grids.
   The replacement of steel spacer grids with grids made from alloys of zirconium is being prepared with the aim of increasing neutron flux, flattening the core power profile and, as a consequence, increasing burn-up and fuel economy and reducing the cost of electric power.
3. Power cycling and power boosting.
   Investigation have recently began into increasing the operating effectiveness of power stations by boosting or varying the power. At the Kola and Rovenki nuclear power stations assemblies have been operating for several refuelling cycles at 105 - 107% of the rated power.

The paper sets out the basic directions of post-irradiation examinations of VVER-440 assemblies. These investigations aim to check possibilities and to provide support for increasing the operating period of assemblies and operation with power cycling and power boosting.

Fuel pin for 4-year cycle	
Fuel/cladding diametrical gap, *mm*	0.27
Fuel enrichment	4.4
Initial gas pressure, *MPa*	0.5 - 0.7
Burn-up in most highly stressed pin, *MWd/kg*	59
Length of refuelling cycle, *h*	28 000

5.5. Proselkov V. N. et al. Operation at increased power of the pilot FAs, consisting of FRs with increased initial helium pressure in the gap. *Atom. Energ.* **67**, p. 101 (1989).

To improve FRs reliability at extended burnup and increased power operation an introduction of FRs with increased initial helium pressure in the gap of value 0.5 *MPa* is foreseen. To proof the operability of such FRs, a 96 pilot FAs (3.6% enrichment, 0.5 *MPa* initial He pressure) are loaded in the VVER-440 reactor core of the 2nd unit of Rovno NPP during the fourth fuel cycle. The fourth fuel cycle has began at 16.08.1985 and ended at 14.04.1986. After the fourth fuel cycle the radionuclide activity (I^{131} and Cs^{134}) was less than $3 \cdot 10^{-8}$ *Ci/l*. The untight FAs (28 inspected FAs, 18 from them with increased He pressure) are not detected.

5.6. Pekka Losonen, Kari Ranta-Puska. Pool Side Inspections and Performance Evaluation of VVER Fuel Rods. ANS LWR Topical Meeting, West Palm Beach, Florida, April 1994, p.113-118.

The results of non-destructive examinations of 24 VVER fuel rods irradiated in Loviisa-2 reactor are presented. The fuel rods are precharacterised and operated for three or four cycles to burnup up to 49 *MWd/kgU*. The pool side examinations, carried out in the spent fuel storage, involve visual inspections, rod length and gap size measurements, clad outside profilometry, etc.

The examined fuel is of the newer type with compressed pellets with chamfered pellet ends and with helium prepressurisation to 0.6 *MPa*. The pellets have central hole. The cladding material is Zr + 1%Nb alloy. The rod lengths show 8.5 *mm* and 7.8 *mm* average increases or 0.33% and 0.31% respectively. At the axial mid-heights the cladding diameters decrease about 40 - 55 μm due to creep. The gap measurements show that the gap is practically closed where local burnups are above 50 *MWd/kgU*. The compressed gap ranged from 35 to 85 μm. The gamma scanning show no Cs-redistribution in the fuel. The axial gamma scans indicate that the pellets have not lost their integrity even in the hottest parts of the rods.

The examinations show that under the conditions of Loviisa the fuel behaves as anticipated at burnups up to 50 *MWd/kgU*. Cladding oxidation is not a limiting factor and rod overpressure is of no concern.

See also abstracts:

2.23

3.4

4.13

6 Fuel Improvement

6.1. Y.K. Bibilashvili et al. Towards high burnup in Russian VVER reactors and status of water reactor fuel technology. ANS LWR Topical Meeting, West Palm Beach, Florida, April 1994.

The report discusses the status of the fuel and fuel cycles of VVER-440 and VVER-1000 reactors in Russia. The work carried out, main results and perspectives are described.

The report gives the major directions of the work presently under way to further improve the design and technological solutions applied to the fuel. The main aim of this work is to improve fuel utilisation and to extend fuel burnup. The results of experimental studies in this field are presented.

The work accomplished recently has permitted to reach three-year and four-year cycles in VVER-1000 and VVER-1000, respectively. The batch averaged discharge fuel burnup for VVER-1000 three-year mode of operation is 43 - 45 *MWd/kgU*. The work is in progress to switch VVER-1000 and VVER-440 reactors to four-year and five-year cycles, respectively. At the same time, the operational reliability of fuel is improved, i.e. the frequency of fuel rod leakage is lowered and is presently at the level of 2 to $3 \cdot 10^{-5}$ per rod.

The principal characteristics of VVER-440 and VVER-1000 reactors and comparison between their fuel operational parameters are given. The failure rate of VVER-440 and VVER-1000 fuel rods, as well as the time dependence of total annual fuel failure rate of VVER-1000 are presented. The changes of averaged total Iodine activity depending on the number of VVER-1000 fuel cycles is shown.

The strategy to reach high burnups was mainly based on decreasing the core fraction of fuel reloaded and, when necessary increasing the fuel enrichment. In this report the improvements, which are under way or projected such as: FAs with Zr-alloy Spacer Grids (SG) and Zr guide tubes instead of Stainless Steel (SS) ones; reduction of Hafnium content in Zr-alloy used as materials of FR cladding, Zr SG and guide tubes; low leakage loading pattern and introduction of integral fuel burnable absorber; use of Zr-Nb-Sn-Fe alloy as FR cladding material, barrier cladding and pellet with additives; increase of specific amount of UO_2 in FR through optimisation of fuel pellet and cladding geometrical parameters; radial fuel enrichment shaping in FA and use of reconstitutable FAs are reviewed too.

To experimentally confirm the feasibility of VVER fuel operation at high burnups a large scope of activities is under way in Russia relevant to testing experimental fuel rods in research rectors MR and MIR. In MR reactor a large amount of lead test assemblies with fuel rods - prototypes of VVER-1000 fuel rods(FR) - were irradiated to burnup significantly higher than the design one of commercial fuel rods. The results of post-irradiation examination (PIE) of one such fuel assembly (FA) are discussed. This FA has reached the average burnup of 70 *MWd/kgU* (peak pellet - about 93 *MWd/kgU*).

Further work relevant to nuclear fuels is aimed at high fuel utilisation and at further increase of fuel rod lifetime to an average burnup of 50 - 55 *MWd/kgU*.

In Russia a large scope of experimental work was carried out to study the behaviour of fuel prototypes irradiated in research reactors to burnups significantly (a factor of 2) higher than the design one of commercial reactors. PIE of a large number of fuel rods of this type shoe that the design and technological solutions relevant to fuel rods and developed for commercial reactors are correct and can well provide for the fuel burnup of 60 *MWd/kgU* and higher.

6.2. Borzakov A.B. et al. Reliability analysis of VVER-1000 fuel rods and fuel assemblies during 3-year cycle. *Atom. Energ.* **74**, p. 482 (1993).

In this paper an analysis of fuel assembly reliability in the case of switching VVER-1000 to three-year cycle is made. Systematic information about the main characteristics of fuel assemblies during transient and steady-state regimes of three-year cycle is given. The time dependence of the number of defective fuel assemblies for Unit 1 of the Kalinin NPP (from 1985 to 1990), as well as for Unit 5 of the Novovoronezh NPP (from 1980 to 1990) is presented. These data were based on the radiochemical analysis of the coolant. The reliability analysis of the VVER-1000 fuel assemblies with higher initial enrichment (4.4 + 3.6%) and average burnup 42.7 *MWd/kgU* (maximal burnup 47 *MWd/kgU*) showed that the probability for work without defects is 0.99993 (for 47 *MWd/kgU* burnup).

6.3. Bibilashvili Yu. K. et al. Search for ways of reducing VVER fuel damage by optimising design and operating conditions. Proc. IAEA TCM on Fuel Failure in Normal Operation of Water Reactors, Dimitrovgrad, Russian Federation, May 1992.

The paper discusses the main directions in improving VVER fuel through upgrading fuel element and assembly design, optimising operating conditions in power reactor cores. The existent designs of fuel elements and assemblies are shown to meet the requirements of operation reliability. At the same time they are prone to further backfitting to significantly improve serviceability, maintainability, neutron economy, to reduce limitations on operation conditions and to increase burn-up and safety.

FAs can as a structural member of a core fail due to the following causes:

- fretting-wear in fuel spacing locations;
- fretting-wear due to debris getting into a fuel bundle;
- deformation and damage of elements due to inadequate accommodation of thermal and irradiation growth;
- deformation of absorber elements of ASS (automatic safety system) as a result of dynamic loads or scramming;
- damages due to a joint action of irregular handling and operation loads in a reactor;

Experience in VVER-440 and VVER-1000 operation shows that the chosen schemes of FA layout, the principle of fuel element spacing ensure the high operational reliability of FAs and their further upgrading.

VVER FAs are backfitted in the following main directions:

- reduction of parasitic neutron absorption through the use of zirconium grids and guide tubes, the use of zirconium of higher purity;
- use of integrated fuel burble absorber - gadolinium;

- increase of efficiency and life-time of control rods;
- maintainability of As (reconstitutable ones);
- increase the reliability of FAs for work under load follow conditions;

It follows from the world-wide experience and domestic investigations that the major factor that limit the fuel serviceability under transient operation conditions is the strength of zirconium claddings under the action of aggressive fission products in presence of tensile stressess. Therefore, to guarantee the strength the designs establish the maximum permissible depth of an accumulated crack that is less than critical depth of its stable growth (60 μm).

Presently regulations for fuels in our country and abroad established 50 μm for an as fabricated defect. Thus, there is only an insignificant margin to the critical depth of 60 μm.

To date in Russia the work is nearing its completion to introduce fuel cladding tubes having an as fabricated defect not more than 35 μm into a commercial production.

Further upgrading the fuel design is aimed at improving its performance to provide its operation to extended burn-ups (the FAs average discharge burn-up of 50 - 55 *MWd/kgU*) and under load follow conditions. At the first stage it is planned to employ bimetallic claddings with their inner surface coated with pure Zr. In future plastifiers containing fuel having a higher creep is suggested for use.

6.4. Dubrovin K. P. et al. Construction improvement of serial VVER-440 FR to extended burnup. *Atom. Energ.* **72**, p. 121 (1992).

Up to now VVER-440 are operated at three-year mode up to average burnup 29 - 31 *MWd/kgU*. Since the start-up of the 2nd unit of Novovoronezh NPP up to the end of 1990 about 44000 FAs, consisting of more than $5.5 \cdot 10^6$ FRs, are charged in these reactors. The relative part of untight FRs is about 0.01%. 650 FAs are experimentally operated during 4 years, 11 FAs during 5 years and 2 FAs during 6 years. After 4 years of operation 10 FAs (from 650) were found to be untight.

In this paper the constructional changes of the serial FR, as well as calculational and experimental proof of FR lifetime extension up to 4 years are given. In the improved FR the following changes are involved: higher initial helium pressure (0.5 *MPa*), using of fuel chamfered pellets, higher fuel enrichment (4.4%). The calculations of the thermomechanical FR characteristics for highest heat rated rod (maximum 325 *W/cm* in the 4th axial node) are performed, using the PIN-04M code. The calculational results for two most conservative variants (the first one is combination of maximal diametrical gap of 0.270 *mm* and minimal initial fuel density of 10.4 g/cm^3; the second one is minimal diametrical gap of 0.12 *mm* and maximal initial fuel density of 10.8 g/cm^3). Some qualitative results of PIE of FA, irradiated during 5 years at the Novovoronezh NPP 3rd unit at average burnup 49 *MWd/kgU* are presented.

The switching of VVER-440 reactor to four-year cycle leads to the decreasing the annually reloaded FAs from 117 to 90 per unit, as well as to decrease the natural uranium consumption with 11 - 12 % [2].

6.5. Yu. K. Bibilashvili, V. V. Novikov, Yu. V. Pimenov, V. N. Proselkov. Factors affecting VVER type fuel behaviour at extended burnup. Proc. IAEA TCM on the Fuel Performance at High Burnup for Water Reactors, Studsvik, Sweden, 5-8 June 1990. IAEA, 1991, p. 205-207.

VVER-1000 fuel has achieved a satisfactory level of operational reliability within burnup of about 43 *MWd/kgU*. It allowed the realisation of the next stage of research and development to achieve the burnup of about 55 *MWd/kgU* using burnable absorbers. Results of experimental studies include the possibility of operation of fuel to higher burnup. The PIE of test assemblies, irradiated in MR (the maximum burnup 76.4 *MWd/kgU*, the average one - 47.2 *MWd/kgU*) showed, that the fuel rod length increased on the average by 1.2 *mm* (initial length is 1 *m*) and the fuel rod diameter increased by 0.01 - 0.03 *mm*. The relative fission gas release was 30 - 45% (the initial helium pressure 2.45 *MPa*). The oxide film thickness at the cladding surface did not exceed 3 μm.

6.6. Berezhnij B. B. et al. Results of experimental FA batch examination in VVER-440 reactor. *Atom. Energ.* **69**, p. 122 (1990).

The VVER-440 reactors have good economic characteristics. But it is possible further to improve these characteristics applying extended fuel burnup by switching VVER-440 from three- to four-year fuel cycle, as well as by power increasing up to 1500 *MW*. To improve the reliability of the FRs operation under these conditions the initial He pressure in the gap between pellet and cladding is increased from 0.1 to 0.7 *MPa*. As a result the gap thermal conductivity is improved, the fuel temperature decreases, the FGR decreases with the burnup. So it is possible to reach the local FRs burnup of about 65 *MWd/kg*. With the aim to test the fuel operability of FRs with increased initial pressure in the gap under operational conditions at the Unit 2 of the Rovno NPP with VVER-440 reactor, from 1985 to 1988 an experimental FA batch examinations were carried out. The pilot batch consisted of 96 FAs with initial enrichment 3.6% U-235 and initial He pressure 0.5 ÷ 0.7 *MPa*. The pilot FRs had the same construction, as the standard ones, except the initial gas pressure. The pilot FAs batch was under operation during 952 EFPD: 239.3 EFPD - in the fourth cycle, 401.6 EFPD - in the fifth cycle, and 301.1 EFPD - in the sixth cycle. The average burnup of the discharge fuel was about 34.7 *MWd/kg* (nearly 18% more than the design value). After the fourth fuel cycle during the reactor shut down, examination of the cladding tightness was performed. There were not any defective FRs. The I^{131} activity was about 10^{-8} *Ci/l*. The averaged over the pilot FA batch value of energy production was with a value of 6 *MWd/kg* more than the design value. So the FRs with initial gas pressure between 0.5 ÷ 0.7 *MPa* can be recommended for use in the VVER-70, -365, - 440 reactors with design fuel cycle.

6.7. Goncharov et al. Test of experimental VVER-1000 fuel rods in the MR reactor. *Atom. Energ.* **62**, p. 312 (1987).

The first VVER-1000 fuel reloads were designed for a two-year cycle mode of operation with 28.5 *MWd/kgU* average burnup. The FRs, used in these reactors, had high operability and reliability (less than 0.02% untightness). Since 1984 first the 5th unit of Novovoronezh NPP is switched to the three-year fuel cycle with average burnup 42.6 *MWd/kgU*. Presently all VVER-1000 units are in process of switching from two- to three-year fuel cycles with average burnup of discharged fuel more than 40 *MWd/kgU*.

The construction of FRs for three-year mode of operation was improved as follow: bigger central pellet hole, decreased initial helium pressure in the gap, increased free volume in the FR, using of chamfered fuel pellets, higher fuel enrichment. The FR characteristics for two- and three-year fuel cycle are given. Bellow some of them are shown:

Parameter	2-year cycle	3-year cycle
Diametrical gap, *mm*	(0.19-0.32)	(0.15-0.26)
Central pellet hole diameter, *mm*	(1.4-1.6)	(2.3-2.2)
Initial helium pressure in the gap, *MPa*	(1.95-2.45)	(1.75-2.25)
Fuel enrichment, %	3.3	4.4

To develop the VVER-1000 FRs a series of constructional and technological works, including the examinations at MR reactor are performed. In this paper some results of such MR examinations are given. The pilot FRs, are different from the standard ones only by the fuel stack length (1000 *mm*). The pilot FAs are consisting of 12 FRs. The FAs are tested at the basic VVER-1000 regime, except one (the 25th FA), which is tested under load follow conditions. The total number of tested FAs was 25, consisting of about 300 FRs. The investigations performed are aimed to improve the construction, as well as to proof the possibility of burnup extending. The experimental data of 13 tested FAs (with burnup equal or more than the design one) are given. Metallographic investigations revealed that the oxide layer thickness on the outer surfaces of the claddings was less 5 μm. The highest hydrogen content in the cladding was 0.008 *wt*%.

The mechanical characteristics of the irradiated FR claddings are given too. The data show, that the irradiation cause an increase of strength cladding characteristics of about 50%.

To study the influence of PCI on the FR operability at the transient regimes the 25th FA was tested under power cycling conditions. The power has changed in the interval of 50% of the nominal one, with the power change speed of about 20 *Wt/cm.min*. The FRs were intact. The FGR in the FR gap was about 50%. A phenomenon of significant decreasing of power, at which fuel melting occurs, was found. Big structural changes of UO_2, including the melting, are observed at the linear power 600 - 700 *W/cm*.

7 Other Papers

7.1. K. V. Simonov, I. Nemes, Computational and experimental validation of VVER-440 fuel pin reliability in load-follow operation of unit 2 of NPP "Paks". Proc. 3rd Symp. AER, Slovakia, Piestany, 27 Sept. - 1 Oct. 1993.

The results of computational and experimental investigations of VVER-440 fuel pin reliability during three-year fuel cycle in load-follow operation of Unit 2 of NPP Paks, Hungary, are presented. The computational validation was made for the traditional scheme of fuel utilisation (out-in-in), as well as for low leakage loading pattern. The "old fuel" (with initial helium pressure 0.1 *MPa*), as well as the "new fuel" (with initial helium pressure 0.7 *MPa*) reliability in load-follow operation was validated. "New fuel" reliability in load-follow operation regimes was investigated according to the different criteria: thermohydraulic, thermophysical and strength criteria.

The investigations accomplished show that fuel pins have necessary strength reserve to be used in load-follow regime during three fuel cycles.

7.2. Saprykin V. V., A. N. Novikov. Characteristics of four-year fuel cycles for VVER-440 reactors with 349 and 313 fuel assemblies in reactor core. Proc. 3rd Symp. AER, Slovakia, Piestany, 27 Sept. - 1 Oct. 1993.

Results of calculational and experimental investigations realised during recent years, improving technology of fuel assemblies manufacturing and FA operational experience demonstrate possibility of four-year fuel cycle introduction into operation of VVER-440 reactors. At present such fuel cycles are developed for reactor cores both with 349 and 313 FAs. Results of calculations show that introduction of four-year cycles would allow to reduce the following indices:

- manufacturing, transportation and processing/storage needs by 23 - 31%;
- effective consumption of natural uranium by 11 - 15%;
- volume of separation work by 6 - 11%.

7.3. Sidorenko V.D, Scheglov A.S. Neutron-physical characteristics of VVER reactor core, affected on the FR performance. *Atom. Energ.* **74**, p. 533 (1993).

In this paper calculational results of some important neutron-physical characteristics, such as radial power distribution in FR, point burnup (at given FR axial layer) and isotope composition (U-235/Pu-239, %) at average FA burnup 48 *MWd/kgU* are presented. These characteristics are calculated, using the TBC code. The results for VVER-440 FR (with 3.6% enrichment, 46 *MWd/kgU* burnup), as well as for VVER-1000 FR (4.4% enrichment, 47.6 *MWd/kgU* burnup) and the results for VVER-1000 FR with burnable absorber (3.6% enrichment with 8% mass Gd_2O_3 from FA, consisting of FRs with 4.4% enrichment, 40.2 *MWd/kgU* burnup) are given, too. The results show, that the burnup in the thin periphery fuel pellet layer is very big (about 100 *MWd/kgU* at average over given axial layer burnup 48 *MWd/kgU*). A relation between fast neutron flux density and FA average linear rating, including FA average burnup, for VVER-440 FR, VVER-1000 FR and VVER-1000 FR with gadolinium is given.

The results obtained were included in the PIN-mod1 code.

7.4. Simonov K. V., Lushin V. B. and Pytkin U. N. Transformation of VVER- 440 on 5 year fuel cycle. *Atom. Energ.* **73**, p. 168 (1992).

The experiment is carried out in Kola NPP. The results of the experiment using 4 year fuel cycle with 4.4% enriched fuel and achieved average burnup 43.4 *MWd/kg* and maximum one - 46.2 *MWd/kg* allow to continue experiments on the third reactor of Kola NPP with a 5 year fuel cycle. 12 fuel assemblies 4.4% enriched with burnup 42.0 *MWd/kg* are used for the work during the 5th year. All of them are with high He pressure up to 0.5 *MPa* under the cladding.

Near the end of the 5th year their duration time in the reactor core is 1567 eff d. The burnup is 49.2 *MWd/kg*, the maximal average burnup in the most stressed cross section is 56.4 *MWd/kg* (in every fuel assembly), and the maximal dept of burnup in the most stressed cross section of the fuel element is 62.6 *MWd/kg*.

The power peaking factor in the core is within the planed limits. The maximal temperature in the fuel centre is reached in the beginning of the 1st year - 936°C and the minimum power is 192 *W/cm*. At the end of the process the temperature decreases to 600°C. Fission gas release under the cladding is insignificant. In the worst combination of technological parameters the pressure under the cladding is 3 *MPa*. The calculations show a possible contact between the fuel and the cladding at 310 eff.days. As a result the inner radius of the cladding will be enlarged with 29.6 *mm* but the functioning of the fuel elements is preserved. The maximum temperature of the cladding reaches 304°C.

As a result can be concluded that the fuel elements 4.4% enriched with He pressure of 0.5 *MPa* under the cladding can also work 5 years for VVER-440.

7.5. Proselkov V.N. and Simonov K.V. Experiments on VVER - 440 fully loaded with fuel of increased enrichment. VANT, *Physika Iadrenih Reactorov*, No. 4, p. 55, (1992).

The results of VVER-440 experiments to determination the physical parameters on activity core VVER-440 fully loaded with fuel of increased enrichment 4.4%.

The calculated results [1 - 3] are compared with experimental data [4]. A good coincidence for the temperature coefficient of radioactivity (1.0 calc, 0.96 exp), effectiveness boric acid, the control rods and the mechanical in condition operating reactor, are shown.

The power reaches the design value at 3 stage: 50, 90 and 100% of the nominal power. Released energy on the activity core is investigated during every stage. On the stage 50% power, is made to control the capability of reactor to self-control. Measured values of the coefficient of radioactivity:

$\alpha_T = -2.8\ 10^{-2}$ %/°C; $\alpha_N = -1.05\ 10^{-3}$ %/*MW*.

Good coincidence with the calculated results:

$\alpha_T^{calc} = -2.88\ 10^{-2}$ %/°C; $\alpha_N^{calc} = -1.19\ 10^{-3}$ %/*MW*

The peaking factors K_q^{max} of assembly, $(K_q * K_k)^{max}$ of fuel rod, $(K_q * K_z)^{max}$ - volume of core, answer the requirements for safety. When the control rods are on the altitude 2.35m the factors have the following values:

$K_q^{max} = 1.50$, $(K_q * K_k)^{max} = 1.73$ and

$(K_q * K_z)^{max} = 2.12$.

7.6. V. V. Yakovlev et al. Incentives for extended burnup for VVER reactors. Proc. IAEA TCM on the Fuel Performance at High Burnup for Water Reactors, Studsvik, Sweden, 5-8 June 1990. IAEA, 1991, p. 24-32.

In the paper steps taken to increase burnup in VVER-440 and VVER-1000 reactors are discussed. Economic incentives to burnup extension above 45 *MWd/kgU* are also analysed. The results of economic burnup optimisation for VVER-440 with 12 and 18 month cycles and VVER-1000 with 12 month cycles up to 65 *MWd/kgU*, performed within the framework of IAEA WREBUS activity, are presented. Main characteristics of the VVER cores as well as the economic conditions used in the study are given. Study of sensitivity to economic parameters is also mentioned. The analysis performed shows reasonable incentives for burnup extension as high as 60 *MWd/kgU*. The optimum burnup is most sensitive to backend costs. Possible burnup limiting factors are also discussed.

7.7. Novikov V.V. Mechanism of zirconium alloy cladding iodine cracking. *Atom. Energ.* **71**, p. 33 (1991).

In this paper an analysis of the iodine stress corrosion mechanism, accounting for cladding stress-strain state characteristics in the top of the growing crack and effect of this state on the iodine diffusion in the destruction zone is performed.

7.8. Proselkov V. N. et al. Exploitation increase of VVER-440 to 18 months. *Atom. Energ.* **71**, p. 466 (1991).

In order to increase the effectiveness of NPP an operational regime with increased operation of 18 month is proposed

The increased operation can be achieved using highly (3.6%) enriched fuel but its average burnup is reduced in the process. For example in a stationary burnup cycle with continuity of 357 eff. days and with 126 loaded 3.6% enriched fuel assemblies the average burnup of U is 23.5 *W/kg*.

In table 1 the main characteristics are given of the projected loading (variant 1), operation of the fuel with every year loading of the fuel assemblies (variant 2), 4 year fuel cycle (variant 3), 2 fuel cycles with increased energy release between the loadings (variant 4 and 5).

If at every loading 25% fuel assemblies are loaded with 4.4% enriched fuel (variant 3) instead of 21% (variant 2), this will prolong the operation from 306 to 348 effective days. Loading of 1/3 fuel assemblies allows 434 days of operation but the burnup is little reduced (variant 4).

The prolonging of the reactor's work at nominal parameters up to 452 eff.days can be done by increasing the number of the reloaded fuel assemblies up to 120 (variant 5):

- 120 fuel assemblies with 4.4% enrichment and with 3 cycles of operation.
- 18 fuel assemblies with 3.6% enrichment of which 11 are in the reactor core 2 cycles, and 7 remain for the 3rd term.

An additional prolonging of the cycle up to 30 eff.days is possible; 7 eff.days form the released reactivity at decreased average temperature of the moderator, and 23 eff. days at the cost of arbitrary power decrease to 80% of the nominal. The work continuity of the reactor with nominal parameters decreases to 440 eff.days.

In the case of reloading for the third time of fuel assemblies with 4.4% enrichment every half year their total staying period in the reactor core is increased to 1300 ÷ 1400 eff.days.

At the Kola NPP safe operation of 36 fuel assemblies with enrichment of 3.6% is registered for a period of 1315 days (3rd reactor 1 ÷ 4 cycles).

In the Novovoronezhkata NPP (3d reactor) a fuel assembly with 3.6% enrichment and 1507 eff. days (5 years) stay in the reactor core reached average burnup 49 MW day/kg. The assembly has been removed from the reactor core without leaking.

In 1991 on the 3d reactor of Kola NPP 12 fuel assemblies with 4.4% enrichment reach power release of 48 *MWd/kg* working in the reactor core 1500 eff.days (5 cycles) with preserved hermetic condition.

7.9. Proselkov V. N., Simonov K. V., Pytkin U. N., Lobov V. I. and Panin *M.* V. Some aspects increasing the effectiveness of the burnup cycle of VVER-440. VANT, *Physika Iadrenih Reactorov*, No. 1, p. 56, (1991).

Several possibilities for increasing the efficiency of the fuel cycle are shown.

A scheme for fuel reshuffling, at which the new FAs are loaded in the central part of the reactor core, and the highly burned FAs are placed in the periphery.

This allow to decrease the amount of escaping fast neutrons from the reactor and to increase the burnup.

Such a regime has been realised in the Kola NPP during the 4th reload on the 3rd reactor. At the beginning of the 4th year 36 fuel assembles with 3.6% enriched fuel are placed from the centre to the periphery of the reactor core. At the end of the 4th year the burnup of these assemblies reaches in average 38 *MWd/kgU*.

Reshuffling of the used assemblies decrease the fuel expenses by about 0.2%.

The experiment has been performed in the Kola NPP when at first loading of the 4th reactor were used 60 burnt assemblies and 12 assemblies with 1.6% enrichment which were already burnt 1 cycle (384 eff.days) on the third reactor.

The average burnup of the fuel assemblies at the loading moment in the 4th reactor has been 12.4 ± 0.4 *MWd/kg*. The average burnup of 1.6% enriched assemblies were 18.5 *MWd/kg*. In such a manner the cycle expenses were diminished with 19.8%.

In the 4th loading of the 4th reactor also 12 assembles (enrichment 2.4%) burnt already 1st (253 eff.days) and 2nd (148 eff.days) cycles were placed in the reactor core.

Increasing the period of the time between loadings.

This regime was applied during the 3rd cycle on the 4th reactor of Kola NPP. The cycle duration was enlarged up to 420 eff.days.

The dependencies on reactivity of the fuel charge from the moderator temperature and the boric acid concentration in the first contour. The measurements of the temperature coefficient of the reactivity were performed in the dynamic regime at moderator temperatures within $140 \div 260°C$. The experimental data were given in tables $1 \div 3$.

The BIPR-7 numerical programs for calculations of the neutron - physical characteristics were used.

The calculations are compared with the experimental data on figures 1 and 2.

7.10. Proselkov V.I. et al. Power increasing of NPP with VVER-440. *Atom. Energ.* **69**, p. 79 (1990)

Presently NPP power increase is applied as an economically effective way to increase the electricity production.

The basic characteristics of VVER-440 reactor core (such as fuel lifetime (EFPD), power in % from the nominal one, maximal FA power, maximal FR linear rating) for Kola and Rovno NPP are given. On the basis of the data presented it can be concluded that the VVER-440 reactor core characteristics at power 107% from the nominal one satisfy the basic design demands. Power peaking factors do not exceed the limits.

7.11. Proselkov V. N. et al. Transition fuel cycles neutron-physical characteristics of VVER-440 with an increased loading fuel enrichment. Physika ladernih Reaktorov, No. 4, p. 52 (1989).

An experiment at Kola NPP with fuel enrichment 4.4% and minimum controlled power level to determine the basic neutron-physical characteristics of VVER-440 and a comparison with the calculated results are presented.

Determined are the differential effectiveness of the boric acid, the integral and differential characteristics of control rods at a temperature of moderator 260°C, moderator and fuel temperature reactivity coefficients. The results are compared with calculated data. The experimental data have a good coincidence with the calculated ones and require no limitation in the operation conditions.

The differential effectiveness of the control rods is an exception. The maximum differential effectiveness of control rods is in the range 190 - 210 *cm* to the base of activity core. The experimental value is 20 - 25% lower than the calculated one, the high speed effectiveness exceed the design value ($0.07\ \beta_{eff}/s$).

The fuel assemblies with 4.4% enrichment of the fuel operate 3 fuel cycles and reach a burnup 35 *MWd/kgU*. The fuel failures were not observed. The mean value of activity of the moderator does not exceed the value $2 \cdot 10^{-5}$ Ci/l

7.12. Astahov S.A. et al. Optimisation of the VVER reactors core for improving the efficiency of fuel utilisation and reactor safety. VANT, *Physika ladrenih Reactorov*, No. 2, p. 71, (1988).

The situation in the field of organisation of the fuel cycle for VVER reactors is considered. The operation parameters of reactor core for VVER and the fuel are given.

Fuel operation analysis of the reactor core VVER-1000 - 5th unit Novovoronezh NPP is carried out. This unit operates through a 3 year fuel cycle with fuel enrichment 4.4%.

In steady state operation conditions the mean burnup is 42.6 *MWd/kgU*, this value exceeding the design value 40.0 *MWd/kgU*. The maximum burnup for approx. 1000 eff.d. in some assemblies is 48.95 *MWd/kgU* for VVER-440 (Novo-voronezh NPP, Kola NPP, Armenian NPP).

The Kola NPP operates with 4 partial reloads for full fuel cycle with design fuel enrichment. One of the units operates in the steady-state condition with mean burnup approx. 36 *MWd/kgU* and a maximum burnup - 39.7 *MWd/kgU*.

The fuel rods with increased pressure (0.5 *MPa*) - improved rods - were tested at Rovenska and Kola NPP.

The calculations carried out demonstrate a good reliability of fuel rods to burnup approx. 55 *MWd/kgU*.

Investigation is carried out to create improved fuel assemblies with Zr grids and guided tubes profiled in radius and height.

It is necessary to use fuel with uranium-gadolinium for decrease of neutron flux on the reactors body.

The improved fuel rods and assemblies are required to increase the reactors VVER type safety. A proposal is made to increase the number of control rods from 61 to 121.

7.13. Solyanij V. V., Yamnikov V. S. Assessment of zirconium alloys loading capacity. *Atom. Energ.* **48**, p. 73 (1980).

A theoretical method for this assessment is described. The calculational results for cladding limiting stresses are presented. A comparison between calculational and experimental results is given.

7.14. Shatskaja O. A. et al. Influence of irradiation on the destruction resistance of Zr - 2.5% Nb alloy. *Atom. Energ.* **47**, p. 18 (1979).

To study the changes of strength characteristics of zirconium alloys during the reactor operation at the first unit of Belojarsk NPP a pilot technological channel was irradiated (1100 EFPD). This channel included a Zr + 2.5%Nb alloy tube. The basic irradiation characteristic of this channel are given. The experimental temperature and fluence dependencies of critical crack opening are showed. A method for calculation of the critical crack opening is presented. The calculational results are given.

7.15. Yamnikov V. S., Malanchenko L. L. Thermal conductivity of gaseous mixture under the FR cladding and its change during burnup. *Atom. Energ.* **42**, p. 322 (1977).

A method for thermal conductivity determination of gaseous mixture in the gap is described. Calculational results for three-component mixtures (He - Kr - Xe and Ar - Kr - Xe) are compared with the experimental ones. For VVER-1000 FR the gap thermal conductivity dependence on initial gas filling, FGR ang burnup are shown.

7.16. Ageenkov A. T., Saveliev V. F. Hydrogen Destroying Impact On Cladding At VVER Fuel Rod Regeneration. *Atom. Energ.* **37**, p. 58 (1974).

Results of hydrogen destructive effect on Zr alloy cladding at fuel rod regeneration are obtained. It is shown, that the effective interaction between the alloys and hydrogen begin at 600°C; at 700 - 800°C for 1h the Hydrogen ratio in the alloy become close to its boundary value: 1.7 - 2.0 *mass%*. Absorption of hydrogen from tube like claddings enlarge its diameter. The composition ZrH 1.8 - 2.0 characterise with low firmness (s < 10 kg/mm^2) and fragileness, high micro hardness (300 - 310 kg/mm^2) and low resistentness in oxygen and air atmosphere up to 700°C. The structure of hydrogen containing alloy is explored by radiographic methods.

See also abstracts:

1.1

2.2, 2.8, 2.10, 2.13, 2.14, 2.19, 2.20

3.1, 3.5

4.10, 4.12

5.3, 5.5, 5.6

Експлоатация, моделиране и експериментално изследване на горивото за реактори ВВЕР

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VVER Reactor Fuel Performance, Modelling and Experimental Support

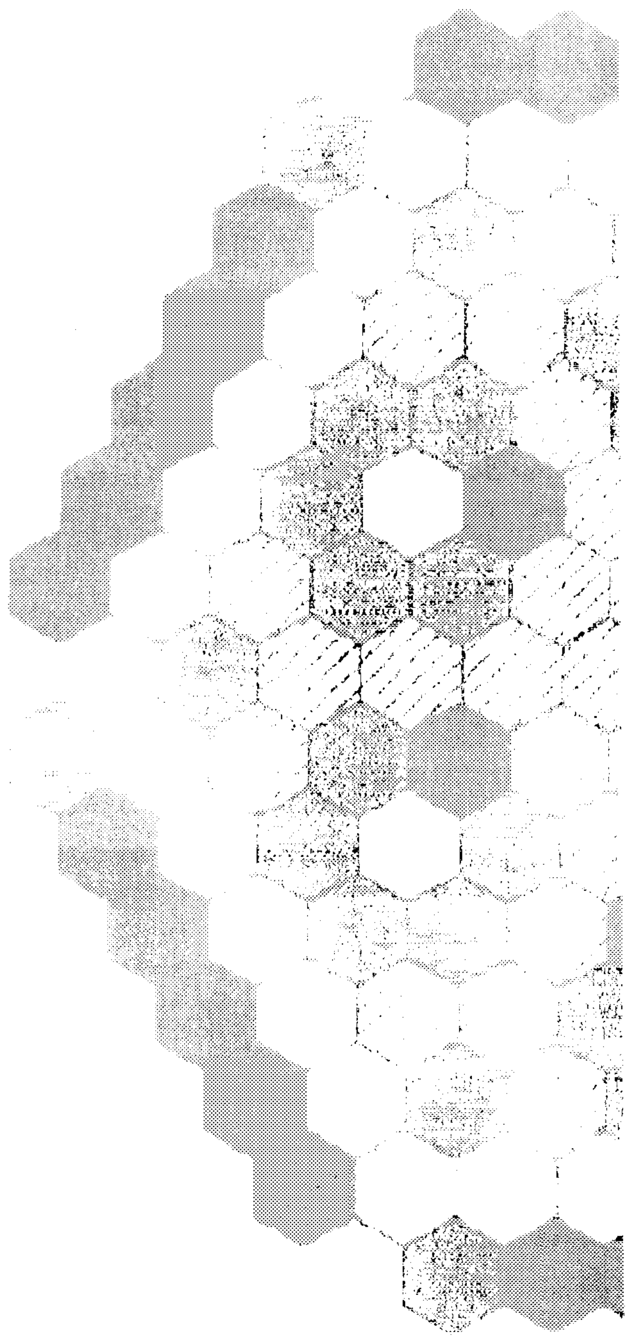
Proceedings of an international seminar, held in St. Constantine, Varna, Bulgaria, on 7 - 11 November 1994

With more than 30 operating VVER units and more than 10 under construction, the VVER fuel market is an important part of the world nuclear fuel market. Once supplied solely by Russia, it is now open to international competition, not that much in the purely financial field, but rather in the field of advanced fuel design and utilisation schemes.

The fuel pellets and cladding are the first barriers to accidental radioactive release, hence another important issue is at hand, considering the general interest in VVER reactor safety.

The international Seminar on VVER Reactor Fuel Performance, Modelling and Experimental Support, organised by the Institute for Nuclear Research and Nuclear Energy of the Bulgarian Academy of Sciences in co-operation with the International Atomic Energy Agency, attracted 70 participants from 16 countries, including representatives of all major Russian plants and institutions responsible for VVER reactor fuel research, design and manufacturing.

The contributions in this book cover (1) VVER fuel performance and economics, (2) fuel behaviour modelling and experimental support, (3) licensing issues and (4) problems of VVER spent fuel - that is, most of VVER fuel life-cycle stages. In addition, a comprehensive collection of abstracts of published bibliographic sources of VVER fuel data is included in the Supplement.



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